An Evaluation of Direct Gamma Dose at the Site Boundary of the Vermont Yankee Nuclear Power Station

FINAL REPORT

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Oak Ridge Associated Universities (ORAU) is a university consortium leveraging the scientific strength of major research institutions to advance science and education by partnering with national laboratories, government agencies, and private industry. ORAU manages the Oak Ridge Institute for Science and Education for the U.S. Department of Energy.

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*** Disclaimer ***

In the course of preparing this report, ORAU scientists, Jeff Chapman, CHP, PE, and Alex Boerner, CHP, provided an independent review and assessment of the means to measure and report site boundary dose from direct gamma-ray radiation at the Vermont Yankee Nuclear Power Station (VYNPS). Beginning in January of 2006, ORAU staff met with engineers, scientists, and program managers from the Vermont Department of Health (VDH), VYNPS, AREVA NP, Inc. and other independent consultants. Mr. Chapman and Mr. Boerner conducted a site walk-down of unrestricted areas outside the property of the nuclear plant, collected available applicable documents, and reviewed historical records. ORAU evaluated summary reports and analyses, examined prior agreements between VYNPS and the VDH, and conducted telephone interviews in support of the report's factual accuracy, and provided recommendations and conclusions on this basis.

ORAU considers this a third-party, independent technical evaluation; however, in all cases, it was beyond the scope of work to:

1) Collect independent measurements on the parameters of interest; and

2) Request raw measurement data for analysis or re-analysis.

To do so would have required a substantial effort to deploy instruments, collect measurement data, reduce the data, and analyze it.

ABOUT THE COMPANY AND THE AUTHORS

Oak Ridge Associated Universities (ORAU) is under contract to the State of Vermont Department of Health to evaluate direct gamma dose at the site boundary of the Vermont Yankee Nuclear Power Station. ORAU is an institution initially formed in 1946 as the Oak Ridge Institute for Nuclear Studies (ORINS) to advance scientific research and education through partnerships with academia, government, and industry. ORAU has made significant contributions in areas emphasizing radiation medicine, science education, and professional training. The two individuals who performed this evaluation, Mr. Alex Boerner and Mr. Jeff Chapman, are certified health physicists (CHP), both with advanced, post-graduate degrees in the nuclear and biological sciences. Mr. Chapman is also a professional engineer (PE). Combined, the authors have over 55 years of experience in designing and implementing radiation safety programs, measurement methods, and public health plans for nuclear facilities and operations around the world. Both authors performed this review objectively, on the basis that an impartial analysis was to be performed from the outset. Author qualifications are provided in Appendix F. Additional general information about ORAU is located on the internet at http://www.orau.org/.

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We gratefully acknowledge the support and efforts of several individuals: from the State of Vermont Department of Health—Dr. William Irwin, Ms. Carla White, and Mr. Larry Crist; from the Vermont Yankee Nuclear Power Station—Mr. Robert Wanczyk, Mr. Samuel Wender, Mr. Norm Rademacher, Mr. David McElwee, Mr. Steve Skibniowsky, Mr. John Dreyfuss, and Ms. Candy Sak; from VYNPS consultants—Mr. Mark Strum, Mr. Jeff Raimondi, Mr. David Keefer and Mr. James Giard; and lastly, Mr. Ray McCandless, the retired former Radiological Health Chief with the Vermont Department of Health. We recognize these people for having provided historical context for this task, the technical documents describing direct gamma measurements conducted at the site boundary, passive TLD monitoring results, and the implementation of methodologies to estimate site boundary dose in real time using the main steam line radiation monitors. In addition, we acknowledge the discussions with instrument calibration and dosimetry scientists at Oak Ridge National Laboratory, Pacific Northwest National Laboratory, Argonne National Laboratory, and the Environmental Measurements Laboratory. We also recognize ORAU staff members who assisted with internal technical and quality assurance reviews of this document—Mr. Eric Abelquist, Mr. Matt Buchholz, Mr. Scott Kirk, and Ms. Ann Payne.

In 2006, four meetings were held in Vermont, including an introductory meeting with the State of Vermont in Burlington, and three technical review meetings in Brattleboro with State of Vermont and VYNPS staff, and VYNPS consultants. In the course of preparing this report, extensive communication transpired to understand the fundamental principles of the various monitoring methods and results. It is because of these people and the result of their efforts that ORAU is able to make recommendations for programmatic future improvements to the monitoring and compliance program for direct gamma dose.

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EXECUTIVE SUMMARY

In December of 2005, Oak Ridge Associated Universities (ORAU) entered into a contract [A] with the State of Vermont to review and evaluate whether Entergy Nuclear Operations Inc. Vermont Yankee Nuclear Power Station (hereafter VYNPS) is compliant with existing annual, direct gamma dose objectives at the site boundary of the plant. The direct gamma dose annual objective and quarterly compliance requirements are written in the State of Vermont regulations [B, C] for protection against ionizing radiation.[1] The Vermont Department of Health (VDH) is the group within the state regulatory authority responsible for Part 5, Chapter 3, Subchapter 1, §5-305(B)(1)(e).

Direct gamma dose at the site boundary is produced primarily by the decay of ^{16}N in the turbine building and as a result, ORAU's task principally involved an evaluation of the monitoring of gamma-rays from ^{16}N production in the VYNPS Boiling Water Reactor (BWR). These high-energy gamma-rays produce what is known as skyshine radiation at the site boundary. The secondary purpose involved an examination of all other direct gamma source contributors, how they are accounted for and whether the proper measurement devices and processes are used to measure this environmental compliance quantity.

The motivation for this independent review was two-fold:

1. A thermoluminescent dosimeter (TLD) that the VDH had deployed at Dosimetry Record location 53 (DR-53) near the west-side boundary recorded a quarterly result in 2004 that was 2 millirem (mrem) above the reporting action limit of 10 mrem. VDH initiated an inquiry.

2. VYNPS had submitted to the U.S. Nuclear Regulatory Commission (USNRC) a request for a 20% power uprate.[2] Given that the ^{16}N production rate increases as a function of reactor power, VYNPS expressed an interest during the planning phase to work with VDH to ensure that the state regulatory requirements could continue to be met.

The VYNPS is required to ensure that the site boundary dose from sources of direct, gamma-ray radiation does not exceed 20 millirem per year. As a proactive measure, when any quarterly dose exceeds 10 millirem, the State of Vermont and VYNPS investigate this impact on satisfying the annual dose objective and identify any unexpected sources of radiation that would cause an annual dose in excess of 20 mrem. The intent is to ensure specific countermeasures are taken on a quarterly basis. The regulation states furthermore that the site boundary annual dose objective is intended to be equivalent to a total body dose of 5 millirem to an individual member of the public located in an unrestricted area.[3]

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[1] State of Vermont, Department of Health, Part 5, Chapter 3, Subchapter 1, §5-305(B)(1)(e). (Available from http://healthvermont.gov/regs/radio_health.pdf)

[2] VYNPS License Amendment 229 (for DPR-28) regarding Extended Power Uprate is posted on the USNRC website at http://adamswebsearch2.nrc.gov/idmws/ViewDocByAccession.asp?AccessionNumber=ML060050024)

[3] VDH needs to issue specific guidance to VYNPS on how the annual dose objective is used and the assumptions implicit in the 5 mrem equivalency. ORAU addresses these issues in Appendix A.

In March of 2005, the VDH informed VYNPS management that measurement results from a State TLD collected in the fourth quarter (Q4) of 2004 at DR-53 exceeded the quarterly reporting limit of 10 millirem. The VDH result was 12 mrem. Furthermore, when the fourth quarter result from VDH was added to the prior three quarters, the annual dose at DR-53 was 24.9 millirem for 2004. The result was provided by VDH, which by their method assumes that the TLD response for exposure and dose equivalent is equal. The 12 mrem result in Q4 of 2004 was the result of subtracting the average of two TLDs used to estimate background. Both background results were noticeably lower values than "historical averages" and thereby increased the net reported response.

Contrary to the VDH TLD results, the VYNPS measurement systems reported neither quarterly nor annual doses that would precipitate a compliance issue. VYNPS believed plant operating conditions resulted in no direct gamma ray dose in excess of the applicable limits at the site boundary and that variability in natural background radiation levels caused the discrepancy between VDH and VYNPS results. Nonetheless, the plant initiated corrective actions to address the State's concerns.

ORAU was contracted to evaluate this discrepancy, the methods and processes used to demonstrate compliance, and to provide recommendations for ensuring that future measurement and notification processes make the compliance process more complete, accurate and effective. The ultimate objective is to continue to protect public health and safety during plant operations. This report presents the results of ORAU's evaluation. Issues of interest encountered and general conclusions reached during this effort are summarized as follows:

1. Review of historical memoranda between the VDH and VYNPS identified no clear and rigorous procedure or agreement for VYNPS to demonstrate compliance with the direct gamma annual dose objective or quarterly limit. Over the years, new terms and definitions were created, personnel turnover occurred, site boundary conditions and locations changed, and measurement methods were modified. The changes were not clearly captured in lines of communication between and among VDH and VYNPS staff. As a result, ORAU believes an uncertainty developed over roles and responsibilities in the context of how results would be reported, and ultimately, how compliance would be demonstrated. Even as late as 2005, the State asked VYNPS how it would monitor for site boundary dose, and at the same time required VYNPS to accept as a premise that the State would monitor for compliance.[4]

   ORAU believes that the State of Vermont annual dose equivalent objectives have not been exceeded, but it would be prudent to take additional action to ensure this will hold true going forward.[5] Specifically, better measurement capability is needed and more refined implementing procedures need to be developed to clearly describe the conditions for which compliance is demonstrated. Much of this effort is administrative and can be treated within a more comprehensive communication plan

[4] VDH letter to VYNPS (Paul Jarris to Jay Thayer), "Measurement of Fenceline Radiation Dose," March 11, 2005.

[5] With the Extended Power Uprate (EPU) and a projected increase of between 26-30% in ^{16}N production, the annual dose equivalent at the site boundary is expected to increase by this percentage.

and agreement between VYNPS and VDH on overall methodology, exchange of data, and "process control/management." Many of the engineering and measurement design principles that guide this type of decision making for environmental compliance are found in the US Environmental Protection Agency (USEPA) publication on Systematic Planning for the Data Quality Objectives process (EPA QA/G-4).[6] The DQO process forces both VDH and VYNPS to understand the compliance points, the measurement process, and the required quality in the measurement process to achieve reliable decision making. The process is used to establish performance and acceptance criteria, which serve as the basis for designing a plan of sufficient quantity and quality to support goals. Once this systematic process is conducted, VDH and VYNPS will agree with ORAU, that no single environmental TLD measurement alone can achieve the level of quality necessary to make environmental compliance point decisions.

2. Plant management and the VDH have had significant discussions on site boundary dose (or dose to the public in an unrestricted area). Traditionally, the emphasis has been on effluent monitoring, but it is clear that as early as 1969, the State oversight office suggested that in order to measure air submersion dose, the dose from gaseous emissions from the plant, a TLD system should be considered. Neither the State nor VYNPS ever believed that sustaining an annual dose objective of less than 20 mrem was difficult to achieve after confirmatory measurements were conducted in 1978, following the installation of a shield wall in 1976. As a result, the annual dose objective did not receive a significant amount of attention until the 2004 dosimeter reporting cycle, when disparate dosimetric results were obtained. VYNPS was very attentive to the issues in 2005 and immediately responded. In addition, because of the anticipated Extended Power Uprate (EPU), VYNPS designed and installed a new turbine shield in May of 2006, to add an additional level of assurance that the dose limits would not be exceeded after the EPU was approved and power levels increased.

   This report addresses this issue and provides recommendations for implementing a comprehensive monitoring program that relies on a graded approach, consistent with ALARA principles: 1) real-time measurements from the Main Steam Line Radiation Monitor method (MSLRM)—as described in the Off-site Dose Calculation Manual (ODCM) Revision 30—should be used to measure and report the greatest fraction of site boundary dose generated from ^{16}N decay. The recalibration measurements of the High Pressure Ion Chamber (HPIC) versus MSLRM rate should be witnessed by VDH; 2) VYNPS should provide evidence that "other source terms" are evaluated and updated on a more frequent and routine basis. This does not need to be a significant effort (because the fractional dose contribution is small) but it needs to provide a sufficient level of rigor to assure the State that it is being performed in a reliable fashion; 3) a passive environmental dosimetry program needs to be implemented by VDH that provides assurance that no significant change to "normal" predicted site boundary dose is exceeded. Under the DQO process, the

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[6] The DQO process document is available online at http://www.epa.gov/QUALITY/qs-docs/g4-final.pdf

TLD measurements results may be treated statistically under process control methods.

The implementation of this refined and improved program will take effort because the net dose contribution of 20 mrem per year is about 20% of natural background radiation (neglecting exposures from radon decay products). The approach for demonstrating compliance should be incorporated into an Enhanced Communication Plan (ECP). In addition, the MSL radiation monitors must be recalibrated, an evaluation of other non-^{16}N produced gamma-ray source terms contributions must be performed more frequently, and statistical methods must be used for the passive dosimetry results to ensure outliers are detected and that the false positive rates (and false negative rates) are acceptable.

Over 90% of the total site boundary dose is from ^{16}N production in the reactor. Fortunately, this source term is predictable and slow to change. When calibrated properly, MSLRM detectors "monitor" in real-time the major component of the site boundary gamma-ray flux and perform the monitoring regardless of how natural background is changing daily. The remaining 10% of the site boundary dose is project dependent, varying with time (day to day, week to week). The methods and procedures currently employed to update and assess this "other" component need to be improved to assure VDH that these processes are accounted for on a more reliable basis. Any measurement method, whether TLDs or otherwise, that rely on two separate measurements to make a net result, need to make use of statistical methods for decision analysis. Methods need to be developed to ensure the integrity of the data is consistent and representative. An example of this measurement process is concurrent TLD measurements at two separate locations: one is a gross measurement at the site boundary and the second, a background measurement in a location(s) deemed the surrogate background location. It is very likely that had better measurement methods and statistical analyses been in place in the fourth quarter of 2004, ORAU's involvement in this work effort would not have occurred.

3. The State of Vermont regulation on radiological protection should be updated. It pre-dates the USNRC regulations. Most importantly, the inconsistencies between the state regulation and the federal regulation cause unnecessary confusion about allowable exposure levels, measurement requirements, and reporting practices for environmental compliance. The State program should take advantage of the wealth of knowledge and approval obtained from USNRC inspections, reviews, and data analysis. The State program should spend its available resources to fill appropriate gaps between what the federal requirements are and what Vermonters need for management of risk and assurance that appropriate attention is brought toward human health. Appendix A of this report provides a detailed examination of the radiation protection regulation including noted difficulties in implementing the requirements, as written. ORAU also notes that, to its knowledge, the site boundary dose requirement for VYNPS is unique relative to all other states and federal agencies. The "extra effort" required to monitor direct gamma dose specifically, in the presence of all other radiation exposure pathways at these low levels, may not be the most cost-effective means to manage radiation exposure risk to the public. A balanced approach over all pathways of radiation exposure is prudent, and should be

considered in the next version of the State regulation.[7] This process of risk management embodies the notion that a common risk baseline is provided in establishing regulatory exemption criteria in §5-304.

4. The State of Vermont regulations cite a common, convenient administrative assumption that one roentgen (unit of "exposure") is equivalent to one rem (unit of "biological dose equivalent"). In most operations, this convenient assumption has no impact on evaluating risk to the public or to radiation workers because it is a conservative assumption. In the science of dosimetry—which examines how the human body responds to radiation exposure—the conversion from exposure to dose equivalent is complicated. Dose equivalent is measured directly in some applications, but only after a substantial effort to calibrate the equipment for "tissue equivalent response." In most cases, gamma-ray radiation is measured in units of exposure and then reported in exposure. If needed, exposure is mathematically converted to dose equivalent by the use of a simple multiple, with the number 1 being the most conservative multiple.

   ORAU has examined very carefully the issue of exposure to dose equivalent conversion factor. Calculations and references are provided in Appendix B. ORAU has determined that the VYNPS conversion factor historically used in the ODCM, i.e., 0.71, was conservative and that direct measurements of this property of the radiation fluence rate result in a factor approximating 0.6. Using a substantially different approach, that is, running computer dose simulations, the authors O'Brien and Sanna determined in 1976 that the conversion factor is 0.56.

   ORAU understands this analysis may not be in favor with conventional practice; nonetheless, this conversion is scientifically valid when converting from one measured unit (flux or exposure) to dose equivalent. VDH will ultimately have to decide whether to accept and approve these conversion factors for use in regulatory compliance purposes because the factors are different from unity (one), as stated in the existing Vermont regulation.

5. VYNPS developed a site-specific method to predict, by a real-time monitoring method, the direct gamma-dose contribution from ^{16}N produced in the reactor. This method, called the MSLRM method, is described in the VYNPS ODCM. This method was developed and implemented primarily because it removed the necessity to determine accurately what representative natural background radiation is, either at the dosimetry record (DR) locations, or at the site boundary.

   VYNPS has cited its confidence in this method for three reasons: the method accounts for over 90% of the total gamma dose rate at the site boundary, the method provides results in real time with electronic data logging, and measures net exposure

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[7] Managing risk against all radiation exposure pathways is discussed in the US EPA's DQO process as cited previously. For radiation protection purposes, the Health Physics Society working group N43 published an ANSI standard in 2002, ANSI/HPS N43.17-2002: "Radiation Safety for Personnel Security Screening Systems Using X-rays," endorsing in Annex A, the risk factor of "5% per sievert (Sv)." This risk factor means that for a 20 mrem annual dose equivalent, the incremental risk is 1 in 100,000 of contracting a fatal cancer in a lifetime.

rate directly, thereby obviating the need to estimate background rates at surrogate locations. The MSLRM detectors are calibrated according to a procedure. The calibration involves making two sets of radiation measurements at different reactor power levels: one set of measurements on the main steam line, the other set of measurements with a HPIC positioned at DR-53. VYNPS draws a relationship, or correlation between the HPIC response (μR/h) and the MSLRM response (~400 mR/h), to fit a curve as a function of reactor power level. ORAU was unable to locate any other BWRs utilizing this approach and consequently, could not compare VYNPS calculations and results with other BWRs. ORAU instead reviewed the MSLRM approach at VYNPS on its own merits and emphasized in its review not only the advantages (strengths) of the MSLRM, but what the MSLRM approach does not estimate—the difference of which is rationalized in the ODCM as the contribution from the north warehouse and the low level waste storage pad. ORAU concluded that the MSLRM is a reasonable technology and likely the best estimator available for determining net direct gamma dose from ^{16}N production.

ORAU recommends that the MSLRM approach be used as the primary method for estimating direct gamma dose from ^{16}N. When the MSLRM detectors are periodically recalibrated, VDH should be present to participate in the calibration tests, review procedural details, perform a review of the specific elements of the quality assurance program, and review measurement control logs. Measurement records of the MSLRM should be sent to VDH monthly for review.

ORAU also recommends that "other direct gamma sources" be evaluated on a more frequent basis. That is, VYNPS should evaluate their current operating procedures involving the potential for creating additional gamma dose at the site boundary.[8] For example, a logbook could be maintained that enables the State to verify that the process of identifying other gamma source terms and accounting for the incremental exposures are current throughout the operating year. At the end of each quarter, the "other source terms" should be added to the MSLRM data. The methods and coefficients for estimating "other source contributions" listed in the VYNPS ODCM Revision 30 (2002) require additional review by VDH. A complete explanation of the derivation of these quantities is needed.

ORAU recognizes that VDH cannot accept, alone, monitoring methods performed by VYNPS (both the MSLRM method and the "other source" identification method are performed by VYNPS staff). To ensure that no gross oversight occurs in the utilization of the aforementioned VYNPS methods, two steps should be taken. First, VDH should review the calibration plans and the procedural modifications to its satisfaction—this is an independent assessment of the monitoring capability. Second, VDH should improve its current independent measurement compliance program. This recommendation leads to an important question: What reliable monitoring device (or devices) should be deployed by the State to ensure that a

[8] ORAU received from VYNPS a revised procedure, effective January 11, 2007, titled "Handling, Transfer and Storage of On Site Radioactive Material," OP 2505, Revision 19. It is not ORAU's responsibility to approve this procedure for meeting the State's requirements, but the general nature of the improvements in this procedure move the compliance program for direct gamma dose at the site boundary in the proper direction.

significant process change in the emitted gamma-ray flux did not occur during one calendar quarter and will not occur for the entire year? As stated before, VDH should review not only the recurring data that VYNPS provides, but also the calibration protocols and the raw data collected during detector calibration. If this independent review is performed well, then ORAU believes that the expense of deploying a real-time HPIC monitor at DR-53 is not warranted. If however, after recalibration of the MSLRM method in 2007, it is concluded that the margin for error in the measurement (with complete measurement uncertainty propagation) is small, then VDH and VYNPS may reconsider the use of a more costly HPIC detector. ORAU reviewed some preliminary data in the absence of recalibration data. With an uprate to 120% power, and the installation of a new turbine shield, the ^{16}N dose equivalent at DR-53 will likely be in the range of 15 mrem per year (100% power, 100% capacity utilization, and a dose conversion factor of 0.6 rem per roentgen). This information may be premature, but for the record, these types of estimates have been discussed. Only through recalibration will the actual values be recorded, and it may very well be the case that the margin of error (the difference between a limit and the measured value) will not be as generous.

Assuming that the margin for error remains acceptable, the State should develop and deploy a more rigorous TLD measurement program. The goal of the program is to provide independent measurement assurance that the better, real-time monitoring methods did not fail "low". Statistical methods need to be developed to analyze the data (gross and background), detect outliers, refine estimates, and ultimately provide a validation of the real-time measurements. Methods need to account for acceptable Type I and Type II error rates (false negatives and false positives), process control statistical approaches, and an understanding in the variability that can be observed between collocated TLDs. Much attention to detail is required: calibration of the TLD, measurement units of the TLD provider, annealing algorithm, fading, control badge use, etc. In short, a single TLD cannot on its own provide a quality assurance level that is required for effective decision making at the 10 mrem per quarter (net) level. A number of environmental intercomparison studies have been conducted over the years and results look promising, but, it is asking too much of the device to make accurate and precise measurements at these low exposure levels. The most significant error is brought about by having to estimate the background radiation rate at the location of interest.

The surveillance program should be phased in, with opportunities to evaluate new passive dosimeters concurrently with conventional methods (for example optically stimulated luminescent dosimeters or "OSLs"). Fortunately, the use of passive dosimeters is cost effective. ORAU encourages that VDH deploy more dosimeters (for better statistics) and that identical dosimetry providers be used by VDH and VYNPS. This recommendation originally raised questions about independence. The fact of the matter is that there are straightforward ways to assure dosimetry results from the same provider are independent. ORAU found that it was extremely difficult (if not impossible) to compare results from the VYNPS Areva laboratory and VDH Global Dosimetry Services provider. Until such time that environmental dosimetry standards are written and approved to deal with such low-level exposures (in the environment and not on a person), then it becomes nearly impossible to determine that two laboratories can achieve the level of control necessary to make effective and reliable decisions at background levels and "slightly above" background levels.

Provisions for outlier detection and dosimetry averaging need to be included as well as a consistent and representative approach to convert from milliroentgen (mR)* [mR "star"] to millirem, or some other agreed upon unit. Calibration of environmental TLDs for dose equivalent is a difficult effort that has not been qualified by a representative national or international standard. There is no critical need to invent new methods for such low level exposures—just reliable and consistent methods. Toward this end, significant consideration should be given to ensure that the same technologies, methods, analytical parameters, etc., be utilized and agreed upon by VYNPS and VDH at these low dose levels.

One final remark on the use of environmental TLDs: most environmental TLDs are deployed to provide very important dose reconstruction data in the event of a significant accident. That is, did the TLD measure 100 mR or 10 R over the last few days? This is a significantly different question than whether the TLD measured 10 mR over a three-month period. In practice the capability of an environmental TLD to measure a net exposure of 10 to 20 mR per quarter is beyond the means of the technology. And so, a <u>single net reading</u> of 11 mR should not be considered and accepted as greater than a 10 mrem action level for the same reason that a net reading of 9 mR should be accepted as being compliant. A single TLD measurement does not provide the statistical power to make this absolute determination (above or below a given action level), but multiple TLDs do.

Multiple TLDs collocated in multiple locations of importance allow outlier data to be detected and rejected for both low and high results. Additionally, statistical tests should be established to manage the quality of the information so that the TLD data will provide an indication that there is reason to believe that the real-time measurement results provided by the MSLRM and "other" sources were in error, specifically, that the real-time results were grossly low. The expertise of a statistical process control engineer may be needed to establish the data quality objectives for this compliance process. The statistical process control problem may be stated with an example. Consider a fully-compliant, independently-verified MSLRM and "other source identification" program that is in place and approved by VDH. In one year, the sum of all measurements yield an "exposure to dose equivalent corrected" direct gamma dose value of 18 mrem, with a reported error at a 95% confidence level of ± 4 mrem. The independent TLD measurement analysis by VDH should be used to reject the reported results as grossly in error (statistically low). This type of quality control analysis is common in the measurement sciences, especially when the measurement quanta of interest (the net signal) is a small fraction of the noise.

Through this review, ORAU found no indication (or statistically significant evidence) that the annual dose objective at the site boundary of 20 millirem was exceeded. Furthermore, no statistically significant evidence has been gathered to suggest that the 2004 Quarter 4 notification limit of 10 mrem above background was exceeded. This finding does not suggest the absence of a problem; rather, it is clear that a number of improvements need to made in the State environmental compliance program and that a communications directive should be put in place between the State and VYNPS.

To address compliance and related issues in the future, ORAU recommends that an ECP be established as a fundamental working framework between the two parties.[9] The ECP should provide sufficient detail on the measurements, data, results, and reporting requirements. VYNPS and VDH should work together to develop the implementation plan for environmental compliance. Implementing procedures need to be written and an improved action plan needs to be formulated to manage the case when data suggests that a compliance point has been exceeded or is about to be exceeded.

The ECP should provide an interim framework from which to demonstrate compliance with the VDH regulation. Over the next several years, the ECP performance results should be folded in to the process for developing updated regulations.

[9] The "ECP" terminology (or equivalent) is recommended by ORAU based on similar communication plans utilized by federal agencies. ORAU suggests that the wording such as a "Memorandum of Understanding" (MOU) be avoided. An MOU implies both parties share identical and equal status in the compliance process.

SCOPE

ORAU conducted an evaluation of the direct gamma radiation emitted from the Vermont Yankee Nuclear Power Station. This evaluation is centered in three areas: 1) real-time monitoring systems for estimating dose at the site boundary; 2) use of passive dosimeters for environmental monitoring of ambient gamma dose, and 3) administrative agreements between the State of Vermont and Vermont Yankee to ensure and demonstrate that annual direct gamma dose objectives are not exceeded.

An introductory ("kickoff") meeting was held in January 2006 with the State of Vermont at their offices in Burlington. Three technical review meetings followed with the VDH and VYNPS at the plant's administrative offices in Brattleboro, including a site walk-down conducted of the west-side boundary. All meeting minutes are available for review as part of the overall deliverable from ORAU.

The principal component of direct gamma radiation detectable at the site boundary originates from the decay of a short-lived radionuclide ^{16}N. The level of direct gamma radiation from ^{16}N varies with reactor power and is "zero" when the reactor is in shutdown mode. Secondary components of direct gamma radiation at the site boundary—indistinguishable from natural background radiation—arise from the daily use of on-site radioactive material storage areas, and less-frequent spent fuel handling operations, use of x-ray equipment for inspection and testing, and movement of radioactive sources. The skyshine dose from ^{16}N produces over 90% of the direct gamma dose, with 10% of the remaining contribution from all other sources. Hence, the focus of this evaluation was on the source term from ^{16}N. This subsequently led the ORAU project team to evaluate the Vermont Yankee methods for estimating this contribution, termed the Main Steam Line Radiation Monitor (MSLRM) method. This method is the "real-time monitoring system" for estimating exposure rate (and dose rate) at the site boundary from ^{16}N production. All other components to site boundary dose are estimated on a periodic basis.

Passive TLD dosimeters have been deployed on-site and around the site from the plant's inception. The purpose of these measurements has historically been to satisfy USNRC license requirements for accident dosimetry; however, over time, both the State of Vermont and VYNPS decided that the measurement results could be used to monitor ambient direct gamma-radiation rates on a quarterly basis. ORAU examined the nature of these past measurements and evaluated the apparent increase in net radiation dose from a State TLD during the fourth quarter of 2004.

Through inspection and review, the USNRC regulates VYNPS operations from requirements and specifications in the plant's operating license, number DPR-28. For demonstrating compliance with State of Vermont Regulations on Radiological Health, VYNPS must ensure that all requirements of Part 5, Chapter 3, Subchapter 1 "Radiation Protection" are met. Within §5-308 "Enforcement", the State follows a protocol for notice of violations. ORAU examined the past administrative practices and agreements between VYNPS and VDH in an effort to improve regulatory compliance in the years ahead.

Given the public interest in the direct gamma dose measurements and evaluation, ORAU has included background information on all three of the primary topics, including the origin of skyshine radiation, the real measurement challenges of estimating a small net dose equivalent of 10 mrem per quarter or 20 mrem per year, and a historical review of the regulatory agreements in place to assure compliance is met. The effect of the extended power uprate is also discussed. No other exposure pathways, including effluent monitoring pathways, have been evaluated.

As stated in the disclaimer to this report, ORAU did not make any independent physical measurements or re-analyze raw data from the measurement devices. The extent of the ORAU analysis consisted of technical reports, memoranda, and environmental dosimetry records provided by VDH and VYNPS. ORAU did not review all instrument calibration records or evaluate the efficacy of raw data reduction methods as this level of review was not believed to be warranted for this task.

ORAU is cognizant of concerns reported in the local media and by Vermont citizens over the past several years concerning the safety of the Vermont Yankee plant based primarily on its age and design. ORAU was not requested to evaluate these issues which are appropriately and continually evaluated by the U.S. Nuclear Regulatory Commission, State of Vermont, and VYNPS staff.

ACRONYMS AND ABBREVIATIONS

ALARA	As Low As Reasonably Achievable
ANSI	American National Standards Institute
ASTM	American Society for Testing and Materials
BEIR	Biological Effects of Ionizing Radiation
BNL	Brookhaven National Laboratory
BWR	Boiling Water Reactor
CFR	Code of Federal Regulations
CHP	Certified Health Physicist (American Academy of Health Physics)
DAW	Dry Active Waste
DES	Duke Engineering Services
DR	Dosimetry Record
ECP	Enhanced Communication Plan
EPU	Extended Power Uprate
ERFIS	Emergency Response Facility Information System (Vermont Yankee)
ft	feet
HPGe	High Purity Germanium
HPIC	High Pressure Ion Chamber
HPS	Health Physics Society
HWC	Hydrogen Water Chemistry
ICRP	International Commission on Radiological Protection
ICRU	International Commission on Radiation Units and Measurements
ISFSI	Independent Spent Fuel Storage Installation
ISOCS	In Situ Object Counting System
keV	kiloelectron volts
km	kilometer
LLW	Low Level Waste
MeV	millions of electron volts
mrem	millirem
MSL	Main Steam Line
MSLRM	Main Steam Line Radiation Monitor

mSv	millisievert
MWe	Megawatts electric
MWt	Megawatts thermal
NaI	Sodium Iodide
NAS	National Academy of Sciences
NBS	National Bureau of Standards (now "NIST")
NCRP	National Council on Radiation Protection and Measurements
NEI	Nuclear Energy Institute
NIST	National Institute of Standards and Technology
NMC	Noble Metal Chemistry
NRC	National Research Council
NVLAP	National Voluntary Laboratory Accreditation Program
NWC	Normal Water Chemistry
^{16}N	Nitrogen-16 (N-16)
OSL	Optically stimulated luminescent (dosimeter)
ODCM	Off-site Dose Calculation Manual
ORAU	Oak Ridge Associated Universities
ORINS	Oak Ridge Institute for Nuclear Studies
PATP	Power Ascension Test Plan
PE	Professional Engineer (registered)
PIC	Pressurized Ion Chamber
PSB	Public Service Board
PUSAR	Power Upgrade Safety Analysis Report
PWR	Pressurized Water Reactor
ReGe	Reversed-electrode Germanium
REMP	Radiological Environmental Monitoring Program
RETS	Radioactive Effluents Technical Specifications
SER	Safety Evaluation Report
S.D.	Standard deviation
SOW	Statement of Work
TEDE	Total Effective Dose Equivalent
TEP	Technical Evaluation Plan (ORAU)

TLD	Thermoluminescent Dosimeter
TLDs	Thermoluminescent Dosimeters
UNSCEAR	United Nations Scientific Committee on the Effects of Atomic Radiation
USDOE	United States Department of Energy
USEPA	United States Environmental Protection Agency
USNRC	United States Nuclear Regulatory Commission
VDH	Vermont Department of Health
VYNPS	Vermont Yankee Nuclear Power Station

1.0 Introduction

1.1 The Vermont Yankee Site

The Entergy Nuclear Operations Inc. Vermont Yankee Nuclear Power Station (hereafter VYNPS) is one of over 100 operating nuclear reactors in the United States and the only nuclear power plant in Vermont. The plant is located in the southeast corner of Vermont, in Windham County, in the town of Vernon. The plant is bounded by the Connecticut River on the east, and privately-owned land in all other geographical directions.

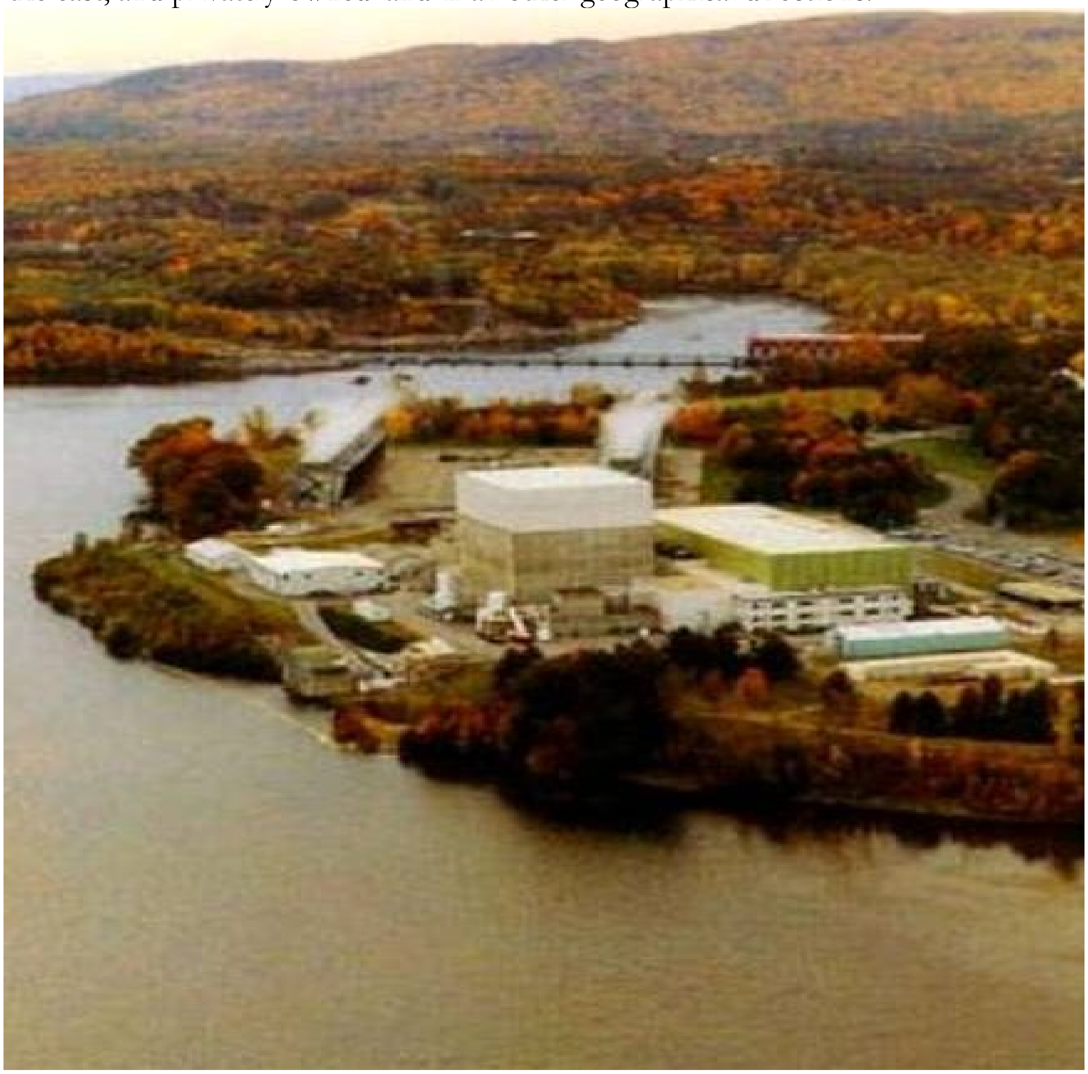

Figure 1. Aerial View of the Vermont Yankee Nuclear Power Station

Figure 1 provides an aerial view of the plant. The plant has a small geographical "footprint" relative to the vast majority of nuclear plants in the country and its physical location borders New Hampshire. The plant began construction in 1967 followed by commercial power production in November, 1972. According to the Nuclear Energy Institute (NEI), the plant satisfies over 72% of the state's electricity needs with hydroelectric power generation a distant second (~20%). The plant's operating license cites a maximum authorized power level of 1593 megawatts thermal (1593 MWt). In 2006, the USNRC approved a power uprate from 1593 MWt to 1912 MWt.

1.2 ORAU's Task

Beginning in January 2006, Oak Ridge Associated Universities (ORAU) met with representatives from the State of Vermont Department of Health (VDH) and VYNPS to discuss issues related to demonstrating compliance with gamma site boundary dose limits established in Part 5, Chapter 3, Section 5-305(B) of the State of Vermont regulations.

The primary objective of the ORAU investigation was to evaluate the VYNPS methods and protocols used to measure and report quarterly and annual site boundary dose from exposure to gamma-rays. The secondary objective was to evaluate the verification methods used by VDH to confirm that the site boundary annual dose objectives or quarterly dose limits are not exceeded. In the course of reviewing measurement data, procedures, analyses, and regulator-licensee memoranda, ORAU determined that process improvements need to be made by both VDH and VYNPS to continue to ensure that state regulatory compliance is assured.

The scope of this project was bounded to direct gamma dose at the site boundary, as conveyed in the statement of work (SOW) to ORAU [A], and specific sections of the state regulation. [B, C] However, elements of the general findings are germane to regulatory compliance with effluent monitoring and control. Because the primary source term giving rise to a direct gamma dose is from the decay of ^{16}N during reactor operation, the ionizing gamma radiation measurable at the site boundary is referred to as "skyshine". All other, less significant source terms were also evaluated, as described in the latest available VYNPS Off-site Dose Calculation Manual (ODCM), Revision 30, dated October 30, 2002. Monitoring methods reviewed by ORAU consisted of the ODCM calculation and procedure, including the Main Steam Line Radiation Monitoring (MSLRM) method, High Pressure Ion Chamber (HPIC) calibration measurements for the MSLRM approach at location DR-53, and environmental dosimetry. All prior agreements and memoranda of understanding available to ORAU between the State and the nuclear utility were also reviewed.

The annual dose objectives for direct gamma-ray exposure at the site boundary, as currently established by the VDH, are lower than federal requirements established by both the U.S. Nuclear Regulatory Commission (USNRC) and Environmental Protection Agency (USEPA). All limits— whether state or federal—are fractions of natural background dose, and as a result, it is difficult to effectively and precisely measure these quantities with absolute certainty. Hence, the treatment of uncertainty is an important consideration for demonstrating compliance. This directly impacts the issue of determining the most representative natural background rate to be subtracted from the gross "signal", to yield the net dose equivalent at the site boundary "above natural background." Furthermore, existing state regulations simplify the exposure to dose equivalent conversion factor to unity (one), thereby causing confusion and the likelihood of misunderstanding between regulatory compliance and the means to actually achieve it.

1.3 Report Contents

This ORAU report begins with introductory material on the regulatory compliance requirements for direct gamma dose and what direct gamma dose is (and is not). Applicable federal and state requirements pertaining to the VYNPS are presented.

Chapter 2 discusses all the contributing factors to site boundary dose from gamma-rays, with the most significant arising from the decay of ^{16}N.

Chapter 3 introduces the topic of natural background radiation and radioactivity, describing first general components impacting the United States population as a whole and then more specifically the natural background in the environs of the VYNPS. This topic influences later discussions on background subtraction methods.

Chapter 4 evaluates the existing regulation which relates site boundary dose to projections beyond the site boundary, for example, to the nearest resident. Relevance to the current situation is described and recommendations are provided.

Chapter 5 is a summary of the various radiation measurement devices that can be and have been used to measure direct gamma dose.

Chapter 6 constitutes a significant chapter in this report, describing the methods that have been utilized to measure direct gamma-ray dose. Primary elements of the existing compliance measurement methods are discussed, with all reported citations included. These include the MSLRM methodology and use of TLDs. Significant detail about the measurement results, when warranted, was moved to an appendix. An example of this is Appendix B, where the details of utilizing high-resolution gamma-ray spectrometer measurements to establish a "flux to dose" conversion factor are provided. Appendix B was a major effort to understand exactly how the measurement results from an ion chamber, or possibly a passive dosimeter, should be calibrated, or converted from measured units to dose equivalent units.

Chapter 7 describes one advantage of the MSLRM method for accounting for natural background radiation, and contrasts this with the dosimetry method of subtracting natural background rates from surrogate background locations.

Chapter 8 briefly describes the calibration of equipment to measure and report in dose equivalent. It was not in ORAU's scope to prescribe solutions for this technical issue, but to provide some ideas of where to look to be able to accomplish this objective.

Chapter 9, measurement uncertainty, provides suggestions to deal with uncertainty when comparing two measured values of the same quantity and their methodologies. How to achieve the implementation of measurement uncertainty into regulatory compliance is an ongoing question, for which the solution in practice resides in an understanding between the regulator and the licensee.

Chapter 10 was initially expected to be an exhaustive analysis of dosimetry results. However, as ORAU reviewed the data, it became clear that the measurement results were small (and below regulatory limits); therefore, attempting to understand random acts of nature or radiation generated randomly from skyshine at such low levels was considered an inefficient use of project resources. As a result, ORAU presents graphical data to show how variable background is with time and location, and that from this data, a suitable averaging method should be used to rule out a net result that is either statistically significantly high or low.

Chapter 11 provides recommendations for establishing an enhanced communication plan (ECP) between VDH and VYNPS with the intent to address compliance issues in the future.

Chapters 12 and 13 provide references that are either programmatic (Chapter 13) and indexed as characters, for example [A], or technical reports (Chapter 14), indexed in numerical order, for example [1].

Chapter 14 provides key definitions.

The report's contents conclude with several appendices designed to provide additional supporting technical and informational details of interest, primarily to the physicists reviewing this information. Topical areas include an evaluation of the State of Vermont's radiation regulations, the "flux to dose" issue, methods for selecting surrogate background subtraction locations, use of environmental TLDS for low level radiation dose measurements, and dealing with measurement uncertainty.

1.4 Direct Gamma Dose

Ionizing gamma radiation is emitted from a variety of sources, including nuclear reactors. During operation, facility employees are protected from exposure to gamma radiation by shielding (e.g., around the turbine, reactor, etc.), or simply by avoiding entry into posted radiation areas. All individuals, whether utility personnel or not, are exposed to this "shower" of ionizing radiation. The intensity of the gamma radiation decreases with increasing distance from the source and as power level is decreased. The intensity level is not linear with either physical process. First, the intensity increases with power slightly more than linear. For example, as the power level increases by 20 percent, the gamma radiation dose increases by approximately 26%. The actual increase can vary during plant operation, but ORAU relied on the 26% value as it was officially stated in the USNRC's Safety Evaluation Report (SER) which cited VYNPS data provided to the USNRC. Second, with distance, the direct intensity decreases as the square of the distance. For example, the direct intensity from a gamma source is reduced to one-fourth of the initial intensity twice as far away from the source (not one-half).

Both variables affecting gamma-ray intensity impact public exposure at the site boundary. Residents located closer to the site boundary are in a larger gamma-ray field than those located farther away. Both levels of exposure, whether at 800 feet (ft) or 1600 ft from the source, are relatively small fractions above natural background— roughly about 20% of natural background (excepting the contribution from radon decay products). Even though the gamma-ray intensity is small, the utility must ensure that the level is below allowable limits of exposure to the public described in the federal and state regulations. Furthermore, it is the responsibility of the utility to ensure that the radiation dose to the public is as low as reasonably achievable (ALARA).

It is the word “reasonable” in ALARA that causes significant differences of opinion between the licensee, the regulators, and the stakeholders. For example, in the State of Vermont regulation, ALARA means “as low as is reasonably achievable taking into account the state of technology at or available to Vermont Yankee and the economics of improvements in relation to the benefits to the public health and safety and in relation to the utilization or atomic energy in the public interest.” In April of 1994, the USNRC published a document, “The Value of Public Health and Safety Actions and Radiation Dose Avoided,” NUREG/CR-6212, to provide guidance to licensees on methods for conducting a numerical analysis that complies with ALARA principles while accounting for cost of the improvement versus risk to human health. Because this report is about regulatory compliance with allowable, chronic low-level exposure to the public from direct gamma radiation, it is important to understand the direct gamma radiation components, to distinguish the direct gamma radiation pathway from the effluent pathway, to understand how direct gamma radiation is measured and suitable methods to demonstrate compliance.

Chapter 2 breaks down direct gamma-ray dose in detail, but as an introduction, the following distinctions are made. Direct gamma dose at the site boundary results from the sum of all gamma-rays emitted from all operations affiliated with the operation of VYNPS. Gamma-rays are emitted from the reactor, the turbine building, spent fuel, on-site storage of radioactive material, radiographic (radiography) operations, and immersion in noble gases. Direct gamma dose is one of the two primary pathways for public exposure.

The other pathway is from the ingestion or inhalation of radioactivity released in the liquid or gaseous effluents. This exposure pathway is explicitly not discussed in this report. Unfortunately, it is this distinction that causes great difficulty and confusion between the State of Vermont and the federal regulations. The State separates the exposure pathways (State of Vermont, Department of Health, Part 5, Chapter 3, Subchapter 1, §5-305(B)(1)(e)), whereas the federal government (the USNRC) establishes standards on the sum total of all exposure pathways. This distinction causes further confusion, not only from the public’s perspective, but for ORAU as the third-party review team conducting this effort.

The sections that follow discuss the pertinent differences between the USNRC and State of Vermont regulatory practices. A unique regulatory situation does exist in the State of Vermont. The State and public cannot make the following assumption: that if VYNPS satisfies its federal license requirements, then it by default, satisfies those of the State of Vermont. Some Vermont state requirements are currently more difficult to comply with than the federal requirements, and these will be discussed in several areas of this report, e.g., Section 1.6 and Appendix A.

1.5 VYNPS Regulatory Compliance with the NRC

To operate a nuclear reactor, VYNPS must comply with Federal and State radiation safety standards. These standards or requirements originate from well-recognized and respected organizations such as the National Research Council/National Academy of Sciences (NRC/NAS) and the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR) who provide the scientific basis. The International Commission on Radiological Protection (ICRP) and the National Council on Radiation Protection and

Measurements (NCRP) use the NRC/NAS and UNSCEAR to establish radiation protection policy. Finally, regulatory agencies in the United States, such as the USNRC and state radiological health agencies, establish their regulations based on these organizations' policies.

On the federal level, VYNPS compliance rests primarily with the USNRC under Title 10 of the Code of Federal Regulations (CFR) Part 20, "Standards for Protection Against Radiation", and Part 50, "Domestic Licensing of Production and Utilization Facilities". Appendix I to Part 50, "Numerical Guides for Design Objectives and Limiting Conditions for Operation To Meet the Criterion "As Low as is Reasonably Achievable" for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluent", is a specific appendix within Part 50 pertaining to effluent releases that all reactor licensees must satisfy.

Other regulations apply as well, such as 10 CFR Part 72 when onsite storage, such as an independent spent fuel storage installation (ISFSI) exists. 10 CFR 20 also incorporates applicable USEPA requirements from 40 CFR Part 190, "Environmental Radiation Protection Standards for Nuclear Power Operations".

The VYNPS utilizes its ODCM as a primary means to satisfy USNRC and USEPA effluent and environmental control limits. The ODCM was first issued by VYNPS in the early 1980's as an USNRC requirement (as was the case for all other existing nuclear reactors) and approved by the USNRC. Once approved, VYNPS initiated changes based on its operating experience and issued revisions. The latest revision, Revision 30, was reviewed by ORAU for this report. The ODCM is subject to the USNRC's routine inspection program. The ODCM provides calculational methods for determining off-site concentrations, off-site doses, and effluent monitor setpoints. VYNPS relies on the ODCM for generating plant procedures and reports, including those required to meet the design objectives found in Appendix I to 10 CFR 50.

A brief description of the primary federal regulations and applicable dose limits follow.

10 CFR 20 (USNRC)

The USNRC requires in Part 20, Section 1301(e), "Dose limits for individual members of the public", that licensees limit radiation exposure to individual members of the public to 100 mrem (1 mSv) total effective dose equivalent in a year from licensed operations. The allowable dose to the public is controlled by the NRC as the sum of all pathways. Licensees must also meet the provisions of 40 CFR 190 as applicable. VYNPS complies with the USNRC public dose limit at and beyond its site boundary through adherence to the dose limits contained in 40 CFR 190.

10 CFR 50 (USNRC)

Compliance with 10 CFR 50, and particularly Appendix I, requires that total body, organ and thyroid dose limits be met for both radioactive liquid and gaseous effluents released to unrestricted areas. These limits are cited in Part 50 and the VYNPS ODCM. The total dose from all such sources cannot exceed 25 mrem (0.25 mSv) in a year for the total body, 25 mrem (0.25 mSv) in a year for organ doses, and 75 mrem (0.75 mSv) in a year for the thyroid.

10 CFR 72 (USNRC)

10 CFR 72, "Licensing Requirements for the Independent Storage Of Spent Nuclear Fuel, High-Level Radioactive Waste, And Reactor-Related Greater than Class C Waste", establishes the USNRC's requirements, procedures, and criteria for the issuance of licenses to receive, transfer, and possess power reactor spent fuel, power reactor-related Greater than Class C (GTCC) waste, and other radioactive materials associated with spent fuel storage in an independent spent fuel storage installation (ISFSI). The ISFSI is of interest in this report due to VYNPS efforts to utilize an ISFSI onsite—efforts that have been documented in several public meetings.

"Criteria for radioactive materials in effluents and direct radiation from an ISFSI or MRS" are established in 10 CFR Part 72, Section 104, which states that:

> "(a) During normal operations and anticipated occurrences, the annual dose equivalent to any real individual who is located beyond the controlled area must not exceed 0.25 mSv (25 mrem) to the whole body, 0.75 mSv (75 mrem) to the thyroid and 0.25 mSv (25 mrem) to any other critical organ…"

One of the principal exposure pathways cited includes direct radiation from ISFSI operations, with operational restrictions and limits imposed, respectively, to meet as low as is reasonably achievable objectives for radioactive materials in effluents and direct radiation levels associated with ISFSI operations.

40 CFR 190 (USEPA)

The USEPA's 40 CFR 190 limits apply to "members of the public," in areas at and beyond the site boundary from sources generated by facility operations. Part 190 is applicable to a "real" (not hypothetical) person, undertaking actual activities in a realistic location, rather than for example, an individual occupying his/her time near a facility's fenceline. The limits also apply to the entire "nuclear fuel cycle," that is, from the mining, milling, and fabrication of uranium into commercial reactor fuel to waste disposal.

40 CFR 190.10, "Standards for normal operations", states that:

"Operations covered by this subpart shall be conducted in such a manner as to provide reasonable assurance that:

> (a) The annual dose equivalent does not exceed 25 millirems to the whole body, 75 millirems to the thyroid, and 25 millirems to any other organ of any member of the public as the result of exposures to planned discharges of radioactive materials, radon and its daughters excepted, to the general environment from uranium fuel cycle operations and to radiation from these operations."

For doses due to effluents, compliance with 40 CFR 190 is met through surveillance programs that assure compliance with the limits of 10 CFR 50, Appendix I. Meeting 40 CFR 190 is not considered difficult for routine reactor operations; an operation involving a significant fuel failure would be required to exceed the limits. It is expected that VYNPS would shut down its nuclear reactor operations to remedy any compliance problems well before 40 CFR 190 limits were reached.

1.6 VYNPS Regulatory Compliance with the State of Vermont

In addition to federal requirements, VYNPS is subject to compliance with both VDH direct radiation and effluent releases. This section discusses the applicable regulations, with emphasis on direct gamma dose, locations where compliance is determined, factors that could impact the measured dose now or in the future, and the history of the 2004 event that precipitated ORAU's involvement as a third party, independent reviewer. Appendix A contains a detailed review of the Vermont radiation protection regulations.

The primary dose "objective" at VYNPS is 20 mrem/yr , as stated in Part 5, Chapter 3, Section 5-305(B)) of the Vermont regulations. [10] ORAU notes that in comparing this value to any other nuclear power plant in the country, it was unable to locate a more restrictive annual dose objective/limit. The basis for this annual dose is not completely known or understood, but apparently originated in the late 1960's/early 1970's when negotiations over the plant's construction and operation initiated.

The VDH radiation regulation (Part 5, Chapter 3, Section 5-305(B)) states in part that "The annual dose objective for the total body of an individual in an unrestricted area due to plant emanations of gamma radiation is 5 millirems. For the purpose of this objective, 20 millirems per year (20 mrem/yr) at any point on the site boundary bordered by land shall be considered equivalent to a 5 millirem dose at the nearest residences in Vermont."

For regulatory compliance purposes, VYNPS must first ensure doses do not exceed 20 mrem per year at the site boundary, irrespective of any assumptions that are used to estimate dose to an offsite individual/residence. Mr. Ray McCandless, a former VDH Radiological Health Chief, stated in a 1999 letter to VYNPS that if the 20 mrem annual limit was met at the site boundary, the 5 mrem annual limit to the nearest residence(s) was also satisfied. Further discussion of this issue is presented in Chapters 4 and 6.

VYNPS location DR-52 is the primary measurement point of interest relative to the 20 mrem annual dose limit at the site boundary. Most experimental evaluations have been conducted, however, at DR-53 because it is the point of highest exposure along the west side fenceline and therefore, the most "conservative" location from an exposure standpoint. Environmental dosimetry data indicate that the DR-52 value is typically about 20% of the DR-53 value. (The VDH has an equivalent measurement location to DR-53 named the "VY Parking Lot Site".) DR-53 is located about 300 ft north of DR-52 and about 40 ft closer to

[10] The annual dose "Objective" is implemented as though it was a limit. VYNPS has asked whether this specific contribution to total dose equivalent could be summed from all other source terms (similar to the USNRC approach). In order to exempt this specific pathway from explicit measurement and control, VDH would need to conduct a dose assessment of the existing Exemptions, §5-304, and baseline the annual dose threshold for which the state requires licensure.

the source term (the turbine building) than DR-52. Figure 2, taken from SVE Associates, depicts a site map with locations DR-53[11] and DR-52. The orange line denotes the fenceline and the red line denotes the site boundary. This distinction is very important because measurements are conducted at the fenceline and compliance is satisfied at the site boundary.

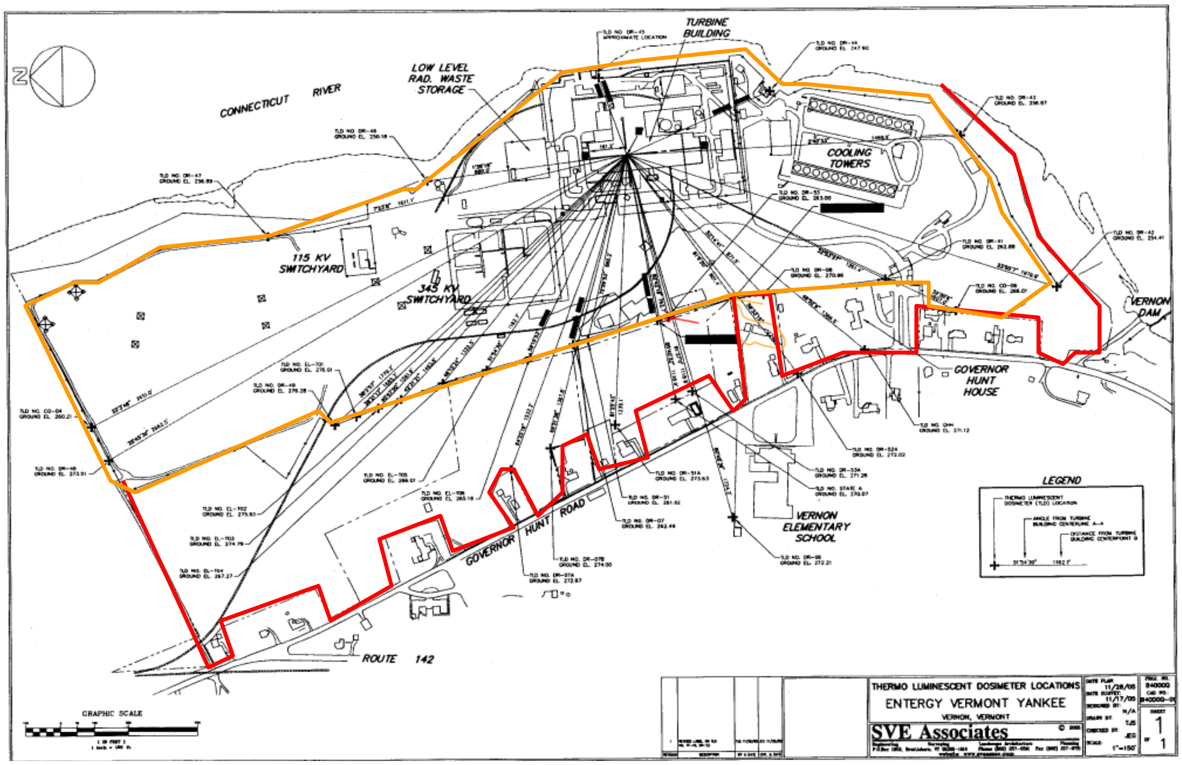

Figure 2. VYNPS Site Map

In 2004, the cumulative dose from VDH TLD measurements collected at location DR-53 was 24.9 mrem per year, including a 12 mrem dose recorded in the fourth quarter of the calendar year. This became an apparent compliance issue as both the quarterly dose and annual dose objectives were exceeded. However, VYNPS TLD measurements did not agree with the State's, eventually resulting in the request to ORAU from the VDH for an independent review and evaluation. The TLD discrepancy was at least partly due to the fact that the VYNPS and VDH did not collocate TLDs at that time (collocation was subsequently initiated in the second quarter of 2005).

Discussion of the compliance issue between all parties at the kickoff meeting in January 2005 immediately centered on the issue of measurement uncertainty. Both the VDH and VYNPS agreed that an allowance for an annual measurement uncertainty was acceptable, and in fact was formally documented in a 1977 letter from the State to VYNPS. Consequently, while the dose objective remains 20 mrem, an effective annual dose equivalent limit of 20 ± 5 mrem per year is currently understood. The VDH has also acknowledged that a 25% uncertainty (20 ± 5 mrem) in the annual dose could logically apply to the quarterly limit as

[11]DR-53 is the primary measurement location for studying and evaluating direct gamma dose. It is located near the west-side fenceline, about 300 feet north of DR-52.

well (10 ± 2.5 mrem). This simplification (or margin of safety) is illustrated by the age old saying used by highway drivers: "You can travel 60 miles per hour (mph) on this 55 mph highway because the state officials allow drivers a speedometer margin of error of 5 mph." Why is it they don't ticket at 50 mph?

The 2004 event resulted in the need for discussion of the inherent uncertainty in TLD results. This uncertainty in TLD measurements is not an issue for routine monitoring in the environs of nuclear power stations where the 100 mrem dose limit is of interest. However, that is not the case for VYNPS. The inadequacy of TLDs at environmental levels is apparent at the VYNPS facility. ORAU recommends that short term, the use of appropriate measurement uncertainties continue for both the annual and especially for the quarterly monitoring results, simply because these passive dosimeters were not intended to be accurate at such low dose levels. Long term, compliance with the 20 mrem dose objective must be established in another manner. This issue will be described in greater detail elsewhere in this report, not only regarding TLDs, but the MSLRM method utilized by VYNPS as one measure of compliance. The need by both parties for an ECP establishing an agreed-upon approach is a key ORAU recommendation. This will improve the methodology(ies) for measuring site boundary dose equivalent to ensure compliance with state regulatory requirements.

In addition to measurement uncertainty issues, interpretation of and compliance with the VDH regulation is subject to several other factors, including residence/occupancy times, and distance and shielding considerations. For example, the use of shielding and occupancy factors impacts the "20 mrem equals 5" issue (see Chapter 4). VYNPS has not utilized occupancy or shielding factors to date, though the plant notes the possibility of their use in the ODCM. During the course of ORAU's involvement with this effort, VYNPS installed a shield above the high pressure turbine in the turbine building (May 2006) to reduce direct gamma dose rates at the site boundary. And in a 2005 VYNPS pressurized ion chamber (PIC) study, "Comparison of Projected Annual Dose at the Blue Trailer when using Vermont Yankee Average Background Dose (DR-14 and DR-16) or Putney Village Background Dose", annual doses on the order of 5-7 mrem per year at the nearest residence were reported by VYNPS. Again, neither occupancy nor shielding factors were utilized. (ORAU notes that there is a precedence for these factors; the State has approved in historical documentation, for instance, the use of occupancy factors at the nearby Vernon Elementary School.)

1.7 Regulatory Compliance Following a Power Uprate

In 2003 VYNPS requested from the USNRC a license amendment for an extended power uprate (EPU). Power uprate requests are quite common in the nuclear industry, involving an increase in the maximum power level at which the reactor operates. The VYNPS request required the submittal of a "Power Upgrade Safety Analysis Report" (PUSAR). Following USNRC review and issuance of an SER, the uprate was approved on March 2, 2006. The SER may be found in the NRC "ADAMS" database.

In its SER, issued in March 2006, Section 2.10 ("Health Physics"), the USNRC identified two factors that could impact public and offsite radiation exposures due to plant operations. These included potential increases in gaseous and liquid effluents released from the site, and an increase in direct radiation exposure from radioactive plant components and solid, radioactive wastes stored onsite. The most significant offsite increases were associated with direct radiation exposures from ^{16}N skyshine and radwaste storage. Each of these factors is discussed in turn.

The VYNPS 20% power increase resulting from the extended power uprate yields a corresponding 20% increase in gaseous effluent releases from plant operations. The USNRC estimated that the total nominal public dose is about 1 mrem annually. Regarding compliance with the federal limits, then, it was concluded (by the USNRC) that even a 20% increase in the effluent dose was well within the dose limits specified in 10 CFR 50 Appendix I. Direct gamma dose from submersion in the noble gases is therefore extremely small during normal operation.

Regarding direct radiation, VYNPS field measurements determined that the plant's west boundary has the highest direct offsite dose where the boundary is bordered by land, approximating 15 mrem, with a large fraction, 13.4 mrem, estimated by the plant from skyshine. Given a power increase of 20%, the ^{16}N skyshine contribution increases non-linearly (by about 26%) with a resulting skyshine dose of 16.9 mrem (13.4 mrem multiplied by 1.26). The dose contribution from radwaste components of 1.6 mrem (15 mrem minus 13.4 mrem) increases to 1.74 mrem. The total estimated dose contribution is therefore 16.9 mrem (skyshine) plus 1.74 mrem or 18.64 mrem/year. This dose meets the requirements of 40 CFR Part 190 (see Section 1.2) as referenced in 10 CFR 20.1301(e).

Following issuance of the final SER, the VYNPS initiated power uprate testing. The USNRC, through its inspection program, reviewed VYNPS operations. A review conducted during the spring of 2006, which included inspection activities following the installation of the turbine shield, evaluated the initial power increase and effects on offsite public dose with respect to 10 CFR 20.1301(e) and 40 CFR 190 public dose limits. The USNRC concluded that the calculation method in the ODCM using the MSLRM was adequate and that offsite doses were within the USNRC and USEPA public dose limits. (A specific discussion on the ODCM and MSLRM is provided in Chapter 6.)

Historically (and according to their MSLRM measurement records), VYNPS has not believed the site boundary dose had ever been exceeded (on either a quarterly or annual basis); however, in the context of the EPU, the margin available to sustain a less than 20 mrem per year average at the site boundary declines because, as noted previously, increased power generation translates to a higher dose. ORAU has assumed that the 18.64 mrem/year dose contribution noted above includes the conversion factor currently applied in the ODCM (page 6-50). This factor is 0.71 mrem/mR. A conversion into units of exposure (mR) would result in an approximate exposure of 26.25 mR (18.64 mrem divided by 0.71.). Using the State's "one to one" roentgen to rem conversion), the fenceline dose objective of 20 mrem would be exceeded. VYNPS has already conducted short term studies of fenceline exposure rates using a PIC during reactor power increases at location DR-53. These measurements directly measure gross exposure rate (in microroentgens per hour). These data are used to augment the existing correlation (or mathematical relationship) between the exposure rate at the main steam line and the exposure rate at DR-53. ORAU's evaluation of data received from VYNPS regarding that study is provided in Chapter 6.

A summary of the applicable regulations appears in Table 1.

Table 1. Summary of Federal and Vermont Radiation Regulations Germane to the VYNPS Site Boundary Gamma Dose Evaluation

Agency	Document	Exposure Pathway	Requirement
USNRC	10 CFR 20.1301(e)	Sum of all pathways	Licensees must limit radiation exposure to individual members of the public to 100 mrem (1 mSv) total effective dose equivalent in a year from licensed operations.
USEPA	40 CFR 190.10	Planned discharges to the general environment	The annual dose equivalent cannot exceed 25 millirems to the whole body, 75 millirems to the thyroid, and 25 millirems to any other organ of any member of the public.
VDH	Vermont regulations Part 5, Chapter 3, Section 5-305(B)	Gamma radiation	The annual dose objective for the total body of an individual in an unrestricted area due to plant emanations of gamma radiation is 5 millirems. For the purpose of this objective, 20 millirems per year (20 mrem/yr) at any point on the site boundary bordered by land shall be considered equivalent to a 5 millirem dose at the nearest residences in Vermont.
USNRC	10 CFR 50 Appendix I	Liquid and gaseous effluents released to unrestricted areas.	The total dose from all such sources cannot exceed 25 mrem (0.25 mSv) in a year for the total body, 25 mrem (0.25 mSv) in a year for organ doses, and 75 mrem (0.75 mSv) in a year for the thyroid.
USNRC	10 CFR Part 72, Section 104	Effluents and direct radiation from an ISFSI or MRS	During normal operations and anticipated occurrences, the annual dose equivalent to any real individual who is located beyond the controlled area must not exceed 0.25 mSv (25 mrem) to the whole body, 0.75 mSv (75 mrem) to the thyroid and 0.25 mSv (25 mrem) to any other critical organ

In summary, ORAU believes that several related issues associated with meeting the 20 mrem annual dose objective require attention:

a. Reducing the uncertainty associated with annual dose limit results

b. Improving methods for measuring net environmental dose

c. Identifying site-specific confounding factors in measuring and estimating site boundary dose

d. Improving methods for measuring background radiation

1.8 Radiation Exposure Versus Dose Equivalent

It seems reasonable to expect that VYNPS would ask the VDH or the USNRC the following question: "Do you regulate us on allowable exposure or dose equivalent?" The public might consider this a completely arbitrary question with little or no ramification. To the contrary, most health physicists (experts in radiation protection) assume for convenience that the unit describing radiation exposure is equivalent to the unit of measure describing radiation dose equivalent. This tradition—that exposure reported in roentgen (abbreviated "R") is equivalent to rem—is approved for use on the basis that this practice over-estimates or is a conservative estimate of the real dose equivalent. That is, for external gamma-ray exposure, the rem, when treated exactly as defined in ICRU 60 or 51[12], is less than the roentgen, by about 40% (with the reduction contingent on the down-scattered energy present at the point of interest).

When a health physicist measures gamma exposure rates with a microroentgen chamber or an ion chamber that is not "tissue equivalent", it is appropriate to convert this value when needed or required by procedure. Otherwise this relationship between roentgen and rem is only of interest to dosimetrists (physicists who study and evaluate the radiobiological effects of radiation exposure and how to best convert a measured quantity to a risk-based quantity, dose equivalent). Personnel dosimetry, for radiation workers, for example, accounts for this important difference through a comprehensive calibration program referred to as NVLAP: The National Voluntary Laboratory Accreditation Program. There is no such program established for environmental dosimetry.

When implemented, the unit conversion factors are normally not a significant issue for debate. This is not the case for Vermont citizens. The State of Vermont defines in Part 5, Chapter 3, Subchapter 1, Section 5-303, Definitions (K) "Rem" as a dose of 1 R due to X- or gamma radiation. The current law provides no allowance for the ability to convert physical quantities of exposure to a risk-based dose equivalent for long-term chronic exposure to direct gamma radiation. This issue became one of the single largest issues with this study: What is the flux to exposure to dose equivalent conversion factor? ORAU studied this matter extensively, taking advantage of prior measurements made using high resolution gamma-ray spectrometers. Seldom is it the case that one is interested in the energy dependent profile of the gamma-ray flux emitted from a nuclear reactor, but this task gave ORAU the opportunity to evaluate what the actual conversion factor is for VYNPS. This extensive analysis was interesting from a dosimetry, physics, and radiobiology viewpoint because the measurements corroborated a completely independent analysis (and method) used by O'Brien and Sanna in 1976—that for a unit flux density of natural background radiation, the conversion factor to dose equivalent is on the order of 0.6. The details of the ORAU analysis are provided in Appendix B. The State questioned the true independence of these measurement results, but ORAU assures all parties that the in-situ gamma spectrometry measurements of gamma-ray flux to dose equivalent were completely independent of the O'Brien and Sanna effort in 1976, even though Areva may have cited the earlier work in its comparison.

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[12] International Commission on Radiation Measurements and Units, Report 60: "Fundamental Quantities and Units for Ionizing Radiation;" and Report 51: "Quantities and Units in Radiation Protection Dosimetry"

In simple terms, a roentgen is not equivalent to a rem. A rem is about 0.60 times that of a roentgen, when considering the conversion from direct gamma-ray exposure (contingent on the down-scattered energy present at VYNPS). Consider a sphere full of air and a sphere full of human tissue. Gamma-rays strike the surface of volume of interest and produce secondary electrons and the residual, positively charged atom from which the electron was removed. The difference between the exposure in air and the dose in tissue is simply an "accounting" matter. For exposure, the ion pairs (or total charge) are counted. For dose equivalent, the absorbed energy of the ion pairs is counted.

The conclusion of this analysis is that the VDH must now decide how to implement this technically justifiable unit conversion, which conflicts with the existing compliance assumptions in the state regulations. Appendix A. provides the details behind this existing conflict.

2.0 Contributing Factors to Direct Gamma Dose at the Site Boundary

At distances in the range of 800 to 1000 ft from the reactor and turbine buildings, the direct gamma dose from reactor operations is predominantly comprised of the skyshine radiation produced as result of activated coolant. The other contributions include direct (unscattered) radiation from the reactor, turbine building, spent fuel, radioactive waste storage areas, and any other operation on-site that produces ionizing radiation. In this chapter, the contributing factors to site boundary dose are presented, starting first with the single most important contributor: skyshine radiation from ^{16}N production in the coolant. This phenomenon, as mentioned earlier, is present only when the reactor is operating. The ^{16}N production rate increases with power level, but non-linearly, at a nominal ratio of a 26% increase in its production for a 20% increase in power.

2.1 Skyshine and Line of Sight

Skyshine radiation is best described through the use of an illustration. In the figure below, ionizing radiation is emitted in all directions from the reactor core and from the coolant. In Figure 3, for example, gamma-rays emitted directly at the source (the turbine building) penetrate the building structure up into the atmosphere and are "air scattered" back to the ground. Gamma-rays emitted in the direction of the point of interest (site boundary fence post in this case) are not scattered and termed the "direct" component. Thus, there is the direct (unscattered) component and the scattered (skyshine) component.

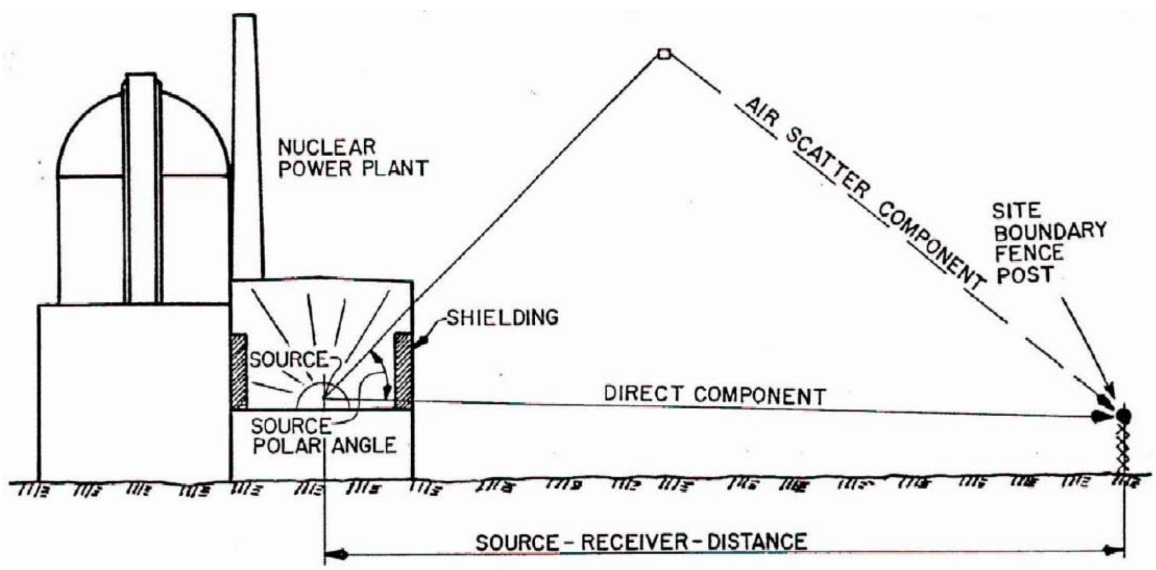

Figure 3. Illustration of Skyshine and Direct Radiation at an example BWR[13]

[13] N. Hertel et al., "A Comparison of Skyshine Computational Methods," slides from ICRS10, International Conference on Radiation Shielding 10, (May 2004).

The direct component of the total gamma-ray flux is calculated using the inverse square law. That is, the direct component decreases with the square of the distance from the source. The direct component is often referred to in the VYNPS documentation as the "line of sight" dose.

The air-scattered component, known as skyshine, is very difficult to calculate, but is measurable with a properly collimated high-resolution gamma-ray spectrometer. VYNPS examines this dose contribution in Section 6.11, *Method to Calculate Direct Dose from Plant Operation*, of the ODCM[14]. ORAU reviewed the source term calculations in the ODCM as well as data reports from the main steam line radiation monitoring system. It is difficult to know for certain what percentage of the total gamma-ray dose is contributed by skyshine flux, but it appears to be well over 90%. Once the other radiation source terms and natural background terms are described in the following sections of this chapter, it will be understood why making an absolute statement about the skyshine percentage is difficult to forecast. Natural background changes with time, weather conditions, and location. Skyshine varies with power level, water chemistry, atmospheric conditions, and topology.

2.1.1 The N-16 Contribution

The primary contribution to the external gamma-radiation dose at the boundary is from the decay of radioactive N-16, which is produced in the coolant via neutron activation[15]:

$$^{16}O(n,p)^{16}N \qquad \{\text{reaction cross section, } \sigma_\gamma = 19 \text{ millibarn}\}$$

This reaction, called an "n,p" reaction, occurs in all nuclear reactors, though the production rate is greater in BWRs than Pressurized Water Reactors (PWR). From the Brookhaven National Laboratory (BNL) National Nuclear Data Center (http://www.nndc.bnl.gov), the half life of ^{16}N is very short, that is, 7.13 ± 0.02 sec, which is about the cycle time of the activation product as it moves from the reactor (where it was produced/activated) through the high pressure turbine, low pressure turbine, condenser, and back to the reactor. As shown in the ^{16}N simplified decay scheme in Figure 4 for this radionuclide, two high-energy photons, 6.05 millions of electron volts (MeV) (with an abundance of 66%), 7.12 MeV (abundance of 5%) are emitted during beta decay (28% of the beta decays are directly to the ground state).[16]

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[14] "Vermont Yankee Nuclear Power Station Off-Site Dose Calculation Manual," Rev 30, October 23, 2002.
[15] Lin, Chien C. "Radiochemistry in Nuclear Power Reactors," Nuclear Science Series, NAS-NS-3119, National Academies Press. (1996)
[16] National Nuclear Data Center, Brookhaven National Laboratory

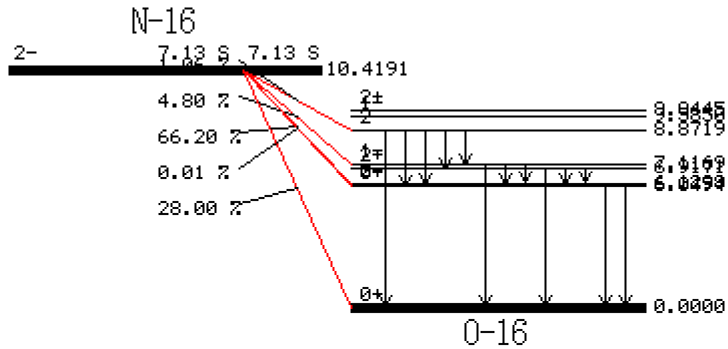

Figure 4. ^{16}N simplified decay scheme

All other activation products are produced in significantly lesser quantities than ^{16}N and the average gamma-ray energies are less than 1.4 MeV. As a result of the large production rate and the highly penetrating gamma rays emitted, ^{16}N is by far the most significant source term contributing to site boundary dose from gamma radiation.

Methods for calculating ^{16}N activity in the coolant of BWRs is provided in Section 10.12 of John Lamarsh's textbook "Introduction to Nuclear Engineering," 2nd edition. Lamarsh presents the theoretical development for calculating ^{16}N activity concentration in the reactor core as α (μCi cm^{-3}), as a function of the average macroscopic cross section for the (n,p) reaction on ^{16}O, $\Sigma_{(n,p)}$ at the average flux density $<\phi(E)>$, and as a function of time in the reactor (t_i) and time outside the reactor, (t_o) :

$$\alpha = \Sigma_{(n,p)} < \phi(E) > \frac{1 - e^{-\lambda t_i}}{1 - e^{-\lambda (t_i + t_0)}} \qquad (\mu Ci\ cm^{-3})$$

Advanced calculations of this origin are computed by the reactor operations group and the reactor manufacturer in order to refine the estimate, particularly, when studies to improve water chemistry are underway. Results of these analyses were reported by Vargo and discussed below.

The 6 and 7 MeV gamma rays penetrate the building structures in all directions. These gamma-rays scatter off the ground, building materials, and the atmosphere. These gamma-rays are capable of reaching essentially infinite distances, but by the time these photons reach a distance of 1000 ft, the mean energy is very typical of natural background radiation.[17] By contrast, "typical" fission product or activation product gamma-rays of less than 1.4 MeV do not penetrate as far, and are essentially negligible at similar distances associated with the site boundary.

As a control measure, there is nothing that can be done to stop or reduce the physical radioactive decay process of the ^{16}N. There are, however, means to reduce the ^{16}N

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[17] Referring back to Figure 3, at VYNPS the direct and skyshine components of the ^{16}N contribution were measured with a collimated high-resolution gamma-ray spectrometer. These results were reported in AREVA, "FINAL Summary Report of *In Situ* Measurements Performed at the Vermont Yankee Nuclear Power Station," EL 145/05-FINAL, (January 26, 2006)

production rate, but methods to achieve this are usually undesirable for other engineering or safety reasons. One reduction method is to reduce reactor power—this is not desirable or necessary alone. Other possible reduction methods are influenced by various water chemistry techniques to reduce corrosion, improve effluent loss, or improve the overall operation and safety of the plant. One paper that discusses these various techniques, and the associated strengths and weaknesses, was published in the Health Physics Journal by Vargo et al. in 1991.[18] Vargo reported that for the James A. Fitzpatrick BWR, ^{16}N concentrations at the reactor vessel steam nozzles were about 40 μCi/g, with an average main steam line monitor reading of less than 1200 mR/h under Normal Water Chemistry (NWC) conditions. With improvements to plant chemistry, using Hydrogen Water Chemistry (HWC), the ^{16}N production was increased and the resulting exposure rate on the main steam line radiation monitor increased to about 1650 mR/h.

Many other engineering evaluations and analyses have been conducted by reactor manufacturers, for example, General Electric Company, over the past 20 years, because it is recognized how important water chemistry is to efficient and safe plant operations. The engineering objective to reduce ^{16}N for the reason of reducing site gamma-dose rate will likely never be discovered or implemented because other, more significant benefits are derived by adjusting hydrogen water chemistry and noble metal chemistry (NMC) to achieve better performance and safety. During discussions with the VYNPS plant chemistry superintendent, it was clear to ORAU that VYNPS was continually studying and seeking improvements to plant chemistry. It appears however, that any improvement to water chemistry increases the ^{16}N production rate. The documented value for this increase (from the USNRC SER), under a 20% increase in power, using state of the art water chemistry, increases ^{16}N production by 26%. In a verbal communication provided to ORAU from VYNPS in April 2006, ^{16}N production would increase by a greater value—28 to 30%.[19]

Because ^{16}N production is a focus of this report, and given that it is the single most significant contributor to site boundary dose, one would conclude that it is a significant safety hazard. In reality, personnel dose from onsite activities is minimized by proper controls, shielding, and ALARA practices. Dose to the public is minimized primarily by increasing the distance from the turbine building to the nearest resident. At distances of 800 ft, the contribution from ^{16}N is a major contributor to the direct gamma dose, likely contributing at least 90% of the total. As stated earlier, it is virtually impossible to quantity all individual components of the gamma dose with complete accuracy. The best measure of this was the *in situ* high-resolution gamma ray spectrometry measurements conducted by VYNPS in 2002, as cited in Appendix B of this report.

2.1.2 All other contributors to site boundary gamma dose

Secondary site boundary gamma dose contributors, both from direct and scattered (skyshine) radiation, arise from the sum of all annual operations on site that produce x-rays and gamma-rays or areas where radioactive material is handled or stored. These operations, unlike the generation of ^{16}N, are not steady state or dependent on reactor power levels. These source terms may vary by day, by operation, or simply, by the inventory of radioactive

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[18] Vargo G., et al. "Implementation of a Source Term Control Program in a Mature Boiling Water Reactor: Health Physics Journal, Vol. 60. No. 6, (June 1991).

[19] Communicated during April 19, 2006 onsite meeting in Brattleboro between ORAU, VYNPS, and VDH.

material being transported on site or stored. These contributors include the following listing, with details presented in the subsections to follow.

North Warehouse (ODCM, §6.11.2)
Low Level Waste Storage Pad (ODCM, §6.11.3)
Spent Fuel Storage Areas (ISFSI)
Temporary Radioactive Material Storage Locations or Staging Areas
Other Industrial Operations, including x-ray equipment radiography
Immersion in low-level releases of noble gases

The site map below, taken from SVE Associates, depicts the source term locations. To assist ORAU and the State of Vermont, SVE associates superimposed the distances from the turbine building to the TLD locations and site boundaries.

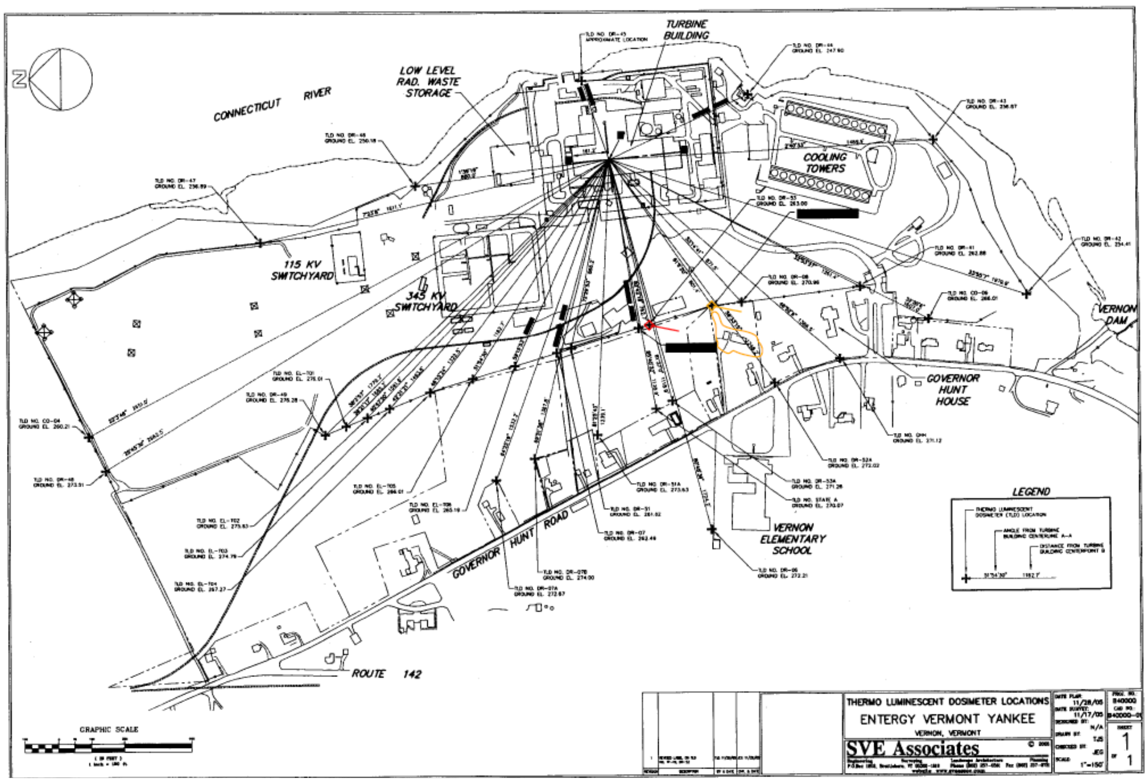

Figure 5. VYNPS Site Map

2.2 North Warehouse

Radioactive materials and low level waste can be stored in the north warehouse. This source term is discussed in the VYNPS ODCM §6.11.2, and an empirical estimate has been derived to estimate the dose contribution at the site boundary. The limitation of the analysis presented is that there are no controls or worst case conditions imposed on the operation in this facility. As a result, the empirical estimate applies under the assumptions described in the ODCM. There are no apparent controls to limit the radiological posting of the facility, radioactive material inventory, or to place an upper bound on total gamma-flux emitted from the facility that could increase (or decrease) the dose contribution at the site boundary.

Though the dose contribution is small, there is no real-time monitoring method or process to make adjustments to this model on a reasonable time basis (for example, if the operational objectives should change significantly). The north warehouse is one of the "other" source contributors that can only be measured and evaluated in combination with all other sources at the site boundary measurement stations.

2.3 Low Level Waste Storage Pad

Interim storage of packed Dry Active Waste (DAW) and spent ion exchange and filter media is permitted in modular concrete storage overpacks on the low level waste (LLW) storage pad facility adjacent to the north warehouse. This source term is discussed in the VYNPS ODCM §6.11.3, and an empirical estimate has been derived to estimate the dose contribution at the site boundary. The empirical estimate for this source term is divided into both the direct and scattered (skyshine) component. Additional estimates are provided for resin liner transfer, as well as a correction for gaps in effective shielding (called radiation streaming).

As in the case of the North Warehouse, the limitation of these analyses is that there are no controls or worst case conditions imposed on the operation in this facility. As a result, the empirical estimate applies under the assumptions described in the ODCM, when the analysis was performed (2002). There are no apparent controls to limit the radiological posting of the facility, radioactive material inventory, or to place an upper bound on total gamma-flux emitted from the facility that could increase (or decrease) the dose contribution at the site boundary. Though the dose contribution is small, there is no real-time monitoring method or process to make adjustments to this model on a reasonable time basis (for example, if the operational objectives should change significantly). The LLW Storage Pad is one of the "other" source contributors that can only be measured and evaluated in combination with all other sources at the site boundary measurement stations.

2.4 Spent Fuel Storage Areas Including ISFSI/Dry Cask

During the course of the ORAU review, VYNPS operations personnel were asked about their plans for an ISFSI dose analysis. The ISFSI inventory will increase with time, but the dose contribution at the site boundary will not increase linearly with inventory. The storage casks are self-shielding and additional shielding may be placed around the area if operations dose rates need to be reduced. These decisions are left to the plant radiological control department. Though not currently known, it is possible that the ISFSI will generate as much direct gamma dose at the site boundary as is produced currently from all other sources accounted for in the ODCM except ^{16}N.

2.5 Other Industrial Operations

Other sources of ionizing radiation on site may vary significantly with time. Examples include nondestructive testing of equipment using industrial radiography, short term transit of highly radioactive materials (activated plant components, spent fuel, etc.), and longer term (weekly) staging of these materials as needed. These "other" sources of direct gamma-radiation are normally controlled by the radiological control department, to ensure plant personnel exposures are low. For dose at the site boundary, there are no specific controls. The assumption is that if the dose can be controlled on site, then the dose contribution at the site boundary is probably low. The difficulty is in not knowing specifically what this dose contribution is at the point of interest.

2.6 Noble Gas Immersion

This is a direct gamma source term that is well documented in the literature and in the VYNPS operations documents. Noble gases are released to the atmosphere in allowable small quantities with release limits based on established license conditions for operation. The noble gases also contribute to site boundary gamma dose. This contribution is relatively small during normal operation.

This dose pathway is referred to as the gamma air dose from noble gases. The principal noble gases are ^{133}Xe and ^{85}Kr. While suspended in the atmosphere (like any noble gas), they decay to the ground state isotope and gamma-rays are emitted. This small contribution to site boundary dose is calculated in the ODCM, Section 6.7, "Method to Calculate Gamma Air Dose from Noble Gases." Two methods are presented. If the results using Method 1 are not of sufficient accuracy, Method 2 may be used, which uses actual metrology data to obtain a more accurate value for χ/Q. χ/Q is a term used commonly for atmospheric dispersion calculations where "χ" (or "Chi") represents the ground level concentration (radioactivity per volume) at a specified point of interest and "Q" represents the emission rate (radioactivity released per second).

2.7 Direct Gamma Dose Monitoring for Compliance

Chapter Six provides the details of the direct gamma monitoring studies and evaluations that have been performed since 1972. But in the context of this chapter, which identifies the various direct gamma-ray source terms, a summary table (Table 2) is provided to explain the various calculations or measurement evaluations that have been performed to better understand the direct gamma radiation field at the site boundary. Some of the coefficients and model assumptions in the referenced documents in the table are outdated.

In order to give the reader an idea of the magnitude of the reported direct gamma-dose values, several reports are of interest. Both Entergy-Vermont Yankee and the State of Vermont publish annual environmental monitoring reports. For example, the 2005 Entergy report provides monitoring results from the REMP (Radiological Environmental Monitoring Program). Table 5.2 of the 2005 report presents the summary results of the VYNPS TLD monitoring, showing the inner ring and outer ring TLD values, as well as offsite station values. The TLD results are not conclusive either way, but generally show that the results are equivalent to natural background, with exposure rate values on the order of 6 μR/h. Keep in mind that the calibration protocol for this average annual exposure rate per hour is a long-term average and that the calibration protocol likely bias the rate low compared to an HPIC detector. The 1999 measurements reported in VYC-2067 for example show a background rate of 9.2 μR/h, with the corresponding function fit to the data as a function of power level, with the exposure rate at DR-53 reported in uR/h as a function of power (MW-electric) as a linear function, with slope 6.44E-03 and intercept of 0.94 (R-squared 0.94).

Table 2. Summary of Source Terms for Direct Gamma Exposure and Evaluation Methods

Source of Direct Gamma Radiation	Evaluation Methods	References
^{16}N Skyshine	Measurements and Calculations • MicroSkyshine Calculations • PIC Measurements • MSLRM Measurements • High Resolution Gamma-ray Spectrometry Measurements $D = 3.39E\text{-}6$ mrem/$MW_e h$ at closest site boundary	ODCM, §6.11.1 VYC-2194 rev1 EL 145/05-FINAL Eq p.A-1, VYC-2194 R01.
North Warehouse	Measurements and Calculations • Measured nominal dose rates at source (term R_S of VYC-2194r1, page A-3)	ODCM, §6.11.2 VYC-2194 rev1
Low Level Waste Storage Pad	Measurements and Calculations • Measured nominal dose rates at source and estimated contribution at site boundary	ODCM, §6.11.3
Spent Fuel Storage Areas (ISFSI)	Insufficient details were available at the time of issuance of this report.	
Temporary Radioactive Material Storage Locations or Staging Areas	Insufficient details were available at the time of issuance of this report.	
Other Industrial Operations, including x-ray equipment radiography	Insufficient details were available at the time of issuance of this report.	
Immersion in low-level releases of noble gas	Measurements and Calculations • Noble gas effluents are monitored. Release limit compliance is assured by measurement. • Gamma immersion dose is calculated in the ODCM.	ODCM, §6.7
SUM OF ALL GAMMA DOSE MONITORED BY PASSIVE DOSIMETERS		

3.0 Natural Background Radiation

All people are exposed to natural background radiation every minute of every day. Natural background radiation comes from a variety of sources, including cosmic radiation, cosmogenic radionuclides, and terrestrial sources (including radon gas and its progeny). These sources are described in this chapter as an introduction to natural background in the environs of the VYNPS.

An understanding of natural background radiation is certainly germane to this report because most regulatory compliance decisions are based on radiation levels in excess of natural background. Vermont citizens are interested in radioactivity releases attributed to VYNPS and the plants' responsibility for measuring and reporting these releases (as it does, for example, in its annual radiological environmental monitoring report). Having knowledge of the factors that affect the contribution from natural background is important. In general, these include diurnal and seasonal effects, rain, snowfall, altitudinal (cosmic ray) effects, terrain, geological composition of the area, etc.

The conflicting environmental dosimetry results between a State TLD and VYNPS MSLRM and TLD results from the fourth quarter of 2004 were due in part to a natural background issue. The VDH utilized background readings to establish a "net" reading for its TLDs which it determined exceeded the 10 mrem quarterly dose objective. VYNPS, on the other hand, did not use a background subtraction method (the plant's approved ODCM permits the MSLRM methodology which eliminates background influences. VYNPS does not employ background subtraction for routine environmental TLD monitoring; rather, an accidental release would have to occur per the plant's ODCM and environmental reports).

3.1 Cosmic Radiation

The earth, and all living things residing on it, is constantly bombarded by radiation from outer space. This radiation primarily consists of positively charged ions from protons to iron nuclei derived from the sun and from other sources outside our solar system. This radiation interacts with atoms in the atmosphere to create secondary radiation, including x-rays, muons, protons, alpha particles, pions, electrons, and neutrons. The immediate dose from cosmic radiation is largely from muons, neutrons, and electrons, and this dose varies in different parts of the world based largely on the geomagnetic field and altitude. This radiation is much more intense in the upper troposphere, at an approximate 10 km altitude, and is thus of particular concern for airline crews and frequent passengers, who spend many hours per year in this environment. As reported in NCRP Report No. 94, "Exposure of the Population in the United States and Canada from Natural Background Radiation", the average total effective dose equivalent (TEDE) from cosmic radiation is approximately 27 mrem (0.27 mSv) in this country and our neighbor to the north. (For the readers of this report, a TEDE essentially incorporates both internal and external doses that are received and converts them into an "effective" dose equivalent relative to the whole body.)

In addition to the immediate (primary) dose from the cosmic radiation itself, there is also an indirect dose due to secondary radiation generated by the interaction of cosmic radiation with atomic nuclei in the atmosphere to generate different radioactive nuclides. Many so-called "cosmogenic" radionuclides (i.e., "originating from the cosmos") can be produced, but perhaps the most notable is carbon-14 (^{14}C), which is produced by interactions with nitrogen atoms. Cosmogenic nuclides eventually reach the earth's surface and can be incorporated into living organisms. The production of these radionuclides varies slightly with short-term variations in solar cosmic ray flux, but is considered practically constant over long time scales of thousands to millions of years. Per NCRP Report No. 94, the annual cosmogenic TEDE contribution is quite low—on the order of 1 mrem (0.01 mSv).

3.2 Terrestrial Radiation

Radioactive material is found throughout nature. It occurs naturally in the soil, rocks, water, air, and vegetation. The major radionuclides in this category include primordial radionuclides of the uranium (U-238), actinium (U-235), and thorium (Th-232) decay series. The word "primordial" can be loosely translated as "from the beginning", inferring (and correctly so) that these radionuclides have very long half-lives, i.e., on the order of the age of the earth. Important decay products of the U-238 series include radium-226 and radon-222. Another commonly cited and ubiquitous radionuclide is potassium-40 (K-40). Collectively, these radionuclides contribute an annual TEDE of 28 mrem (0.28 mSv).

3.3 Radionuclides in the Body

The parent radionuclides of the uranium, actinium, and thorium decay series and their progeny, K-40, and other radionuclides can be inhaled or ingested into the body. NCRP 94 estimates the annual TEDE contribution from these radionuclides to be 40 mrem (0.40 mSv).

3.4 Inhaled Radioactivity

By far the largest contributors to inhaled radioactivity are the progeny (decay products) of radon gas. Radon-222 originates from the decay of radium-226, which in turn is a member of the uranium (U-238) series. NCRP 94 estimates the radon TEDE contribution at 200 mrem (2 mSv) annually.

3.5 Summary Contribution from Natural Background Radiation

Table 3 summarizes the contributions from natural background radiation. As reported in NCRP Report No. 94, natural background sources collectively contribute an annual TEDE of 300 mrem.

Table 3. Estimated Total Effective Dose Equivalent Rate for a Member of the Population in the United States and Canada from Various Sources of Natural Background Radiation

SOURCE	TEDE (mrem/yr)
Cosmic	27 (0.27)[a]
Cosmogenic	1 (0.01)
Terrestrial	28 (0.28)
Inhaled	200 (2)
In the Body	40 (0.40)
Rounded Total	**300 (3)**

[a]Numbers in parentheses represent dose equivalent values in mSv/yr.

Of the total TEDE listed in the table, about 80-100 mrem is attributed to direct gamma-ray exposure.

3.6 Contributions from "Background" Radiation

In contrast to natural background, another category exists that ORAU refers to simply as "background" radiation. It originates from artificial (man-made) sources such as medical x-rays, nuclear medicine procedures, consumer products (exposure from television sets, smoke detectors, etc.), fallout from historic atmospheric weapons testing, and contributions from the nuclear fuel cycle. According to NCRP Report No. 93, "Ionizing Radiation Exposure of the Population of the United States", these sources result in an annual TEDE to the population in the United States of about 60 mrem (0.60 mSv). No further discussion of these sources is required in the context of this report.

Figure 6, from the USNRC website (adapted from NCRP Report No. 94) compares the relative contributions from all natural background and background radiation sources.

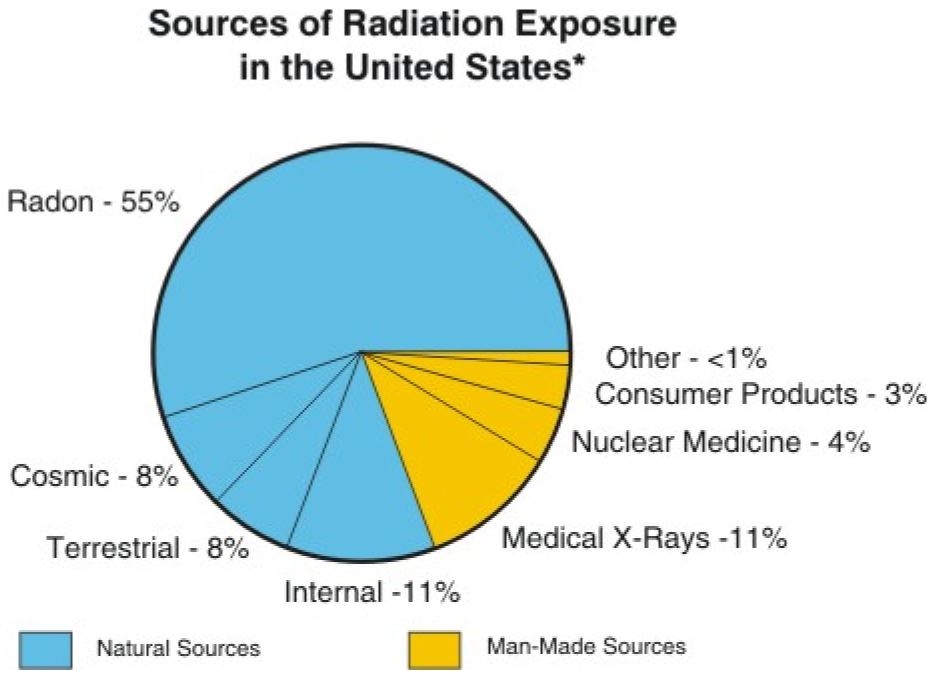

Figure 6. Sources of Radiation Exposure in the United States (adapted from NCRP No. 94)

3.7 Natural Background Gamma Radiation in the Vicinity of VYNPS

Various components of natural background and their corresponding dose contributions have been discussed previously in the context of the entire United States. Natural background radiation in the environs of VYNPS, emphasizing the direct gamma-ray exposure pathway, is discussed in this section.

As noted above, most regulatory-based decision levels for allowable public dose examine compliance endpoints at fractional exposures <u>above natural background on an annual basis</u>. Therefore it is important to examine and determine natural background radiation levels around the plant (and practical uncertainties inherent in assessing background) and explain the VDH regulatory annual dose objective (20 mrem) in this context. Ultimately, it is a challenging problem to deploy cost-effective measurement methods to detect and report public exposures at this level.

ORAU reviewed the VYNPS "Radiological Environmental Monitoring Program" (REMP) for the calendar years 2004 and 2005 which discuss general aspects of natural background radioactivity in the environs of the plant and relates those aspects to the plant's responsibilities to ensure it can detect the "appearance or accumulation of any radioactive materials in the environment caused by the operation of the station". The VDH annual environmental reports have a similar objective.

3.7.1 Factors Affecting Natural Background Exposure Rates

Chapter 6 provides specifics on measurements of direct gamma ray dose at VYNPS and details several factors influencing the natural background in the environs at VYNPS. Many factors do in fact influence the natural background; general information related to some of these factors is included below. These include:

●Weather

- Rainfall

Rainfall has a definite impact (and perhaps the greatest impact) on natural background at the VYNPS. For example, exposure rates measured by a PIC at the plant will "spike" by 4 to5 μR/h above the typical levels recorded by the plant (approximately 10-11 μR/h) within just a few minutes following a rainstorm. The increase can certainly be at least partially attributed to the association of radon progeny with moisture; per an article by G. Klemic in the 1996 International Congress on Radiation Protection entitled "Environmental Radiation Monitoring in the Context of Regulations on Dose Limits to the Public", this effect can increase radiation levels by up to a factor of two and last for several hours. The argument has also been made specific to the VYNPS situation that the increase could be due to the interaction of ^{16}N(skyshine) with moisture; however, similar increases have been observed at locations well away from the plant.

- Snowfall and frozen ground

As Vermont citizens can attest, snowfall is readily prevalent. Snowfall impacts the natural background in various ways, influencing the gamma ray component that would be recorded on the VDH and VYNPS monitoring TLD stations. For example, as with rainfall, snow falling to the ground may bring with it radon progeny, increasing the exposure rate temporarily. ORAU concurs with general comments provided in the VYNPS REMP 2005 report regarding the influence snowfall has on TLD results.

Wet or frozen ground generally results in a reduction in dose recorded by a TLD, but this is not a consistent effect. One variable is the depth of frozen ground which retards the escape of radon gas.

A review of TLD data from VDH and VYNPS covering several recent years is provided in Chapter 6. The effect of snowfall on TLD results is discussed.

● Altitude

With increasing height, the natural background rate from cosmic radiation increases. NCRP Report 94 cites a doubling of the dose equivalent rate for every 6560 foot increase in elevation. To put in perspective, an individual living in Denver, Colorado at an altitude of 5280 ft would receive approximately 50 mrem annually from a cosmic ray dose. The Van Pelt report (1971) cited in Chapter 6 of this report discussed cosmic ray background effects for VYNPS. At an approximate 300 foot elevation above sea level for the VYNPS environs, the estimated cosmic ray background was about 32 mR/y. While far less variable than the terrestrial contribution, the solar cycle that occurs on an eleven year basis increases the exposure rate.

●Terrestrial Gamma Radiation and Geology

As noted above, the U-238, U-235 and Th-232 decay series and the radionuclide K-40 contribute to the terrestrial background component. Van Pelt (see Chapter 6) estimated the annual exposure from the uranium and thorium series and K-40 to be 44 mR.

The geology of the southern portion of Vermont varies depending on the specific area of interest. Depending on the composition, the concentrations of naturally occurring radionuclides and resulting dose equivalent rates will vary significantly. And, in fact, it is not uncommon to analyze samples from geological formations containing higher radioactivity content than samples collected from the VYNPS. Background TLDs deployed in or in close proximity to the towns of Vernon, Putney, and Wilmington would be impacted by a varying geological composition containing quartz, gneiss, metamorphosed sediments, and gray slate to name just a few. The Vermont Geological Survey at the Agency of Natural Resources is one source to provide geological information and the relationship to naturally occurring radiation and radioactivity in the State of Vermont.

●Terrain

The terrain of the area affects the natural background and resulting TLD measurement results. The VYNPS has essentially flat terrain with some hills and ridges. A prominent berm located on the western side of the plant, near TLD location DR-53, affects the resultant readings relative to relatively flat terrain found at other monitoring locations. DR-53 is located on a fenceline above relatively flat terrain; however, the berm rises right behind it. The exposure/exposure rate is impacted by the geometry at this location.

●Fallout Contributions

Fallout radioactivity is primarily associated with cesium-137 (^{137}Cs). Barring a nuclear accident, the contribution from this pathway and radionuclide is minimal. Its contribution is captured through the VYNPS and VDH annual environmental monitoring programs.

●Variability Regarding Inside Versus Outside TLD Locations

Environmental TLDs are routinely located in outdoor locations. While TLD deployment at these locations can be affected by one or more factors, indoor locations are also affected by factors such as building construction and shielding effects. The rationale for indoor deployment can be rationalized as, for example, at the Vernon Elementary School where TLDs are placed inside (and outside) the school. (Results from these locations routinely reveal significantly different (higher) exposure rates inside the school.).

The VDH has deployed for several years background TLDs inside the Putney Town Clerk's office and within an air sampling station at a second background location in Wilmington. ORAU recommends these TLDs be re-deployed to outside locations—in the absence of a technical or other (e.g., vandalism issue) basis—in order to generate what ORAU believes would be a "best practice" and generate more realistic environmental results.

●Background Dosimeter Locations

Background dosimeter locations used by the VDH are located in Putney and Wilmington, Vermont. During its initial site visit in January 2006, ORAU observed the Putney and Wilmington background TLD monitoring locations. Two VDH TLDs have been deployed for several years inside the Putney Town Clerk's office. Following the 2004 fourth quarter TLD results, the VYNPS and VDH agreed in early 2005 to collocate a TLD on a utility pole outside the Historical Society office (Figure 8). As noted above, ORAU advocates the continued use of this location in lieu of VDH deployment inside the building. In the absence of weather, vandalism, or other factors, the Wilmington TLD should be re-positioned external to the air sampling station.

VYNPS utilizes two control dosimeters, identified as DR-4 and DR-5, located 11.3 kilometers (km) south southeast (SSE) of the plant in Northfield, Massachusetts and 16.5 km north northeast (NNE) of the plant at Spofford Lake, respectively. VYNPS background locations are found in Hinsdale, New Hampshire (noted as "DR-14" and "DR-16").

Background TLD results for the VDH and VYNPS monitoring stations are provided and discussed in Chapter 6.

Figure 7. TLD background monitoring station – town of Putney, VT

Figure 8. Closer View of TLD background monitoring station – town of Putney, VT

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4.0 Relating Site Boundary Dose to Dose at the Nearest Residence

From a direct gamma-ray dose perspective, the VDH radiation protection regulations cite two dose values of interest: The 20 millirem per year dose objective "at any site boundary, bordered by land" and the 5 millirem per year dose objective "at the nearest residences in Vermont". Under the existing State of Vermont regulation, direct gamma radiation dose equivalent objectives are established "above background radiation". The relationship of these two values goes back many years. Of the two, the 20 mrem is the primary regulatory compliance objective. Historically, the "20 mrem equals 5 mrem" relationship originated at a time when the nearest resident was located much further away from the site boundary (several residences existed along Governor Hunt Road) than the current situation discussed herein.

ORAU believes this regulatory issue is of significant interest in the context of the site boundary dose evaluation and an important component of our independent review. This section reviews the distinction between the VYNPS site boundary and fenceline and the historical chronology of the "20 mrem equals 5 mrem" relationship, discusses related dose reduction methods, and offers conclusions based on the presumption that the current situation (i.e., resident does not move/sell the property) will remain for the indefinite future.

4.1 "Site Boundary" versus "Fenceline"

It is important to distinguish the "site boundary" from the "fenceline" as there has been both confusion and misunderstanding regarding the distinction between them. The two are not equivalent except at certain locations, including that for DR-52. As provided in Section 15 ("Definitions") of this report:

> The "site boundary" is the area of land surrounding VYNPS that is contiguous, and owned and operated by Entergy Nuclear Operations Inc. under its existing NRC license. **The site boundary is not equivalent to the fenceline.** VYNPS has been purchasing property along Governor Hunt Road while maintaining the existing location of the fenceline at locations DR-53 and DR-52.
>
> The "fenceline" at VYNPS is a security fence established for the purpose of keeping intruders from entering the plant. Other security systems augment the appearance of a barbed-wire fence. **The fenceline may or may not be equivalent to the site boundary.** In most cases, the site boundary (property owned and operated by VYNPS) is extended a significant distance from the fenceline.

4.2 VYNPS 1998 Self Assessment and Event Chronology

In late 1998, VYNPS conducted a "best practices self-assessment", a common practice for many organizations as part of their quality assurance program. VYNPS radiation protection staff were reviewing fenceline TLD data when an observation was made that there may be an issue affiliated with the assumptions historically used to relate site boundary dose rate to dose rate at the nearest residence. The observation was written into a subsequent VYNPS condition report which summarized the issues that were noted during the field activities

("Condition Report CR-VTY-1998-02205 Nearest Site Resident Exceeding Dose Assumptions").

The condition report directly impacted VYNPS in its efforts to determine whether it *fully* complied with the VDH radiation regulations—not as it related to meeting the site boundary dose limit for gamma radiation (in this particular instance), but rather, the dose to the nearest resident. Correspondence on this matter between VYNPS and the VDH ensued.

Based on ORAU's review of the available correspondence, the following chronology took place:

1. VYNPS issued a condition report questioning the assumption that the 20 mrem per year dose at the TLD measurement location DR-52 (a fenceline location equivalent to the site boundary) was equivalent to 5 mrem per year to the nearest resident.
2. This condition report was issued because VYNPS radiation protection staff noticed during their self-assessment that a trailer had been moved closer to the site boundary/fenceline, sometime before the fall of 1998. (The trailer is depicted in Figure 11.)
3. Recognizing that a trailer was situated closer to the fence, VYNPS concluded that the dose rate at the new location was probably not four times lower than the fenceline dose based on an assumed inverse square law calculation (i.e. a 20 mrem to 5 mrem extrapolation).
4. However, on February 10, 1999, Ray McCandless (then the Chief of the State of Vermont Division of Occupational and Radiological Health) issued a reply to inquiries from VYNPS, stating that "A measured or calculated exposure at the site boundary fence of the location in question of 20 milliRems equivalent or less will satisfy the regulatory requirement." The State reinforced its position that as long as VYNPS maintained a site boundary dose below 20 mrem per year, compliance with the 5 millirem to the nearest resident regulation was met.

In ORAU's view, there are two separate, but related issues brought about by the "positioning" of this trailer:

1. The intent by the State of Vermont, in the early 1970's, concerning the basis for the rationale that implies: the licensee need only assure that the annual dose rate at the site boundary (bordered by land) not exceed 20 mrem. And, furthermore, that this measure of compliance would in turn ensure that an individual in an unrestricted area of the plant (i.e. outside the site boundary) would receive less than 5 millirem per year. From first principles and ORAU's direct observation during a site walkdown of this area, it is difficult to conceptualize a habitable situation where the dose equivalent rate at the fence is 20 mrem per year and at a distance of 30 ft further away from the plant, it is 5 mrem per year, unless we account for individual stay times (occupancy factors) in the area and/or a shield is juxtaposed between the site boundary and the residence. To ORAU's knowledge, the dose to the nearest resident was never and has yet to be measured with radiation detectors that would provide a reliable result. Rather, the 5 mrem per year objective has been extrapolated from known measurements at the fenceline.

2. If the State had made this declaration in the 1970's (that 20 mrem at the fence equals 5 mrem at the residence), given a "reasonable" distance from the fence to the nearest residence of 100-200 ft, then the physics of radiation transport would hold, that is, 20 mrem at the fence should translate to less then 5 mrem at the residence.

4.3 ORAU Site Walkdown

In January 2006, at the beginning of its work effort, ORAU conducted a walk-down of the VYNPS west side site boundary, encompassing TLD locations DR-52 and DR-53, and the nearest residence. Observations were made about the location and description of the nearest resident relative to DR-52 (primarily) and DR-53 (secondarily). Figure 9 is taken from SVE Associates with the TLD locations of interest at the west side boundary circled and highlighted.

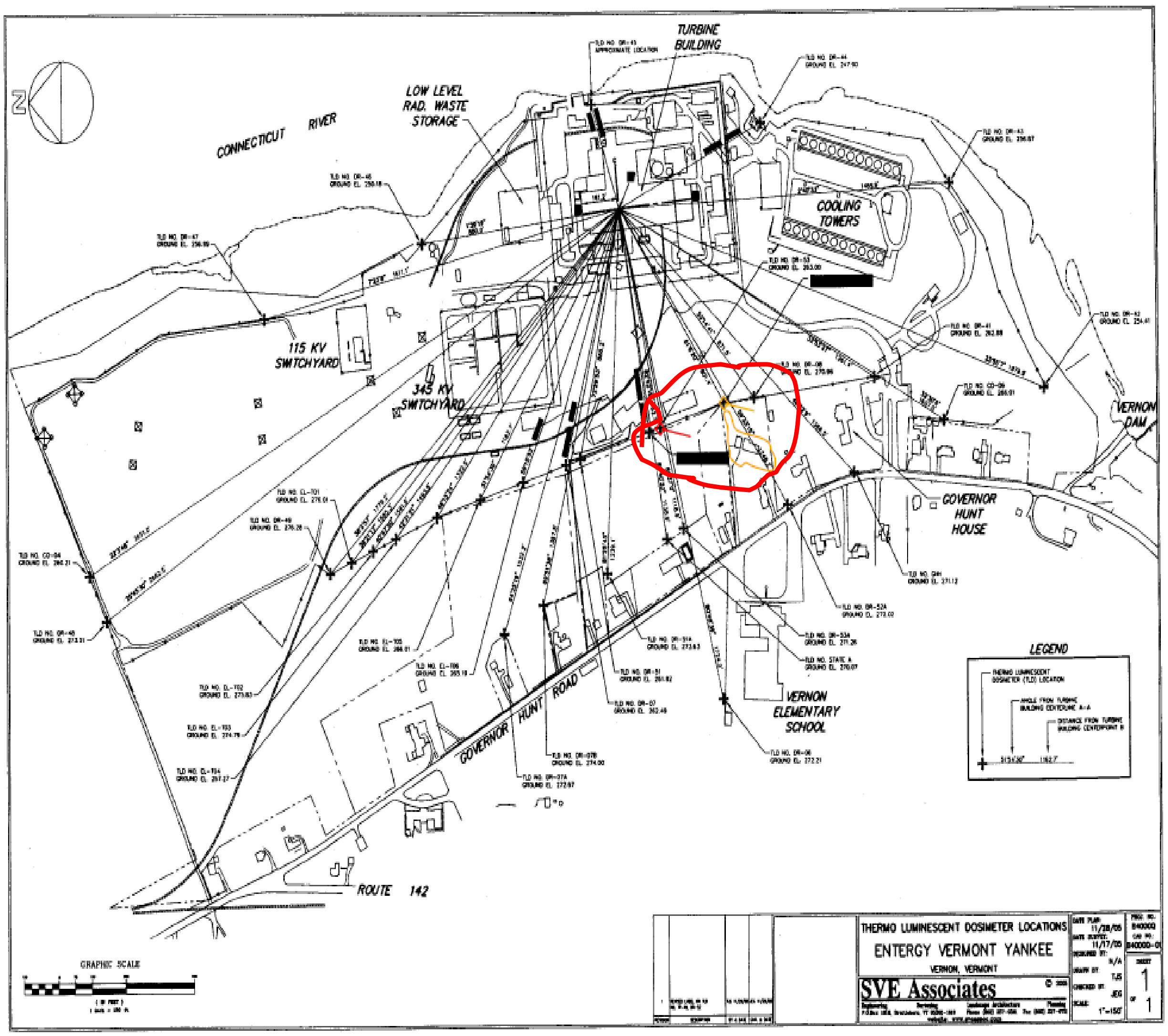

Figure 9. Thermoluminescent Dosimeter Locations

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In examining the figure above, the residence is located nearest to DR-52. The residence was estimated to be located approximately one-third of the distance between the fence and Governor Hunt Road, or about 150 ft from the site boundary, which in turn was about 800 ft from the turbine building. ORAU acknowledges exact distance measurements were not taken during the site walk-down.

What does not appear in Figure 9 is a trailer with a resident occupying the property. The trailer is located where the "green circle" is shown on Figure 10 (on the right). The trailer is located about 30 ft from the west-side boundary, and hence significantly closer to the turbine building than the house at 150 ft from the site boundary. This is an important distinction.

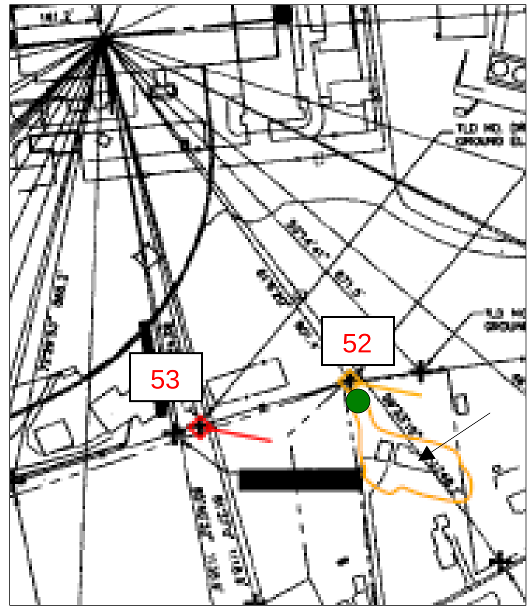

Figure 10. Trailer location near DR-52

Figure 11 was taken from the driveway of the "nearest residence" looking east towards the VYNPS. The plant's fence is located 30 ft on the other side of the blue trailer. The house is located directly to the right of this photo. From this vantage point, the trailer and house appear to be approximately 30 ft and 150 ft from the site boundary, respectively.

Figure 11. Trailer housing the nearest resident

The following photograph (Figure 12) was taken by ORAU at the northeastern corner of the blue trailer, looking east towards the plant. The turbine building (the green, multi-story structure in the background) is located about 830 ft away. The VYNPS fence is about 30 ft from the far corner of the trailer shown in this photo, with the TLD station, DR-52, located just behind the fence. Multiple TLDs are mounted on a metal pole. The red arrows show the locations of the TLDs, which appeared to be positioned at 0.3 m, 1 m, and 2 m heights.

ORAU noted the change in elevation between the fence and the trailer—while not measured during the walk-down, the elevation of the dwelling appeared to be about 15 ft higher than the location of the fence.

Figure 12. Location of DR-52

In contrast to DR-52, DR-53 is the location reported to have the highest dose rate. DR-53 is located about 100 yards north of DR-52 and the blue trailer. Figure 13 shows DR-53, with a tent covering a measurement and test area set up by VYNPS. A PIC can be seen near the fence, and near the TLD, acquiring time-dependent measurement data as part of an ongoing VYNPS study. (Refer to Chapter 5 for general information on PICs.)

Figure 13. Location of TLD Location DR-53

4.4 Exposure Reduction Principles Related to "20 Equals 5"

The issues noted in 4.2 above introduce several interesting factors related to typically encountered exposure reduction principles. Discussed in further detail below, these include time, distance, shielding, and ALARA. The ALARA philosophy is integrated into the current federal and Vermont regulations.

4.4.1 Time

Exposure to radiation is reduced by simply reducing the time spent near the site boundary. In the context of the nearest offsite resident (currently located in a trailer within about thirty ft of the plant's western boundary) and meeting the 5 mrem dose, the fraction of time spent in/near the trailer over a 24 hour, 365 day period is of interest—relative to being physically well removed from the trailer and plant environs to undertake other daily activities. This introduces the possibility of assigning an "occupancy factor", a factor that ranges from 0 to 1 (with "1" indicating the individual/family resides in the trailer all year without interruption.) To date, VYNPS has not incorporated an occupancy factor (to ORAU's knowledge) though their possible use is cited in the ODCM. Occupancy factors, when utilized, have historically been associated with healing arts facilities. ORAU acknowledges their use and potential application at VYNPS; however, implementation of such a factor, when warranted, should be a component of the ECP between the VDH and VYNPS.

4.4.2 Distance

Another common radiation protection tenet is that with increasing distance from a radiation source, exposure and dose rates decrease. Depending on the type of source, the decrease may occur rapidly, that is, with the square of the distance for a "point" source (inverse square law) or in a linear manner (a "line" source). Proximity to the source of radiation is the single most important factor in determining actual radiation dose equivalent.

As noted, the nearest offsite resident is currently located within about 30 ft of the VYNPS western boundary/fenceline in a trailer. However, historically, this was not the case; the nearest resident was located a greater distance from the boundary in a house. For example, the majority of "nearest residents" live along Governor Hunt Road. In 1998, VYNPS observed that the distances historically used to describe the "nearest resident" had changed, leading to questions regarding full compliance with the intent of the VDH regulation. It was noted then that complying with the 5 mrem dose to the nearest resident may be more of a challenge.

When estimating radiation exposure rates as a function of distance from a radiation source, it is very important to know exact distances between the location of interest and the source. It is also very important to understand the geometry of the source and the scattering media between the source and the location of interest. And, for the case of skyshine radiation, it is important to note that no analytical function is easily calculated to provide estimates for exposure rate as a function of distance. As a result, a simple inverse square function is not applicable. The approximate distances of interest were estimated during the ORAU site walk-down of the western boundary on January 5, 2006:

Turbine Building to TLD-DR-52:	801.4 ft (from SVE drawing, TLD mounted on fence)
Fence to nearest corner of trailer:	30 ft (visual observation from top of berm)
Turbine building to trailer:	831.4 ft (801.4 ft + 30 ft)
Fence to nearest corner of house:	150 ft (estimated from SVE drawing)
Turbine building to house	951.4 ft (801.4 ft + 150 ft)

Distance is the most important factor for extrapolating an estimate for actual dose received. Citing the results published in 1976, VYNPS published a curve (based on measurement data) showing the drop-off in the net exposure rate (X) with distance (d) in feet as:

$$\boldsymbol{X\ (net) = 100 * 10^{\left(-d/492\right)} \qquad \left(\mu R\ h^{-1}\right) \qquad (\textit{circa}\ 1976)}$$

This known physical effect was originally used by VDH to conclude, for example, that a 20 mrem annual dose received by a fictional person living at the fenceline every minute of every day for one year, would equate to a real maximum dose of 5 mrem to the nearest resident located pre-1998 about 150 ft from the fenceline.

4.4.3 Shielding

VYNPS has utilized shielding on multiple occasions to reduce the offsite dose from its operations. In 1976, the plant constructed a shield wall on the west side of the turbine for this purpose. In 2006, a shield was installed above the turbine in an attempt to reduce the skyshine contribution. Both situations are examples of the ALARA philosophy where technical and societal (public impacts) were introduced.

While VYNPS has not done so, and emphasis on regulatory compliance remains with meeting the site boundary dose based on historical agreements between VYNPS and Vermont regulatory officials, a "credit" for the shielding provided by the offsite trailer is a consideration when the 5 mrem dose objective is discussed. ORAU is not aware of a situation pertaining to any other nuclear plant in this country where an offsite resident lives so close to a plant's site boundary. If required in the future for compliance purposes, ORAU believes that a reduction in the person's dose to a man-made radiation contribution from VYNPS—due to the shielding afforded by the trailer's composition—is acceptable. However, ORAU does not believe the magnitude of that factor would be significant or extremely beneficial based on our visual observations during the initial site walk-down in January 2006. As warranted, ORAU recommends the ECP be used as the vehicle to determine an appropriate shielding reduction factor acceptable to the State and VYNPS.

4.5 Local Background Estimation

The preceding discussions in this section and Chapter 3 reiterate the importance of understanding the direct gamma ray contribution from natural background because the applicable federal (USNRC) and VDH radiation regulations (refer to Chapter 1) are based on meeting cited dose limits/objectives that represent a "net" reading above background. The gamma ray background at the trailer housing the nearest resident (and near vicinity) is estimated based on the TLD results from DR-52 and DR-53, located on the west side boundary— not from measurements taken inside or near/at the trailer. In addition, a PIC study conducted in the summer of 2005 indicated that the calculated annual dose at the blue trailer would be between 5 and 7 mrem. This dose estimate depended on the VYNPS and VDH background stations, respectively, without taking into account the EPU and any consideration for residence (occupancy) time and shielding provided by the trailer.

4.6 Conclusions

ORAU concurs with the McCandless (VDH) 1999 response that meeting the 20 mrem dose objective is the primary (controlling) dose objective, in part because it is not the fenceline that has moved—only the location of the nearest resident relative to the fenceline. The 5 mrem dose objective is a legacy of prior years when ORAU presumes an assumption was made that the increased distance of several residences along Governor Hunt Road from the fence would not cause any concern with meeting this secondary dose objective. (Refer to Appendix A for a detailed discussion of the Vermont regulations.) One avenue for consideration would be the conduct of a dose assessment covering a minimum of one year, employing the placement of dosimeters at the trailer and on each resident residing within. As noted previously, the use of occupancy factors for this particular situation is also worthy of consideration.

5.0 Instruments and Detection Devices for Measuring Direct Gamma-ray Dose at the Site Boundary

This chapter introduces the various radiation measurement devices that can be and have been used to measure direct gamma dose at the VYNPS. These include pressurized ion (ionization) chambers, high resolution gamma-ray spectrometers, and thermoluminescent dosimeters. Chapter 6 then provides information tailored more specifically to instrument applications utilized by VYNPS, VDH, and the USEPA.

5.1 Pressurized Ion Chambers

Ionization chambers ("*ion*" chambers) are radiation detection instruments that directly measure exposure rate in real-time. Ion chambers consist of a pressurized gas-filled volume. Gamma rays passing through this volume interact with the molecules of the gas, creating ion pairs. An electrical voltage potential is applied across the gas volume, causing the freed ions to be collected by anode and cathode electrodes. Associated electronics, utilizing properties of the fill gas, relate the rate of charge collection (coulombs per second) to units of exposure rate (e.g., microroentgen per hour). No information is gained regarding the energy or type of radiation causing the ionization, but in practice environmental PIC detectors measure only the gamma-ray component of ionizing radiation.

In general, ion chambers may be grouped into two types for typical use as: 1) high range (high exposure rate) detectors; and 2) low-range (very low exposure rate) and long-term environmental monitoring devices. The first category often employs hand-held instruments used for radiation protection purposes. Exposure rates encountered range nominally from several tenths of a milliroentgen per hour (mR/hr) to 100 roentgen per hour (R/hr) and higher. In contrast, environmental "pressurized" ion chambers measure ambient exposure rate in the microroentgen per hour (μR/h) range, and are easily able to distinguish fluctuations around natural background levels of 10 μR/h.

Fill gases come in a variety of forms, and may be either "normal" air, purified air, argon, or organic (carbon-based) gases. A goal in the design of most ion chambers is to make the combination of the wall material and fill gas as tissue-equivalent as possible.

Most hand-held ion chambers have fill gases that are either vented to the atmosphere, or are at atmospheric or near-atmospheric pressure. Additionally, hand-held ion chambers must be small enough to be carried and used in practical situations. The combination of a small gas volume and low pressure of a hand-held ion chamber results in relatively low measurement efficiency, especially compared to the low exposure rates encountered in environmental measurements. Hence hand-held ion chambers are normally used only for relatively high-range, radiation protection purposes (mR per hour and greater), though there are a few new hand-held models capable of measuring at background levels. Common hand-held ion chamber specifications are provided at the website link below, including the Victoreen Model 451B and 451P (the high pressure version):

http://www.promis-electro-optics.com/doc/cardinal_medical_imaging/RS2.pdf

In order to measure very small exposure rates and to operate under a wide range of environmental conditions, ion chambers intended for environmental measurements are designed differently from hand-held ion chambers. The sensitive volume is larger than in hand-held ion chambers, and is filled with a high pressure fill gas (up to 25 atmospheres). The larger volume and high pressure of the fill gas create a much larger mass for the gamma radiation to interact with, and result in much higher measurement efficiencies. These types of ion chambers are appropriately referred to as a pressurized ion chamber or as an HPIC (High Pressure Ion Chamber). In the last ten years or so, it is the HPIC which has been used to measure and report ambient radiation exposure rates, and in particular, at VYNPS, the evaluation of exposure rate at the facility fenceline, DR-53.

Pressurized ion chambers have several notable advantages for use in monitoring environmental radiation levels. The energy response is uniform for most energies found in the environment. They measure environmental exposure rates directly. The sensitivity is excellent; pressurized ion chambers can measure a fraction of natural background. They can also be used to measure and electronically record exposure rate and integrated exposure.

Several disadvantages of environmental HPICs are the relatively high cost, the greater size and complexity, maintenance and calibration issues; and the time required to set up and make measurements. In this regard, these detectors are normally used to conduct high-precision and accuracy measurements over short-term, experimental studies, rather than continual (24 hour/day) measurements.

A specific type of HPIC commonly used for environmental monitoring is the General Electric Reuter-Stokes series. These detectors were originally developed to allow radiation protection personnel to study and evaluate very low exposure rates, and observe the fine details of ambient background radiation as a function of terrain, topology, and with time. Reuter-Stokes environmental monitoring systems are typically offered as an all-in-one system, complete with the detector itself and the associated electronics and data collection system, with communications protocols, either serial RS-232 or over a wide area network. Reuter-Stokes detectors typically have good response abilities, with an advertised accuracy for the RSS-131 model of +/- 5% at 10 μR/hr. VYNPS used the RSS-131 model ion chamber during its recent studies for the power uprate, as described in the SER, Section 7, "Fenceline Dose Monitoring with the PIC."

Specifications for the widely used GE Reuter Stokes HPIC (Model RSS-131) can be found online at:

http://www.gepower.com/prod_serv/products/radiation_monitors/en/downloads/rss131.pdf

Most environmental monitoring applications do not require full time monitoring of ambient radiation. Real-time measurement and analysis of the radiation field is normally only needed when the radiation rates may change abruptly or the operation which produces the radiation behaves in a rapid (and somewhat unpredictable) manner. HPIC detectors have been used traditionally for evaluating the status of cleanup activities in decommissioned nuclear facilities.

The environmental HPIC detector is not difficult to calibrate. The quantity measured, exposure rate, is by far the most straightforward radiation interaction quantity of interest to measure. The detectors are designed and manufactured in environmentally-robust enclosures. The pico-ammeters used in existing systems are very accurate to measure the small currents produced from the formation of ion pairs in the cavity. As a result, if the intended measurement is to determine exposure rate at a given location and over fine time periods (30 second intervals), the HPIC system is the best choice. Converting from exposure rate to a dosimetric quantity is a matter which requires a significant effort, as discussed in Appendix B. If a dosimetric quantity measurement must be obtained in real time, the tissue equivalent proportional counter is an excellent choice, though it is not discussed in this report because it has never been deployed at VYNPS to evaluate dose equivalent rate.

5.2 High Resolution Gamma-ray Spectrometers

In contrast to ion chambers, which measure total exposure rates without recording individual radiation interaction events, high resolution gamma-ray spectrometers record the energy deposited in each detected radiation interaction event. This provides the ability to evaluate the energy spectrum of the incident gamma radiation, which in turn allows identification of specific radionuclides according to their known gamma-ray energies. A benefit of this ability to discern the energies of incident radiation is that natural background radiation can be distinguished from man-made radiation. Mathematical algorithms exist for several instruments to convert the measured energy spectrum to exposure rate and dose equivalent rate (see Appendix B, for example).

Gamma-ray spectrometers are classified in two groups: low resolution and high resolution. Sodium iodide (NaI) gamma-ray spectrometers are low resolution, but are the most widely available. NaI spectrometers have the advantages of being relatively inexpensive, portable, and having a high detection efficiency. However, the energy resolution is poor, so gamma-rays of similar energies can be very difficult to distinguish from each other, if possible at all. Prior to operation of the VYNPS in 1972, Van Pelt conducted an extensive set of NaI measurements in and around the site to document "natural background" conditions. The results of these spectrometer measurements are provided in Chapter 6.

The best technology currently available for high resolution gamma-ray spectrometry is based on high purity germanium (HPGe) crystals. HPGe gamma-ray spectrometers have excellent energy resolution, and are able to discern gamma-ray energies normally to within one to two kiloelectron volts (keV). Disadvantages include the high cost of the detectors, and the requirement that the detector be cooled by liquid nitrogen prior to and during any measurement. HPGe gamma-ray spectrometers come in a variety of configurations dependent on the measurement objectives and conditions. A special type of HPGe detector is the reversed-electrode germanium (ReGe) detector. An ReGe is an HPGe detector in which the electrodes are reversed from a conventional coaxial detector. There are two advantages to this electrode arrangement—reduced window thickness and radiation damage resistance.

High resolution gamma-ray spectrometers provide laboratory grade measurements to be conducted in the field (*in situ*). In 2002, VYNPS conducted an extensive study using a variety of collimated/shielded HPGe detectors to refine prior estimates for the contributions of the direct gamma-dose: cosmic, terrestrial, direct line of site, and skyshine. These estimates for the makeup of the direct gamma radiation field at the fenceline can only be performed with this type of measurement device. By collimating the detector and arranging the orientation in certain ways, estimates for the direct ^{16}N component, the skyshine component, and the cosmic component can be measured. By pointing shielded/collimated detector towards the earth, each of the primordial radionuclides can be directly measured to estimate the terrestrial natural background component. Finally, with the gamma-ray spectra produced from such measurements, conversion to dosimetric quantities, such as dose equivalent in mrem, is possible. The details of the application of HPGe measurements at DR-53 are presented in Chapter 6.

Calibration of HPGe detectors is more complicated than the HPIC detectors discussed in Section 5.1 when the results are going to be used to quantify the activity concentration of naturally occurring radionuclides detected in soil. However, when calibrated to study the differential energy flux of gamma-rays impinging the detector, the calibration need only ensure the gain is set sufficiently high to detect higher-energy gamma rays and that the multi-channel analyzer has been calibrated for photon energy. For this purpose, the detector output is differential energy gamma-ray flux, which can then be converted to essentially any radiation measurement unit of interest, either exposure (roentgen) or dose equivalent (rem).

The two primary suppliers of high resolution gamma-ray detectors and associated electronics are Canberra Industries and Ortec/Ametek. The VYNPS *in situ* studies of 2002 utilized the Canberra ISOCS (in situ object counting system) detector system, which includes a set of collimators, back-shields, and electronics.

5.3 TLDs

A thermoluminescent dosimeter (or "TLD") is a passive measurement instrument for measuring the cumulative exposure, primarily from gamma-rays, typically over a 3-month to a one-year period of time. When calibrated for tissue equivalent response and deployed accordingly, the TLD measures dose equivalent, and is commonly used as the dose of record for radiation workers. TLDs are also routinely deployed for environmental monitoring by nuclear facilities, mostly as a dose reconstruction tool following a significant accidental release of radioactivity from the facility. Thus, for environmental monitoring purposes, the TLD is commonly deployed to measure cumulative annual exposure well over 25 mR to 1000 mR.[20] In the environment, and specifically at Vermont Yankee and its environs, TLDs are placed at specific locations of interest on a quarterly monitoring cycle. They are then collected and returned to a contracted analytical laboratory for reading and analysis while a new set of TLDs is positioned. In contrast to ion chambers, TLDs do not monitor in real time. The results of the cumulative measurement period are normally delayed by 30 days from the last day of the monitoring period.

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[20] Intercomparison studies and experimental evaluations of environmental TLDs have shown that reliable detection of about 25 mR (gross) is achievable {[20]Klemic, G, et al., "Results of the Tenth International Intercomparison of Environmental Dosemeters," Rad. Prot. Dosimetry, Vol. 58, No. 2, pp 133-142, (1995).

TLD designs vary, but each TLD normally consists of several radiation sensitive elements, often called "chips", which are on the order of several millimeters in size. As radiation interacts with each chip, some of the energy deposited is stored in a retrievable form. When the TLD is read in a reader designed by that manufacturer, each element is heated (various heating methods are used) and the stored energy is released as light. The light is measured as it is released, with the amount of light directly proportional to the total energy deposited in the element over the monitoring period. The cumulative exposure to each element over the monitoring period can then be calculated using known calibration and response functions.

There are several types of TLD materials utilized. Typical materials include calcium fluoride (CaF_2), calcium sulfate ($CaSO_4$), lithium fluoride (LiF), and lithium borate ($Li_2B_4O_7$) with trace quantities of other elements (commonly copper or manganese) added to modify its performance. The material used for the elements has a very large impact on the sensitivity and relative response to different types and energies of radiation. Oftentimes, a thin metal layer is placed in front of one or more elements to attenuate (shield) gamma-rays below a certain energy. By using different thicknesses and types of metals for these thin layers, each element can be made to respond to gamma-ray energies above a different energy cutoff. By comparing the relative exposures to several elements in a TLD, information is gained about the energy spectrum to which the TLD was exposed. The atomic number of the casing material will also strongly affect the energy response and the low energy cut off. High-sensitivity TLDs usually over-respond to low energy gamma-rays, requiring the use of calibration factors to correct this situation.

For twelve years, an extensive international environmental TLD intercomparison program was directed by the Environmental Measurements Laboratory in New York. Specific TLDs intercompared include: LiF(nat):Mg,Ti; ^{7}LiF:Mg,Ti; LiF:Mg,Cu,P; $CaSO_4$:Tm; CaF:Mn or Dy; and Al_2O_3:C. In practice, the calcium sulfate chip has been the most widely used for environmental dosimeters, with LiF-based chips gaining in popularity (particularly now with the introduction of copper as the doping element). G. Klemic's article published in Radiation Protection Dosimetry should be examined for the results of the 10th intercomparison.[21]

The results of these intercomparison studies were used to formulate draft performance specifications for an ANSI standard that has yet to be published, and probably never will because the interest in making such precise, accurate environmental dosimetry measurements at low levels of exposure is waning—simply put, the financial resources required to meet this level of performance and reliability do not exist. Much of this extensive set of work in the 1990s was cancelled after the 12th intercomparison study. As will be explained below, this level of effort is in sharp contrast to the practices employed for accreditation of TLD providers for personnel dosimetry.

During the course of the ORAU evaluation, it was noted that the VDH and VYNPS have utilized different TLD vendors and manufacturers. Specifics regarding these vendors, manufacturers, the TLD materials used in those dosimeters, uncertainty in the measurement

[21]Klemic, G, et al., "Results of the Tenth International Intercomparison of Environmental Dosemeters," Rad. Prot. Dosimetry, Vol. 58, No. 2, pp 133-142, (1995).

results, and ORAU's conclusions and recommendations regarding the environmental monitoring programs implemented by both parties, respectively, are discussed in Chapter 6.

In recent years, another dosimeter with a different theory of operation than a TLD was introduced for primary use in an occupational workplace setting for monitoring of personnel radiation dose. Known as an optically stimulated luminescent (OSL) dosimeter, it has more recently begun to find its way into limited use as an environmental dosimeter. An OSL contains aluminum oxide (Al_2O_3), doped with carbon, or Al_2O_3:C, a crystal. The OSL was introduced by Landauer Inc., and is marketed with three advantages over the TLD: sensitivity in the 1 mrem range (versus 20-25 mrem for the TLD), wider energy response, and greater stability. The lower sensitivity specification will likely make the OSL an attractive choice for environmental monitoring at natural background levels, though the copper doped, LiF TLD appears to have the sensitivity of the OSL.

Calibration and performance testing of TLDs is very complicated. But, there has also been a tremendous amount of development and evaluation of detector response and qualification for application to personnel dosimetry. Because so many TLDs are distributed worldwide for personnel monitoring (hundreds of thousands), a rigorous qualification program has been developed to ensure that TLD vendors meet minimum qualification requirements. For United States commercial applications, the NVLAP program is used. This extensive calibration and qualification process enables personnel dosimetry results to be provided in units of dose equivalent.

This same level of effort for environmental applications has not been required, and as a result, environmental TLD results can be and are reported in a variety of different units. And until a calibration and qualification program is developed for the application of environmental dosimeters for assignment of public dose or population dose, there is no motivation to apply this significant amount of effort going forward. The fact that environmental TLDs do not need to be deployed in a consistent way throughout the country leads to significant confusion in the interpretation of results between and among measurement groups. For example, VYNPS and VDH use different TLD vendors. Because there is no uniform "best practice" requirement or qualification program, each vendor selects the algorithms it chooses to run in the analysis of the glow curve and the reduction of the data. Only when national and international intercomparison programs are implemented with a defined reporting structure, can "real" intercomparisons be accomplished.

In practice, the calibration and qualification protocols between personnel TLD dosimetry and environmental TLD dosimetry are best illustrated by example. If three NVLAP-qualified TLD vendors place TLDs on a radiation worker for three months, run the dosimeter analysis and report dose equivalent (mrem), the result will agree within the nominal performance specifications within the national standards and NVLAP protocols. The agreement should be excellent, e.g., to within 5% of one another. The extensive calibration protocols required by the personnel dosimetry accreditation process ensure excellent reliability. If on the other hand, three NVLAP-qualified TLD vendors place their environmental TLDs at the same "natural background" location for three months, the reported units will likely be different because the calibration protocols are not normalized, tested, and evaluated to the same degree of reliability. It has not been necessary. For example, the analysis of VDH and VYNPS environmental dosimetry results shows a constant bias: the VDH results are consistently 25% greater than the VYNPS results. In addition, the variability between the two TLD systems is different as well. If environmental TLD measurements are to be used to qualify a quarterly site boundary dose greater than 10 mrem, then a significant amount of attention should be placed on device calibration, qualification, and testing, which heretofore has not been required of any nuclear operation.

This level of effort, while possible, has long been regarded as not plausible. Hence, environmental TLDs have traditionally been deployed to provide trending data; to ensure that no significant unplanned radiation exposure event occurred that would have otherwise gone undetected; and in the event of an accidental release of radioactivity, the ability to evaluate the plume of radioactivity, radiation levels, and project accident-related doses to the public. Because TLDs are relatively inexpensive, they are typically deployed in nearly hundreds of locations around operating nuclear facilities.

In this chapter, it is particularly important to understand the capability and limitations of environmental TLDs and the fact that a qualification program does not exist because it is the lack of such formality that led to an observation by VDH in 2004 that VYNPS may have exceeded the quarterly and annual dose objective at the site boundary. ORAU then was asked to evaluate the authenticity of this result and provide some insight as to why the VYNPS TLD data, the VDH TLD data, and the other monitoring approaches did not agree. In fact, there is little insight to be gleaned when measuring such low levels of ambient radiation because the fine details that would have to be implemented to make this assessment were never in place.

The major providers of TLDs and TLD readers are:

Panasonic
http://www.panasonic.com/industrial/other/other_components_radiation_measurement_home.htm

This website provides excellent summary information, including a table of dosimeter types and performance, technical data, and issues dealing with spurious results. Panasonic claims reliability of use at exposures as low as 10 mR/month. VYNPS subcontracts environmental dosimetry to the AREVA NP laboratory in Westborough Massachusetts, which provides Panasonic TLDs. The TLDs are deployed at the monitoring locations by VYNPS staff, collected by VYNPS staff, and sent to AREVA for TLD reading, analysis, and reporting. The TLDs use calcium sulfate elements, the UD-804A series, for measuring gamma-ray exposure. The AREVA dosimetry laboratory is NVLAP-accredited for personnel dosimetry, code 100524-0.

For VYNPS results, AREVA presents quarterly results in μR/hour.

Global Dosimetry Solutions (GDS)
http://www.dosimetry.com/

This website provides, like Panasonic, excellent summary information on the Harshaw TLDs and readers it provides. But unfortunately, it does not provide information on the Panasonic-based TLDs and readers that are supplied and used by VDH. A common environmental dosimeter is the TLD-100, comprised of two calcium fluoride and two lithium fluoride elements. GDS claims that the TLD-100 is capable of detecting to low levels of 5 mR/month. GDS is also NVLAP accredited for personnel dosimetry, code 100555-0. VDH subcontracts environmental dosimetry analysis to GDS, following a transition from Proxtronics, Inc. last year. The four-element Panasonic TLD-814, with E1, a $Li_2B_4O_7$ chip; and E2-E4, $CaSO_4$ chips, are used. GDS claims that the performance of TLD-814 is at the 5 mR/month level.

> For VDH environmental dosimeters, GDS reports results for each of the three elements in mR* per quarter. GDS indicates that mR* is used to indicate that "no control exposures have been subtracted, and only one element, reader and fade corrections have been made." Additional important footnotes in dosimetry reports account for "unusual element results," and "element damaged," demonstrating that statistical treatment of the data is required to account for issues that arise in deployment and analysis.

Landauer Inc.
http://www.landauerinc.com/

Landauer is by far the largest single provider of dosimeters. Landauer has commercialized the aluminum oxide (Al_2O_3:C), OSL dosimeter under the Luxel trade name. Landauer claims many advantages of this dosimeter over conventional TLDs or film badges, and continues to be a market leader in personnel dosimetry. The application of OSLs to ruggedized environmental conditions is progressing, but it appears at present that very few users have deployed OSLs for ambient monitoring in the environment. Landauer is also NVLAP accredited for personnel dosimetry, code 100518-0.

The NVLAP program plays an important role in assuring quality for personnel dosimetry providers. While there is no NVLAP accreditation process for environmental dosimetry, many performance criteria for personnel dosimetry are applicable, though incomplete. This fact is illustrated, for example, by reviewing the accreditation categories: whole body, and extremity (not environmental). All NVLAP accreditation requirements and performance metrics are managed by the U.S. National Institute of Science and Technology: http://ts.nist.gov/Standards/214.cfm.

5.4 Micro-R Meters

A very common hand-held device is the micro-R (microroentgen) meter. It's a very sensitive gamma-ray detector that measures very small exposure rates (μR per hour), much like the HPIC detector mentioned in Section 5.1. The micro-R meter uses a NaI scintillation detector. The output is conditioned by an amplifier. A low-level threshold is set to reject events considered noise from the photomultiplier tube. The meter has a visual scale and an audible output. These detectors are excellent for locating a misplaced source or conducting a rapid survey of an area to ensure no gross radiation source was inadvertently left in place. Because the response is temperature sensitive and the detector is NaI (not tissue equivalent or even air as in the case of the HPIC detector), these detectors are best used for "indication only."

One can be mislead that, because the scale reads in μR/h, the reading from this device should be identical to a reading from the HPIC detector placed next to it. This is only true if both devices are cross-calibrated for the experimental situation of interest (a rare event).

A common micro-R meter, the Model 19, is manufactured by Ludlum instruments:

http://www.ludlums.com/product/m19.htm

As described in the ORAU report of the site walk-down conducted January 5, 2006, a Model 19 instrument was held at waist level to measure ambient exposure rates while the reactor was operating at full power. ORAU conducted these measurements just outside the fenceline in the vicinity of DR-52. Results ranged from 12 to 13 μR per hour. These results understandably lead to a question as to whether this translates into a radiation level on an annual basis exceeding the VDH site boundary dlose objective. ORAU's answer to this is that for this instrument, if background had been taken in this location while the reactor was at zero power and the exposure rate determined to be greater than 9.6 μR/h, then the time-integrated net exposure for one year would be less than 20 mR at a gross exposure rate of 12-13 μR/h as measured. This example analysis however is flawed because a micro-R meter should not be used for this level of analysis, but rather, for indication only. Attempting to compare micro-R meter results with HPIC results and converting to dosimetric quantities must be done very carefully.

6.0 Methods and Results for Measuring Direct Gamma-ray Dose at VYNPS

This chapter presents a summary of the methods used and currently in use to measure direct gamma-ray dose at the site boundary. Results from studies and evaluations are also presented in summary form, with citations included. This is an extensive review because it is important to understand the development of various methods for determining net dose equivalent from direct exposure to gamma radiation.

As early as 1969, when construction of the plant began, direct gamma-ray dose to the public was being addressed. Pre-operational radiation monitoring evaluations were made to understand natural background levels. Many follow-up evaluations were made both by the state and by plant health physicists to understand direct gamma dose, quantify the radiation levels, and ensure that annual public dose objectives were not exceeded.

In ORAU's review of historical technical reports, direct gamma-ray dose has been extensively studied, measured, and evaluated. Recently, the direct gamma-ray dose pathway has continued to take on more interest, largely because of a power uprate license amendment issued by the USNRC to VYNPS.[22]

ORAU reviewed many monitoring reports and results related to the direct radiation pathway. It was not ORAU's objective to look for flaws in the monitoring methods, but simply to report on the methods and results of the past, all of which are categorized by independent review as to whether they constitute "best practices." An example of a best practice occurred in 2002 when VYNPS conducted an evaluation of site boundary dose, using a shielded/collimated high resolution gamma ray spectrometer to directly measure the various components of site boundary dose. To ORAU's knowledge, this type of evaluation had never been performed operationally at any other facility in the world. The 2002 measurement evaluation was necessary to establish a better understanding of the components of background radiation and the higher gamma-ray energy component from ^{16}N.

A significant number of measurements have been performed to evaluate natural background radiation and to evaluate net gamma-ray dose to the public from VYNPS operations. The USNRC and the State of Vermont have reviewed these results since the reactor was placed into operation.[23] **Through this review, ORAU finds no indication (or statistically significant evidence) that the annual dose objective at the site boundary of 20 millirem has been exceeded.** This conclusion extends through the thermoluminescent dosimetry results in 2004 that were apparently higher than historical averages. While this conclusion can be reached, it is also apparent that VYNPS was approaching (not exceeding) the objective for members of the public who resided near the west site boundary on Governor Hunt road.

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[22] On March 2, 2006 the USNRC issued license Amendment 229 to VYNPS operating license DPR-28, increasing the maximum authorized power level from 1593 MWt to 1912 MWt.
[23] For example, refer to David Scott's (VDH) extensive set of monitoring reports (1974, 1977).

The goal of this chapter is to present specific methods used and results reported in order to reach three conclusions:

1) On an annual basis, no statistically significant evidence exists to suggest that the site boundary dose equivalent objective was exceeded, even given the apparently higher than normal TLD measurement recorded by the State of Vermont in the fourth quarter of 2004. This conclusion is reached with the assumption used in the VYNPS methodologies developed in the 1990s that the exposure to dose equivalent conversion is no greater than 0.71;

2) On a quarterly basis, methods used to make an evaluation of compliance at 10 mrem above background are of insufficient statistical precision. Existing TLD measurement practices cannot statistically resolve between an actual net dose of 7 mrem and 12 mrem (the “net” readings above background) in order to make a reliable, statistically valid, determination of compliance. Proper selection of the quarterly background rate for each TLD location also remains a problem. On the other hand, the VYNPS main steam line monitoring results show that the ^{16}N skyshine component was less than 10 mrem per quarter, since development. Integrating measurements at either DR-52 or DR-53 show long-term trends of having satisfied this criterion, but it cannot be claimed with complete certainty that in the fall of 2004 an exceedance of 12 ± 3 mR actually occurred; the integrating dosimeter measurement procedure and process needs improvement going forward; and

3) Annual site boundary dose is small, about 20%-25% of natural background (inhalation dose from radon progeny excluded), but, has been increasing toward the 20 mrem per year dose objective as a result of power uprate testing and improved plant water chemistry.[24] If the VYNPS measurement method results are converted to units of exposure, the dose equivalent result that is currently being reported must be divided by 0.71. ORAU’s analysis (Appendix B of this report) shows that the multiple should be about 0.6.

On the basis of this review and looking toward future operating conditions, further engineering efforts may need to be examined to ensure site boundary dose is less than 20 mrem per year. Future operating plans will put pressure on this limit, and the measurement methods to demonstrate compliance will require better accuracy and precision.

6.1 Historical Perspective

Direct gamma measurements have been made continuously on the site since late 1969, prior to operation of the plant. The measurement methods and results are presented in three separate time periods:

[24] Improved plant water chemistry techniques for corrosion control tend to increase ^{16}N production, thereby increasing site boundary dose from skyshine. ^{16}N production also increases with power level, as discussed in the USNCR license amendment 229 for extended power uprate.

- The Early Years (1969-1978)
  Several reports were written between the State of Vermont, VYNPS, the USEPA, and the USNRC on the issue of direct gamma dose. Prior to operation in November of 1972, background measurements were made. Preliminary analyses were conducted to predict contributions from ^{16}N at the site boundary. Early measurement results necessitated engineering efforts to construct additional shield walls around the steam turbines to further minimize dose concerns. Once the shield walls were installed and measurements were conducted to confirm that direct radiation exposures satisfied the annual objective, a site boundary PIC was taken out of service, as communicated in a 1978 memo by Desrosier. This was the last word on the matter until 1998.

- The 1980s to early-mid 1990s
  Since there was no correspondence discovered covering this time period, ORAU assumes that the memo sent by Desrosier to Moody in 1978 brought all discussion about site boundary direct gamma dose to a successful conclusion. Meanwhile it does appear that Scott from the State was very active briefing the State of Vermont public health department on how the overall state compliance program should function. D. Scott's efforts appear to have reached an impasse in 1977, when his review was sent to the Governor of Vermont. No records exist between 1978 and the mid 1990s on the issue of direct gamma dose at the site boundary.

- The 1990s
  In the 1990s, a few operational changes were underway. Plants (including VYNPS) began to evaluate potential improvements to water chemistry in an effort to prevent or minimize corrosion. At the same time, the population density near the plant began to change—some nearby residents moved away from the plant, others actually moved closer to the site boundary. Ambient radiation monitoring was performed around the country using environmental TLDs. A draft ANSI standard was written, but never issued, on how to deploy environmental TLDs for accident analysis. In 1998, a self assessment conducted by VYNPS radiological control engineers observed that the nearest residence assumption used historically might require modification. VYC-2067 was published in late 1999, taking advantage of the HASL-305 methods to estimate site boundary dose as a function of reactor power. There were several comments raised by the USNRC on VYC-2067 methods, all of which were addressed in the final approved document in January of 2006 (Appendix E was attached to the 1999 draft describing the minor calculation change).

- 2000 to Today
  From activities occurring in the late 1990s brought about by improvements to plant water chemistry, VYNPS undertook several new measurement campaigns to improve understanding of the energy dependent profile of skyshine radiation and the composition of site boundary dose (dose from natural terrestrial gamma radiation, dose from cosmic-induced gamma rays, and gamma-rays produced on-site from nuclear operations). The 2002 *in situ* method, using a high resolution gamma-ray spectrometer, was an extension of the NaI(Tl) measurements that Van

Pelt made in 1974. In conjunction with these site boundary measurements, a new method was developed to monitor in real-time, the site boundary dose from ^{16}N-produced gamma-rays. This new method was termed the Main Steam Line Radiation Monitor (MSLRM) method, which was implemented into the 2002 version of the Off-site Dose Calculation Manual (ODCM). The design concept behind the MSLRM was two-fold: 1) enhance site boundary gamma-ray dose estimates by making and logging results in real time; and 2) measure and report net increase in dose directly, without relying on an independent measurement of natural background levels at the point of interest. Real-time monitoring and implicitly removing the contribution from natural background provided two advantages to the previous methods that relied on radial point measurements of exposure rate, as a function of reactor power. Surveillance TLDs continued to be deployed to validate that gross errors did not occur for the monitoring of ambient gamma radiation. The condition report from 1998 as well as the necessity to improve water chemistry provided the motivation to make additional analyses. The regulatory climate had also been changing over time, introducing various ideas on how to monitor for and demonstrate compliance with allowable public exposure to radiation. New terms included the critical group, critical receptor, nearest resident, maximally exposed individual, and population dose. These terms and ideas were discussed internationally and nationally, but to ORAU's knowledge were never formalized between the state and the utility.

Throughout the entire operating period, environmental thermoluminescent dosimeters have been deployed in and around the unrestricted area. Various types of TLDs have been deployed both by VYNPS and VDH. TLDs served three purposes: 1) provide an accurate dose reconstruction of releases in the event of a design basis accident; 2) identify any long term radiation dose trends in and around the facility; and 3) identify any otherwise undetected release (a fail-safe validation and safety precaution). As is the case with environmental TLDs in general, the TLDs were never calibrated to dose equivalent, which leads to significant misunderstandings when intercomparing measurement results between multiple TLD vendors or other direct measurement methods.

6.1.1 The Early Years (1969-1978)

The primary set of references pertaining to this period of reactor startup, in <u>chronological</u> order is:

McCandless, R. "Preoperational Surveillance Program for VYNPS," **September 26, 1969.**

Van Pelt W.R., "Environmental Radiation Survey of Vermont Yankee Nuclear Power Station," **December 1971.**

Vandenburgh D.E (VYNPS) [Letter to Dr. P. A. Morris (USNRC)], "Modeling of Direct Gamma Dose," **April 28, 1972.**

McCandless, R. [Letter to H. Ashe (VDH)], "State Radiation Standards for Gaseous Radioactive Stack Releases," **November 17, 1972.**

Scott, D.M. (VDH), "Environmental Radiation Surveillance at Vermont Yankee Nuclear Power Station, Special Field Study Direct Gamma Radiation from Vermont Yankee," **September 1974.**

- This report submitted to Director, State of Vermont Agency of Human Services, Dr. J. R. Froines, **October 2, 1974.**
- This report submitted to VYNPS Plant Superintendent, Mr. B. Riley, **October 3, 1974.**

Lewis, H.S. (VYNPS) [Letter to Wallman (VDH)], "Gamma Radiation Shield Wall," **October 15, 1976.**

Brinck W., et al., "Special External Radiation Field Study at the Vermont Yankee Nuclear Power Station," Health Physics Journal, Vol. 32, **April 1977.**

Scott, D.M. (VDH), "The Operation of Vermont Yankee and Public Health and Safety: An Approach to State Evaluation," **July 18, 1977.**

Froines J. (VDH) [Letter to R. Snelling (Governor of VT)], submittal of a report from David Scott on "The Operation of Vermont Yankee and Public Health and Safety: An Approach to State Evaluation," **July 18, 1977.**

Desrosier A.E. (VY), [Letter to Moody] "Fenceline Environmental Radiation Monitor at Vermont Yankee – *with analysis showing that the site boundary radiation monitor was no longer needed in order to show that direct gamma dose requirements were being me*t," **May 15, 1978.**

Concurrent with the local efforts in Vermont, the Atomic Energy Commission's Health and Safety Laboratory in New York[25] had published two important technical documents, which operations staff could (and did) use for further evaluation and understanding of the direct radiation exposure from ^{16}N produced in BWRs:

HASL TM 72-1, "Environmental Gamma Radiation from Nitrogen-16 Decay in the Turbines of a Large Boiling Water Reactor," **February 1972.**

HASL-305, "Determination of ^{16}N Gamma Radiation Fields at BWR Nuclear Power Stations," USAEC, **May 1976.**

On September 26, 1969, Ray McCandless from VDH wrote that the environmental surveillance program should include the use of TLDs for measuring direct gamma dose from immersion in noble gas. [J] As he states at the time "TLDs should be selected on the basis of selectivity and versatility." In 1969, TLDs were a relatively new technology. Most radiation workers at the time were issued film badges. It is interesting to note, in retrospect,

[25] HASL was the pre-eminent national laboratory for developing methods to measure environmental radiation. HASL stands for the Health And Safety Laboratory. It was created in 1947 and since inception, resides in New York City. It was transferred to the US Department of Energy after the AEC, but now is managed by the Department of Homeland Security.

that TLDs offered sensitivity that film badges could not (and would not) match. Mr. McCandless provided the early concepts for the VDH environmental monitoring program. Most importantly, he directed the collection of baseline environmental radiation data to establish the net increase in radiation levels from operation of the reactor.

1971

The Van Pelt report (1971) provided all necessary measurement results for establishing baseline radiation rates for the VYNPS and the surrounding area. The major results determined from the ORAU review are:

1. At about a 360-ft elevation above mean sea level, the cosmic ray background was estimated at 31.8 mR/y [26].

2. Using a 4-in x 4-in NaI(Tl) gamma-ray spectrometer, spectral unfolding was used to estimate the contributions to background from fallout (primarily ^{137}Cs), and terrestrial gamma radiation emitted from naturally occurring ^{40}K and the ^{238}U, and ^{232}Th decay series. With average PIC measurements showing background exposure rates on the order of 80 mR/y and the cosmic component of 31.8 mR/y , the NaI(Tl) measurements estimated the terrestrial contributions sufficiently well to explain the net difference between the PIC measurements and the estimated cosmic component:

^{40}K	15 ± 6 mR/y	(random error reported as 1.96σ, 95% confidence level)
^{238}U	10 ± 4 mR/y	
^{232}Th	20 ± 8 mR/y	

3. Intercomparison studies were performed with two different Reuter Stoke model pressurized ion chambers (PICs). Each chamber was calibrated using different methods and used by different groups, thus providing an excellent set of data for intercomparing the degree of precision between two separate devices. Subtle differences using these PIC detectors were evaluated, including measuring only cosmic radiation (by making the measurements in an aluminum boat on the lake), and by measuring both the terrestrial and cosmic components. Another important result was that two PIC measurement calibration schemes for each of the two detectors resulted in a net 8% bias between instruments. Given that the agreement was within ±10%, the detector intercomparison was exceptional because it captured variability between two groups of physicists, two different detector models[27], and two different calibration schemes.

4. At the time, physicists had not studied in significant detail the relationship between dose equivalent and exposure; Van Pelt's 1971 report, simply stated that 1R = 0.95 rem. Hence, it is reasonable to expect that the writers of the state regulation would have thought it appropriate to convert, for simplicity, using a one to one conversion.

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[26] Lowder et al. "Cosmic Ray Ionization in the Lower Atmosphere," J. Geophys. Res., 71 (19), p. 4661, **1966.**

[27] Two different model PICs were used, but both were similar designs from Reuter Stokes.

The Van Pelt report established baseline exposure values for 12 measurement locations onsite and 9 locations offsite. The locations were identical to the locations where TLDs were to be placed after startup. The figure below shows that the natural background exposure rates are no different on-site from offsite. The estimated mean cosmic-ray component (31.8 mR/y) is shown on the figure. The mean background rate including both on and off site TLDs is 80 ± 16 mR/y, with the error estimate reported as 1.96σ (2-sided 95% confidence limit), where σ is computed as the standard deviation of the 21 measurements.

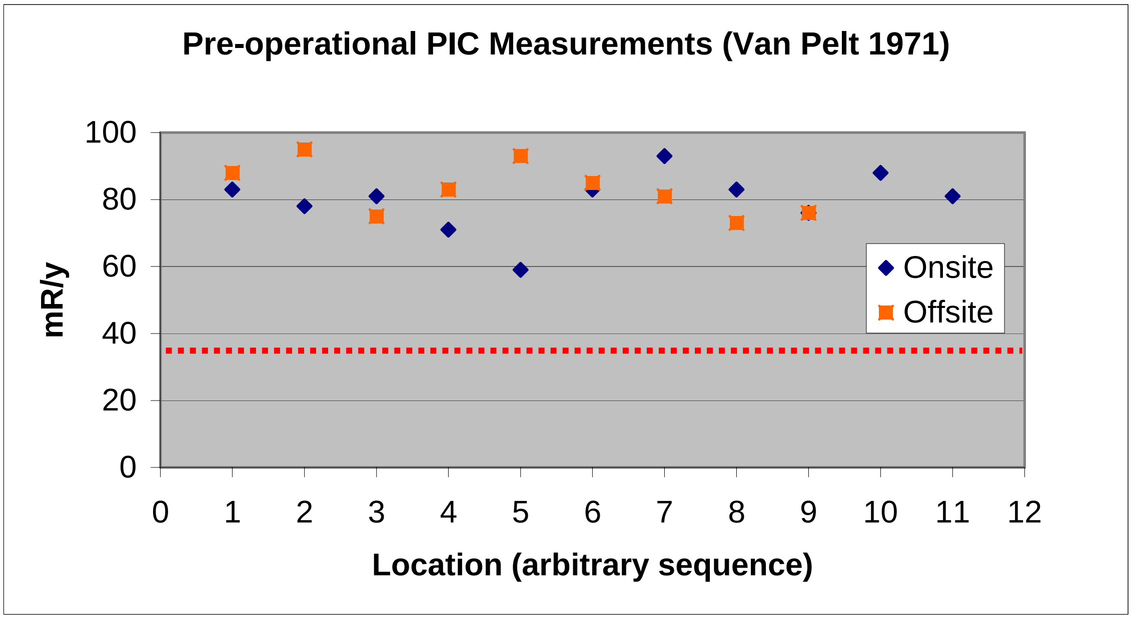

Figure 14. Van Pelt Background Radiation Measurements (12 onsite, 9 offsite, 31.8 mR/y cosmic, 1971)

Regarding the data in the figure above, one of the 12 data points on-site was not reported and the on-site measurement in location number 5 is significantly lower than the other measurements. Location 5 could probably be statistically rejected from the dataset.

The Van Pelt measurements established the baseline background level at 80 mR/y with a standard deviation in the 21 independent measurements of 8 mR/y (10%). Using the calculated variation from the 21 measurements, it can be stated that the true background radiation level is between 64 and 96 mR/y at the 95% confidence level. This estimate for variability is probably smaller than the actual variability because the measurements did not capture all of the diurnal and seasonal variation that is known to occur over longer periods of time.

Van Pelt also used a NaI(Tl) spectrometer to measure the individual contributions from natural background from terrestrial radionuclides. The bar chart below shows the annual exposure contribution from ^{40}K, ^{238}U, and ^{232}Th; an estimate for cosmic contribution (31.8 mR/y); the sum of all estimated components; and an independent measurement total using the PIC. The sum of each estimated component was very close to the integrated PIC measurement (77 mR/y vs. 80 mR/y).

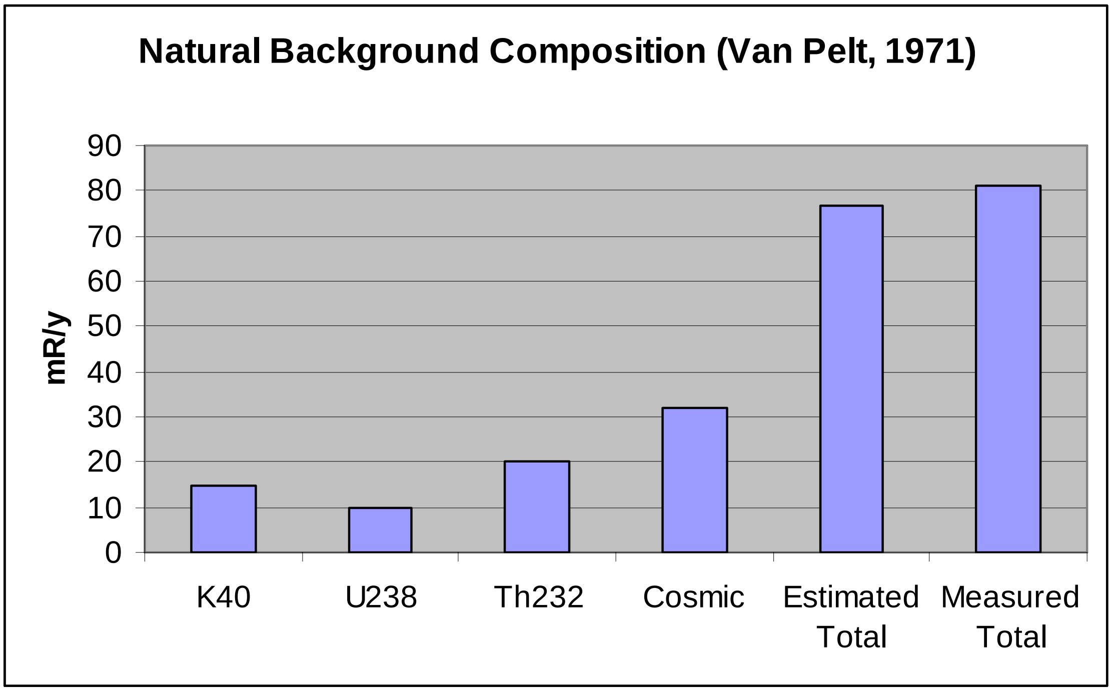

Figure 15. Natural Background Measurements at VYNPS in 1971 (Van Pelt)

It is these set of baseline measurements conducted before reactor startup that leads to the ORAU statement that the annual dose objective of 20 mR is on the order of 25% of natural background. Furthermore the net objective of 20 mR is equal to the variation of independent measurements at the 95% level of confidence (± 16 mrem stated above). Hence, making any measurement to describe the net difference in means (gross from background) at this level of reliability is statistically challenging. The most effective way to reduce variability is to increase n, the number of independent measurements. Statistical issues on this matter are described in Appendix D of this report.

1972

In April of 1972, prior to startup, VYNPS and the USNRC were calculating estimates of the net contribution of direct gamma dose from the reactor/turbine building. In particular, it was of interest to estimate ^{16}N dose to the nearest neighbor and to the nearby school, Vernon Elementary School, located 1650 ft away. Mr. Vandenburgh, from VYNPS wrote Mr. Morris, of the USNRC, April 28, 1972, providing calculational results obtained by benchmarking measurements from a similar BWR, Oyster Creek. Results showed then that the nearest neighbor would receive 4 mrem per year and Vernon Elementary, about 2 mrem per year. Both of these estimates used a combined shielding and occupancy reduction factor of 2. Thus, these calculations were showing an uncorrected rate of 8 and 4 mrem per year, to the nearest neighbor and Vernon Elementary, respectively. Vandenburgh states in his memo that the USNRC analysts were too conservative at 20 mrem per year and an occupancy of 0.2. Though ORAU never saw the USNRC analysis, Vandenburgh is remarking that USNRC calculations indicated the nearest neighbor could receive a dose of 100 mrem per year at 100% occupancy (20 mrem/0.2 occupancy). The Vandenburgh correspondence is important for two reasons: 1) occupancy and shielding factors were presumed; and 2) the USNRC calculations were incorrect, over-predicting the actual ^{16}N

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contribution significantly. Had the State of Vermont read this memo, it may have set a slightly different direct gamma dose standard, given the uncertainty in the net dose contribution estimated by calculation and as benchmarked at Oyster Creek. It also appears in this memo that any issues about the radiation units (mR, mrad, or mrem) were not important enough to warrant discussion. In the final analysis, the calculations of 1971/1972 significantly over-predicted direct gamma dose.

1974

Reactor startup occurred in November of 1972, a few months after VYNPS notified the USNRC of its direct gamma dose projections. Meanwhile, David Scott, a radiological engineer from the State of Vermont, was planning to measure and evaluate the direct dose contribution from operation of the plant. Mr. Scott published his monitoring results in September of 1974 (about two years after startup) and submitted some important memos to the Director of Health, State of Vermont. Scott's memo of October 2, 1974, informed Dr. Froines of the State that the USNRC regulations were limited because there were no provisions explicitly for direct gamma dose, only for allowable effluent concentration. He encouraged Dr. Froines to include a state limit for direct gamma exposure into the regulation for radiological health. He presented the measurement results of his field study to both the State (Dr. Froines) and to VYNPS (Mr. Riley). These field measurements were an excellent summary of measured conditions during the first two years of plant operation; reflecting measurements by the USEPA, VYNPS, and the VDH.

PIC measurements were made by a USEPA team after VYNPS had made measurements on January 11-12, 1973. TLDs were also deployed during this work, not to report absolute results, but to become part of the long term ambient radiation surveillance program. VDH also made PIC measurements.

The measurement locations for Scott's Study are shown in the figure below, all of which are located mostly along Governor Hunt Road, near the school. These locations are about 600-700 ft further from the turbine building than the current measurement locations of interest, DR-52 and DR-53, located near the fenceline of the restricted area. The fenceline measurement locations DR-52 and DR-53 are located on the map for reference, but were not evaluated during this 1974 time period.

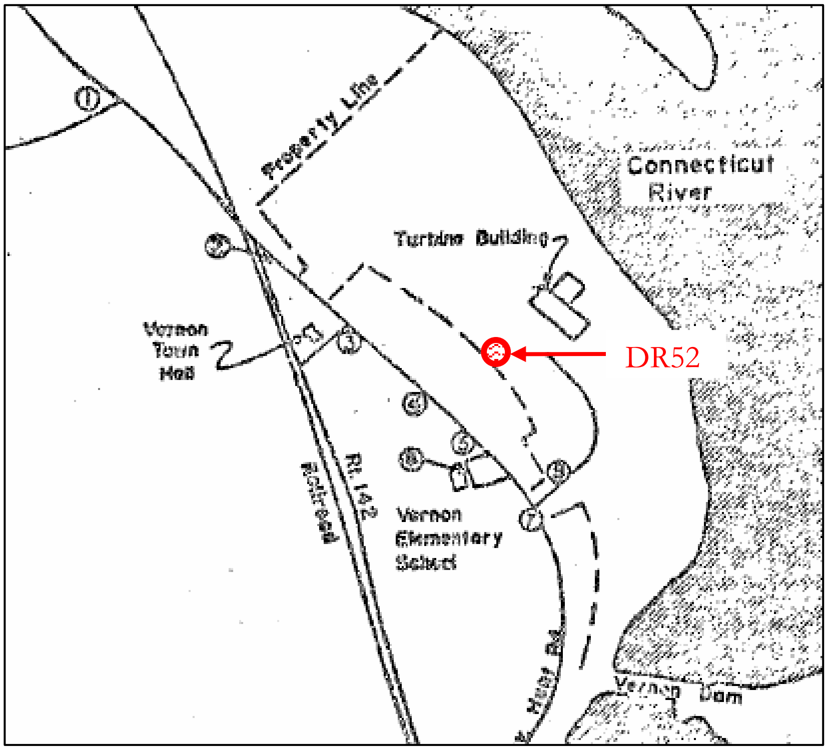

Figure 16. USEPA Measurement Locations (1973, from D. Scott, 1974)

The USEPA results, presented in Table 1 of Scott's report, as a result of the PIC measurements included the following:

1. A background reading of 7.2 μR/h (equivalent to 63 mR/y). This is lower than the 9 μR/h result from Van Pelt but still within the expected distribution of results within the 95% confidence interval about the mean.

2. Of the eight measurement locations, Location 5 (school parking lot) had the highest gross exposure rate of 8.4 μR/h. Scott appeared somewhat concerned that exposure rates above background could be measured as far away as the school. ORAU reminds the readers of this report that this conclusion relies on an accurate assessment of what the background is at each location for each of the measurements. In the case of this study, the background radiation rate assumed was 7.2 μR/h (63 mR/y), based on measurements made during a plant shutdown in April, 1974, though the "table 2 and table 6 background results" from that report shows the average to be about 8 μR/h. What does this all mean? The margin for error in any measurement device to resolve net changes on the order of 1.5 - 3 uR/h is very difficult. Assuming background for a specific location will result in false positives and false negatives when the net rate is such a small fraction of background radiation. The resultant question then is "what was the background radiation rate in the measurement location of interest?"

Scott's report included three major result sections: Appendix 1 (USEPA); Appendix 2 (VYNPS); and Appendix 3 (VDH). The special study that the USEPA conducted was later published in the Health Physics Journal (Brinck, 1977). Specific attention was given to the nearest residence.

There are number of interesting intercomparisons that can be made about the measurements in Scott's report, but the most important graph that can be extracted is the following. The first full-power operation was Jan 11-12, 1973. During this period, all three independent groups were able to either directly measure the gross exposure rates with PICs and/or TLDs; all three recognized that it would be difficult to meet a 20 mR/y objective at locations near the site boundary, along Governor Hunt Road. They also measured Vernon Elementary school, showing that the school exposure was about 5% above background.

Part 1 of the report, which cites the EPA's result, shows net exposure rates in some locations along Governor Hunt road in excess of 20 mR/y, at 85-90% power. The figure below is the VDH and VYNPS data plotted as a function of distance from the Turbine Building. The gross and net labeled data was measured by VYNPS using a PIC. ORAU used a 80 mR/y background to estimate net. The other data plotted is the VDH data, also collected with a PIC. These were the first "full power" measurements. While the VDH data and VYNPS data are different, they both point to site boundary dose in excess of the dose objective. The distance where DR-53 is located (762 ft from the turbine building) is shown on the graph of exposure rate versus distance. The net rate is 80 mR/y less than the gross rate, as used by ORAU to evaluate "net above background."

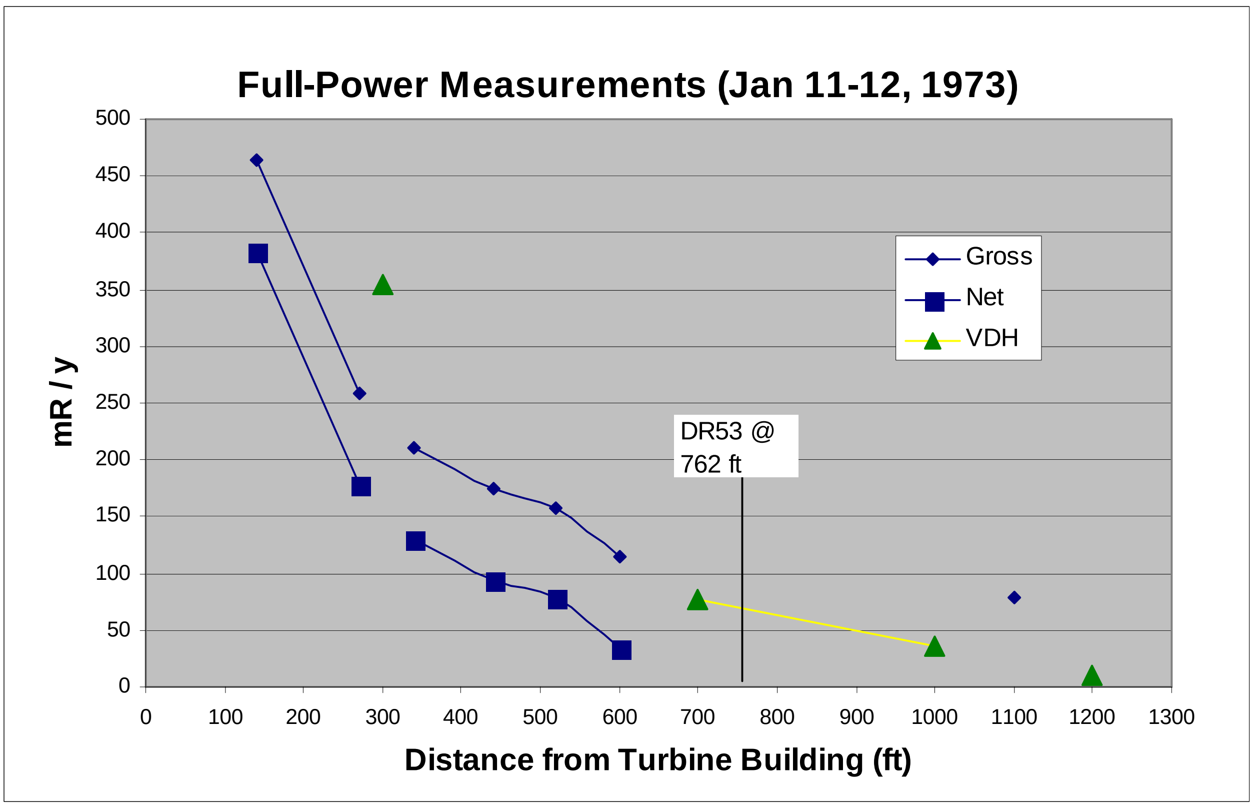

Figure 17. Comparison of VDH and EPA Measurements at Full Power (Jan 11-12, 1973)

The graph above shows several key points: The VDH PIC data was substantially greater than the USEPA data and that the projected annual exposure east of Governor Hunt Road and west of the fenceline could be in the range of 40-60 mR/y net above background, if the reactor were to operate at 100% power and 100% capacity. It appears that all parties agreed that a shield wall should be constructed for ALARA purposes, regardless of what the State dose objective was or would later become.

Some TLD data was presented by the EPA (integrated over an entire quarter). The TLD data did compare favorably with the PIC projections when the error estimates were used to compare the difference in mean values. What this says is: the measurement to measurement variability at such low rates was sufficiently large so as not to conclude that the mean values were different.

In January of 1974, the USEPA sent VDH a memo with instructions and a reminder to make sure that TLD measurements were accounting for self-dosing, to develop calibration factors, to account for transport dosing, and to determine how many TLDs to co-locate in any given measurement location—<u>USEPA recommended three collocated TLDs</u>.

Attention was placed on the type of shield wall that should be built around (not over) the turbine. ORAU understands that a turbine wall was fabricated in mid-1976. In October of 1976, Lewis (VYNPS) wrote a letter to Wallman (VDH) to mention the successful installation of a shield wall and the result of measurements that show **"**<u>at 80% annual operation, the maximum exposure increment would be 20 ± 5 mR/year</u>.**"** This stated result was based on short-duration full-power tests showing <u>24 mR/y</u>. The state's reply to this Lewis to Wallman memo came in the form of another D. Scott report, July 1977: "The annual dose objective to an individual in the unrestricted area is 5 mrem. Measurements made [as communicated in the October 1976 memo to Wallman] indicate that the site boundary dose is 20 ± 5 mrem. This satisfies the objective of 5 mrem to the nearest residence."

> **Unlike the extensive set of measurements made by Van Pelt (1971) and Scott (1974), the memos and reports from 1976-1978 after the shield wall was installed, <u>did not</u> transmit data—only results. VYNPS (Lewis to Wallman, 1976) stated that the shield wall would enable VYNPS to meet the annual dose objective of 20 mrem. The State never replied to VYNPS that the measurements were acceptable. But, in reading ahead to Scott's 1977 report, he was tracking the progress on the "after shield" measurements, particularly during a 1977 refueling outage. Scott published a report in July of 1977, stating on page 31 that the dose objective at the site boundary met the 5 mrem per year objective to the nearest residence. ORAU could not locate the data to support these results; it was not until 1978 when VYNPS' Desrosier wrote Moody, that a log-linear graph was displayed, showing the 1973 and 1977 results of exposure rate versus distance from the turbine building; ORAU never located this supporting data, upon which decisions were made regarding the acceptance of the shield wall as having satisfied the dose objective. More information about this data, as reported by Desrosier in 1978 is provided chronologically below.**

1977

In 1977, the USEPA measurements from 1973 were published in the *Health Physics Journal* (Brinck et al.). Up until this 1977 publication, the only place the 1973 measurement results were presented was in Scott's 1974 paper (Part 1). It was also during this period of time the shield wall was installed. So, it's important in the overall timeline not to confuse measurement results pre-operation, startup and operation (no shield), versus operation (with shield).

The 1977 *Health Physics Journal* publication provided significantly more detail of the 1973 pre-shield measurement results provided in Scott's 1974 report, including:

- The EG&G TLD (TL-15), CaF_2:Mn,[28] recognizing that since TLDs are calibrated in arbitrary units, calibration factors were developed to convert to exposure (mR). At that time in history, ORAU is fairly certain that adequate corrections for energy dependent response were incomplete (this type of work is numerically challenging). TLDs were deployed in the field for only 2-4 weeks (not a very long integration time to obtain "good statistics"). Three or four dosimeters were placed at each measurement location. The Brinck paper addressed how problems with self-dosing (background in the dosimeter materials), and fading affected results.
- TLD measurements for background varied between 5.3 and 9.9 μR/h.
- The PIC measurements showed background of 8 μR/h, within statistical agreement from prior results.
- The PIC measurements showed elevated readings on January 12-13, 1973 from passage of a noble gas cloud. A maximum instantaneous reading of 34 μR/h was observed. <u>These rapid changes in exposure rate reported do require further understanding, and to what extent noble gas releases are controlled/minimized.</u> The USNRC has looked at the monitoring of the noble gas release, and has stated in the EPU SER, that the noble gas submersion dose is about 1 mrem per year to the public. What was reported in 1973-75 as an issue has since been mitigated.
- No major effort had been undertaken to capture the variability in natural background brought about by snow cover, rain, differential pressure changes, and terrain/topology/geology of the local measurement area. Brinck mentioned that accounting for this seasonal variation would be important if station operation net doses of 5 mR/y needed to be detected.

The snow pack effect is illustrated in Brinck's measurement results below. Snow cover in 1973 increased the exposure rate. As will be reported in another chapter, background changes do not always follow the same trend. For example, an analysis of TLD data performed by ORAU shows that in most cases, background during the winter months is less (not more). One must take great care to understand the nature of background variability and general trends in order to make any assessment about net dose above natural background. The case is illustrated here where 1973 data showed snow cover increasing background, while later data showed the opposite effect, particularly when the effect can lead to effective 1 μR per changes (given a dose objective of 2.3 μR/h).

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[28] These dosimeters were referred to as "bulb" dosimeters because they looked like a small flashlight light bulb.

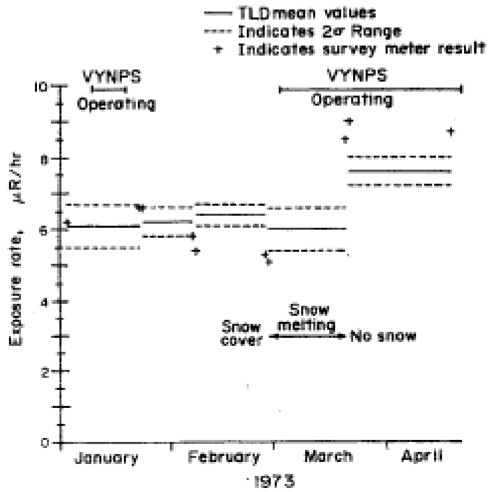

Figure 18. TLD Measurements During Snow Cover (Brinck et al, 1973)

After the installation of the shield in 1976, D. Scott published his 1977 report which was then sent from Dr. Froines to Governor Snelling. As stated above, Scott accepted the 1976 VYNPS results that the dose at the site boundary was less than 20 ± 5 mrem per year (page 31 of his report), which references the letter to Wallman in 1976.

1978

The first 6 years of operational evaluations ended in 1978 with an internal VYNPS report from Desrosier to Moody, showing PIC measurement results made in 1973 (without a shield wall) and in 1976-78, after construction of a shield wall around the turbine. The primary purpose of Desrosier's memo was to inform Moody that the PIC monitor placed near the fenceline was no longer necessary to demonstrate compliance. He further noted that a formula had been developed to calculate site boundary exposure as a function of electrical power (but this formula was not provided in the memo). It appears that the Desrosier memo was specifically addressing potential exposures in the Vernon Elementary School and generally showing a new semi-log fit of exposure rate as a function of distance from the turbine building.

The difficulty with reviewing the Desrosier memo is that it does not provide any raw measurement data in tabular form, but rather the results of a semi-log data fit (origin of measurement data unknown) and a statement that "the continuously recording environmental radiation monitor is no longer needed." ORAU has reconstructed the data fits and provides the following results from these PIC measurements. It is this Desrosier memo (and data) that essentially provided proof that operation of the plant would meet the annual dose objective, even at 100% power. There is no evidence that VDH reviewed and approved this analysis for compliance. This notwithstanding, the figure below shows several important, consolidating points:

1. All measurements from 1973 and from 1976-1978 were made with a PIC.

2. The measurement points extend from the turbine building in a westerly direction toward Vernon Elementary School. TLD location DR-53 is on this line of sight, and the 1973 measurement locations 4 and 5 are near this line of sight, on Governor Hunt Road.

3. The 1976-1978 PIC results are about 70-80% less than the 1973 results because a shield wall had been constructed in 1976.

4. The January 11-12, 1973 measurements using a PIC detector were published in Scott's 1974 report as well as the USEPA report by Brinck in the Health Physics Journal (1977). The 1973 measurements were conducted at 100% power, and according to table 1 of Scott's report a constant background radiation rate of 7.2 μR/h (or 63 mR/y) was subtracted from the gross measurements to yield a curve fit for net exposure rate above background. ORAU estimates that a fit to the 1973 data is a log-base 10 function of the form:

$$\boldsymbol{X\ (net) = 263 * 10^{\left(-d/564\right)} \qquad \left(\mu R\ h^{-1}\right) \qquad (circa\ 1973)}$$

where d is the distance from the turbine building in <u>feet</u>, and X is the exposure rate in μR per hour. This equation can be solved for d, the distance at which the net exposure rate (at 100% power and 100% capacity) would exceed 2.3 μR/h, the equivalent annual objective of 20 mR:

$$\boldsymbol{d = \log\left(2.3/263\right) * (-564) \quad = \quad 1160\ ft \qquad (circa\ 1973)}$$

5. After the shield wall was built, measurements taken in the same locations as in 1973 show a fit that reduces the intercept and increases the slope of the semi-log function to:

$$\boldsymbol{X\,(net) = 100 * 10^{\left(-d/492\right)} \qquad \left(\mu R\ h^{-1}\right) \qquad (circa\ 1976)}$$

The 1978 Desrosier memo and report states that during a refueling outage, background measurements were collected at the same locations as the gross measurements. However, the background results are not provided.

Solving for d (in feet), the result is that the fenceline is the boundary where the annual objective is met.

$$\boldsymbol{d = \log\left(2.3/100\right) * (-492) \quad = \quad 806\ ft \qquad (circa\ 1976)}$$

6. The equations developed by ORAU in parts 4 and 5 above show that the shield installation successfully reduced "projected" annual exposure rates to less than the annual objective of 20 mR. ORAU believes that the data collected in 1976 was the basis for the utility to justify removing the PIC monitor from the site boundary.

7. All data, up to that time were the results of attempts to measure the net increase in exposure rate at the school (1650 ft), but the sensitivity of the PIC is such that beyond the fenceline, the net increase must be extrapolated.

8. The locations DR-53 (763 ft), DR-52 (800 ft), Governor Hunt Rd (1200 ft), and the school (1650 ft) are superimposed in Figure 9 along with the exposure rate of 2.3 μR/h, which corresponds to the annual rate of 20 mR.

9. The most important conclusion to reach is that the shield wall significantly reduced the exposure rate and that the limit of 2.3 μR/h was not exceeded beyond location DR-52/DR-53. It is assumed at this point, in 1978, that the USNRC and the state viewed these analyses as having satisfied the ALARA and annual dose objective requirements.

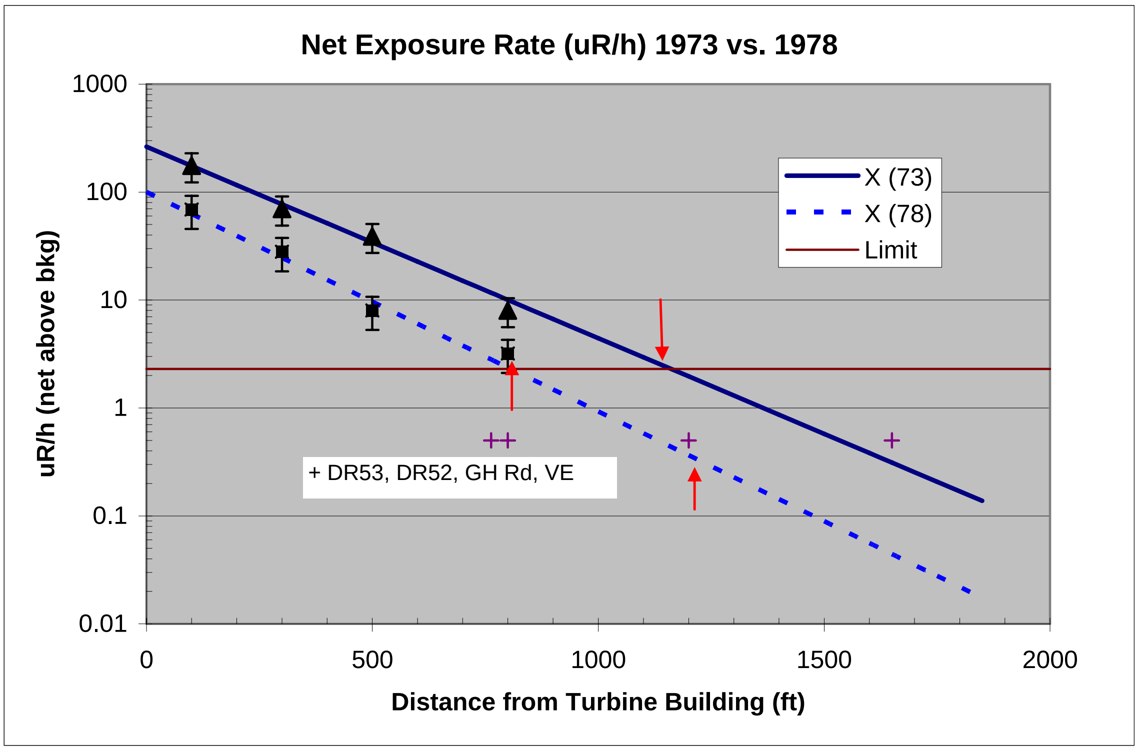

Figure 19. Net Exposure Rate as Function of Distance from TB (1973 vs. 1978)

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Summary of Measurements and Results (1969-1978)

Category	Conclusions
1. General	1.1 The State, Atomic Energy Commission (AEC), and VYNPS worked closely together conducting radiation measurements and intercomparing results. The state was very active to make sure that the regulations were written properly, and particularly, to ensure that requirements not expressly covered by the AEC were addressed by the state. For example, this included direct gamma dose at the site boundary 1.2 The results of preliminary calculations performed by the AEC for the dose contribution from ^{16}N were significantly greater than later measured, i.e., high by a factor of four. This overestimate underscores the difficulty in calculating such quantities. Even today, these calculations, either using a monte carlo code or discrete ordinates computer code, are difficult. 1.3 The newly-formed USEPA worked closely with the utility and state to measure ambient radiation levels. The USEPA had direct access to the physicists and equipment used by the national environmental measurements laboratory in New York. The studies that were performed at VYNPS were instrumental in later publishing recommended monitoring methods (the HASL reports). 1.4 There was considerable discussion on the use of occupancy factors and shielding factors applied to the calculation of direct dose equivalent to members of the public, including the nearest neighbor and to Vernon Elementary School. 1.5 The accepted practice in health physics at that time was to convert from roentgen to rem using the multiple, 0.95. As a result, most practitioners simplified this to a 1:1 conversion. It was not until 1976 (Sanna) and 1985 (Kocher) that a better understanding between R and rem was introduced and understood. 1.6 In 1974, D. Scott from the State published a summary document with three appendices of measurement results: USEPA, VYNPS, and VDH. The first appendix by the USEPA was later published in the Health Physics Journal by Brinck (1977). Pre-operational tests and measurements conducted after startup showed that a side shield would need to be installed to reduce direct gamma dose. A shield was installed in 1976, resulting in a 70%-80% reduction in total gamma dose.

Category	Conclusions
2. Instruments	2.1 With such an interest in measuring ambient radiation, the best available equipment was used: Pressurized Ion Chambers for background evaluation (Van Pelt), for studies at full power (Jan 11-12, 1973), and several follow-up measurements. It appears that reliable Keithly pico-ameters were used to measure the small current output and thus, all PIC measurements are believed to be quite accurate (except for the measurements close to the turbine building, where the energy dependent flux density is more heavily influenced by a higher-energy component). A Shonka, muscle equivalent ion chamber was used by Brinck (USEPA), but it is unclear how well these measurements compared with the Reuter Stokes ion chamber used by all other measurement physicists. The only measurement intercomparison was shown in Table 5 of Scott's 1974 report. At the school, the following net results were reported: TLD (8±6), PIC1 (10±3), Shonka (6±3), and PIC2 (3-4) mR per year. These results demonstrate not only that the measurement methods are not that precise at such low net exposure rates but secondly, that the background variation has not been treated the same way in all cases. TLDs (CaF2:Mn) were used only by the USEPA laboratory. Exposure durations were short. The intercomparison of results between a PIC and TLD by the USEPA (Table 4 of Scott, 1977), compare at the 95% confidence interval, using a t-value for n=4. The difference in means was not significant because the error term for each of the two measurements was 25%-50% of the mean. NaI gamma-ray spectrometers were used by Van Pelt to measure the various contributions to gamma exposure from terrestrial radionuclides. Van Pelt estimated the cosmic-ray background at 31.8 mR/y. Generally, it was well known that additional care would need to be taken with TLD measurements, if there was to be any hope of making an ambient net measurement of 20 mR per year. Calibration of TLDs was also a problem. The PIC measurement was the most accurate instrument with reported agreement to within 8%, even with different models, users, and calibration schemes (^{137}Cs or ^{226}Ra)
3. Methods	3.1 Flux to dose conversion factors at the time were assumed to be 0.95. 3.2 Before operation of the reactor, Van Pelt made 12 measurements onsite and 9 offsite to characterize the natural

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Category	Conclusions
	background distribution. 3.3 After startup, several sets of measurements were made at Vernon Elementary and along Governor Hunt Road. 3.4 Log-linear least squared fits were used to estimate exposure rate as a function of distance from the turbine building. In 1978, a plot of data was presented as evidence that the real-time monitoring PIC could be taken out of service. 3.5 Discussions took place on the assumptions regarding the application of occupancy factors and local shielding. 3.6 Shielding installed in 1976 reduced exposure rates by as much as 80%.
4. Natural Background	4.1 With much attention placed on the dose to school children, Vandenburgh writes to the AEC citing background measurements conducted in and around the Vernon Elementary School: 80 mrem/y outdoors in the schoolyard and 105 mrem/yr inside the school. This example of indoor/outdoor background variability underscores the complications of basing regulations and regulatory guidance on exposure and dose equivalent above natural background. Natural background dose is dependent on whether the critical person is indoors our out of doors, what the building materials are, and so on. 4.2 Natural background radiation was measured by a number of different groups with various instruments. The mean exposure rate measured is probably a "reliable" value, but the estimate of variability is likely too low, because the measurements were not collected to estimate diurnal or seasonal effects, nor the variation from local topology/geology. It appears reasonable to use the following values for describing background radiation in and around VYNPS: 80 ± 8 mR/y (at one standard deviation) 9 ± 0.9 μR/hr (at one standard deviation) Measurements have been reported with average values as small as 7.2 μR/hr and as large as 10 μR/hr.
5. Pre-Shield Results (1973)	5.1 <u>Calculational</u> models for predicting dose rate as a function of distance from the turbine building were biased high by about a factor of four. Once actual measurements could be made at full power, results were presented to the AEC stating that the calculational predictions were high.

Category	Conclusions
	5.2 A number of measurements were collected at full power along the residences of Governor Hunt Road, showing that the exposure rate above background was on the order of 20 ± 3 mR/y. However, some estimates were showing at various power levels/capacity factors that the exposure rates to the Augustinowitz residence could be 23 ± 3 mR/y. The Augustinowitz residence was located east of Governor Hunt Road, slightly south of a line drawn between the turbine building and the Vernon elementary school. 5.3 A plot of the measurement data in 1973 (which was later published by Brinck) shows the area east of Governor Hunt Road as generally exceeding 20 mR/y, but no greater than 25-30. Whether it was this area, or an interest in reducing exposure rates at the school, VYNPS installed a shield in 1976. 5.4 As D. Scott (1974) noted, the VDH measurements reported in Table 8 were much greater than the results of VYNPS or USEPA. No explanation was provided. In summary, Scott concludes essentially that "the data shows that the turbine dose can be distinguished from background up to 1700 feet" away. The difference in the measurement results can be seen in the graph presented above, Figure 17, "Comparison of VDH and EPA Measurements at Full Power (Jan 11-12, 1973)". When ORAU reviewed the data, admittedly, the three sets of measurement were difficult to interpret because the location designations were different and the PIC devices themselves were different.
6. Post-Shield Results (1976-1978)	6.1 Elevation (relative to the turbine building) at measurement locations is important. Skyshine contribution varies with elevation relative to the turbine building. 6.2 The shield wall reduced exposure rates by about 70-80%. 6.3 By an ORAU analysis, the exposure rate at DR-53 was estimated with a semi-log fit of the form, with d in feet: $X = 100 * 10^{\left(-d/150\right)} \quad \left(\mu R\ h^{-1}\right)$ 6.4 VYNPS showed by a figure in the Desrosier memo (1978) that the exposure rate per year would be less than 20 mR/y at the location of DR-53, though the measurement data are not referenced. 6.5. Based on an analysis in 1978, the PIC monitor was removed from service. 6.6 Reference is made to a calculation that relates exposure rate as a function of power level, but it was never provided to ORAU. Only

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Category	Conclusions
	the graph of exposure rate with distance was provided in the Desrosier memo.
7. State Policy	7.1 D. Scott was instrumental at providing measurement data and results to his supervisor, Dr. Froines. 7.2 D. Scott, in 1977, wrote another memo to Froines which was forward to Governor Snelling, asking essentially how the state should regulate the utility separately from the USNRC. 7.3 Exposure rate to dose equivalent was thought to be a 0.95 conversion, so the assumption of 1:1 was made and adopted in the state regulation. 7.4 There is no evidence that the state received or replied to Desrosier's memo of 1978 that the PIC was to be removed, and that by installing the shield wall, direct gamma dose objectives could be met.

A current VYNPS dosimetry map (by SVE associates) has been edited by ORAU to show the approximate locations of the early measurements (1973-1978).

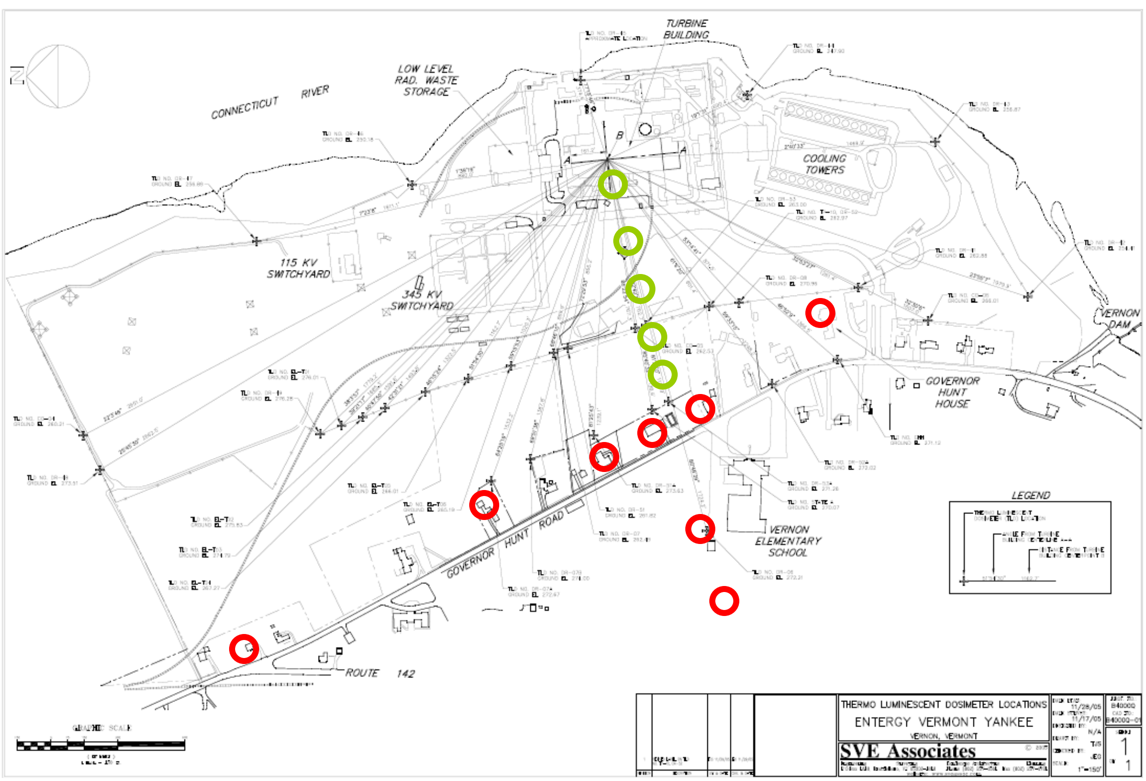

Figure 20. Measurement Locations of Direct Gamma Exposure (1973-1978)

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6.1.2 The 1990s

Once the PIC measurements of 1978 were conducted as a function of distance from the turbine building and Desrosier communicated that the shield wall successfully reduced gamma exposure to 20 mR per year at 800 feet from the turbine building, no documented communication between VYNPS, the USNRC, or the state on this particular matter was generated for nearly 20 years. The conclusion from this "reporting/documentation" gap is that all parties were satisfied with the methodology and the results therein. As part of the environmental surveillance program, VYNPS continued to deploy TLDs to trend environmental radiation data and to perform accident reconstruction, in the unlikely event of a significant release. There appears to be very little TLD data exchanged between the state and the utility, and as a result, there were no conclusive validation measurements to review. The TLD data that were briefly reviewed (from VYNPS and VDH) showed "environmental rates at natural background levels" or "less than values," both of which provide no defensible means for knowing whether the exposure rates were within regulatory tolerance. What can be stated is as follows: if the measurements of 1978 were accurate and the plant did not undergo any significant changes in ^{16}N production, the exposure rate at the fenceline was likely no greater than 20 mR per year. There appears to be no way of independently demonstrating compliance. One of the "lessons learned" for this project is that going forward the state should be involved with the measurements that are conducted, review data as it is generated, and have the opportunity to verify calibration schemes and results.

On the international and national scene, a number of developments did occur over this period that eventually would lead to changes regarding how direct gamma radiation measurements should be performed, the requirements for making these measurements, and methods for reducing data and presenting it in units of dose equivalent:

- Many technical papers were published on ^{16}N production in BWRs and the interaction physics of skyshine radiation, including:

  HASL TM 72-1, "Environmental Gamma Radiation from Nitrogen-16 Decay in the Turbines of a Large Boiling Water Reactor," **February 1972.**

  HASL-305, "Determination of ^{16}N Gamma Radiation Fields at BWR Nuclear Power Stations," USAEC, **May 1976.**

  De Sousa C., Eichholz GG, "Dosimetry of Gamma Rays from Nitrogen-16 Decay," Int J Appl Radiat Isot.; 36(12):981-4. **December 1985.**

- A better understanding of gamma-ray interaction in human tissue was underway, including the development of new physics models to describe the science of dosimetry. Several publications that became useful for describing how exposure should be related to dose equivalent include:

ICRU 47: Measurement of Dose Equivalents from External Photon and Electron Radiations, **1992**

O'Brien K., Sanna R., "*The Distribution of Absorbed Dose-rates in Humans from Exposure to Environmental Gamma Rays*," Health Physics Journal, Vol. 30, pp. 71-78, **(January 1976)**.

Kocher, D. C.; Sjoreen, A. L, "*Dose-rate Conversion Factors for External Exposure to Photon Emitters in Soil*," Health Physics Journal, Vol. 48, **(February 1995)**.

- For personnel monitoring, TLDs began to replace film badges. In conjunction with personnel dosimetry, TLDs were studied extensively for application to environmental monitoring. International environmental dosimeter intercomparison programs began in 1974 and continued into the 1990s.[29]

- Pressurized Ion Chamber performance was improved, including the development of the High Pressure Ion Chamber (HPIC) and the mathematical means to measure tissue equivalency.

- Portable high resolution gamma-ray spectrometers (HPGe detectors) and associated electronics (multi-channel analyzers) were developing at a rapid rate, enabling radiological engineers to make precise measurements of energy dependent gamma-ray fluence.

Meanwhile, at the Vermont Yankee site, a few notable accomplishments were made in the late 1990s, with respect to the direct gamma-dose issue, summarized in the subsections below.

1. M.S. Strum's report on September 4, 1998, describing a short duration TLD exposure at DR-52. The result of this test to measure the net change at full power versus zero power, was used to make a calculation that the annual dose at DR-52 would be 11.9 mrem (100% power and capacity). In addition, the report referenced a 1997 estimate (VYC-1817) that accounted for all other gamma radiation source terms. The result for all other sources accounted for a net dose of 1 mrem; thus the total dose at DR-52 was estimated at 13 mrem.

2. A December 21, 1998, "Condition Report" was issued by VYNPS. Radiological engineers observed that the nearest resident had moved closer to the west side boundary. This observation triggered a discussion about the state assumptions used for estimating gamma dose to the public versus gamma dose at the site boundary.

3. On December 21, 1999, a technical publication was released by VYNPS to describe recent measurement results on the west site boundary. This report is entitled VYC-2067 "Evaluation of N-16 Dose Contribution at the West Site Boundary."

[29] Klemic, G. et al., "Results of the 10th International Intercomparison of Environmental Dosemeters," Radiation Protection Dosimetry, Vol. 58, No. 2, **(1995)**.

6.1.2.1 Strum to Voland (September 4, 1998)

An opportunity to evaluate, by TLD, the change in exposure between full power and zero power presented itself in the second quarter of 1998. During the first two months of the second quarter, a refueling outage was underway. Strum had the idea to co-locate a second TLD at DR-52 and measure the difference in the first TLD collecting data during the entire quarter and the second TLD collecting data during the last 30 days of the second quarter. An estimate of the full power TLD response was then given by the following relationship, when corrected for exposure duration and separating zero power contribution from full power contribution:

> Net TLD Response per 30 days ≈ TLD2 (30 days at full power) – TLD1 (90 days; 60 days zero power, 30 days full power)

Though Strum conducted a statistical test to compare the difference in means, concluding that there was a statistically significant difference between the full power and zero power measurement, ORAU finds that the approximations could lead to the incorrect result of a net ^{16}N contribution at DR-52 of 11.9 mrem per year. A few ORAU remarks about this analysis:

1. The experiment protocol could have been improved at the beginning of the refueling outage so that more than one TLD could be placed at DR-52 for 3 months, and more than one TLD could be collocated for the last month of the quarter, and at full power. The power of the test would have been much greater had at least 7-8 TLDs been collocated for each time period of interest.

2. It is not very clear how the TLDs were calibrated. Therefore it is not known precisely how the TLD vendor (Framatome ANP) reported the results in μR/h as well as the corresponding ±1 standard deviation (S.D.) values. ORAU did confirm the t-test results using the data provided (page 2 D. Voland, REG 98-132)

3. Random and systematic error introduced in the different TLD integration times was not addressed, but is very important for the analysis. Mention of the normalization from a 100% capacity factor to a 70.9% capacity factor is provided, but errors in this normalization were not propagated into the final assessment.

4. The conversion factor for mrem/mR was greater than 1:1.06.

5. Strum introduces some approximations for calculating the non-^{16}N produced source term (from the other fixed sources) and the background. For the data that was collected, the assumptions were essentially the only option for calculating three contributions to gamma-dose: natural background, ^{16}N produced, and other sources (low-level waste storage and the North Warehouse). However, when making these assumptions at such low levels of exposure above natural background, TLDs are simply not the first instrument choice for making this determination.

6. This having been said, Strum did report the following estimates of the three components at DR-52:

a. 7.22 μR/h (natural background + fixed sources)
b. In order to separate out the fixed source contribution, Strum assumes that reported monitoring results in a prior report, VYC-1817, are valid, but 20% further away from DR-52 than the site boundary. He concludes that the fixed source term is 0.12 μR/h, so that natural background is 7.1 μR/h.
c. 1.28 μR/h (full power ^{16}N at DR-52)
d. The 1.28 μR/h value is multiplied by hours in a year and 1.06 mrem/mR to yield the dose equivalent of **11.9 mrem/y** from ^{16}N at DR-52, 100% power, 100% capacity).
e. Strum concludes that the total dose equivalent at DR-52 is the sum of the ^{16}N component and the other fixed sources: 11.9 + 1.54(0.69) = **13 mrem per year**. The 1.54 mrem*0.69 factor is a correction for the distance between the other sources at DR-52, using an inverse square relationship.

ORAU examined these calculations and found no incorrect calculations. However, there were simply too many assumptions and approximations used for analyzing one set of collocated TLDs, to make any statistically significant definitive statement about the net response from ^{16}N and from other fixed sources. The TLD test was inconclusively negative, showing that DR-52 was nearly half of the state dose objective. While ORAU noted in §6.1 above, "The early years", the accurate PIC measurements from 1978 predicted at the site boundary of DR52 a site boundary dose just at the annual dose objective of **20 mrem/y (from ^{16}N alone).** ORAU believes it is unlikely that the TLD measurements of 1998 were that much better than the PIC measurements of 1978. The actual exposure rate (not dose equivalent rate) was likely less than 20 mR per year (100% power, 100% capacity)**,** but the TLD measurements were not the best measurements for making this determination.

What was learned by Strum's report is that in 1997 an analysis had been made (VYC-1817) to estimate the contribution of "other fixed sources" including the low level waste storage area and the north warehouse. Furthermore, a new estimate of 13 mR per year had been derived which was lower than the 1998 PIC measurement of roughly 20 mR per year. Site radiological engineers were definitely interested in ensuring the limit was not exceeded, and in so doing, decided to conduct a self assessment in late 1998, which lead to the issuance of a condition report.

6.1.2.2 The Condition Report of 1998 (CR-VTY-1998-02205)

In December of 1998, VYNPS radiological engineer David P. Tkatch conducted a self assessment of the radiological conditions along the site boundary. A condition report was written: Condition Report CR-VTY-1998-02205 Nearest Site Resident Exceeding Dose Assumptions," (50 pages), **Dec 21, 1998**. This condition report had relatively little to do with the measurements and analyses that had been conducted (the emphasis of this chapter); however, it did lead to significant discussion between VYNPS and VDH on the assumptions used to calculate dose to the public. ORAU received copies of the documents that were generated from this condition report, an extensive list by any measure:

1. Vermont Yankee Event Report 98-2205: Nearest Site Resident Exceeding Dose Assumptions, (6 pages), **Dec 21, 1998.**

2. VDH letter to VYNPS (Ray McCandless to Deborah Voland), "A measured or calculated exposure at the site boundary fence of the location in question of 20 milliRems equivalent or less will satisfy the regulatory requirement," **Feb. 10, 1999.**

3. Memorandum from M.S. Strum to D. Voland, "N-16 Dose Estimate at TLD DR-52 (West Site Boundary Line)," REG 98-132, (12 pages), **Sept. 4, 1998.**

4. Memorandum from Jo-Ann Pelczar to D. Voland, "Proposed ODCM Revision 23 – Rev. 1," REG 99-048, (3 pages), **April 26, 1999**.

5. "An Excerpt from the Vermont Yankee Nuclear Power Station Off-Site Dose Calculation Manual Revision 23," Section 3.11, Method to Calculate Direct Dose From Plant Operation, (2 pages), **May 12, 1999.**

6. Memorandum from Jo-Ann Pelczar to D. Tkatch, "Evaluation of N-16 Dose Contribution at West Site-Boundary," REG 99-120, (6 pages), **December 30, 1999**.

7. "Corrective Action CR-VTY-1998-02205, Address Nearest Residence in ODCM," **July 1, 1999.**

The relevant measurement and detection conclusions cited in this series of documents were found to be the following. Regulatory implications are provided in Chapter 4 of this ORAU report separately. Major conclusions:

1. In May of 1999, after the 1998 condition report, Revision 23 to the ODCM was made, revising the calculation to the nearest resident, though ORAU is not aware of what the distances were changed from or to. Equation 3-27 in the ODCM, which describes the maximum dose over some time interval, Δt, from ^{16}N decay in the steam of the turbine is related to gross electric power over that same time interval.

   For the "maximum west site boundary" the equation was:

   $$H(\Delta t) = 3.17E-06 * Gross\ Electric\ Power(MW_e - h) \qquad (mrem)$$

   For the maximum residence (SW site boundary with respect to turbine)

   $$H(\Delta t) = 2.63E-06 * Gross\ Electric\ Power(MW_e - h) \qquad (mrem)$$

   In Revision 23 of the ODCM, ORAU is not aware of the origin of the data for this mathematical relationship (though believes it is from the data that eventually was reported in VYC-2067). However, if one takes the data in the graph from Desrosiers' memo in 1978 (assuming it was produced at full power), and estimates the corresponding distances, the coefficients 3.17E-06 and 2.63E-06 look reasonable as long as the roentgen is not converted to rem at a multiple other than one.

In summary form, some example data is provided in the table below.

Table 4. ODCM Power to Site Boundary Conversions (v23, 5/12/1999)

Description	Power Level (MWe)	Number of Hours	Maximum West Site Boundary Dose (mrem)	Maximum Residence Dose (mrem)
100% power, 100% capacity, 1 year	535	8766	14.9	12.3
90% power, 100% capacity, 1 year	482	8766	13.4	11.1
80% power, 100% capacity, 1 year	428	8766	11.9	9.9
100% power, 90% capacity, 1 year	535	7889	13.4	11.1

2. From the reports provided, ORAU concludes that the 1978 tests showing 13 mrem per year in the vicinity of DR-52 (with a one to one conversion) is in agreement with the results in the table above. However, it is not clear from the data how well this quantity was known (precision/accuracy) and whether the 1998 estimate had been adjusted by an exposure to dose equivalent conversion factor of less than 1 (for example, 0.87, 0.71, 0.60).

3. On December 30, 1999 (REG 99-120), a memo from Pelczar to Tkatch at VYNPS suggested that the coefficients listed in the ODCM Revision 23 (May 1999) equation were low. Her recommendation, based on a calculated fit to the data, was to adjust the DR-52 (Maximum West Site Boundary) by 7% (from 3.17E-06 to 3.39E-06). However, a concomitant increase in the maximum residence was not suggested. The new coefficient was placed into Revision 26 of the ODCM.

   The increase of 7% in the coefficient increased the bounding case estimate (100% power, 100% capacity, 1 year) from **14.9 mrem to 15.9 mrem at DR-52**.

4. In July of 2000, a VYNPS "commitment review" (index number ER-982205_03) presented the results of other TLD data. Four TLDs were collocated with DR-52 (TP-110, TP-11, TP-31, and TP-32). Because of the number of TLDs deployed, an estimate for precision could be provided. However, because the irradiation periods were different, once again it is difficult to say with any defined precision that this estimate was any better than the one before. Nevertheless, the 1998 calculation for dose to nearest residence (presented above):

$$11.9 + 1.54(0.69) = \textbf{13 mrem per year} \text{ (nearest site residence)}$$

was modified to an estimate of :

$$15.46 + 1.54(0.69) = \textbf{16.52 mrem per year} \text{ (nearest site residence)}$$

5. In the same memo from Pelczar (REG 99-120), the PIC measurements were described during the outage and power-up of 1999. The PIC measurements identified the location of the greatest exposure (at full power) along the fenceline between DR-7 and DR-8. At DR-52, further measurements were collected with the PIC as a function of reactor power, provided by the Yankee Emergency Response Facility Information System (ERFIS). The results reported in the memo were:

    a. The background at DR-52 was 9.2 μR/h.

    b. A curve fit of the data, describing the net exposure rate at DR-52 as a function of reactor power is:

    $$X = 6.44E - 03 * Power\,(MW_e) \qquad \left(\mu R\ h^{-1}\right)$$

    c. This curve is reproduced below from the original report. The results are important. The error terms should have been accounted for in the residual error terms of the fit so that confidence intervals could be determined. Results show that for 100% power (535 MWe), 100% capacity, the annual exposure at DR-52 would be between 20-25 mR, nearly double the 1978 measurement results. This 1999 result is corrected for the measured background rate <u>at that time</u> (at zero power) of 9.2 μR/h. This data was further used in VYC-2067, which is described in the next section.

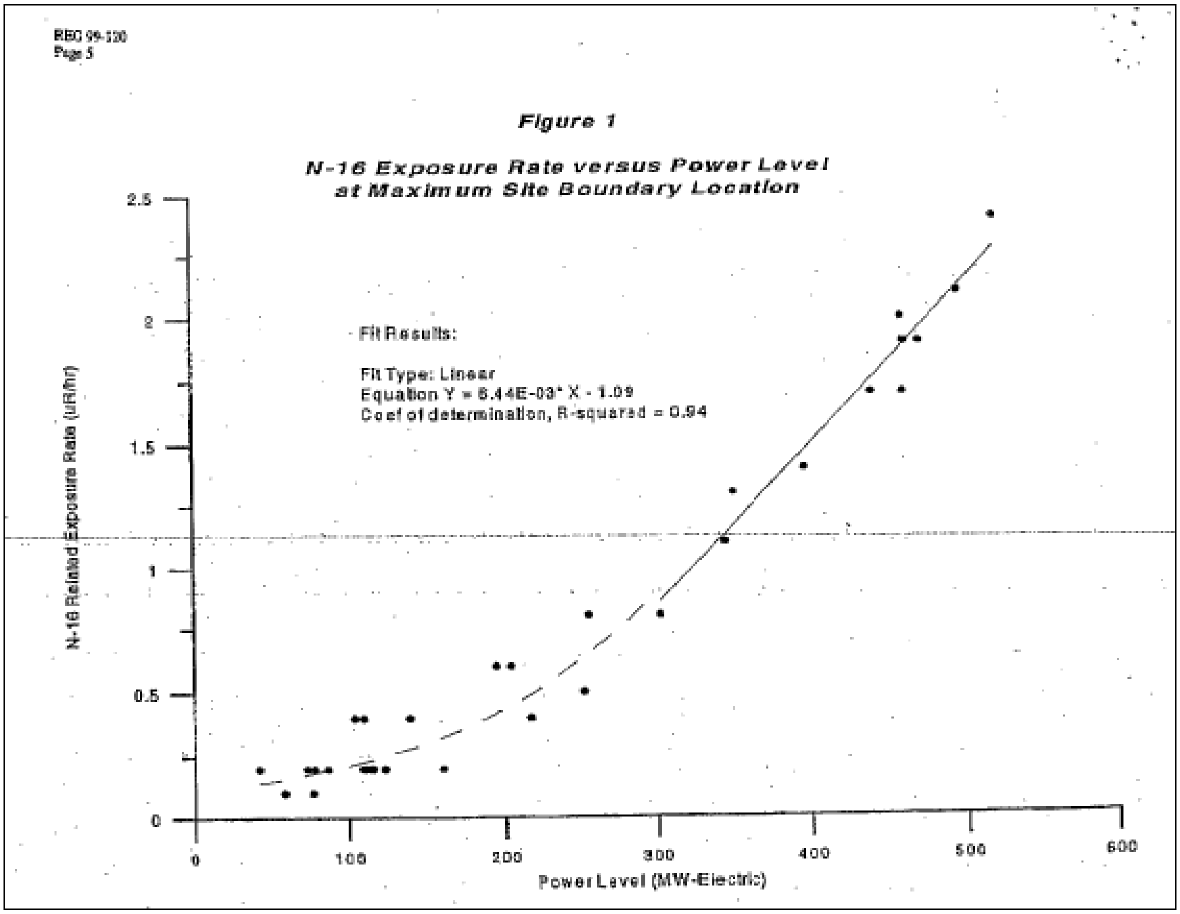

Figure 21. Curve Fit of PIC Data at DR-52 (Pelczar 1999)

6.1.2.3 VYC-2067 (1999)

A combination of radiological engineering reviews and discussions was ongoing by the late 1990s. The suite of measurements and calculations that were performed culminated in the release of an important technical document published by Duke Engineering Services (DES):

Duke Engineering Services, "Evaluation of N-16 Dose Contribution at West Site Boundary," **VYC-2067, December 28, 1999**.

An important point needs to be made about this document because its final release into the overall methodologies and current day estimates plays an important role. The report was released December 28, 1999, essentially providing the interim guidance until such time that the MSLRM approach could be tested, evaluated, developed, and placed into Rev 30 of the ODCM, October 30, 2002. During this time however, the USNRC provided significant comments to VYNPS on the measurements and calculations provided in VYC-2067. The comments were addressed by VYNPS and included as Appendix E of the VY-2067, which was released January 23, 2006. ORAU recommends that Minor Calculation Change (MCC) form ENN-DC-126 R5, which became appendix E of VYC-2067 be read and understood. ORAU reviewed VYC-2067 without the benefit of the USNRC comments and found independently that the measurement data could have been collected in a more robust fashion and that the error propagation was incomplete, including the statistical methods used to describe the net response above natural background. ORAU is not going to revisit the USNRC's comments in this report. Rather, ORAU accepts all the comments and the Appendix E reply from VYNPS as complete. ORAU will address the revised calculations here, and present in summary form. The reader should keep in mind that while this review of VYC-2067 was ongoing between VYNPS and the USNRC, VYNPS had decided to implement a new approach (termed MSLRM) because it provided more accuracy than the VYC-2067 method of relating dose rate to power level. The MSLRM method is presented in the next section. With this in mind, it is not understood why both the USNRC and VYNPS went into such detail of comments and corrections to the 1999-developed, VYC-2067 method. ORAU reaches a conclusion that because the VYC-2067 method was beginning to push the 20 mR limit (but not the mrem limit), a number of people were becoming interested in the results.

Using calibrated PIC detectors, the purpose of VYC-2067 was two-fold:

1. Measure the exposure rate profile along the west-side fenceline between DR-7 on the north to DR-8 on the south. Identify the specific location where the exposure rate was greatest; and

2. At the location of greatest exposure, measure the exposure rate as a function of reactor power.

ORAU generated an existing dosimetry map to include the measurement locations studied. Unfortunately, VY-2067 only provided "location descriptions" as "feet from DR-8" or "feet from the parking lot TLD." The map below (Figure 22) is what ORAU believes to be the locations of the measurements, between DR-7 and DR-8. ORAU believes as stated on the map that the "parking lot" location is identical to the location marked as DR-53. Notably, there was a significant amount of confusion about where these measurements took place, particularly as one reviews data that is now seven years old. The map below shows the distances (to scale) from DR locations indexed in the report, as well as the area "50 feet south of the parking lot" where the greatest exposure rate readings were obtained.

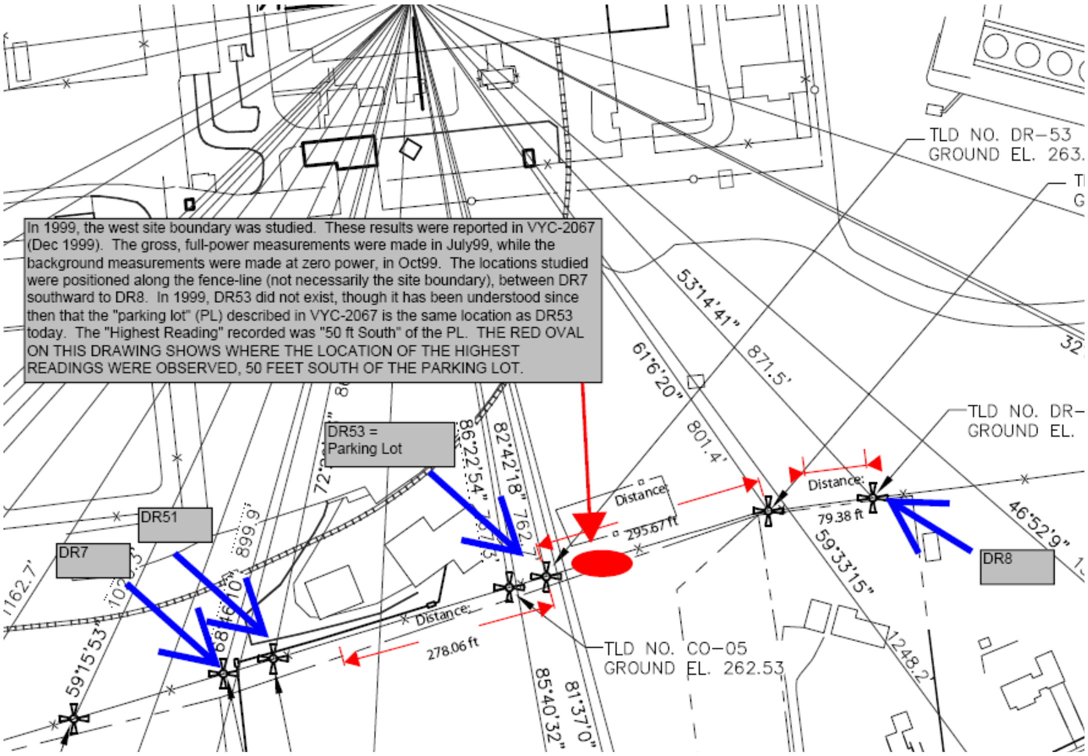

Figure 22. PIC Measurement Locations (VYC-2067)

Gross radiation measurements were conducted at these locations using a calibrated Reuter-Stokes RS-111 pressurized ion chamber. In VY-2067, Figure 2 is a plot of the gross measurement results, which ORAU has regenerated here for this report. A couple of conclusions from this plot include:

1. The data is plotted from south (DR-8) to north (DR-7), with the reactor at full power.
2. The notation for the measured locations is either plus or minus the DR location in feet.
3. The background measurement used was 9.2 μR/h, so the plot starts at this approximate rate.
4. DR-52, where the nearest resident resides, reads 11.8 μR/h
5. The maximum exposure rate is located at DR-53-50ft, at 12.5 μR/h
6. The new engineering building is located between DR-53-25 and DR-53 (according to VYNPS records).
7. With a background of 9.2 μR/h, the net exposure rate at the maximum point is 3.3 μR/h (**equivalent to 29 mR per year, 100% power, 100% capacity**).
8. Using a background of 9.2 μR/h, the measurement results at DR-52 are **20%** less than DR-53, which is an important conclusion because most measurement studies were performed at DR-53.

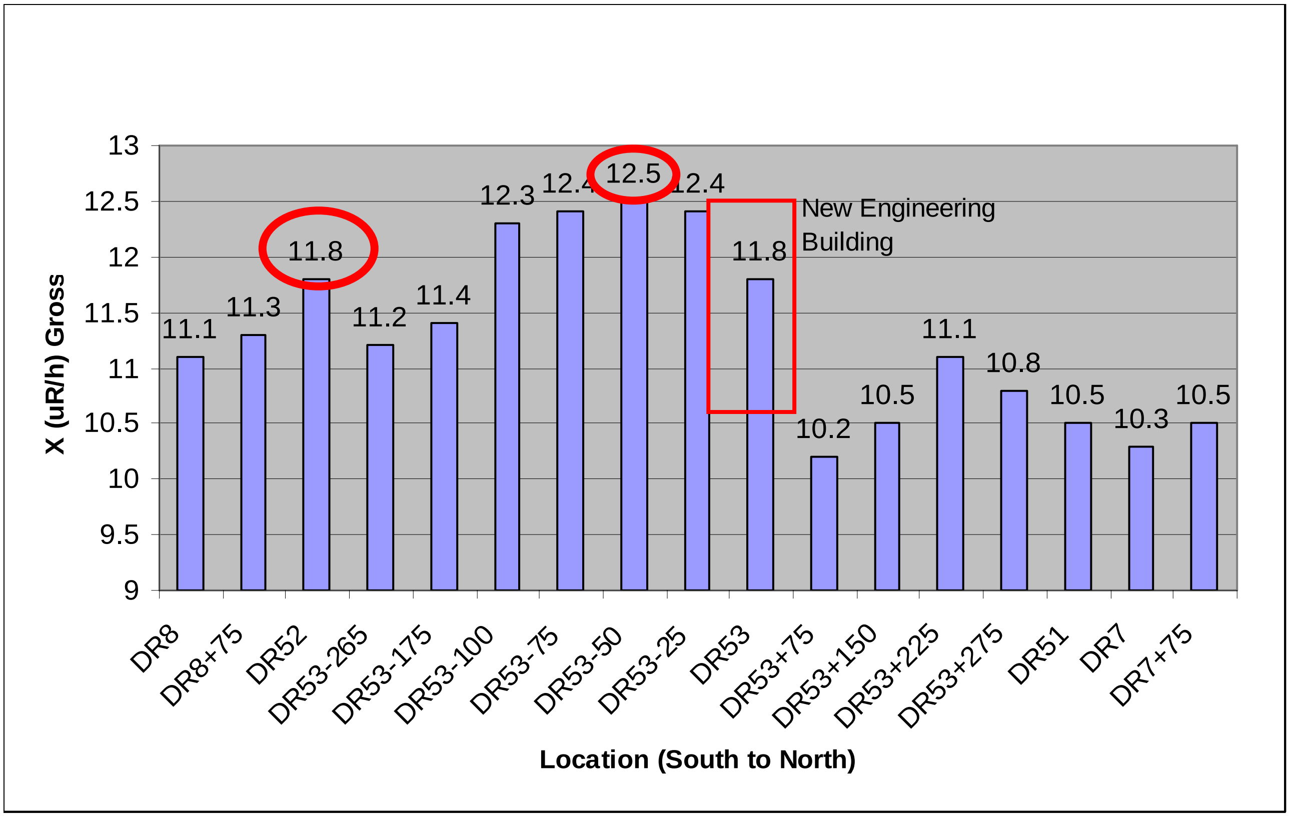

Figure 23. Gross Exposure Rate Measurements Along West Side Fenceline (1999)
(modified in different format from Figure 2 of VYC-2067)

The gross exposure rate measurements along the fenceline were collected at full power in June of 1999. In October of 1999 (10/29/99), measurements were collected at the "maximum location" referred to as 50 feet south of the Parking Lot, which is where ORAU has assumed DR-53 is today. The measurements using the same PIC as used before provided are shown in Table 2 of VYC-2067, which are plotted in Figure 3 of VYC-2067. The PIC data plotted in Figure 3 are the same data reproduced earlier in this report, showing an expression for exposure rate as a function of reactor power:

$$\boldsymbol{X = 6.44E-03 * Power\,(MW_e)} \qquad \left(\mu R\ h^{-1}\right) \qquad \textbf{(@ } \boldsymbol{DR}\textbf{52, 1999)}$$

This equation includes a correction for actual background measurement of 9.2 µR/h.

ORAU reviewed what later became Appendix E of VYC-2067 in January of 2006 and agrees in every way with the questions raised by the USNRC. Several mistakes were made in the original document. There were several relevant questions posed on instrument calibration, uncertainty propagation, unit conversions from roentgen to rem (the original concept was to multiply 0.87 * 0.60 for a multiplier of 0.52).

The conclusion that can be drawn from this is: there was clear indication that at the maximum location of the site boundary (DR-53), the time-integrated exposure rate for one year, assuming 100% power and 100% capacity would exceed 20 mR per year. If the utility had continued to assume that one roentgen was equivalent to one rem, then the corresponding dose equivalent would exceed the dose objective at that location. These measurements, incidentally, did not include a contribution estimate from "other fixed sources." Rather, VYC-2067 address only the ^{16}N component affiliated with operation of the reactor.

Fortunately, VYNPS recognized that determining the natural background radiation rate at the location of interest over the exposure interval of interest in the presence of a gross response was central to making an accurate assessment of "net exposure or dose above natural background." Studies and evaluations up through 1999 were limited by coming up with an accurate evaluation of background. As a result, VYNPS carried out two significant measurement campaigns prior to the release of Revision 30 of the ODCM (10/30/2002): 1) high resolution gamma-ray spectrometric studies to evaluate the contributions of direct radiation, skyshine radiation, terrestrial radiation, and cosmic radiation; and 2) deployment of a new detector system mounted on the main steam lines from the reactor to the turbine.

It appears that "negative" surveillance data from the TLDs placed in and around the plant provided an assurance that no major release had occurred. However, no one, neither the state's environmental dosimetry program nor the utilities', was carefully evaluating the performance of these TLDs to measure and ensure any sense of compliance at quarterly or annual dose conditions. It was not until 2004, in fact, that an apparent "positive TLD" raised a flag that the site boundary exposure rate could have been exceeded. As ORAU has now learned, this TLD result was a weak indicator that there was an issue—in fact, it was the 1999 set of PIC measurements conducted that should have raised a flag that the exposure limits were increasing above the 1973 and 1978 baseline data. The state either never read VYC-2067, or never responded that it was read and approved.

6.1.3 2000 to Today

As stated, a number of issues came to light in the late 1990s. Radiological engineers at VYNPS knew from a set of PIC measurements that the projected rates were leaving little margin for measurement error in order to comply with the site boundary dose quarterly limit and the annual dose objectives. An impetus to study these radiation rates was also motivated by the fact that new water chemistry tests were going to prove successful to reduce corrosion, but at the same time increase ^{16}N concentration in the steam.

The primary measurement studies that have occurred were documented in the following reports:

Duke Engineering Services, "Summary Report In Situ Measurements Performed at the Vermont Yankee Nuclear Power Station," EL 018/02, **February 13, 2002.**

[This report was edited and reissued as an AREVA document : FINAL Summary Report of In Situ Measurements Performed at the Vermont Yankee Nuclear Power Station," EL 145/05-FINAL, (January 26, 2006)]

Duke Engineering Services, "Vermont Yankee Site Boundary Direct Dose Determination Methodology," VYC-2194 Rev. 0, **March 20, 2002.**

Duke Engineering Services, "Vermont Yankee Site Boundary Direct Dose Methodology," VYC-2194 Rev. 01 – DRAFT (ENN-DC-126 R5), **November 28, 2005.**

The data presented in these reports provided a better understanding of the following direct gamma measurement issues. Consequently, demonstrating compliance with the state requirements from direct gamma dose at the site boundary would be best estimated by:

1. Improvements in the quarterly and annual estimates for direct gamma dose at the site boundary are achieved by measuring in real time (i.e. not a passive TLD dosimeter that reports exposure "after the fact"); and establishing a method for inherently accounting for natural background radiation. This method is now known as the MSLRM method.

2. Estimating all other fixed sources of direct radiation beyond the ^{16}N component. This list of possible sources is provided at the end of Chapter 2 of this report.

3. Understanding the components of natural background and how to subtract natural background from a gross response.

4. Understanding, by direct measurement at VYNPS, how to convert energy dependent fluence rate into units of exposure, absorbed dose, and dose equivalent.

This collective set of knowledge was included, to a large degree, into the most recent version of the ODCM, revision 30, October 30, 2002. The ODCM, as stated by VYNPS, relies on two separate methods to arrive at site boundary dose:

1. First and foremost, the MSLRM methodology is used exclusively to estimate ^{16}N-produced direct gamma radiation at the site boundary; and

2. Separate measurements and calculations to estimate the contributions from other sources.

The MSRLM method is a reasonable approach for estimating the ^{16}N component at the site boundary; it is much better than the previous method of estimating exposure from the time-integrated reactor power level (VYC-2067). ORAU has a few outstanding questions about derived coefficients describing the other "fixed source" contribution, and regardless of the outcome of this derivation, ORAU believes while the "projected" methodology is sufficient for planning, it is insufficient for making fine time-dependent adjustments to the possible set of "source terms" that could present themselves in the course of daily operations.

In the ensuing sections, important measurement results are provided for the MSLRM technique, and the mathematical constructs reported in the ODCM Revision 30. Results from the field measurements of the *in-situ* study are presented in Appendix B, whereby a flux to dose conversion was developed based on methods established in ANSI/ANS-6.1.1.

Finally, it is very important to keep in mind that the use of TLD dosimeters by VYNPS was not (and is not) intended for making a determination of compliance. Why? Because

VYNPS raised a valid question: how does one estimate natural background contribution at the measurement location of interest, and estimate it well enough to make a determination of a net quarterly increase of 1.15 μR per hour (1.15 μR/h is the effective reporting limit (10 mR) per quarter integrated over the entire quarter). Meanwhile, through all of this development, the state had continued to believe that their environmental dosimetry program was of sufficient quality to measure these low net results, which was (and is) a false sense of assurance. A "negative" TLD result is not capable in a statistical sense to prove or disprove an absolute quarterly limit of 10mR (or 10 mrem) has been exceeded. And thus it is the responsibility of both the State and VYNPS to understand the limitations of existing surveillance program methods and to ensure that the measurement methods deployed will function in such a way as to meet data quality objectives. A collection of data should be reviewed and analyzed each quarter so that a more robust, statistically significant assessment can be made on whether a limit was exceeded or not. Also, no agreed upon approach was reached between VYNPS and VDH on relating actual public dose from a site boundary dose measurement, including occupancy factors and local shield corrections.

6.2 VYNPS ODCM and MSLRM

The MSLRM calculational methodology constituted another major issue evaluated by ORAU for its adequacy and appropriate implementation. The MSLRM method is an integral component of the overall methodology described in the ODCM, Revision 30, and is currently used to satisfy USNRC radiological monitoring requirements. While the MSLRM is the single largest component, the ODCM gives a more complete description for sources of gamma radiation other than from ^{16}N at the site boundary. The VDH requested that ORAU include and evaluate all sources impacting on the site boundary dose. Chapter 2 describes these other "fixed" sources.

Due to their long-standing use, TLDs and pressurized ion chambers (PIC) fall in the category of conventional field measurement methods. The MSLRM, on the other hand, is considered by VYNPS to be a technology advancement. As noted previously, it is clear that the MSLRM is a unique application given the fact no other BWR, to ORAU's knowledge, utilizes an MSLRM method. The MSLRM methodology is intended to eliminate background as a confounding influence in meeting the assessment of whether a net quarterly dose equivalent at the site boundary exceeds 10 mrem. VYNPS has identified justification for the application of the MSLRM calculational methodology based on Part 5, Chapter 3, Section 5-303 ("Definitions" ["ALARA"]) of the State of Vermont regulations.

During one of the onsite technical meetings, VYNPS and AREVA staff provided an overview to ORAU pertaining to this methodology, including its purpose, instrument locations and operational characteristics, the correlation of MSLRM and PIC readings, and the effect of noble metal chemistry (NMC) and hydrogen injection on the monitors. Questions regarding the type, geometrical characteristics, and energy response of the four main steam line radiation monitors were raised and discussed. VYNPS provided ORAU with MSLRM calibration data for review and a major reference (draft VYC-2194) as the primary document for correlating MSLRM and PIC readings and the MSLRM approach and offsite dose. VYC-2194 was published by Dr. John Hamawi. ORAU received not only the mark-up copy of this report, but also all the raw data files, converted into Microsoft Excel.

The MSLRM method is a reasonable method for measuring, in real-time, the site boundary dose from ^{16}N production. It is the most accurate method for this purpose because background is accounted for initially when the reactor is at zero power. The MSLRMs, because they are operated at such relatively high radiation levels (>100 mR/h), are not sensitive to background fluctuation. As a result, an estimate for the net dose contribution from ^{16}N is estimated without an explicit (and concurrent) measurement/estimate of the background. MSLRM captures on the order of 97% of the total dose contribution at the site boundary, but does not capture contributions from other time-dependent processes described in the ODCM as "other fixed sources." Also, the MSLRM assumes that the gamma-ray flux at the boundary from skyshine is constant; in fact, the skyshine is dependent on atmospheric conditions.

The ODCM provides conservative methods for establishing estimates for other sources of gamma-ray flux not resulting from ^{16}N production. One example includes a description of the method to predict the exposure rate contribution at the site boundary from a storage cask. Each cask's contribution is considered separately even when physically shielded by other casks. ORAU noted that VYNPS has previously provided testimony (in 2005) before the Public Service Board (PSB) regarding predicted (hypothetical) exposure rates from onsite cask storage. These predicted exposure rates were reported as adding a dose of approximately 0.025 mR per month (0.075 mR per quarter) to the site boundary.

Revision 30 of the ODCM does not currently address dry fuel storage, but is expected to be revised later in 2007 to include this dose contribution.

> One question that was discussed during the ORAU review related to how "good is good enough", for these types of measurements. ORAU asked the VDH if given any measurement process used by VYNPS (<u>even assuming a perfect system/process, without bias or error</u>) used to demonstrate compliance with a regulatory limit, what key characteristics of that system would be considered satisfactory such that "independent verification by measurement" would <u>not</u> be required? VDH replied that there was no single measurement system that could be deployed by VYNPS to demonstrate compliance with the state radiation regulations limit<u> that would not require</u> independent measurement by the VDH.

This having been said, ORAU reviewed in its entirety VYC-2194 and the ODCM to reach the following conclusions:

1. The calibration method for relating MSLRM rate to DR-53 exposure rates is sound.
    a. The PIC monitor placed at DR-53 adequately measured background radiation when the reactor was at zero power.

    b. The PIC monitor placed at DR-53 adequately measured gross gamma radiation levels and the results were properly indexed in time to the ERFIS data. By indexing in time to the ERFIS data, a calibration curve could be generated. The ERFIS data set contains the following important outputs:

M120 Steam Line RM A (mR/hr)
M121 Steam Line RM B (mR/hr)
M122 Steam Line RM C (mR/hr)
M123 Steam Line RM D (mR/hr)
M002 SJAE Off Gas Activity (mR/hr)
C024 SJAE Off Gas Activity Release Rate (uCi/sec)
C047 Core Thermal Power Level (MWT)
G002 Generator Gross (MWE)
C183 Barom. Pressure - 15 min. Average (in. Hg)
M010 Air Temp (deg. F)
C192 15-min. Precip. Total (inches)
M126 RB Vent Exhaust RM A (mR/hr)
M127 RB Vent Exhaust RM B (mR/hr)

c. From this data all instrument adjustments were made (including PIC measurement correction for barometric pressure and temperature), and regression functions were established to relate the MSLRM reading to the PIC reading as a function of time (and reactor power).

Figures 24, adapted from Figure 3.3 of VYC-2194, shows a typical MSLRM detector response from zero power to full power. This graph shows that all four monitors are in very good agreement with each other and that, in general, the MSLRM exposure rate monitors read on the order of 400 mR per hour.

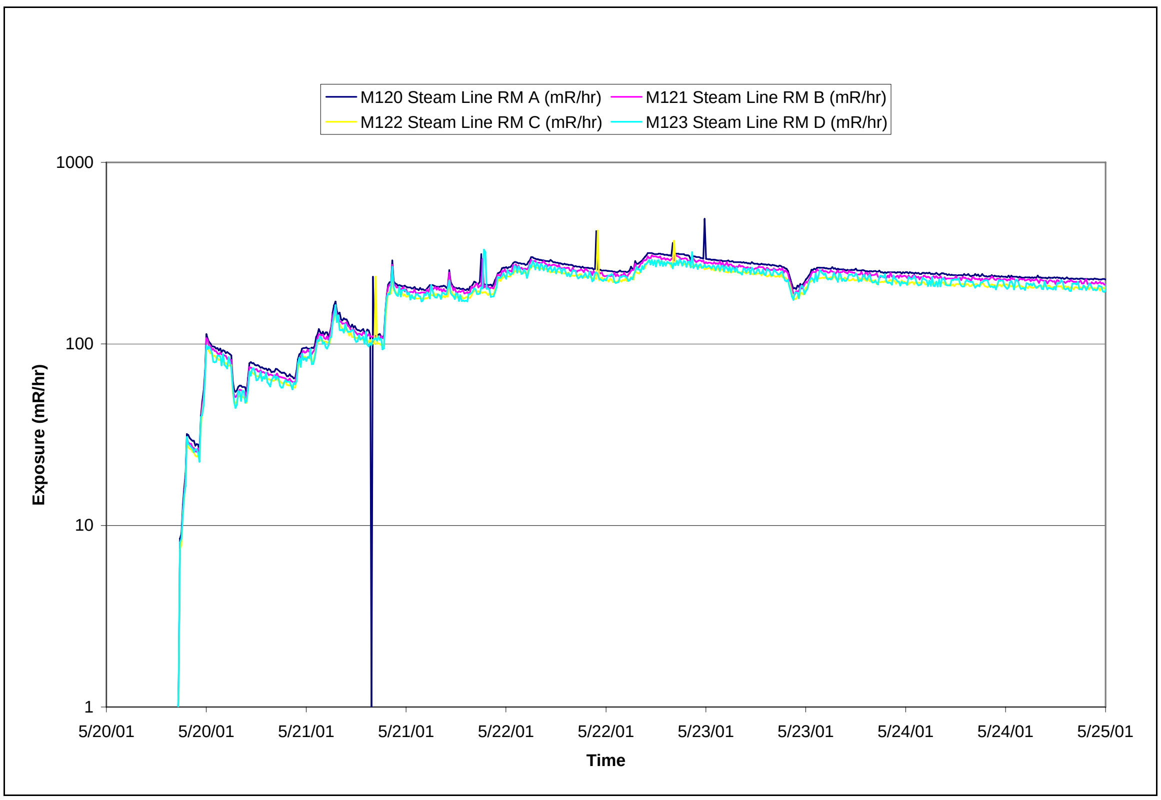

Figure 24. Steam Line Monitor Response

While the MSLRM detectors are reading 400 mR per hour, the site boundary (DR-53) PIC detector generates an exposure response on the order of what was presented in the report, and regenerated below:

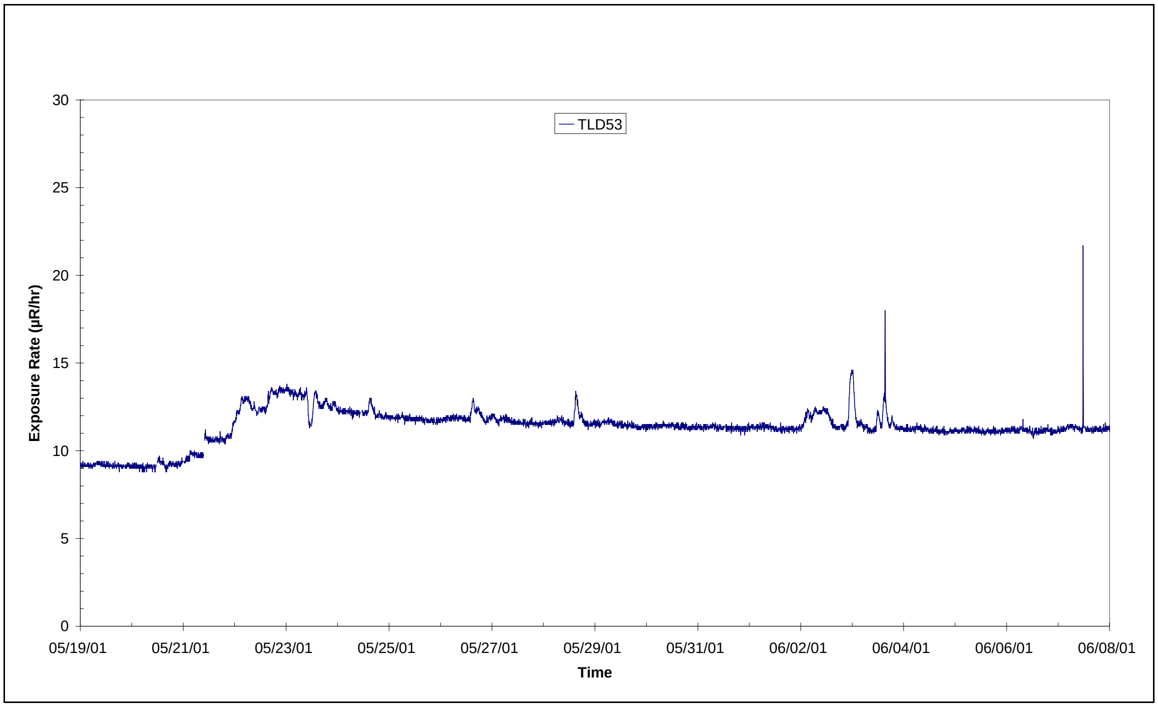

Figure 25. HPIC Detector Response Located at DR-53

From these response characteristics, a calibration curve is established. This parameter, referred to in the ODCM is the K_{N-16} parameter, has a value of 1.31E-05.

Looking at the final equation in the ODCM, the expression that calculates site boundary dose, D_d, from ^{16}N production in the core is given as follows:

$$D_d = K_{N-16} K_{tissue} K_{calib} D_{MSLRM}$$

Parameter	Description	ORAU Analysis
K_{N-16}	A factor determined by calibration of a PIC measurement placed at DR-53 and the average of the MSLRM detectors placed on the main steam line. In VYC-2167 this factor was determined to be 1.31E-5.	The calibration equation for establishing this factor is correct.
K_{tissue}	The exposure to dose conversion factor. In 2002, the ODCM uses as a factor 0.71 mrem/mR.	The origin of this factor is not specifically known. ORAU evaluated the set of in situ, high resolution gamma ray spectrometry measurements, and concludes that a nominal value for this parameter is 0.60 (see Appendix B)

Parameter	Description	ORAU Analysis
K_{calib}	A calibration factor for the detector	This calibration factor is reasonable based on the data reviewed. A more extensive review of the actual calibration protocol would likely lead to a few additional recommendations. For the moment, ORAU agrees.
D_{MSLRM}	The average reading of the four MSLRM detectors	This is the dependent variable in the calibration equation. A separate equation is shown to examine the dwell time of the MSLRM detector – the dwell time (or integrating time) for this measurement is adequate.

Recommendation: The MSLRM approach is a reasonable real-time monitoring scheme for estimating the site boundary dose at DR-53. However, the approach for estimating other (less controlled) means of direct exposure is less clear. Most importantly, it is difficult to control all the possible sources of direct gamma radiation. For every single operation on-site that involves the use of material that emits gamma rays, or radiation generating machines, a dose assessment review would need to be conducted to determine the level of contribution to the net boundary dose that particular operation would produce—ORAU believes this type of procedural practice should be avoided.

6.3 VYNPS in situ Measurement Studies

In the discussion regarding the development of the MSLRM technique, it is important to note that VYNPS was collecting additional data about the energy dependent flux profile at the site boundary. Important questions included:

- Is energy dependent flux from skyshine significantly different from natural background, that is, is there a significant contribution to flux between 1 MeV to 7 Mev?
- How can this information be used to better calibrate PIC detectors and TLDs for the energy dependence, that is, how can the data be normalized to dose equivalent?
- What are the various contributions to the measured signal (terrestrial, cosmic, skyshine, and direct line of site).

The development of VYC-2194, which included results from EL 145/05 (the *in situ* measurements) were concurrent. The development cycle, which included a significant technical review and comment from the USNRC is as follows:

VYC-2194 (MSLRM Approach)	**In Situ (HPGe Measurements)**
Rev 0 – 2002	Rev 0 – EL 018/02 –February 2002
Rev 1 – 2006 (draft)	Rev 1 – EL 018/02–December 2005
	Final – EL/145/055–Jan 26, 2006

The draft 2006 version of VYC-2194 had an extensive set of comments from the USNRC included as Attachment E. ORAU accepts this review as a judicious review of the results and agrees with the nature of the questions posed by the USNRC and replies from VYNPS.

ORAU had technical discussions with the Framatome/Areva engineers responsible for conducting high-resolution gamma-ray spectrometry measurements at the site boundary and the methods to calculate a spectrum-weighted response for dose equivalent following NMC (noble metal chemistry) and HWC (Hydrogen Water Chemistry) changes. ORAU reviewed this data, which included a review of ICRU Report 47, "Measurement of Dose Equivalents from External Photon and Electron Radiations".

The American National Standard Institute/American Nuclear Society ANSI/ANS-6.1.1, "Neutron and gamma-ray fluence-to-dose factor*",* was cited as a basis for the roentgen to rem conversion factor utilized by VYNPS at the plant. The adequacy of the VYNPS selected conversion factor of 0.71 was one of ORAU's primary evaluation issues in this work effort. ORAU subsequently determined that the methodology and approach is consistent with the ANSI/ANS-6.1.1 methodology and recommendations and recommended a lower conversion factor. The details behind this conclusion are found in Appendix B. With the May 2006 installation of a turbine building shield, which occurred following the ORAU flux to dose evaluation, the energy-dependent photon spectrum should be re-assessed during the plant's next outage.

The *in situ* high resolution gamma-ray spectrometry data could be used further, as alluded to in the document. There are energy dependent response functions presented for the TLD and for the PIC. These response functions could be used for calibrating the tissue-equivalent response for any detector, against the "as measured" spectral data. This suggestion is a very technical matter, but, if the true measurement endpoint is dose equivalent, the *in situ* data can be used to further the cause and improve the estimated conversion. Further discussion regarding the calibration of TLDs for tissue-equivalent response is provided in Chapter 8.

ORAU understands that VYNPS intends to continue the use of PIC measurements for real-time measurements until the relationship between PIC and MSLRM readings is established in concert with the Power Ascension Test Plan (PATP), consistent with commitments made to the USNRC. The gamma spectrometer measurements are no longer required; these measurements were conducted years ago, and only need to be conducted to provide a flux to dose conversion factor. If plant conditions are such that a spectral shift has occurred, a subset of these high-resolution gamma-ray spectrometry measurements could be repeated.

Once the PATP measurements are completed and a calibration of the PIC versus MSLRMs occurs, VYNPS intends to discontinue the use of the PIC detectors. The VDH has stated their interest in seeing the PIC measurements continue for an indefinite period. ORAU recommends the calibration data be reviewed by the State prior to approval and the issue of ongoing (or not) PIC measurements be resolved via the suggested ECP.

An extensive review of the *in situ* data for purposes of deriving the flux to dose conversion factor is presented in Appendix B. The reader is encouraged to review this significant effort in order to understand why the "isotropic" flux approximation was selected over the "rotational," geometry, thereby reducing the 0.71 multiplier to 0.60.

6.4 VDH and VYNPS Environmental Monitoring Programs

The VDH and VYNPS have implemented environmental monitoring programs prior to the operation of the nuclear power plant in 1972. During the course of this work effort, ORAU became knowledgeable with each program, with the area of emphasis on the use of instrumentation for measuring direct gamma radiation. ORAU has reviewed historical annual monitoring and surveillance reports provided by both VDH and VYNPS; visited deployment locations; and interviewed staff from both groups, including the former (now retired) Chief of the VDH, Mr. Ray McCandless.

The environmental reports are available to the public at the Brooks Library in Brattleboro, VT. VYNPS issues annual Radiological Environmental Operation Reports (REMP) and the state of Vermont issues annual Environmental Radiation Surveillance Reports.

- The VYNPS reports are written to satisfy technical specification 6.6.E as required in the USNRC operating license, DPR-28. The collection of data, the analysis of the measurement results, and the report are performed within accordance of the plant quality assurance program, required for environmental compliance, under 40 CFR 190. Specifications for the measurements and the analysis are provided in lower-tier operating procedures or the ODCM. There is a substantial amount of rigor in preparing these reports.

- The VDH reports are prepared as a good faith measure to report results of environmental sampling and TLD results. It is unclear, however, what specific requirements the annual reports are written to meet. There are no cited operating procedures or quality assurance plans, nor are there cited performance metrics against which the results are evaluated. There is not substantial amount of rigor in preparing these reports.

As described in the regulations, Chapter 1, Table 1, the utility issues the annual report to satisfy overall environmental requirements for allowable public dose in 10 CFR 20.1301(e). For direct gamma dose at the site boundary, the VYNPS ODCM states that the MSLRM method will be used to measure gamma dose due to ^{16}N and additional calculational estimates will be used to define all other, less important contributions. These methods, which make use of calibration curves to relate detector response of PIC measurements made at the site boundary to MSLR measurements. It is clearly pointed out in addition that environmental TLDs are deployed in rings around the plant to:

- "Provide an early indication of the appearance or accumulation of any radioactive material in the environment
- Provide assurance to regulatory agencies and the public that the station's environmental impact is known and within anticipated limits
- Verify the adequacy and proper functioning of station effluent controls and monitoring systems.
- Provide standby monitoring capability for rapid assessment of risk to the general public in the event of unanticipated or accidental releases of radioactive material."[30]

Specifically, the VYNPS environmental TLD measurements are not conducted in order to prove or verify that the state quarterly limit of 10 mR has not been exceeded. VYNPS states, and correctly so, that using a TLD to reliably measure an small exposure above background places too much emphasis on what background was over the monitoring period. As a result, VYNPS developed the background-adjusted MSLRM method to measure direct dose. The differences between the implied TLD program requirements of VYNPS and VDH make for great difficulties in comparing environmental TLD data without a requirements-based, qualified program between the two groups.

On the other hand, VDH implies, though it is not written, that the state of Vermont TLD monitoring program will be used to assure the public and its stakeholders that the limits are satisfied. Yet historically, the TLD measurement efforts have not been conducted under a requirements document, a quality control program, or operating level procedures. While it is possible to make 10 mR/quarter net measurements with environmental TLDs, this level of detection sensitivity and accuracy can only be accomplished by implementing a rigorous quality program, using recommendations put forth by Klemic's reports[31] and draft ANSI standards. Consistent algorithms, data reduction, and statistical analysis methods should be implemented to ensure quality results. Historical VDH environmental reports do present analytical measurement results, but because there was and is no implemented qualification program for these measurement devices, the results could be interpreted in a variety of different ways, leading to confusion between VDH and VYNPS. The purpose and goals of VDH environmental monitoring program need to be prepared; specifications for instrument and method qualification need to be written; and documentation needs to be prepared to describe whether the VDH measurements verify or validate the results provided by VYNPS, or are the final measuring stick for compliance. The lack of rigor in the VDH monitoring program lead to misunderstandings in 2004, that lead to this ORAU evaluation of direct gamma dose.

A good environmental program should be initially designed using guidance provided by the USNRC, meeting the intent of USNRC Regulatory Guide 4.1 which provides general guidance on establishing a pre-operational and operational program. Regulatory Guide 4.13 discusses dosimeter placement. In brief, the "4" series of USNRC Regulatory Guides is devoted to environmental topics. In its discussion with VDH and VYNPS staff, ORAU

[30] The environmental surveillance and monitoring design requirements are listed in the Entergy Vermont Yankee 2005 REMP, May 10, 2006.

[31] Klemic, G. "Environmental Radiation Monitoring in the Context of Regulations on Dose Limits to the Public," 1996 International Congress on Radiation Protection, April 14-19, 1996.

determined that pre-operational background data is not being used, though it was collected (see Chapter 6 discussion on historical measurements).

Environmental programs are impacted by many factors. Putting aside the administrative aspects, e.g., cost of dosimeters and staffing considerations, a few of these include variability in natural background caused by weather and several other factors described in Chapter 3. "Lost data" can result from vandalism when a dosimeter is stolen from a deployed location. Operation of a PIC involves significant technical and maintenance issues.

Each program is described below with emphasis on TLDs in this section. ORAU analyzed TLD data from the VDH and VYNPS environmental monitoring programs for the period from 1995 through 2005.

ORAU evaluated this data to:

- Evaluate the consistency and variability of background and non-background locations; and

- Identify trends and patterns (either anticipated or unanticipated)

A general comparison was also made (since there were insufficient co-locations) of the results from the VDH and VYNPS monitoring programs, though because the TLD programs are different and implied measurement objectives/specifications are disjoint, ORAU advises against attempting to draw any definitive conclusions from this data that one measurement protocol is better from the other or that there are biases in one measurement approach that are not accounted for in another. ORAU spent many hours with the physicists from VYNPS, VDH, and the dosimetry vendors (AREVA ANP and GDS) attempting to unfold the results. The fact of the matter is, the programs were implemented only to identify trends in low level background measurements. VDH may have intended to use these TLD measurements as if they were the defining accurate and precise measurements for evaluation against regulatory requirements for direct gamma dose, but the fact of the matter is the deployed TLD program was never put through a qualification program.

Based on the data provided to ORAU, it appears that no locations were jointly monitored by both VDH and VYNPS until 2005. Therefore, ORAU did not have the sufficient data to compare results from both group's monitoring programs for the same locations. ORAU recommends that this data be re-evaluated once several more years of data is available for these new locations. Also, a great many monitoring locations were added to the VDH monitoring program in 2006, nearly doubling the number of locations monitored by VDH TLDs. Data for these new locations were excluded from the ORAU analyses because there was only one data point for each of the new locations (first quarter 2006).

ORAU noted that the VDH and VYNPS currently utilize different TLD vendors and dosimeter results are provided in distinct ways. Regarding vendors, an ideal result from the ORAU evaluation would have been that since VDH and VYNPS TLDs were ostensibly measuring the same radiation field, similar, if not identical results would be obtained, thus generating confidence in an ongoing intercomparison. Based on the data provided to ORAU, however, this was not the case. Even when looking at long-term averages over a large number of locations (to account for natural variability), the average response of the two types of TLDs used by VDH and VYNPS was not equivalent.

ORAU's overall conclusion was that the VDH TLDs responded higher, and with greater variability, than the VYNPS TLDs. VDH relative standard deviation typically ranged from 10 to 20% for each location over the 10 year period. The VYNPS relative standard deviation typically ranged from 5 to 10% for each location over the 10 year period. Some of the variability was no doubt due to actual variations in the background radiation level. However, if this were the only factor, then the relative variability (%) of the two types of TLDs would be expected to be roughly the same. Since this was not the case, it would appear the VDH TLD results were varying significantly more than could be attributed to normal background fluctuations. While not known with complete confidence pending a greater in-depth evaluation, ORAU believes the higher variability is likely due to the type of TLD used in the VDH monitoring program and the procedure/algorithm used to account for: transit dose, self-dosing, glow-curve analysis, fading, control badge correction, background subtraction for net analysis, environmental ruggedness, data anomalies, and the performance metrics for use described in ANSI N545-1975, the draft ANSI N13.37, and draft ANSI N13.29, including the qualification of:

- Minimum quantifiable dose (MQD)
- Angular response of TLD
- Energy dependence of TLD
- Uniformity and reproducibility of TLDs
- Linearity, including low-dose linearity
- Environmental stability (light, moisture, air, temperature)
- Outdoor field tests
- H*(10) dose equivalent calibration
- Identification of QC TLDs (controls, blanks)
- Element averaging and outlier rejection
- TLD reader capability and performance spec (written into contract with vendor)
- Calibrating on a batch basis.
- Participation in intercomparison programs and internal audits
- Siting and packaging of dosimeters in the field (height, security, safety) and how to deal with stolen or damaged devices. TLDs can be eaten by animals, stolen, or environmentally damaged.
- Controlling and measuring the "storage and control" dose
- Measurement of the transit dose
- Development of procedures and a QC/QA program

The "good practices" list for qualifying TLDs is extensive. But, if the TLD measurement alone is to be used as the gatekeeper for whether the state limit was exceeded or not, then this level of effort is required. For this reason, ORAU is advising a graded approach: use the MSLRM real-time measurements to ensure that the primary component of the direct gamma radiation is monitored. Augment the MSLRM method with a proceduralized process to identify "all other sources" on a regular basis (rather than simply updates to the ODCM). Put in place some type of ALARA measure within RADCON that if a "hot operation" is to take place onsite in the yard, that these infrequent operations be documented with a quick estimate of site boundary dose. And finally, the TLDs serve as a last line of defense to ensure that the data is trending properly and that no significant change in the process has occurred. Statistical methods will need to be developed for reducing the number of TLD data, averaging the background distribution, and the ability to identify spurious reading and qualify them as such (both low and high). The other significant issue that should be managed is TLD reporting units.

Regarding the measurement units, results from VDH and VYNPS are reported in different units. This makes a comparison difficult. The VDH reports their results in millirem/quarter; the VYNPS in microroentgens/hour (uR/h) apparently based on their contracted vendors protocol even though both use the Panasonic 814. Conversion between the two units introduces unnecessary complication and opportunity for flawed comparisons between the data sets. One of the complications is having an understanding of what constitutes a "standard" quarter. The TLD analysis software used by the VYNPS vendor computes the exposure rate based on the number of hours in 91 days, given a provided start/stop time. In order to compare VDH and VYNPS results, the data needs to be normalized to the same reporting unit. In Quarter 4 of 2004, for example, VDH used a 103 day cycle (covering the period October 13, 2004 to January 24, 2005). In addition, the cycle time can vary each quarter depending on factors such as receipt of dosimetry data, field placement date, and dosimetry processing shipment date. A significant difference in 103 and 91 days could result in divergent results without a proper accounting of this variable.

Regarding weather effects (snowfall), the 2005 VYNPS Environmental Operating Report states that the lowest background levels occur during the winter months. ORAU concurs with this conclusion, as well as with the probable cause being snow cover attenuation of radon emanation and direct radiation from naturally occurring radionuclides in the soil. This pattern was apparent when comparing results for the quarters within each individual year; however, when the results were averaged over the 10 year period (covering our analysis), there was no discernable difference between the average results for first, second, third and fourth quarters.

Another pattern that emerged was that the relative standard deviation for the first quarter results over the 10 year period was typically factor of 2-3 higher than for the third or fourth quarters. This pattern appeared in both the VDH and VYNPS TLD data. The higher relative standard deviation for first quarter results was almost certainly due to weather (e.g, (snow) effects mentioned above. Varying levels of snow cover from year-to-year cause large variability in the first quarter monitoring results.

VYNPS collocates specified TLD locations with the VDH and collects them at the same time. ORAU recommends the continued use of collocated TLDs as a means to foster improved consistency in their respective measurement methodologies. The details can be determined as a part of the recommended ECP.

With these programs, and particularly this task, the distinction between "demonstrating compliance" (e.g., VYNPS through adherence to quality programs) and "independent verification/measurements" by the VDH (i.e., TLDs) came into play. VYNPS noted that meeting the NRC requirements were one option that could be considered and should suffice and the plant should not have to rely on the State's TLD results to establish compliance with quarterly/annual limits.

6.5 VDH TLDs

It has already been stated that the purpose and measurement specifications for environmental dosimetry using TLDs was different between VDH and VYNPS. VDH expects better accuracy and precision in the measurement protocol than is achievable today. VYNPS expected the TLDs to provide general trending analysis and accident dose reconstruction. While both VYNPS and VDH used the same TLD (Panasonic 814), different providers were (and are) used. Thus the qualification of the TLDs and specifications for calibration, analysis, corrections, and reporting of the results was inconsistent. Inter-comparing results in the absolute sense is of dubious value, but the trending should be similar, for what data was provided. TLD data that does not trend reveals QA/QC issues that will need to be resolved in the future, and definitely requiring proceduralization of statistical, data-reduction methods.

> *Note: ORAU evaluated a substantial amount of environmental TLD measurement data from both the VYNPS and VDH reports issued over the past 10 years. No graphs or plots of the data reveal any specific trending patterns. As a result, these figures were not duplicated from the annual reports submitted by both VYNPS and VDH.*

ORAU's analysis of the VDH TLDs for the time period in question revealed that, even after converting to identical units, the VDH TLD results were consistently about 25% greater than the VYNPS TLD results, indicating a calibration and analysis bias. This conclusion was based on the following:

When averaged across the entire 10 year period, the VDH results averaged about 19.5 milliroentgen/quarter (gross). The VYNPS results averaged approximately 7 microroentgen/hr. Using a conversion of 8760 hours/year * ¼ year per quarter * 1 milli-R/1000 micro-R, the VYNPS average correlated to approximately 15 milliroentgen/quarter (gross).

Regarding the impact of subtracting an average background to determine net reading, the average background radiation level is determined by the VDH as the average of two control locations. This average background reading is subtracted from gross readings at indicator TLD (vs. control TLD) locations to yield the net reading, which is assumed to be the above-background dose resulting from plant operations. The underlying assumption/necessity is that the background level at the control TLD locations is similar to the background level at the indicator TLD locations. "Similar" is the key word, and is an important qualifier for the level of confidence to put into any results. Background levels will never be identical at any two locations, and also vary across the site by location and by time, as discussed in Chapter 3. VYNPS recognized that the analysis of net TLD results was brought into question primarily because the uncertainty in the background assumption drives the overall uncertainty in the net TLD analysis. For this reason, VYNPS suggested the background-corrected MSLRM method.

As would be expected with a large number of monitoring locations, the results for some indicator locations were higher than the average background, yielding a positive net reading, while other readings for other locations were lower than the average background, yielding a negative net reading. This phenomenon was a result of both the normal background fluctuations and the statistical variability of the TLD response. A slightly positive net reading is not necessarily indicative of an exposure above background rates, while a slightly negative net reading is not necessarily indicative of instrument failure. Based on the VDH TLD data, these normal fluctuations can be as large as several milliroentgen per quarter. Thus, the "apparent" margin for error, when compared against a small quarterly limit of 10 mR, is excessively small. If direct gamma dose is to be measured accurately at the 10 mR per quarter level in the presence of a natural background level of a nominal 20 mR per quarter background, then precision methods must be used to make a reliable judgment. The same can be said for meeting the 20 mR per year objective in the presence of an 80 mR per year natural background level. Now consider not the mean rates, but the variability of these mean rates over time, for which statistical hypothesis testing is introduced to determine whether a limit has been exceeded.

In the big picture, these slight variations around an average background value would not be an issue; compliance would be expected to be determined from the long-term average of the results. However, since the 20 mrem limit in Vermont is so low compared to these normal, random fluctuations, a few positive fluctuations can push what is essentially a background reading significantly to the regulatory limit. These fluctuations, or variability, must be introduced for the hypothesis test of whether a limit was exceeded. Furthermore, terms and conditions must be agreed to between VYNPS and VDH as to how these statistical results should be processed and reported.

There are several programmatic elements of the VDH TLD deployment for the past ten years that appear to ORAU as an evolving process. For example, the identification of locations to place TLDs, where to collocate TLDS with VYNPS, how to combine data from nearby TLDs, how to subtract background, the specifications for the TLD vendor, managing the control dosimeters, reducing data, accounting for lost/damaged data, and identifying procedures and QC plans are ongoing activities. VDH should programmatically evaluate the intentions of its TLD monitoring program; identify program needs; and reliability, precision, and accuracy requirements in order to improve surveillance capabilities and to augment the ORAU-reviewed monitoring and reporting methods provided by VYNPS. The programmatic evaluation will specify how many TLDs to locate, where to locate them, how high to locate them, when to collect them, how to collect them, how to process, how to calibrate and report. The general methods need to be the same for VDH and VYNPS so that reliable measurement intercomparisons can be made.

ORAU read several of the environmental surveillance reports issued by the State of Vermont and finds the following observations of interest:

- Up through 2004, the summary table near the beginning of the report provides the following entries for TLD (mR/quarter): Historical Range (0-12.5) and Result (0-12.5). It appears to ORAU that VDH simply looked at historical ranges to ensure that the report for the year of interest fell within the range of historical (gross) values.
- In 2004, VDH introduced a new concept to convert gross TLD reading to net TLD readings, stating "The exposure above background is the exposure of each location minus the average of the exposures at Putney Town Clerk's Office in Putney and the Vermont State Highway garage in Wilmington. As is now known, failure to understand the nature and variability of background in these locations, account for statistical aberrations at background locations and site locations, lead to an original notion that the plant had exceeded its limits.
- The TLD data presented in the reports begins with a brief summary which includes the statement "the annual exposure at the boundary of Vermont Yankee is less than 20 ± 5 mrem." It is interesting to note that the state uses an uncertainty estimate for reporting the limit. Furthermore, it is interesting that the state converted in the measurement results from mR* to mrem. ORAU could not determine that a validated and verified calibration approach for the TLDs to respond to dose equivalent, was utilized.

6.6 VYNPS TLDs

The VYNPS "Annual Radiological Environmental Operating Report" for the year 2005 was reviewed which summarizes the plant's yearly submission requirements to the USNRC for its "Radiological Environmental Monitoring Program (REMP). One specific aspect of the REMP is the deployment and subsequent collection and analysis of TLDs for the purpose of establishing direct gamma radiation levels. (Refer to Chapter 5 for a general discussion of TLDs and other equipment used to measure direct gamma radiation.)

VYNPS deploys TLDS in the field at specifically designated locations on a continuous basis for approximately 3 months at a time (quarterly basis), then collects the TLDs and contracts with an external vendor, AREVA ANP Environmental Laboratory, to analyze the results. The TLDs are positioned quarterly within monitoring "zones" to assess where the plant may be impacting the environment. Two primary zones exist: 1) An "Indicator" zone; and 2) "Control" zone.

VYNPS currently utilizes a total of 53 TLD monitoring stations to achieve the objectives discussed in section four of the annual report, "Program Directives." While allowance for changes in the monitoring locations is permitted per the ODCM, none of these locations were changed in 2005. Location DR-53, of interest in this report, was added as a site boundary dosimeter in 1999. Interested reviewers of the 2005 VYNPS REMP can find the specific locations of these 53 stations in Table 4.3 and illustrations in Figures 4-4 through 4-6, respectively. VYNPS, and other nuclear plants in this country, determine these locations through an evaluation of several factors, including meteorological conditions. Twenty-one (21) TLDs are positioned within an inner ring; 16 in an outer, incident response ring; 14

posted at or near the site boundary; and two “control” stations located beyond 11 kilometers of the plant and therefore, assumed to not be influenced by plant emissions.

Summary results for 2005 are provided in Table 5.2 of the REMP. Results are provided in units of microroentgens per hour for the inner ring, outer ring, offsite station with the highest “mean” reading (Location DR-36), and the control TLDs. The REMP identified that the lowest TLD readings occurred during the winter months. Monitoring station DR-45, a site boundary location located northeast (on the “back side”) of the plant, was reported as having the highest average exposure rate. Due to its location between the plant and the Connecticut River, the dose potential to a member of the general public would be low.

6.7 Examination of the 2004 Q4 VDH TLD Measurement Results

In January of 2005, VDH received fourth quarter results from it dosimetry vendor, Global Dosimetry Systems. Upon examination of the data at the location denoted “VY Parking Lot,” and subtracting relatively new estimates for natural background radiation rates at Putney and Wilmington.

Taking the average mR* values from the Proxtronics report, fourth quarter of 2004, the following analysis was performed:

The VY Parking Lot TLD (co-located with DR-53) gross reads:	29.5
The average of Putney and Wilmington gross reads:	17.55
The net VY Parking Lot TLD is the difference:	11.95

The 11.95 exceeded the 10 mR per quarter limit. An investigation was initiated.

In addition, the sum of all four quarters for 2004 was 24.9.

These simple calculations do not include propagation of error, which lead to the Q4 result of 12±3, but if one uses the reported net mean and reported error terms for each of the four quarters:

Q1	5.3 ± 0.5
Q2	1.5 ± 0.7
Q3	6.1 ± 0.9
Q4	12.0 ± 3

And the independent error terms summed in quadrature, the annual result would be 24.9 ± 3.2 mR

The primary argument behind the fact that these net results are incorrect is based on an analysis of the background rates in Putney and Wilmington. From the 2004 VDH dosimetry report the gross TLD results are presented:

	Q1	Q2	Q3	Q4
Putney Town Clerk	21.7	27.7	23.8	16.2
Wilmington	24.1	30.7	28	18.9

When comparing these measurements against all other reported values, the Q4 background values of 16.2 and 18.9 are reported low. There are many possible analysis explanations for this, but it's no more than a hypothetical guessing game as to what really occurred. Suffice it to say, the Q4 results for describing the background rates on site were probably in error on the low side, which in turn created an inflated result for net exposure at DR-53. Whether or not this is true or not will never be known because no robust statistical analysis and data reduction methods had been developed to select the best background, look for outliers, identify problems, and use filtering/averaging to identify a "true signal above noise." This is why ORAU has repeatedly stated that no statistical evidence has been presented to suggest that an exceedance actually occurred, in the presence of all other supporting data provided. The result of this investigation has in turn not been an identification and confirmation of a direct dose problem, but rather, an evaluation of what measurement methods need to be in place to ensure that the direct dose is reliably measured and reported so that the result can be effectively compared to the regulatory limit.

Since these observations, VDH has placed additional TLDs in strategic locations, attempting to collect more data for intercomparison, and evaluate detector reliability and variability. However, the follow-up has not occurred in a systematic fashion, for example, defining the quality of the measurements, the specifications, and the requirements so that definitive decision making can be achieved reliably. Programmatic elements need to be defined first, and agreed to within the framework of an ECP with VYNPS.

7.0 Background Subtraction Methods

As stated in the introduction to this report, regulatory limits and dose objectives are developed and implemented on the basis of net radiation dose above background. These requirements have been and always will be problematic when the net allowable dose is a fraction of the background rate and the background rate variability is high. In statistical terms, consider a mean background rate of 10 μR/h. For one calendar quarter, this would produce a total exposure of 22 mR. Further consider that the relative standard deviation of measurements made over the calendar quarter is 20%[32], so that the standard deviation is 0.20 * 22 mR = 4.4 mR. The specification for the calendar quarter is that the measurement device must be capable of detecting 10 mR over a background distribution with a mean of 22 mR and a standard deviation of 4.4 mR.

This simple example illustrates the statistical difficulty in making this decision. Even if ideal instrumentation could be constructed to measure the actual background rate at the site boundary, the natural variability is high (rain, snow, sun spots, solar flares), making the statistical proof of an exceeded dose objective very challenging. This feature of "mother nature" is probably why there are historical memos and documents between VDH and VYNPS that 20 ± 5 mR satisfies the annual objective, that since the measurement result is 19 ± 3 mR for the year, this also satisfies the objective, or that until 2004, the State would report a historical range of annual gross TLD measurements (0 to 30 mR) and conclude that the recent measurement year fell within that range. Clearly, the subtraction of background and accounting for the propagation of error in both the gross and background estimates is the most significant challenge in making this determination. VYNPS recognized that background variability was high when reviewing PIC data during observable changing environmental conditions (rain, snow). This led them to consider a new approach for monitoring site boundary dose by removing the background contribution, i.e., the MSLRM method. To make the measurement situation further complicating, skyshine contribution to site boundary dose (the most significant component) not only varies with power level as described, but also varies with environmental conditions (photons scatter differently as a function of atmospheric conditions).

In the statement of work to ORAU, VDH inquired what could be done to improve background subtraction methods: statistical smoothing, measurement intervals for smoothing, *in situ* studies to evaluate the most representative surrogate background locations, background averaging, and statistical methods. These concepts have never had to be implemented in practice at any other site in the country. However, there are numerous published TLD inter-comparison studies and several draft ANSI standards that have been prepared from which better decision making can be made. In the near term, statistical methods need to be developed to ensure erroneous/outlier data is detected and removed from the data set: one or two outlier TLD readings that look erroneous probably are. Statistical methods need to be developed to identify and manage results. A decision tree needs to be assembled for the conditions under which a noncompliance is identified with confidence.

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[32] A review of 10 years of TLD data provided by VDH and VYNPS reveals the following. For VDH, ten years of TLD data (40 measurement points) shows a relative percent standard deviation of between 17-22% for 10 TLDs. For VYNPS, the percent relative standard deviation is about 10%.

Because of these issues with "background identification and subtraction," the following three point plan should be implemented:

1. The MSLRM approach should provide the first, most reliable form of monitoring site boundary exposure.

2. "Other" sources of direct radiation should be evaluated, measured and documented on a routine basis, say quarterly. In most cases, the review could be rapid with a result of "no modifications or changes this quarter."

3. Utilize TLD results to trend the background and conduct measurements to ensure that no significant change in the process occurred that was otherwise undetected. TLDs could account for the one time, radiation-producing operation that was not recorded by the program for estimating exposure contribution. Fortunately, the most significant contributor to site boundary dose is a relatively steady-state source, as a function of power, meaning it is easily predicted and managed. There are no surprises: On any given day skyshine is not going to suddenly increase by a factor of 2. This is actually what enables the MSLRM approach to yield effective results and the fact that background is automatically adjusted (see Section 7.1).

The ORAU recommended plan for moving forward was briefly emphasized in this chapter because background measurements and subtraction of these measurements is central in the decision making process for evaluating compliance with the state regulation.

7.1 Intrinsic Background Subtraction in the MSLRM Method

VYNPS developed the MSLRM method because by definition, it measures radiation readings at the MSL and infers net site boundary exposure. Section 6.2 of this report provides a complete description of this method which is described by VYNPS in technical report VYC-2194.

The figure in Section 6.2 shows real-time PIC data collected May 19 through June 8, 2001, for a PIC stations at DR-53 (later designation became TLD-53). In parallel with this data of gross gamma exposure at DR-53 as a function of time, reactor power was adjusted and measurements of radiation flux at the main steam lines were acquired. A calibration curve was fit through (0, 0), that is, when the MSLRM detectors read zero (the reactor is at zero power), and the PIC detectors, which continue to read background levels, are set to zero response. This calibration removes background from the equation and allows VYNPS to report net exposure rate at the site boundary.

This background subtraction method is appropriate, though there are some minor limitations:

1. The PIC detectors placed at DR-53 are not measuring "only" background when the reactor is at zero power. There is a small contribution from "other" sources. While the other sources may contribute only 1 μR/h, this overestimates true background and results in a smaller net response. VDH should look at this when the MSLRM calibration is performed again to account for the shield that was installed (May 2006).
2. The calibration of PIC response at site boundary vs. MSLRM reading assumes that skyshine is constant as function of power. There are environmentally-induced fluctuations that are not captured by the calibration model.

These limitations would not be discussed if the allowable levels of direct gamma exposure were greater. The background subtraction method is better than any other option available today, which relies on the identification of a surrogate background location (or locations) from which to estimate background concurrent with reactor operation.

7.2 VDH and VYNPS Background Subtraction

The State's environmental monitoring program employs TLDs placed at designated locations around the VYNPS. Background TLDs are located in Putney and Wilmington. As noted by ORAU in its review of the VDH TLD data provided primarily in its annual environmental reports and through discussion with VDH staff, the VDH employed a historical range of measurement results against which quarterly TLD results were compared. If the gross measurement results provided by the TLD vendor were within the historical annual range (0-30 mR), the state reported no issues were identified. Beginning in 2004, the State changed their methodology and began utilizing the Putney and Wilmington locations to determine a net reading. This led to VDH notification to the VYNPS in March 2005 that the fourth quarter of 2004 had exceeded the 10 mrem reporting limit, and that the overall annual dose for 2004 exceeded the State's dose objective of 20 mrem at the West Fenceline (as measured by the VDH Parking Lot TLD).

The VYNPS subsequently issued a condition report and initiated corrective actions including quarterly TLD reporting to the State, the identification of two environmental TLDs (because the State had two background locations) in close proximity to the west fenceline (DR-14 and DR-16), and implementation of background subtraction by the plant to independently assess net exposures due to its operations. VYNPS concluded its background subtraction methodology resulted in net readings below the State's 10 mR/quarter and 20 mR/year. These aforementioned activities are described in the VYNPS white paper "Development of Environmental TLD Background Locations to Aid in the Determination of Plant-Generated West Fenceline Dose Contribution".

Even with the actions taken by VYNPS (which continue as of the issuance of this report), resolution of this issue was not reached and led to ORAU's involvement as a third party to evaluate this compliance issue. As stated in this report, the state TLD program had no approved data reduction procedure or statistical methods to test the quality and reliability of the data. The state had no qualified program to make these determinations, and in fact come to the conclusion that the background readings in Putney and Wilmington were abnormally low (or erroneous), thereby creating a greater net exposure estimate. As stated in Chapter 6, qualified methods should be developed to manage the data to support these critical decisions.

7.3 Background Subtraction using Surrogate Locations

The average background radiation level is determined as the average of two surrogate "control" locations. This average background reading is subtracted from gross readings at indicator TLD (vs. control TLD) locations to yield the net reading, which is assumed to be the above-background dose resulting from plant operations. The underlying assumption/necessity is that the background level at the control TLD locations is similar to the background level at the indicator TLD locations. "Similar" is the key word, and is an important qualifier for how much confidence to put into any result. Background levels will never be identical at any two locations, and also vary across the site by location and by time, as described in the spatial and temporal sections of "natural background."

As would be expected with a large number of monitoring locations, the results for some indicator locations are higher than the average background, yielding a positive net reading, while readings at other locations are lower than the average background, yielding a negative net reading. This phenomenon is a result of both the normal background fluctuations and the statistical variability of the TLD response. A slightly positive net reading is not necessarily indicative of an exposure above background rates, while a slightly negative net reading is not necessarily indicative of instrument failure. Based on the VDH TLD data, these normal fluctuations can be as large as several milliroentgen per quarter.

7.3.1 VDH and VYNPS Historical TLD Locations

Background flux (or exposure rate) was measured at Location DR-53 during the 2001 shutdown using the GE Reuter Stokes PIC while the plant was at zero power. Location DR-53 is a "conservative" location due to the fact it is not located at the actual site boundary or at the "closest resident" location. The same PIC was then used to measure natural background radiation around the plant in an effort to locate and describe a surrogate background area (or reference background location). Based on these measurements, Hinsdale, NH was found to yield similar results and is used by VYNPS as their choice for an appropriate background.

ORAU reviewed VDH background radiation data.   The VDH has selected background measurement locations in Putney and Wilmington, VT. The background readings at Putney appear to be similar to Hinsdale, NH (used by VYNPS) which are similar to measurements made at the plant site boundary during zero power. Readings at Wilmington vary more than the other locations and are greater than Putney. Over time (a calendar quarter, for example), the Wilmington background measurements cause a negative bias in the net results. This variability in the datasets needs to be understood before a suitable measurement program can be implemented that will provide reliable results in making the determination of dose at the site boundary. These biases and variability differences should be well understood and normalized accordingly prior to use in decision making.

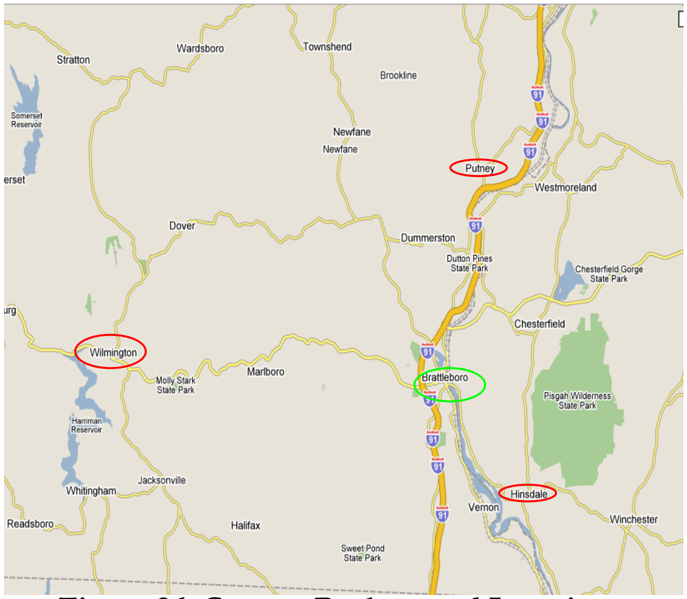

Figure 26 Current Background Locations

7.3.2 Considerations for Selecting Surrogate Locations

In order to make representative TLD measurements of background at locations surrounding VYNPS, the following should be considered:

1. Is the underlying topology for the surrogate location similar to the topology at the fenceline monitoring location? In particular, if there is a large berm at the fenceline, then this effect should be considered for the background measurement. The background location should have the same topology.
2. Is the underlying geology and concentration of primordial radionuclides in the soil the same for each location?
3. Is the measurement height from the ground the same at both locations? Are both in environmentally similar conditions?
4. Are the locations free of nearby buildings or structures that could contain natural radioactivity (^{40}K in building materials)?
5. Are proper control badges placed to account for measurement periods before and after the TLD is "measuring over the quarter of interest?"
6. Are the surrogate background locations free of radioactive traffic or radiation producing machines (no cancer treatment patients, no radioactive material shipments, radiography, x-ray machines)?

7.4 Use of Pre-Operational Data

Per NRC Regulatory Guide 4.1, pre-operational data is recommended to establish background radiation exposure rates (and radionuclide concentrations in soil) prior to plant start-up. In the case of the VYNPS, historical data collected during the 1969-1972 timeframe resulted in average background radiation rates significantly higher than expected and consequently, not ideal for background subtraction. ORAU determined that this was the case based on its review of available data and discussions with both VYNPS and the former Chief of the VDH directly involved in the design of the State's pre-operational program and evaluation of the data subsequently collected. It appears clear that background data from this time period is not adequate to be used alone, however, by looking at the Van Pelt report discussed in Chapter 6 as well as the later report by Brinck in the Health Physics Journal, the measurement results can be used to provide information on background variability. This may help establish the framework for the statistical tests that are required for the decision making process on whether a collection of measurements demonstrated that a noncompliance occurred.

8.0 Calibrating Instruments for Dose Equivalent

Prescribing solutions for calibrating equipment to measure and report in dose equivalent was not specifically in ORAU's scope of work. Accordingly, this chapter is relatively brief. However, it is an important topic and ORAU does provide information on accomplishing this objective. Any method employed to verify the dose equivalent rate at the VYNPS site boundary requires a quality program and specification of the methods used to achieve such a program.

This issue is important for one reason: measurements historically conducted at the site boundary utilized instrumentation (PICS and TLDs) calibrated for and measuring exposure rates. Mathematical conversions from the literature or from *in situ* high resolution gamma-ray spectrometry measurements were developed to modify the exposure readings to dose equivalent. An approach to measuring dose equivalent directly avoids the "secondary mathematical unit conversions." Up to this time, however, no instruments have been deployed for measuring dose equivalent directly. It can be done and requires much more attention to the details of the calibration protocol and of course, the selection of instruments that are "tissue equivalent."

8.1 Fundamental Quantities in Radiation Protection

There are numerous fundamental quantities and units utilized in the field of health physics/radiation protection to characterize an external direct radiation field and the resulting dose. The introduction of and precise definitions for these quantities arises principally from reports issued by the International Commission on Radiation Units and Measurements (ICRU) and the ICRP. Often cited and commonly encountered examples include the quantities "exposure", "absorbed dose", and "dose equivalent". Using conventional (English) units, the corresponding units for these quantities are the "roentgen", "rad" and "rem", respectively. (In the SI system, the corresponding units are the "coulomb/kilogram", "gray", and "sievert", respectively.) The English units appear in the VDH radiation regulations and have been described in several other chapters of this report; the roentgen and the rem, i.e., the conversion of one to the other, is of particular interest in this chapter because the state has simplified the regulation by assuming a one to one conversion, which, in the absolute sense of fundamental radiation protection quantities, is incorrect.

Other more recent quantities described in ICRU and ICRP reports include the "ambient dose equivalent" and "directional dose equivalent". According to G. Klemic (U.S. Environmental Measurements Laboratory), these operational quantities are recommended as reasonable approximations for dose equivalent in environmental monitoring [V]. She makes the initial point that use of these operational quantities would avoid complications encountered due to advancements in our knowledge of radiation biology; at the same time, however, the acknowledgment is made that measurements made to verify compliance with the dose limits have to relate to the quantities specified in the regulations.

8.2 Practical Implementation

Practical implementation of these quantities in operational or environmental applications requires knowledge of factors that influence the choice of instrument or measuring device, the appropriate readout units and the accuracy of the measurement results. Some of these factors include the radiation type, incident radiation energy, energy response, angular dependence, whether a rate or integrated reading is required, etc. For example, where energy response is a concern, the instrument/measuring device either needs to have a flat (energy independent) response or a conversion factor needs to be applied. The instrument of choice requires an appropriate readout unit, such as exposure rate (e.g., mR/hr, μR/hr) used by PICs or dose equivalent (mrem/hr), as in the case of tissue equivalent proportional counters (TEPCs) described in Chapter 5. In the case of passive dosimeters such as TLDs, an integrated reading (for example, over a calendar quarter) in mrem is an anticipated result, but it can only be achieved through calibration protocols that are currently not employed for environmental dosimeters (but are for personnel dosimeters).

8.3 Calibrating the PIC for Dose Equivalent

Pressurized ionization chambers are routinely used to record environmental *exposure rates* in contrast to dose equivalent units. Calibration of these devices, such as the GE Reuter Stokes PIC typically involves the use of a NIST traceable calibration source in a dedicated room. A "shadow shield" technique is employed to account for background and room scatter with the objective of verifying the instrument's measured exposure rate response is within an acceptable statistical error of the calculated ("true") exposure rate response. Note that the calibration does not involve an exposure rate to dose equivalent rate conversion. In typical environmental applications where exposure rates are measured, an exposure to dose equivalent conversion would not be required. ORAU is unaware of a situation where this conversion in the environs of a nuclear power station was required for this particular instrumentation.

Some efforts have been made in PIC design to calibrate for the purpose of a "tissue equivalent" response, but ORAU observes these developments to be more experimental and developmental than practical. Exposure rate measurements using PICs could be adjusted by a cross-reference to concurrent measurements made with a TEPC counter. For example, when the PIC exposure rate reads 1 μR/h, the TEPC may read 0.8 mrad/h, providing a direct correlation "in the field." There are no direct calibration methods known for calibrating PIC detectors to dose equivalent, and as a result, any group wishing to convert units is relegated to a mathematical adjustment.

8.4 Calibrating a Passive Dosimeter for Dose Equivalent

As was the case for Section 8.3, routine use of environmental TLDs does not involve calibration for dose equivalent units. (For the vast majority of nuclear plants required to meet only the 100 mrem public dose limit, a true conversion into these units is not needed, that is, one roentgen equals one rem. Meeting the State's lower dose objective, which is cited in dose equivalent units, is another matter.) Currently, based on standard practice, dosimetry suppliers for both the VDH and VYNPS provide their results in exposure units (milliroentgen) or exposure rate (microroentgens per hour), respectively. While both vendors are NVLAP accredited personnel dosimetry providers, a similar accreditation for environmental dosimeters dose not currently exist.

ORAU notes that the use of different reporting units complicates the situation, making it more difficult for the VDH to compare VYNPS results against the site boundary quarterly and annual dose objectives. Consequently, ORAU recommends that the VYNPS utilize the more conventional "mR" unit.

Klemic notes that TLD materials exist that are "nearly" tissue equivalent. These include lithium fluoride (LiF), lithium borate ($Li_2B_4O_7$) and beryllium oxide (BeO). Though not a dose equivalent application, appropriate filters can be added, resulting in basic spectrometric uses for the detection of high energy ^{16}N gamma rays from BWRs such as VYNPS. Calibrating a passive TLD or OSL dosimeter for dose equivalent is an effort that requires attention to a lot of detail discussed in ANSI standards under N13:

- Angular response
- Energy response
- Tissue equivalency of the element and the filtering system (H*(10) dose for example)
- Algorithms for interpretation of the glow curve
- Backscatter from a phantom (30x30x15cm methylmacryolate slab)
- Environmental impacts (temperature, moisture, soil)

There are draft ANSI guidance documents and several research and development papers available for providing supplemental information for these calibrations, but ORAU does not recommend "pushing the state of the art" when no other operating facility is asked to study these details at the practioner's level.

8.5 Methods for Converting Exposure to Dose Equivalent

Acceptable methods are required for calibrating instruments to report results in dose equivalent (versus exposure). The instrument can either be calibrated for dose equivalent as described above or the output can be converted from one measurement unit to another, as described in detail in Appendix B.

8.6 Flux to Dose Equivalent Measurement Results at VYNPS

Appendix B provides an in depth examination of the energy dependent flux to dose equivalent evaluation conducted by ORAU. Based on the date provided to ORAU, which involved gamma ray spectrometry measurements conducted by a VYNPS contractor, a conversion factor of 1 roentgen equivalent to 0.6 rem was determined to be scientifically valid for the measured isotropic, energy dependent gamma-ray fluence rate at the site fenceline.

9.0 Measurement Uncertainty and Impact on Regulatory Compliance

This chapter on measurement uncertainty provides suggestions to deal with uncertainty when comparing two measured values of the same quantity and their methodologies. Implementation of measurement uncertainty in regulatory compliance is an ongoing issue, for which the solution in practice resides in an understanding between the regulator and the licensee. From ORAU's prior experiences, Appendix E provides examples of how measurement uncertainty was applied for the successful national programs in the nuclear sciences.

One of the essential issues faced in this task is this: The regulator specifies maximum allowable limits for radiation dose to the public. In Vermont, VYNPS must satisfy one set of federal standards and a different set of standards at the state level. Measured values of gross concentration or gross radiation flux are reported and compared to natural background so that a net value can then be compared against the corresponding limit. Measured values, both gross and background, are reported as a mean with corresponding standard deviations (or error term comprised of both random and systematic error). When the objective for the measurement device is to make a differential measurement between gross and background that is small relative to the physical detection limit, then the random and systematic error terms must be understood and accounted for. This effort of understanding and accounting for the various measurement features is a primary reason national standards are prepared.

For the direct gamma-radiation dose requirement for VYNPS, it is a known fact that the "margin for error" is small. From Chapter 6, it can be concluded that the west-side site boundary exposure rate (at DR-53 and DR-52) on an annualized basis is probably on the order of 20 mR, which when corrected for dose equivalent is 14 mrem (using the VYNPS-proposed conversion factor of 0.71). With the annual dose objective of 20 mrem at the site boundary, the measurement devices in total must be able to account for a small differential of 6 mrem, in the presence of a fluctuating natural background rate twice the margin of compliance.

ORAU recommends the following to assist in the development of methods for measuring and determining for compliance:

- When the MSLRM monitors are re-calibrated and new mathematical functions are fit to MSLRM output versus PIC output at DR-53, error functions for the fit should be developed.
- Propagate the error terms for the contribution of "other" sources described in the ODCM and described in this report (Table 2 of Chapter 2).
- Conduct an error analysis on the TLD results. In order to remove the bias terms currently observed between the VYNPS vendor and the VDH vendor, use the same vendor and ensure the vendor has no knowledge of the indexing scheme to identify which TLDs were collocated, i.e., ensure independent measurements are conducted between VYNPS and VDH.

The scientific error analysis should examine:

  - Agreement between collocated TLDs. Attention should be paid to TLD positioning, environmental enclosures, height above ground, local topology, snow cover, etc.
  - Correlations with PIC measurements conducted at the specified locations.
  - Element algorithms, averaging, and outlier rejection.

- Combine all error terms in an overall uncertainty management program that examines the total measurement uncertainty in the various methods (a monte carlo analysis or deterministic uncertainty analysis can be performed).
- The result of the uncertainty analysis is that the state can be assured that the measurement process, which includes data reduction and reporting, is effective in providing realistic decision criteria. That is, there is not a false sense of compliance or a false sense of noncompliance.

Once a better understanding on the total uncertainty in the decision-making process has been identified, the decision-making mathematical equations should be developed and peer reviewed. A graded approach must be used in the decision-making approach, utilizing the sum of all information known about the radiation levels at the site boundary. This is the only way the public can be assured, with confidence, that the limit was (or is) not exceeded. Test plans should be prepared and the data examined. In the ORAU statement of work, a question was asked about the use of two different background results reported on page 15 of the DES *in situ* report, 68 mR versus 80 mR per year. This is a very important question posed by VDH, but until the effort is made to understand in more detail the calibration methods, conduct confirmation measurements, and conduct an uncertainty analysis, an assessment cannot be made at this time. This is the type of assessment that is referred to in this chapter.

10.0 Historical Review of VYNPS Compliance with the Direct Radiation Dose Limit

Chapter 10 was initially thought to be an exhaustive look at records analysis of dosimetry results. ORAU did review the data, creating spreadsheets and examining statistical trends. Our evaluation concluded quickly that the measurement results were small (and below regulatory limits). Because the direct gamma radiation rates are generally predictable (at least 90% of the component arises from ^{16}N), then the only "sudden increase" can be explained by one of two issues:

1) Some type of radiation generating operation took place on the site that increased the gross TLD measurement result (a real event undetected by the MSLRM approach); or

2) The background measurements were low, thereby inflating the net result.

ORAU concludes in the case of the TLD measurements for the 2004 fourth quarter that no conclusive evidence exists that the net result was as high as originally reported. This is because the background radiation rates were statistically lower than previous values reported for both the Putney and Wilmington locations.

Upon the extensive review of the methods and results presented in Chapter 6, ORAU concludes that there is no conclusive evidence to suggest that the actual site boundary dose equivalent annual objective has ever been exceeded at the 20 mrem level above background. The annual exposure rate at the site boundary has likely varied within the range of 22 to 28 mR. By converting these values by the conversion factors 0.71 rem/R or 0.6 rem/R, the annual objective is shown to have been met.

Chapter 6 describes the significant amount of attention that has been focused on the measurement and evaluation of site boundary dose. The measurement values, from 1972 to today, vary more ORAU believes because of variability in natural background, than variability in the process that actually generated the site boundary dose. This explains why several scientists measured and estimated net results from 7 mR per year to 12, 16, and up to 26 mR. ORAU focused a significant amount of attention during the onsite meetings with the VDH and VYNPS on analysis of data, and have now concluded that the variability in background and the magnitude of changes to background are simply far too great in order to capture the absolute "true result". All that can and should be said is that the exposures to date are low and below the regulatory objectives. ORAU also emphasizes that the margin for error in the past is larger than the margin for error going forward due to the EPU, when about 26% more ^{16}N will be produced in the reactor at 120% power.

One mathematical practice that appears to be flawed is that the background distribution is under-sampled. More TLDs need to be distributed for determining the background. This conclusion and the determination of how many TLDs to distribute will be known once the total measurement uncertainty analysis is completed, as described in Chapter 9. Conducting statistical t-tests for inter-comparisons of data sets has been used by VYNPS in the past. For example, the Strum REG 98-132 memo concludes that the hourly rate of 8.13 μR/h for a 1-month TLD irradiation (at full power) is statistically different from 7.54 μR/h for a 3-month TLD irradiation (2 months at zero power and 1 month at full power). These statistical methods (namely the t-test) need to be peer reviewed and proceduralized for future inter-comparisons. Most importantly, more data needs to be collected in order to reduce variability; up to this point, decision making has been occurring with no statistical rigor and with a variability that is far to large when the mean of the gross distribution is near the mean of the background distribution. It is because of this historical feature in the data sets that ORAU cannot conclude VYNPS exceeded the VDH dose objectives.

11.0 Recommendations

The executive summary of this report provides overall recommendations for improving communications between the VDH and VYNPS, for selecting the measurement methods to be used for determining whether plant gamma-ray emissions exceed allowable limits, and for suggested improvements to the regulatory framework within the State of Vermont Department of Health.

This chapter augments the executive summary and extracts from each of the chapters suggestions for improving the regulatory framework for determining compliance with the site boundary dose limit.

General

A single environmental TLD measurement should not be used as an accurate measure of compliance against a net limit 10 mrem per quarter. Multiple TLDs should be deployed. Statistical process control methods need to be implemented into an overall environmental assessment plan by VDH. The methods need to be described in procedures.

The state should assist in the recalibration of the MSLRM method in 2007. The state should observe the measurements and the procedure, inspect calibration records, and verify the fitted detector response of MSLRM response versus HPIC detector response at DR-53.

> Section 6.1.2: One of the "lessons learned" for this project is that going forward the state should be involved with the measurements that are conducted, review data as it is generated, and have the opportunity to verify calibration schemes and results. For example, the VDH should observe the calibration of the PIC detectors and the MSLRM monitors during the next refueling outage, and confirm at each power level during startup the VYNPS reported PIC reading. Duplicate measurements by a state PIC (if available) may also be warranted.

The State should review and approve the procedural improvement to identify "other gamma-ray sources." A logbook should be available for inspection by the state that "other" sources are appropriately identified and cataloged.

Consistency in terminology is needed. For example, an established, regulated document should be used to define the site boundary such as the definition of "site boundary" as defined in the VYNPS Final Safety Analysis Report (FSAR).

Section 6.1.2.3: Minor Calculation Change (MCC) form ENN-DC-126 R5, which became Appendix E of VYC-2067, should be reviewed.

Section 6.3: With the May 2006 installation of a turbine building shield, which occurred following the ORAU flux to dose evaluation (see Appendix B), the energy-dependent photon spectrum should be re-assessed during the plant's next outage.

Enhanced Communication Plan (ECP)

This is a key ORAU recommendation. The need exists for both parties to establish a documented framework in the form of an ECP, with sufficient detail to ensure that regulatory compliance is achieved. Examples of where the ECP would be useful include:

● Determining an appropriate shielding reduction factor, as warranted, and acceptable to the VDH.

● VDH review of VYNPS calibration data prior to approval and the issue of continuing PIC measurements. This has been an issue because VYNPS has stated that once the PATP measurements are completed and a calibration of the PIC versus MSLRMs occurs, VYNPS intends to discontinue the use of the PIC detectors. The VDH has stated their interest in seeing the PIC measurements continue for an indefinite period. ORAU believes that if the state is provided ample on-site inspection of the measurements, and shown that the model equations can predict site boundary dose from 16N, then ongoing PIC measurements may be discontinued on the basis of cost-to-risk-to-benefit ratios.

● Regarding the "flux to dose" issue, ORAU made an extensive evaluation of high-resolution gamma-ray spectrometry measurements conducted by VYNPS in 2002 at the site boundary. ORAU recommends that VDH and VYNPS incorporate into the ECP either the ORAU recommended dose equivalent conversion (adjustment) factor stated in Appendix B (1 R = 0.6 rem) or an acceptable alternative that should be applied at the site boundary. The measuring equipment could either be "calibrated" for dose equivalent or an administrative factor applied. This issue has a specific impact on amending Part 5, Chapter 3, Subchapter 1, Section 5-303 Definition (K) of the VDH regulations.

●Utilizing a "defense in depth" approach using real-time monitoring to evaluate the cumulative dose per year, and then use a set of passive dosimeter measurements only to reject the real-time monitoring results as not having measured all gamma-radiation emitted from the site.

●From ORAU's perspective, neither the VDH or VYNPS has ever really known or agreed to the terms of the compliance requirements, including what methods were/are acceptable, what set of measurements (or methods) would be used to demonstrate compliance, and importantly the process for reporting data/information (including time lags for TLD post processing), how real-time information (MSLRM) can be used to augment passive/delayed results, and the process for rejecting outlier data

●Re-evaluating the ECP within a year or two, determining what is working and not working, making adjustments accordingly, and eventually migrating the over-arching elements of the ECP into a new regulation that is more consistent with the federal requirements.

Main Steam Line Radiation Monitor (MSLRM)

Executive Summary: The MSLRM approach should be used as the primary method for estimating direct gamma dose from ^{16}N, but additional calibration details and quality assurance and control logs should be reviewed before full approval by the VDH. ORAU recognizes that the VDH has stated that it requires its own compliance program and does not possess an MSLRM-equivalent methodology. Therefore, in association with the MSLRM, a passive TLD measurement program remains necessary for both parties. However, the TLD program should be refined to not only estimate all other direct gamma dose contributors—small contributions in most cases—but also to detect source terms not otherwise identified by the MSLRM.

Section 6.2: The MSLRM approach is a reasonable real-time monitoring scheme for estimating the site boundary dose at Location DR-53. However, the approach for estimating other (less controlled) means of direct exposure is less clear. Most importantly, it is difficult to control all the possible sources of direct gamma radiation. For every single operation on-site that involves the use of material that emits gamma rays, or radiation generating machines, a dose assessment review would need to be conducted to determine the level of contribution to the net boundary dose that particular operation would produce—ORAU believes this type of procedural practice should be avoided. Minor source contributions to the site boundary dose should be added and subtracted as they arise, including on-site storage operations and handling of radioactive materials.

Thermoluminescent Dosimeters (TLDs)

Section 1.6: For the near term, the application of appropriate measurement uncertainties should continue for both the annual and especially for the quarterly monitoring results, simply because these passive dosimeters were not intended to be accurate at low dose levels. Long term, compliance with the 20 mrem dose objective must be established in another manner.

Section 3.7.1: The VDH has deployed background TLDs inside the Putney Town Clerk's office and within an air sampling station at a second background location in Wilmington. ORAU recommends these TLDs be re-deployed to outside locations—in the absence of a technical or other (e.g., vandalism issue) basis—in order to generate what ORAU believes would be more realistic environmental measurements.

It is important to design into the measurement process a suitable number of passive dosimeters (at the site boundary and at the background locations) and to ensure that statistical outlier methods are used to provide a means to reject erroneous data.

<u>VDH Improvements to Existing Dosimetry Program</u>

The following recommendations, based on current elements of the draft ANSI N13.37 and N13.29 standards on environmental monitoring/dosimetry and testing, are provided as a potential model for environmental monitoring and dosimetry. (ORAU recognizes that guidance in implementing specific aspects of an environmental monitoring program is lacking and many variables exist in this area).

- Develop terms and conditions for demonstrating compliance. This includes notification schedule, response times, and follow-up activities when data suggests that a dose objective has been exceeded. All relevant terms must be defined (site boundary, fenceline, exposure, dose, dose equivalent, nearest resident).

- Develop and implement a robust/reliable independent VDH measurement program that satisfies performance objectives. These include:

    - Extending its independent passive dosimetry program to meet acceptable precision and accuracy specifications, with allowance for rejection of outliers and appropriate statistical tests.

    - Determining how to handle measurement uncertainty against a regulatory limit.

    - Developing a consistent approach to performing background subtraction.

    - Calibrating directly (or properly utilizing the *in situ*, spectrum weighted average) for reporting of <u>dose equivalent</u> (flux to dose conversion) in mrem/quarter and mrem/year.

12.0 References-Programmatic

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[B] The State of Vermont, Regulations for Radiation Protection, Part 5, Chapter 3, Subchapter 1, section 5-305(B)(1)(e), Direct Gamma Radiation, page 9. (Available from http://healthvermont.gov/regs/radio_health.pdf)

[C] The State of Vermont, Regulations for Radiation Protection, Part 5, Chapter 3, Subchapter 1, section 5-303(K), Definitions (rem), pages 2 and 3.

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[F] VYNPS letter to VDH (Jay Thayer to Paul Jarris), "Measurement of Fenceline Radiation Doses-Revision to March 30, 2005 Letter," **March 31, 2005**.

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[U] Froines J. (VDH) to R. Snelling (Governor of VT), "State Involvement in Nuclear Power Issues and submittal of a report from David Scott, **1978.**

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[2] Sáez-Vergara J.C**.** *Recent Developments of Passive and Active Detectors used in the Monitoring of External Environmental Radiation*, Rad. Prot. Dosim. 92:83-88 **(2000)**

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Note: See Appendix B for additional technical references used in assessing flux to dose conversion factors.

14.0 Definitions

ALARA	ALARA means "As Low As is Reasonably Achievable," which is the approach to radiation protection to manage and control exposures to the work force and to the general public to as low as is reasonable, taking into account social, technical, economic, practical, and public policy considerations. ALARA is not a dose limit, but is a process which has the objective of attaining doses as far below the applicable limits as is reasonably achievable.
ANSI	The "American National Standards Institute". A non-profit, privately funded organization that coordinates the development of voluntary national standards. The Institute, supported by over 1,000 companies, 250 technical, trade, labor and consumer organizations, and some 30 government agencies, encourages accredited organizations to develop standards and approves those developed under a consensus process as American National Standards.
Air submersion dose	See "submersion dose"
Boiling Water Reactor (BWR)	A type of light-water nuclear reactor developed by the General Electric Company in the mid 1950s. Water is used to conduct heat away from the nuclear fuel and to "thermalize" neutrons, i.e., reduce their kinetic energy, which is necessary to improve the probability of fission. Steam from the heated water produced in the reactor core goes to a turbine and from there to a generator to produce electricity. Electrical power is then transmitted via a power grid to the end user.
Cosmic Radiation	High energy radiation produced primarily by the sun and secondarily from deep space (the cosmos and the galaxy). Cosmic radiation interacts with the atmosphere of the earth and the earth itself to produce a measurable amount of ionizing radiation that exposes all living things. The level of cosmic-produced, ambient ionizing radiation increases with altitude up to about 80,000 feet. At sea level, the average annual cosmic radiation background produces a dose equivalent rate of between 30 and 40 mrem per year. Cosmic radiation, along with radiation from primordial radionuclides, is a contributor to natural background radiation. The intensity of cosmic radiation is highly variable, depending on solar flare activity, sun spot activity, local atmospheric conditions, and the local composition of the earth (collision density and reflectivity). Cosmic radiation intensity is thus time dependent and spatially dependent. That is, the fluence rate of cosmic radiation varies with time and location. Refer to UNSCEAR 2000 and NCRP 93 for further

discussion.[1,2]

Direct Gamma Radiation

This terminology originates in the State of Vermont regulations [Department of Health, Part 5, Chapter 3, Subchapter 1, §5-305(B)(1)(e)] establishing an acceptable radiation dose limit at the VYNPS site boundary. Direct gamma radiation is the total flux of gamma radiation produced on the site of VYNPS. The primary source of direct gamma radiation is from the turbine building. Secondary sources are from any daily operations on the site, including noble gas generation, radiography, spent-fuel shipments/handling, and on-site storage/ handling/ transport/disposition of radioactive materials. Direct gamma radiation is not contamination. It is ionizing radiation that is "broadcast" like a radio wave from the site. The direct radiation level "dies off" quickly with distance from the source, as it is absorbed in the atmosphere and ground. The regulatory compliance point for the level of direct gamma radiation applies at the site boundary.

Dose Equivalent

A unit of radiation dose that accounts for the difference in biological risk due to the type of radiation that delivered the dose. Dose equivalent is normalized such that a given dose equivalent has the same biological risk, whether delivered by alpha, beta or gamma radiation.

The unit for dose equivalent is rem. It is common practice to express dose equivalent in $1/1000^{th}$ of a rem, or millirem (abbreviated mrem).

Electronic Data Loggers

An automated system for recording and storing data in electronic form. Electronic data loggers are commonly used to record data from radiation detection systems in real-time for later analysis.

Energy-dependent Flux

The number of photons of a specified energy striking an area of interest (cm^2) per unit time (sec). Energy dependent flux is reported in units of photon cm^{-2} sec^{-1}.

See also "flux energy-dependent flux".

Energy Resolution

A measure of the ability of a radiation detector to correctly identify the precise energy of the radiation being measured. High purity germanium detectors, for example, have the capability to

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[1] United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR) 2000 Report Vol. I, Sources and Effects of Ionizing Radiation, New York, UN (2000) (available from: http://www.unscear.org/unscear/en/publications/2000_1.html)

[2] US National Council on Radiation Protection and Measurements, NCRP Report No. 93, *Ionizing Radiation Exposure of the Population of the United States,* Bethesda, Maryland, USA. (1987)

resolve gamma-ray energies within 1-2 keV of each other. Conversely, sodium iodide detectors, can only "resolve" gamma energies separated by several keV.

A common standard used to describe the energy resolution of a detector is the full-width at half-maximum (FWHM). The FWHM describes the degree to which the measured energy of the radiation varies from the actual energy of the radiation.

Enhanced Communication Plan (ECP) A formalized plan established between the regulator (VDH) and the licensee (VYNPS) to enhance communication between the principal parties to reach defined objectives. The plan should define roles, responsibilities, schedule, milestones, etc. and be updated on an established frequency.

Exposure A gamma-ray radiation measurement quantity describing the number of ion pairs created per unit volume of air. Defined in terms of electrical charge, exposure is normally the quantity directly measured by some types of radiation detectors (e.g. ion chambers). Exposure, defined only for air, can be correlated to dose in human tissue through known conversion factors. Thermoluminescent dosimeters are one example of an exposure detector.

See also, "roentgen" and "Thermoluminescent Dosimeter"

Fenceline The fenceline at VYNPS is a security fence established for purposes of keeping intruders from entering the plant. Other security systems augment the appearance of a barbed-wire fence. The fenceline may or may not be equivalent to the site boundary. In most cases, the site boundary (property owned and operated by VYNPS) is extended far out from the fenceline.

Flux (or Fluence Rate) The number of photons (over all energies) striking an area of interest (cm^2) per unit time (sec). (Unit: photon cm^{-2} sec^{-1}).

See also "energy-dependent flux".

Flux to Dose Conversion The mathematical process of converting an energy-dependent flux measurement (or calculation) to a dosimetric quantity. A dosimetric quantity, for example, is dose equivalent. Dose equivalent is normally the quantity of interest for regulatory compliance purposes and is the quantity of interest within the State of Vermont regulation, Part 5, Chapter 3, Subchapter 1, Section 5-303.

Gamma Radiation One of several examples of ionizing radiation. Gamma radiation originates from the nucleus of an unstable atom, can travel potentially infinite distances in air, and can penetrate the human body. It is the ionizing radiation of most interest in the ORAU evaluation of site boundary dose at VYNPS.

Geiger-Mueller Detector

The Geiger-Mueller detector is one form of a class of radiation detectors called *gaseous detectors* or simply gas detectors. A Geiger-Mueller detector has several advantages, including its common use in personnel and equipment surveys and purchasing cost. Prime disadvantages include its inability to identity the radionuclide or the energy of the radiation being measured.

Geiger-Mueller detectors are primarily used to detect particle radiations (alpha and beta). Geiger counters can also be used to detect gamma radiation, although this process is inefficient due to the fact that the density of the gas in the device is low, allowing most gamma photons to pass through undetected.

Geiger-Mueller detectors consist of an inert gas-filled tube configured in different shapes that briefly conducts electricity when a particle or photon of radiation temporarily makes the gas conductive. The instrument amplifies this conduction which is then translated into a "count" (a general unit of radioactivity) and often an audible "click" one readout device.

Half-Life

The amount of time required for the radioactivity of a readionuclide to decay to one-half of its initial amount. For example, nitrogen-16 has a half-life of 7 seconds. Every seven seconds, the amount of radioactivity is reduces by half. After 14 seconds, to one-fourth, and so on. As a "rule of thumb", after about 7 to10 half-lives have passed, essentially no radioactivity remains.

HPGe Gamma-ray Spectrometer

A solid state radiation detector used to measure discrete-energy photons. The spectrometer is capable of identifying specific radionuclides or source terms. The HPGe (high purity germanium) gamma spectrometer is the best available instrument for measuring energy-dependent flux.

HPIC

High Pressure Ion Chamber. See "Ion chamber" for definition.

Hydrogen Water Chemistry

In boiling water reactors (BWRs), the modification of coolant chemistry by feedwater hydrogen addition. It is considered a viable option to mitigate the inter-granular stress corrosion cracking problems common to operating BWRs.

ICRP

The "International Commission on Radiological Protection". An independent, international group of experts from a wide range of scientific disciplines which have published recommendations for the protection of radiation workers and the public from ionizing radiation for more than 50 years. Their recommendations are widely used as the basis for the formulation of national and international radiological protection standards and regulations. For example, the radiological protection regulations in the United States are based in part on the recommendations of the ICRP.

***In situ* Measurement** A measurement of radioactive material conducted "in place." An *in situ* measurement is the opposite of collecting a sample of the material and analyzing it for radioactivity in a laboratory. *In situ* measurements of energy-dependent gamma-ray flux have been conducted at the fenceline of VYNPS using an HPGe gamma-ray spectrometer.

Ion chamber A radiation detector that measures exposure rate (roentgen per hour). Ion chambers may be grouped into two types: 1) hand-held, normally high range detectors; and 2) long-term environmental monitoring at very low exposure rates (low-range). Hand-held ion chambers are used for radiation protection purposes when the exposure rates encountered range nominally from several tenths of a milliroentgen per hour (mR/hr) to 100 roentgen per hour (R/hr) and higher. In contrast, environmental ion chambers measure ambient exposure rate in the microroentgen per hour (μR/h) range. Nominal exposure rates on the east coast average approximately 10 μR/h. The low sensitivity of the detector is achieved by filling the sensitive volume with a high pressure fill gas, and hence, is referred to as a pressurized ion chamber or as an HPIC (High Pressure Ion Chamber). It is the HPIC that is specifically of interest in this task.

Main Steam Line Radiation Monitor (MSLRM) A system comprised of four, high-range ion chambers placed adjacent to the main steam line; and the associated acquisition electronics. The output from the detectors is recorded in real time by instrumentation in the VYNPS control room. The MSLRM data is interpreted by fitting a calibration function of the MSLRM response to measured exposure rates at the fenceline measurement location, DR-53, during various power levels. By measuring in real time the radiation level at the main steam line, the exposure rate at the site boundary is inferred. The real time measurement data is recorded over one year to demonstrate site boundary dose limits are not exceeded.

Minimum Quantifiable Dose (MQD) The minimum delivered dose at which the measurement process results in a dose with a specified relative standard deviation (often specified at 10%). The MQD is a performance measure of interest when the goal of a measurement is comparison with another value, rather than detection, such as comparison to a radiation dose limit or to a natural background radiation level.

Multiple Dosimetry	The practice of using more than one dosimeter on an individual's body to obtain information regarding radiation exposure (ANSI N13.41-1997). This concept, while addressed for personnel dosimetry, is of interest in environmental dosimetry because it addresses the fact that radiation flux and dose rates can vary significantly with location, position above the terrain, and other operational issues. Multiple dosimetry concepts give some expert judgment and evaluation latitude to the measurement problem.
NaI Gamma-ray Spectrometer	Similar to an HPGe spectrometer, but of significantly lower energy resolution. Algorithms exist for several instruments to convert the spectrometer output to exposure rate and dose equivalent rate.
Natural Background Radiation	The photon flux at the measurement (or compliance) point of interest that is produced by natural phenomena: cosmic radiation and primordial radionuclides in surface soils and geologic formations. Natural background radiation varies with time and location. The time variation is due to two processes: 1) cosmic radiation rates vary significantly with solar flares, sun spots, and deep space phenomena; and 2) the weather—a low pressure system will cause more radon gas to be released from soil thereby increasing local background conditions. The exposure rate from natural background radiation is typically about 10 uR/h, which is about 90 mR/yr.
Natural Background Reference Area	An area or multiple areas where the natural background radiation conditions are equal to the background conditions at the point of interest, e.g. the site boundary. The reference area should be free from all manmade sources of radiation.
Natural Background Subtraction	Any measurement of gamma-ray radiation flux at the site boundary is a gross measurement that is a sum of the radiation emitted from plant operations (^{16}N included) and natural background. Natural background subtraction is the mathematical process of subtracting the natural background rate, typically determined from the natural background reference area, from the gross rate to arrive at the net rate, for which regulatory compliance is evaluated.
^{16}N	A radionuclide produced in the coolant of a boiling water reactor (BWR) via the interaction of neutrons with ^{16}O. It has a very short half life. When the reactor is at zero power, ^{16}N is nonexistent. There is a proportionality between ^{16}N production and power level. During its rapid decay, ^{16}N emits a highly energetic gamma-ray that is detectable at the site boundary. It is the single largest contributor to exposure rate at the site boundary.

NCRP	The "National Council on Radiation Protection and Measurements". The NCRP is a U.S. organization, charted by the U.S. Congress in 1965, which seeks to formulate and widely disseminate information, guidance and recommendations on radiation protection and measurements which represent the consensus of leading scientific thinking. NCRP recommendations are widely used as the basis for the formulation of radiological protection standards in the United States. The NCRP evaluates information provided by the International Commission on Radiological Protection, the Federal Radiation Council, the International Commission on Radiation Units and Measurements, and other national and international organizations, governmental and private, concerned with radiation quantities, units and measurements and with radiation protection.
Nearest residence	The nearest residence is defined as that person (or persons) whose primary residence is located nearest to the VYNPS fenceline or site boundary. This person (or group of persons) is considered to be the maximally exposed "member of the public" to direct, gamma-radiation from VYNPS.
Noble gas	During the fission process in a reactor, noble radioactive gases are created as fission products. Noble gases are chemically inert. Noble gases released from a reactor produce a slight increase in local gamma-ray radiation from direct immersion. Immersion in noble gases produces less than a few mrem per year at the locations adjacent to VYNPS.
Noble Metal Chemistry	Noble metal chemistry refers to the branch of inorganic chemistry that focuses on reactions involving metallic elements that are resistant to corrosion or oxidation, unlike most base metals. Examples include gold, silver, tantalum, platinum, and palladium.
Normal Water Chemistry	Water chemistry is the field of chemistry relating specifically to water and aqueous solutions. A specific focus of water chemistry is reactivity of water towards alkali metals; alkaline earth metals; halogens; hydrides; methane; oxides; and oxygen ions.
Occupancy Factor	A factor that describes the number of hours per year that an individual (or critical group) is exposed to ionizing radiation at the exposure rate and point of interest. Occupancy factors provide a more realistic estimate of actual dose received because the exposure rate of interest is multiplied by the number of hours exposed and not the cumulative time assuming a

	continuous exposure. The use of occupancy factors is addressed in publications such as NCRP Report 49, "Structural Shielding Design and Evaluation for Medical use of X Rays and Gamma Rays of Energies Up to 10 MeV".
Optically stimulated luminescence (OSL) dosimeter	Like a TLD, an OSL dosimeter is a passive measurement instrument for measuring the cumulative exposure from gamma-rays. OSL is a relatively new technology that is replacing TLDs for monitoring dose to radiation workers, but not necessarily for environmental monitoring. Like TLDs, OSLs are placed on the person or location of interest for 3 months to a year, and then returned to an analytical laboratory for reading. In contrast to ion chambers that have electronic data loggers, OSLs do not monitor in real time. OSL dosimeters are available only from Landauer under the trade name "Luxel", essentially replacing their film-badge dosimetry.
Pressurized Ion Chamber	See "Ion chamber".
Primordial Radionuclide	A radionuclide or chain of radionuclides that make up the earth's crust, soil, and geologic formations. Radiation from primordial radionuclides, along with cosmic radiation, contributes to natural background radiation. Primordial radionuclides include the radionuclides of potassium, thorium, and uranium (and the decay products).
Quality Factor	A factor introduced by ICRP-26 (1977) by which the absorbed dose (rad) is multiplied to account for biological damage. For gamma-rays, the quality factor is one.
Quarterly Average Dose	This is a compliance term unique to the state of Vermont regulation in Part 5, Chapter 3, Section 5-305(B)(1)(e) Direct Gamma Radiation. The intention of this term is to provide a numerical value for the total (time-integrated) direct gamma radiation dose equivalent for one calendar quarter (3 months). It is not, as the name implies, the annual dose divided by four, which would actually give a numerical value for the average dose in a calendar quarter.
rad	The dosimetric quantity/unit established to describe absorbed dose, that is, energy absorbed per unit mass.
Radiation (Ionizing Radiation)	Energy that is radiated or transmitted in the form of subatomic particles or electromagnetic waves. Ionizing radiation means radiation that has sufficient energy to produce ionization in substances through which it passes. Ionizing radiation results from a variety of sources, including the decay of unstable atoms and nuclear reactions such as in a nuclear power reactor.

Common types of ionizing radiation include particulate radiation, such as alpha particles, beta particles, and neutrons, as well as non-particulate electromagnetic radiation, such as x-rays and gamma rays. Discrete packets of electromagnetic radiation are referred to as photons.

In contrast, non-ionizing radiation refers to radiation that does not have sufficient energy to cause ionization. Examples of non-ionizing radiation include microwaves, infrared light, radio-frequency (RF), and lasers.

rem The dosimetric quantity/unit established to describe dose equivalent—the product of absorbed dose and the radiation quality factor.

roentgen (R) The traditional unit of radiation exposure. One roentgen is equivalent to 1 electro-static-unit (esu) per cm^3 of dry air at standard temperature and pressure, or 2.58E-4 coulombs per kilogram. The roentgen is abbreviated with a capital "R".

Reversed-Electrode Germanium (ReGe) Gamma Spectrometer A special type of HPGe detector in which the electrodes are reversed from a conventional coaxial detector. There are two advantages to this electrode arrangement: reduced window thickness and radiation damage resistance.

Site Boundary The site boundary is the area of land surrounding VYNPS that is contiguous, and owned and operated by Entergy under the existing NRC license. The site boundary is not equivalent to the fenceline. There has been a great deal of confusion and misunderstanding regarding the distinction between the site boundary and the fenceline. VYNPS has been purchasing property along Governor Hunt road while maintaining the existing location of the fenceline at DR-53 and DR-52. Under existing State of Vermont regulation, direct gamma radiation dose equivalent limits are established "above background radiation" and "at any site boundary, bordered by land."

Skyshine Term used to describe the radiation transport process of gamma-rays scattering off the atmosphere back to earth. The degree of scatter back to the earth is dependent on atmospheric conditions and thus, varies with time. Skyshine is a term used to describe both neutron and gamma-ray interactions, but for the situation at VYNPS, it is only the gamma-ray component that is of interest; no neutron component exists.

Source Term The types and amounts of radioactive materials or radiation contributing to the quantity of interest.

Submersion Dose	An external radiation dose resulting from being surrounded or covered (submersed) in air containing radioactive material.
Terrestrial Radiation	The radiation, or gamma-ray flux above the surface of the earth, produced from the decay of terrestrial, primordial radionuclides
Thermoluminescent Dosimeter (TLD)	A thermoluminescent dosimeter (or "TLD") is a passive measurement instrument for measuring the cumulative exposure from gamma-rays, typically over a 3-month to a one-year period of time. When calibrated for tissue equivalent response and deployed accordingly, the TLD measures dose equivalent, and is commonly used as the dose of record for radiation workers. TLDs are deployed for environmental monitoring, mostly as a dose reconstruction tool following a significant accidental release of radioactivity from the facility. Thus, for environmental monitoring purposes, the TLD is deployed to measure cumulative exposure well over 100 mR to 1000 mR. TLDs, as passive devices, are placed on the person or location of interest for 3 months to a year, and then returned to an analytical laboratory for reading. In contrast to ion chambers that have electronic data loggers, TLDs do not monitor in real time.
Tissue Equivalent Response	The response of an ionizing radiation measurement instrument such that it is equivalent to the interaction response in tissue.
Total Effective Dose Equivalent	A unit in health physics that provides the dose "effectively" received by the whole body by summing the individual dose contributions from internal and external sources to individual organs.
Turbine Shine	This is a term introduced by Duke Engineering Services (DES) in their 2002 report EL 018/02 that describes the flux of N-16 gamma-rays emitted directly from turbine building without scatter. The flux of gamma-rays emitted from the turbine building and detectable at the site boundary consists of two components: the direct (unscattered) turbine shine, and the (scattered) skyshine/groundshine. DES also refers to the Turbine Shine as Line Of Site (LOS) radiation.
Uncertainty	The value that indicates the range around the measurement result in which the desired or true value most likely lies, most often expressed as +/- a number of standard deviations. The uncertainty is a function of systematic error (accuracy), random error (precision), and the error in estimates of unknown values. There are many variants within the measurement sciences on how to express total measurement uncertainty (TMU), total propagated error, and common random error. Often times,

institutions will report only the relatively small random error of a radiation measurement. Relative uncertainty increases as the direct gamma radiation rate decreases, thereby explaining the wide variation (or difference) between two similar measurements, for instance, background and gross. As a result, identical measurements can yield substantially different results at low radiation levels. It is how uncertainty is expressed and accounted for that plays a major role in comparing "equivalent" radiation measurements.

Unrestricted Area Land surrounding VYNPS that is not controlled by VYNPS for purposes of regulatory compliance, safety, or security.

UNSCEAR The "United Nations Scientific Committee on the Effects of Atomic Radiation". UNSCEAR was established by the General Assembly of the United Nations in 1955. Its mandate in the United Nations system is to assess and report levels and effects of exposure to ionizing radiation. Governments and organizations throughout the world rely on the Committee's estimates as the scientific basis for evaluating radiation risk and for establishing protective measures.

Appendix A. Applicable State of Vermont Regulations on Radiological Health

There are four specific sections of the State of Vermont Regulations on Radiological Health that pertain to the issue of direct gamma site boundary dose at VYNPS. Because these sections emphasize different aspects of this issue, they are cited separately. It is also important to recognize that other radiological limits, beyond direct gamma radiation, apply to the regulation, but are not covered here. [1]

> A reasonable place to start is the definition of dose equivalent (rem) and its relationship to other radiation measurement and dosimetric quantities, the rad and the roentgen (abbreviated "R"). For purposes of regulatory compliance, the recommended unit conversions are found in the regulation, Part 5, Chapter 3, Subchapter 1, Section 5-303 Definition (K). Appendix B of this ORAU report presents in much more detail acceptable unit conversions, consistent with concepts introduced by the ICRP, NCRP, and ANSI standards. Appendix B also presents an extensive review of the data and methods used by VYNPS to measure dose equivalent using an HPGe detector.
>
> The second important section of the regulation describes actions the licensee must take if the quarterly average dose at the site boundary is greater than 10 mrem or greater than 20 mrem. There are no specific actions to take based on an annual measurement and whether it is greater than the "annual objective" of 20 mrem. The direct gamma radiation compliance requirements are in Part 5, Chapter 3, Section 5-305(B)(1)(e), with the corresponding actions to take in Section 5-305(B)(2). This appendix describes the decision rules to show clearly what changes need to be considered for a future rewrite of the regulation, or in the short term, for an interim ECP between the State and the licensee.
>
> The third section is the requirement by the State to the licensee that national reports and documents published by the NCRP and NBS (now known as NIST) be used in demonstrating compliance with the radiological health regulations.
>
> The fourth is section 5-304, Exemptions. The State of Vermont, like all other states, provides a list of operations or quantities of radioactive materials that are exempt from regulation.

This appendix examines each of these three issues with ORAU commentary.

ISSUE 1: Part 5, Chapter 3, Subchapter 1. Section 5-303. Definitions (K) "Rem"

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[1] State of Vermont, Department of Health, Part 5, Chapter 3, Subchapter 1, §5-305(B)(1)(e). (Available from http://healthvermont.gov/regs/radio_health.pdf)

"(K) "Rem" means a measure of the dose of any ionizing radiation to body tissue in terms of its estimated biological effect relative to a dose of one roentgen (R) of X-rays. A commonly used submultiple of the rem is the millirem (mrem):

One millirem (mrem) = 0.001 rem

For the purpose of this regulation any of the following is <u>considered to be equivalent</u> to a dose of one rem:

1. A dose of 1 R due to X- or gamma radiation.
2. A dose of 1 rad due to X-, gamma or beta radiation.
3. A dose of 0.1 rad due to neutrons or high energy protons.
4. A dose of 0.05 rad due to particles heavier than protons and with sufficient energy to reach the lens of the eye."

ORAU Commentary: When written in the 1970's, these approximations were convenient from a practical standpoint and commonly used. In fact, today, the unit conversion is still commonly encountered because it simplifies policy and compliance, and it results in a conservative compliance point. In the science of dosimetry however, then (and now) it is accepted by the ICRP, the ICRU, and the NCRP that in fact 1 roentgen is *not* equivalent to 1 rad is *not* equivalent to 1 rem. It is permissible and desirable for routine radiation protection practice to measure these quantities and convert, from one measurement method to another, the quantities of roentgen, rad, and rem. A licensee should be allowed to measure and report regulatory compliance dose equivalent in units of rem (dose equivalent). Dose equivalent is the measurement quantity of most importance to radiological health and safety because it reflects not only the amount of energy absorbed by the body, but also the risk from that dose by describing explicitly, biological effect.

Regarding the dose equivalent conversions provided under Definition (K) of the regulation, only item one (gamma radiation) is of relevance in this report because it affects the outcome between a measured value and the dose equivalent response. Items two through four are understood and accepted by the scientific community. Item two cites a quality factor of "one" for gamma and beta radiation. Item three infers that neutrons of an unknown or uncharacterized spectrum use a default quality factor of ten. Item four assumes a quality factor of 20 for the rad to rem conversion factor for heavy ions (e.g., protons).

ISSUE 2: Part 5, Chapter 3, Section 5-305(B)(1)(e) Direct Gamma Radiation

The existing regulations are written such that the direct gamma radiation limits and associated actions are presented together with the effluent monitoring limits, making it difficult to separate these respective limits from each other. To partially remedy this, the sections of the regulation pertaining to specific citations about direct gamma radiation have been denoted in boldface. Other than that change, the regulation is cited verbatim.

"(e) Direct Gamma Radiation

1) The annual dose objective for the total-body of an individual in an unrestricted area due to plant emanations of gamma radiation is 5 millirems. For the purpose of this objective, 20 millirems per year at any point on the site boundary bordered by land shall be considered equivalent to a 5 millirem dose at the nearest residences in Vermont.

2) If any site boundary, bordered by land, quarterly average dose exceeds 10 millirems above background radiation, the actions described in **(B)(2)** shall be taken.

3) If any site boundary, bordered by land, quarterly average dose exceeds 20 millirems above background radiation, the actions described in (**B)(3)** shall be taken.

(B)(2) If the radioactive materials discharged from Vermont Yankee exceed the rates, concentrations or quantities as defined in Subsections (B)(1)(a) 4), (B)(1)(b) 6), (B)(1)(b) 7), (B)(1)(c) 4), (B)(1)(d) 4), or **(B)(1)(e) 2)** of Section 5-305 of these regulations, Vermont Yankee management shall, as soon as it is evident that the quarterly average of any discharge will exceed these levels:

(a) Make an investigation to identify the causes of such release rates or radiation levels.
(b) Define and initiate a program to reduce such releases to within the objectives defined in (B)(1)(a) 1), (B)(1)(b) 1), (B)(1)(c) 1), (B)(1)(d) 1) and **(B)(1)(e) 1)**.
(c) Report these actions to the State of Vermont Board of Health within 14 days of the date it became evident that the levels listed in (B)(2) would be exceeded, but in no event later than 10 days after the end of the calendar quarter; the report shall include submission of the plan for corrective action, to be approved by the Board of Health.
(d) Implement the approved plan with all reasonable speed.

(B)(3) If the radioactive materials discharged from Vermont Yankee exceed the rates, concentrations, or quantities defined in Subsections (B)(1)(a) 5), (B)(1)(b) 8), (B)(1)(b) 9), (B)(1)(c) 5), (B)(1)(d) 5), or **(B)(1)(e) 3)** of Section 5-305 of these regulations, Vermont Yankee shall take the following actions as soon as it becomes evident that the quarterly average of discharges will exceed these levels, but in no event later than the last day of the calendar quarter in which the average discharge exceeds these levels.

(a) Make an investigation to identify the causes of the discharge which exceeded the levels listed in (B)(3) above, and initiate a program designed to insure that future discharges will be maintained at or below the levels listed in (B)(2) above.

(b) Immediately report the quarterly average discharge rates to the Vermont Board of Health, together with the action taken or proposed to be taken to achieve immediate reduction of the discharges.

(c) Within 14 days, but in no event later than 10 days after the end of the calendar quarter, report the actions described in (B)(3)(a) above to the Vermont Board of Health for the Board's approval."

ORAU Commentary: ORAU has formatted the regulation as an annotated decision table that more clearly shows the reliance on quarterly measurement results, and not on annual measurement results. This format also shows that an interim ECP should be put into place to more definitively state the time limits for reporting, and reasonable actions to be taken therein.

ORAU is not submitting an improved action plan, but is simply pointing out that the current actions that would be taken by the VDH are more consistent with violations or accidental overexposures imminent to health and safety, rather than for chronic low-level exposure to the public.

Table A.1. Annotated Decision Table (Quarterly Measurement vs. Annual Objective)

QUARTERLY MEASUREMENT	ACTION (DIRECT GAMMA CASE), annotated
If quarterly average exceeds **10 mrem** above background radiation → Go to §(B)(2)	(B)(2) … Vermont Yankee management shall, as soon as it is evident that the quarterly average will exceed 10 mrem: (a) Make an investigation to identify the causes of such radiation levels. (b) Define and initiate a program to reduce such releases to within the objectives defined for the annual dose case of 20 mrem, **§(B)(1)(e) 1)**. (c) Report these actions to the State of Vermont Board of Health within 14 days of the date it became evident that the levels listed in (B)(2) would be exceeded (10 mrem), but in no event later than 10 days after the end of the calendar quarter; the report shall include submission of the plan for corrective action, to be **approved** by the Board of Health. (d) Implement the approved plan with all reasonable speed.
If quarterly average exceeds **20 mrem** above background radiation → Go to §(B)(3)	(B)(3) …Vermont Yankee shall take the following actions as soon as it becomes evident that the quarterly average of discharges will exceed these levels, but in no event later than the last day of the calendar quarter in which the average discharge exceeds these levels. (a) Make an investigation to identify the causes of the discharge which exceeded the levels listed in (B)(3) above, and initiate a program designed to insure that future discharges will be maintained at or below (10 mrem per quarter). (b) Immediately report the quarterly average discharge rates to the Vermont Board of Health, together with the action taken or proposed to be taken to achieve immediate reduction of the discharges. (c) Within 14 days, but in no event later than 10 days after the end of the calendar quarter, report the actions described in (B)(3)(a) above to the Vermont Board of Health for the Board's approval.
ANNUAL OBJECTIVE	**ACTION (DIRECT GAMMA CASE), annotated**
The annual dose objective for the total-body of an individual in an unrestricted area due to plant emanations of gamma radiation is 5 millirems. For the purpose of this objective, 20 millirems per year at any point on the site boundary bordered by land shall be considered equivalent to a 5 millirem dose at the nearest residences in Vermont.	There are no imposed actions to take on the basis of an annual measurement. Section (B)(2)(b) refers to the annual objective, but no specific action is required. (B)(2)(b) states essentially that when a quarterly average of 10 mrem is exceeded, the licensee should "initiate a program to reduce such releases to within the objectives." The objectives cited are the annual objectives in **(B)(1)(e) 1)**.

ISSUE 3: Part 5, Chapter 3, Subchapter 1. Standards. Section 5-305(C)

"(C) Persons within the scope of this regulation, other than as described in Section 5-305 (A) and (B), shall control all sources of radiation by using the applicable recommendations contained in the reports of the National Council on Radiation Protection and Measurements and the National Bureau of Standards handbooks as standards and bases for calculations."

ORAU Commentary: The licensee is required to use widely accepted, current methods to calculate regulatory compliance quantities, as it should be (at least to the extent that the methods are consistent with existing USNRC guidance). ORAU also notes that the regulation as written requires an editorial revision in that sources of radiation are not "controlled" in and of themselves by utilizing the cited reports.

Summary Commentary

1. While practical, straightforward, and routinely cited, the current regulatory position that a roentgen, rad, and rem are equivalent is incorrect. The unit of most importance to relate risk and human health from direct exposure to ionizing radiation is the dose equivalent (rem). Methods exist and should be used to account for the spectral weighted average of the energy-dependent flux at the location of interest in order to measure and/or calculate dose equivalent. Provisions under section 5-305(C) are already in place to allow the use of accepted methods as published by the NCRP and NBS (now NIST, the National Institute of Standards and Technology).

   Measurement methodologies should be put into place which convert or calibrate for the desired regulatory unit, rem, instead of assuming a one-to-one correspondence. ORAU recognizes that there is a fairly significant cost to implement an environmental monitoring program, using dosimetry or real-time monitoring, but VYNPS should be allowed the flexibility to implement these advanced methods if agreed upon by VDH.

2. Provisions for exceeding a given regulatory limit (whether administrative or by engineering controls), including proper notifications and remedies, need to be clearly defined. The fact that there is not an action-based response to be taken on an annual basis is a significant omission within the existing regulation. Note that the current regulations allow, without notice or control/review, an annual dose limit approximating 40 mrem, essentially doubling the 20 mrem/year "objective." How could this occur? The licensee need not report or take action when the quarterly dose equivalent is below 10 mrem. Hence, four consecutive quarters at 9 mrem per quarter (the nearest whole integer dose to 10 mrem) is not reportable, and would result in an annual dose essentially twice the annual dose objective.

   The regulatory "value" of citing quarterly limits for such low, direct gamma-ray exposures should be discussed. ORAU believes that quarterly limits should be used only as an "administrative tool" to ensure that annual limits will not be exceeded (in advance of potentially exceeding the annual limit). Most importantly, it is the annual dose limit that should be regulated. The quarterly dose limit and action guides should be used more for process control than absolute regulation.

All decision levels need to be clearly defined, with a clear action plan if and when a limit is found to have been exceeded. Measurement uncertainty and quality control of the measurements, including background variation, needs to be defined in the implementation plan (and stated as clearly as possible in the regulation). Within §(B)(2) and (B)(3), the action steps are currently too lengthy and confusing. These steps should be shortened and clarified. Notification intervals and follow-up requirements should follow guidance from the NRC. An interim compliance protocol should be written in the ECP between VDH and VYNPS. Some guidance on this issue may be provided from the document ORAU received from VYNPS, "State Statute Comparison," and from current USNRC compliance requirements for direct gamma radiation exposure to the public.

3. The Vermont regulations are not clear on who conducts the radiation measurements, how they are to be performed, and to what extent compliance can be achieved by calculation. Specifically, it is not clear that the State of Vermont will conduct compliance measurements, whether the State will verify (by direct measurement) and confirm VYNPS measurements and calculations, or whether the State will conduct assessments of the VYNPS methodologies.

4. Part 5, Chapter 3, Section 5-305(B)(1)(e)(1) states: "The annual dose objective for the total-body of an individual in an unrestricted area due to plant emanations of gamma radiation is 5 millirems. For the purpose of this objective, 20 millirems per year at any point on the site boundary bordered by land shall be considered equivalent to a 5 millirem dose at the nearest residences in Vermont."

   ORAU believes that the approximations and assumptions inherent in this policy are difficult to enforce and confuse the overall objective of the requirement. This section should be rewritten to include, for example, that historical documents between the State and VYNPS do in fact discuss provisions for occupancy factor, not only for the nearest resident, but also the school located nearby. Reasonable occupancy factors should be considered for inclusion in the compliance protocol (ECP) between the State and VYNPS. The use of occupancy factors is warranted when rational arguments can be made that describe the critical group's exposure pathway from the direct gamma-radiation source term; occupancy factors are used routinely in the design and construction of medical diagnostic and treatment centers that use x-ray machines, CT scanners, PET scanners, and other x and gamma-ray radiation generating devices. The direct gamma radiation from ^{16}N production falls in the same category.

The figure below is a simplified (existing) regulatory decision process (refer to previously cited annotated decision table) to limit/control exposure to direct gamma radiation at the VYNPS site boundary. This simplified flow diagram illustrates deficiencies in the existing protocols.

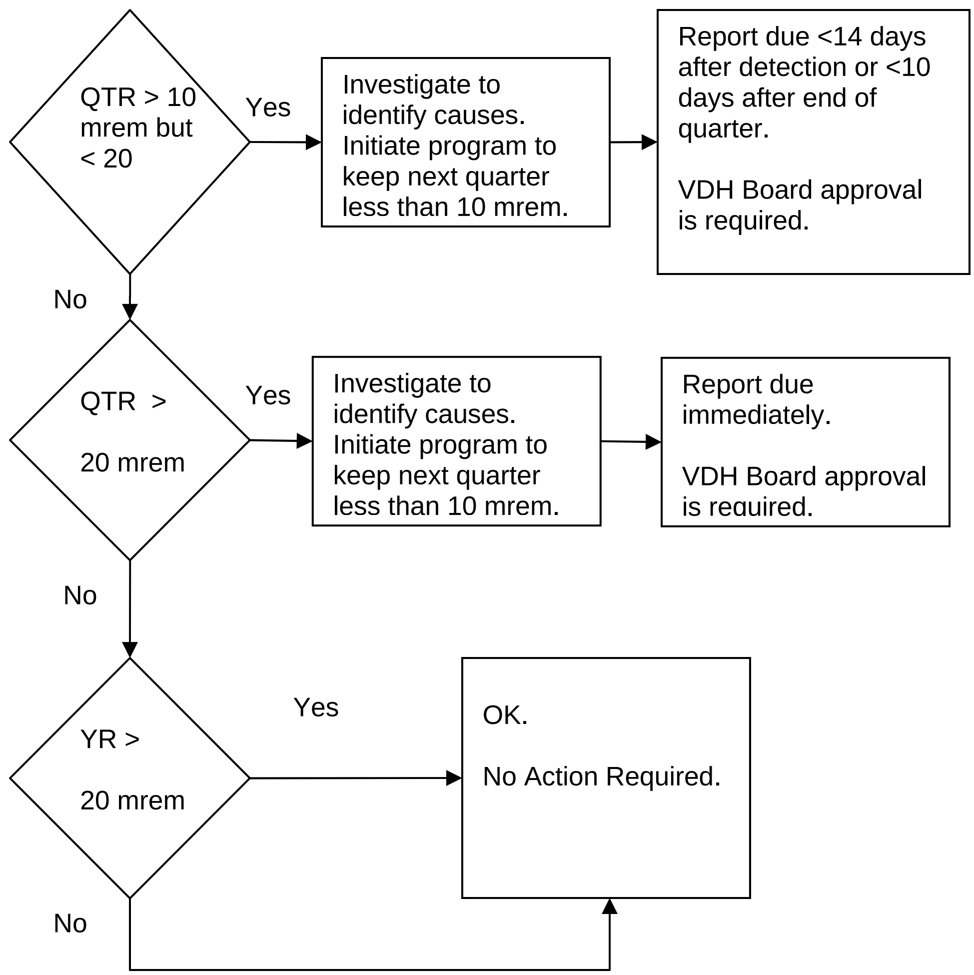

Figure A.1. Simplified Flow Diagram of Existing VDH Regulation for Complying with the External Gamma-dose Limit at VYNPS

From the flow diagram, the following conclusions are reached:

1) There is no significant difference in the corrective action plan between exceeding the quarterly limit of 10 or 20 mrem, which then brings into question the necessity for checking compliance over such a short time interval. Also, if passive dosimeters are used for demonstrating compliance, the reporting requirement of 10 to 14 days, is likely too short in practice to allow for retrieval of the dosimeter, sending to the laboratory for analysis, and making the proper analysis and background subtraction. A more manageable timeframe is on the order of 30 days.

2) There is no action to take if the annual limit is exceeded. This is a fundamental flaw in the existing regulation, as captured in this flow diagram, unless it was the intent of the authors (1972) to state an objective of 20 mrem and an effective limit of 40 mrem per year. As noted previously, it is possible to have four consecutive quarters with reported

doses below the quarterly limit, with a total dose exceeding the annual objective, yet no action by the licensee would be required; and

3) For some measurement methods, the reporting time and follow-up requirements are considered by ORAU as too aggressive. Refer to Conclusion #1.

ISSUE 4: Part 5, Chapter 3, Subchapter 1. Standards. Section 5-304. Exemptions

The State of Vermont exempts from regulation several materials, machines, and conditions that produce ionizing radiation. During discussions with VYNPS and VDH, ORAU was asked "Is it possible to create a state regulatory environment more consistent with the VYNPS USNRC license that evaluates dose to the public from a sum of all sources? And that furthermore, accounts for a similar "dose baseline" used to establish exemption requirements?"

ORAU Commentary: ORAU reviewed the exemption requirements from the State of Vermont regulation. As the VDH begins to write the next generation of regulations, it should conduct a dose assessment of all possible pathways for public exposure in order to establish a dose baseline from which to exempt the various materials, machines, and conditions producing ionizing radiation. NUREG-1717, "Systematic Radiological Assessment of Exemptions for Source and Byproduct Materials," (2001), is a reasonable place to start for method evaluation. Within the context of the exemptions, it may be possible and feasible to argue that the direct gamma dose requirement be folded into the overall plan for limiting dose to the public, rather than distinguishing it from the other pathways.

Appendix B. The Energy Dependent Flux Profile from Skyshine at VYNPS and the Results of Direct Measurements of Flux to Dose Conversion

In Appendix A of this report, "Applicable State of Vermont Regulations on Radiological Health," the application of existing State of Vermont (State) regulations and options for licensees to demonstrate compliance with the "direct gamma radiation" requirement is discussed. The appendix is written to assist the State in formulating a plan for future refinements to the regulation and for ultimately, refining existing agreements. In the near term, the ORAU recommended ECP prepared between VYNPS and the State should define the measurements to be performed, the interval over which the measurements are to be performed, who will perform the measurements, and the timeliness of measurement results. Several discussions have occurred between VYNPS, VDH, and ORAU in this regard, including of course, the methodologies described in the VYNPS proposed methods in the ODCM.

One important question centers on whether the measurement results should be reported in exposure (roentgen) or dose equivalent (rem). The existing State regulation, Part 5, Chapter 3, Section 5-305(B)(1)(e) Direct Gamma Radiation, is written in units of dose equivalent (rem). Accordingly, it is a reasonable practical matter to allow a measurement protocol to be put in place such that the measurement data is converted to dose equivalent. In 2002, VYNPS reported on a set of measurements conducted in 2001 using a high resolution gamma ray spectrometer, from which a spectrum weighted dose equivalent was then calculated in May of 2006. The approach developed by VYNPS was implemented at the recommendation of ORAU and is consistent with the historical American National Standard, ANSI/ANS-6.1.1-1991. The method should be adopted as an acceptable approach for flux to dose equivalent conversion. ORAU has reviewed the results presented by VYNPS and finds the results acceptable for consideration by the State in its implementation of the regulatory requirement in units of dose equivalent (rem). ORAU recommends an agreement to this effect should be placed within the ECP.

Introduction

The State of Vermont Regulations on Radiological Health, Part 5, Chapter 3, Section 5-305(B)(1)(e) Direct Gamma Radiation, provides limits for quarterly average dose equivalent at the site boundary (millirem per quarter). Furthermore, Part 5, Chapter 3, Subchapter 1, Section 5-303, Definitions (K) "Rem" is defined as:

> *"(K) "Rem" means a measure of the dose of any ionizing radiation to body tissue in terms of its estimated biological effect relative to a dose of one Roentgen (R) of X-rays. A commonly used submultiple of the rem is the millirem (mrem):*
>
> *One millirem (mrem) = 0.001 rem*
>
> ***For the purpose of this regulation, any of the following is considered to be equivalent to a dose of one rem:***

A dose of 1 R due to X- or gamma radiation.
A dose of 1 rad due to X-, gamma or beta radiation.
A dose of 0.1 rad due to neutrons or high energy protons.
A dose of 0.05 rad due to particles heavier than protons and with sufficient energy to reach the lens of the eye.

It is not uncommon in practice to simplify the unit expressions and assume that the roentgen, the rad, and the rem are equivalent for photon radiations (as described above in the State regulation). For radiation safety and protection purposes, this simplification results in reduced cost and implementation of a conservative policy. The quantity "dose equivalent" for gamma rays is always a fraction of the quantity "exposure"; hence, assuming that they are equal to one another builds in a level of conservatism. This level of conservatism (for environmental gamma rays) was reported by O'Brien and Sanna as over 40%, meaning specifically that the total body exposure to absorbed dose conversion factor is 0.561 rad/R.[1] Kocher later suggested that a conversion of about 0.6 applies for the infinite-plane uniform source geometry.2

In other words, O'Brien and Sanna noted that the absorbed dose to various organs in the human body was approximately 60% of the exposure, due to the body's self-shielding.
In the science of radiation dosimetry, it is appropriate to consider the following: If the regulatory limit or control limit is expressed as a dose equivalent, the measurements and methods leading up to the issuance of quarterly or annual reports should all be calibrated to report dose equivalent (rem). In radiation protection dosimetry, it is the dose equivalent that is paramount to measuring and reporting human radiation dose. Exposure is a convenient quantity for measurement purposes. These important principles are expanded upon in this report.

Radiation Measurements of Direct Gamma-ray Fluence Rate, Exposure Rate, and Dose Equivalent Rate

For radiation protection purposes, measurements of gamma ray fluence rate are normally performed by an instrument called an ionization chamber. For environmental applications, this device measures exposure rate in microroentgens per hour, abbreviated, μR/hr. The prefix, μ, represents the multiple of the primary unit, micro, meaning 1×10^{-6} or 1 millionth. It is important to understand this notation because all natural background gamma ray measurements are in the μR/h range, which translates almost by a factor of 10 to mR/year, by unit conversion: $\left(1\ \mu R/h\right)\left(24h/d\right)\left(365.25d/y\right) = 8.8\ mR/y$

High Pressure Ion Chambers (HPICs) are commonly used to measure exposure rate. General Electric-Reuter Stokes manufactures the most widely used environmental monitoring instruments for measurement of low-level exposure rate. For example, the Model RS-111 uses an 8-liter chamber filled with 25 atmospheres (atm) of argon, is omni-directional, and exhibits a "flat-energy" response. This essentially means two things: 1) The detector responds to an incident gamma ray flux, independent of angle of incidence; and 2) The number of ion pairs created in the chamber is independent of energy. Another way to state these detection parameters is to say that the detector response is independent of gamma ray angle and energy, respectively. The design of this type of instrument allows for very accurate measurements of exposure rate, i.e., the number of ion pairs created of either sign per cubic centimeter of air at standard temperature and pressure (STP). This measurement is not to be confused with a measurement of dose equivalent (rem).

Natural background radiation at 1 meter from the ground is approximately 10 μR/hr, which is equivalent to nearly 100 mR/y. Striking the face of the detector is a gamma ray fluence rate which is energy dependent. That is to say, the gamma rays emitted from naturally occurring radioactive material in the soil and from cosmic ray interactions in the atmosphere range in energy from a few keV to tens of MeV. The majority of gamma ray energies from natural background are in the range of about 50 keV to 2 MeV[3]—the energy spectrum which became the basis for the flux to dose conversion first calculated and reported by O'Brien and Sanna[4], showing a roentgen to absorbed dose conversion of 0.561 (refer to Table 5 of that reference). Their paper, published in 1976, laid much of the groundwork for what eventually became the approved method established in ANSI/ANS-6.1.1-1991.[5] Furthermore, it informed the radiation protection community that environmental radiation dose equivalent from exposure to terrestrial and cosmic gamma rays may be reported too high (by nearly 50%), by choosing to use a one-to-one correspondence between the roentgen and rem.

Gamma ray emissions from VYNPS, like any nuclear power station, are emitted over a wide range of energies, and as mentioned previously, the most penetrating gamma ray component is that from N-16, which emits two gamma rays at approximately 6 and 7 MeV, respectively. By the time these gamma rays scatter off the atmosphere and ground thousands of times, and reach the site boundary, the original gamma rays are significantly down-scattered into the nominal background range of 50 keV to 2 MeV. While the N-16 contribution to the gamma ray fluence rate is directly measurable at the site boundary, the energy spectrum is not significantly different from natural background (at large distances from the turbine building).

At location DR-53, many gamma ray fluence rate measurements have been conducted, using a HPIC, expressing the output in units of exposure rate (μR/hr). To measure fluence rate (photons per square centimeter per second) requires additional effort and unit conversions. As a result, this implicit conversion from flux to exposure is accomplished by designing the detector in such a way that the detector response is independent of gamma ray energy and angle of incidence. This requirement drives the HPIC detector design and signal processing. While this is extremely important, it still does not address the question of flux to dose equivalent.

In the practice of radiation protection, it is frequently the case that the exposure rate (an easily measured quantity) is related to dose equivalent rate by multiplying exposure rate by the number "one" because, as stated above, it is easier, less costly, and safely conservative. It is not the most proper treatment of dosimetric quantities, but has been accepted as common practice.

There are a number of ways to measure dose equivalent directly and then establish a multiplicative factor to convert the response of a given radiation detector into a "tissue equivalent" response. The reference dosimetric, tissue-equivalent, sphere has been defined by the International Commission on Radiation Units and Measurements (ICRU), Report 33[6]:

The unit ICRU sphere as a set of measurement conditions for which the tissue equivalent detector response should be applied. The "ICRU sphere" is defined as a sphere of 30 cm diameter made of tissue equivalent material with a density of 1 g cm^{-3} and a mass composition of 76.2% oxygen, 11.1% carbon, 10.1% hydrogen and 2.6% nitrogen…used as a reference phantom defining dose equivalent quantities."

For "laboratory grade" dosimetric physics studies, these precise concepts must be clearly described so that fundamental interaction and absorbed dose properties of the detector representation can be made. This concept of tissue-equivalency was further described by the ICRU.[7, 8] The principles of these conversions from one unit to another have been studied by a number of people and institutions. It is understood by dosimetric physicists that incorporating these measurement principles should be performed with a great deal of care. It was not until 1991 that ANSI/ANS-6.1.1 was approved by the American National Standards Institute and adopted for use by the community. The standard has not been updated since its original release, thereby resulting in its current designation as a "historical" standard. However, in the interim, other papers have been published to describe results of computational models and monte carlo simulations—one example is cited here.[9]

In 2006, ORAU recommended that VYNPS and VDH consider implementing ANSI/ANS-6.1.1-1991 on measurement data that had been already collected in 2002. ORAU recommended that the measured results could be used in calibrating future measurement systems, whether passive dosimeters, active HPIC monitors, or for that matter, MSLRM. The goal behind this suggestion was to enable VYNPS and VDH to convert between gamma ray fluence rate, exposure rate, and dose equivalent rate. There is no interest in understanding or expressing absorbed dose (rad) in this case.
Dr. Nolan Hertel, now at the Georgia Institute of Technology, has published many papers on spectral unfolding and dose equivalent measurements, and was the working group Chairman of ANSI/ANS-6.1.1-1991. For further reading, see three of his publications in Radiation Protection Dosimetry.[10, 11, 12] In addition, ORAU has recently been involved in a project with Dr. Hertel and was able to discuss the mathematical constructs of the process of converting from flux to dose. ORAU acknowledges his contribution to these methods and our evaluation of the flux to dose issue at Vermont Yankee.

Converting from Energy Dependent Fluence Rate to Dose Equivalent

In the course of the ORAU review, VYNPS had made a set of high-resolution gamma ray spectrometry measurements at a location ORAU is calling the "environmental radiation test station" (DR-53). The test station has been used by VYNPS to conduct direct gamma ray radiation measurements. DR-53 is situated near the fenceline. ORAU is sensitive to the fact that the relative location of DR-53 to the site boundary and most importantly the "nearest resident" is not equivalent. This difference in distance and the impact of any measurements conducted at DR-53 extrapolated to other nearby locations is a point of important interest that will be discussed separately. Though this distinction regarding distances is important for describing annual dose equivalent rate, it does not sufficiently impact the spectral shape and the conversion factor from fluence rate to dose equivalent (at these distances). This is not the case at greater distances, for example, 3000 feet, from the turbine building.

The set of measurements, as noted during one of the ORAU/VYNPS/VDH onsite review meetings, was ideal for implementing the physical and mathematical constructs established in ANSI/ANS-6.1.1-1991. Very accurate, energy dependent gamma ray flux measurements had previously been made at DR-53 that, when applying the principles in ANSI/ANS-6.1.1-1991, would yield directly measured results of dose equivalent (rem). There is no better methodology to measure energy dependent gamma ray flux than with a properly shielded/collimated high-purity germanium detector. Once this data set was known to exist, ORAU suggested that an evaluation of the data set be performed. VYNPS agreed that the data could be and should be used to obtain this otherwise difficult physical property of the radiation field, especially since significant attention had not previously been focused on the amount of downscatter in the original source term contribution from approximately 6- and 7-MeV gamma rays emitted from N-16 (at VYNPS).

Citing the methods described in ANSI/ANS-6.1.1-1991, the measurement property of interest is dose equivalent, expressed in the unit "rem". Of particular interest is the spectrum weighted average conversion factor. Given a multi-group energy flux measurement, each energy bin (or energy interval from E_l to E_u) of interest, the dose equivalent for the i^{th} energy interval is calculated as the integral of the product differential flux times the flux to dose factor, $h_E(E)$, divided by the group flux:

$$< h_{E_i} > \quad = \quad \frac{\int_{El}^{Eu} h_E(E)\phi(E)dE}{\int_{El}^{Eu} \phi(E)dE}$$

Equation 1

where the mean dose equivalent over the i^{th} energy bin, $<h_{Ei}>$ is the spectrum-weighted average of the dose equivalent over the i^{th} energy bin of interest. The total dose equivalent is then this average value for the i^{th} energy bin summed as the product of the group fluxes (Equation 2), and shown in Equation 3.

It is the energy group structure of the flux described for each i^{th} interval that is represented as:

$$\phi_i = \int_{El}^{Eu} \phi(E)dE$$

Equation 2

The energy group structure and h_E values are provided in Table 3 of ANSI/ANS-6.1.1-1991, Fluence-to-Dose Factor (h_E) for Gamma Rays Incident in Various Geometries on an Anthropomorphic Phantom (plotted below). VYNPS measured $\boldsymbol{\phi(E)dE}$ with a gamma ray spectrometer and then used the Table 3 values to calculate <h_{Ei}>. The total dose equivalent, summed over the differential energy spectrum, is then calculated as:

$$H_E = \sum_i < h_{E_i} > \phi_i$$

Equation 3

The mean h_{Ei} values are provided in the standard, and plotted below, in μrem-cm^2 as a function of photon energy (MeV), from Table 3 of ANSI/ANS-6.1.1-1991.

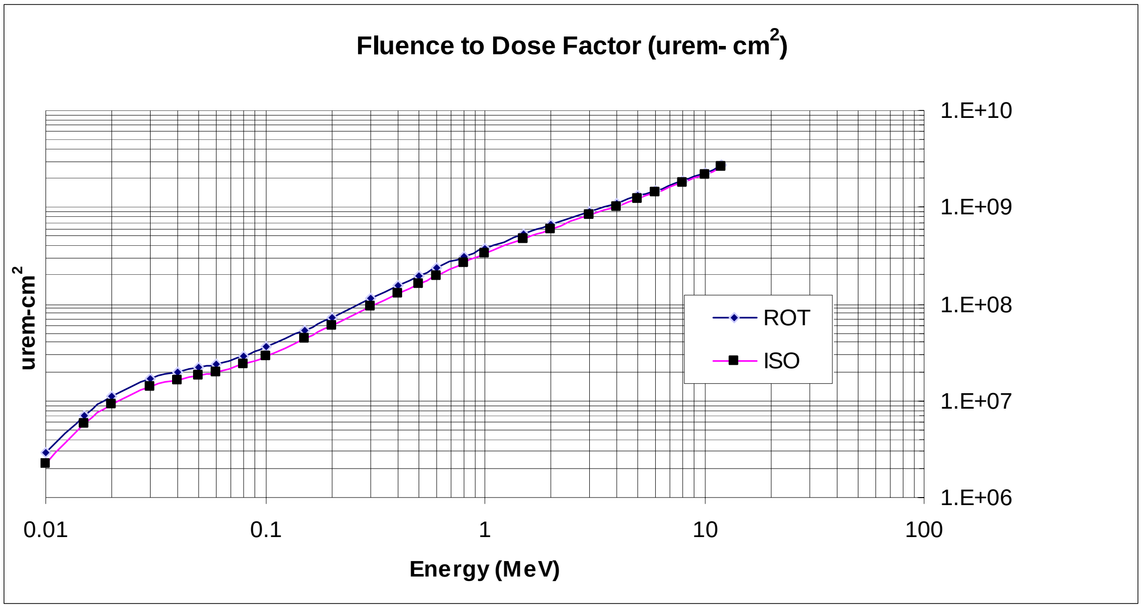

Figure B.1. ANSI/ANS 6.1.1 Fluence to Dose Factors for Rotational (ROT) and Isotropic (ISO) Geometries

Figure 2 demonstrates that the dose equivalent per unit fluence conversion factors decrease with decreasing energy (i.e., the down-scattered N-16 energy that would be encountered with increasing distance from the plant site).

ANSI/ANS-6.1.1-1991 also provides an approved method for estimating dose equivalent as a function of energy. A fourth-order function, using the coefficients provided in the ANSI standard (represented below as the "C_i's") is:

$$h_E(E) = 10^{-4} \exp\left\{C_0 + C_1 \ln(E) + C_2 \ln(E)^2 + C_3 \ln(E)^3 + C_4 \ln(E)^4\right\}$$

Equation 4

where h_E is expressed in microrem per square centimeter, and the photon energy, E, in MeV. Note that the multiplier, 10^{-4} was modified from the ANSI/ANS-6.1.1-1991 value of 10^{-12} to convert from the SI unit, sievert (Sv), into the unit expressed here, microrem.

ORAU recommended that the high-resolution gamma ray spectrometry measurements conducted in 2002 be reanalyzed using this methodology.

Measurement and Analysis Results

In April of 2006, ORAU participated in several discussions with the technical staff at VYNPS to evaluate the *in situ*, high-resolution gamma ray spectrometry measurements conducted in 2001 and reported in February of 2002[13] and in final form in 2006.[14] On April 11 of 2006, Raimondi and Keefer submitted, for review by ORAU, the results of their analysis.[15]
The high purity germanium (HPGe) detector measurement spectra collected in 2001 at TLD-53 (DR-53 environmental radiation test station) were reanalyzed. The measurements were performed in such a way as to estimate the components of the background radiation field: cosmic, terrestrial, and the primary source term itself, skyshine. The components and estimates for background and above background are discussed elsewhere in this report, but, related to the spectrum flux to dose conversion, Raimondi and Keefer implemented the approach summarized above as explained in full detail in ANSI/ANS-6.1.1-1991.

While ORAU did not conduct an independent set of measurements or independent analysis of the raw data, ORAU confirmed the following from the report provided:

The high-resolution gamma ray spectrometer data appear reasonable. The measured spectra from "net skyshine" show a characteristic down-scatter contribution (a hump) at 60 keV falling off logarithmically up to 500 keV. No evidence of a high-energy photopeak at approximately 7 MeV was detected (as expected). An acceptable measurement methodology was used to focus on the cosmic ray component of the flux, and subtract it from the combined skyshine plus cosmic ray component. This is best shown as the spectral data, excerpted from the Raimondi/Keefer report:

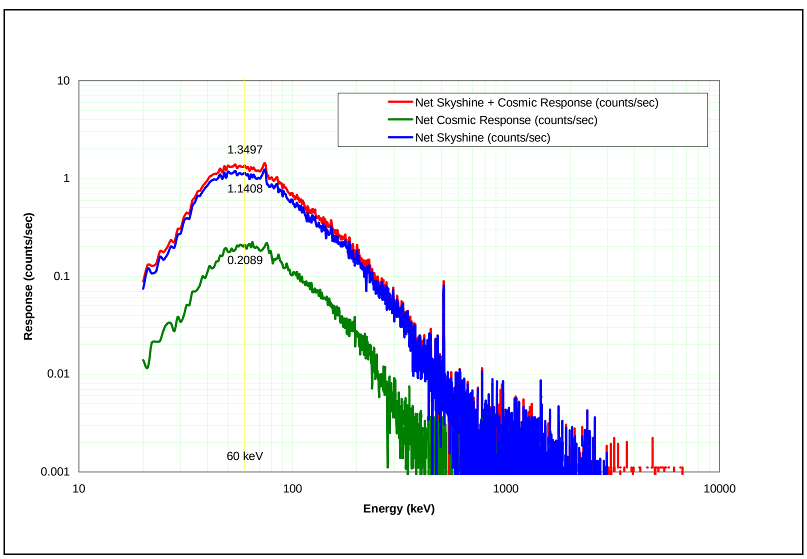

Figure B.2. VYNPS Measured Gamma Ray Spectra at DR-53

On the spectral plot, it is noted that the red spectrum is cosmic plus skyshine, the green is natural background, and the difference (blue) is net skyshine. The data shows that indeed the high-energy component from skyshine is detectable above the 300-keV natural background cut-off—a significant contribution of the skyshine flux is observed from 300 keV up to 1000 keV, as expected. ORAU also notes that the “gain” of the spectrometer was set sufficiently high to detect the high-energy components of the flux from N-16 decay (it has been observed through experience that measurement errors are made by improper gain settings). It is also noted from the measurement that the natural background contribution to the total flux at the peak of 60 keV is 15% (from the graph: 0.2089/1.3497=15%). This spectral plot shown above is at 100% power, per the AREVA document EL/145-05-FINAL.

ORAU did not review the raw data files, but did confirm in discussion with technical staff, that the energy binning structure formulated in ANSI/ANS-6.1.1-1991 was used with the resultant $<h_E>$’s computed properly. Results for isotropic and rotational fluxes were provided.

All calibration certificates provided in EL/145-05 were reviewed and are acceptable.

The fit coefficients (C_0, C_1, C_2, C_3, and C_4) from ANSI/ANS-6.1.1-1991 were properly implemented in the polynomial expansion for calculating dose equivalent as a function of energy.

There are a couple of decisions that ORAU supports in this analysis. Agreements between VDH and VYNPS that need to be considered:

Spectrum-weighted dose equivalent results have been provided (microrem per hour) on the basis of the ANSI/ANS-6.1.1-1991 methods and based on the 2001 measurements. ORAU believes that there is no significant rationale in repeating these HPGe measurements to account for a power rate increase. Within the uncertainty tolerances and the natural variation tolerances, power uprate will not significantly change the spectral characteristics—only the magnitude. Furthermore, Raimondi and Keefer have cross-referenced the new calculations with environmental ion chamber measurements of exposure rate to arrive at the most important conversion of interest: dose equivalent per unit exposure (rem per roentgen). This ultimately, is the conversion factor that the current VDH regulation states should be 1, “for purpose of the regulation.” ORAU has reviewed the methods and results and concurs with the Raimondi/Keefer results, as written. The only other assurance ORAU can provide in the analysis is to request the raw CAM files from the spectrometer, and independently run the analysis from within ORAU software. ORAU accepts the measurement and analysis uncertainty that is introduced when, as ANSI/ANS-6.1.1-1991 states, “the spectral shape used to obtain the weighted mean fluence-to-dose factor should be identical to that used to obtain the group-averaged cross sections” is not precisely available, and is thus approximated.

ANSI/ANS-6.1.1-1991 provides multiple options to describe angular, energy-dependant flux: Frontal (Anterior-Posterior, AP); Rear (Posterior-Anterior, PA); Lateral (LAT), Rotational (ROT), and Isotropic (ISO). The physical properties of skyshine radiation are such that the isotropic angular flux distribution (ISO) is the best representation. Raimondi and Keefer used coefficients for ROT and ISO and provided the results. It is well understood that the conversion factor for an isotropic angular flux distribution will yield a smaller conversion factor than for a rotational angular flux. This in fact, may lead to a regulatory dilemma on whether to use the most conservative approach. In light of this, however, from a health physics and radiation protection standpoint, and a computational physics view, it is more accurate and representative to select the ISO distribution because this is the best description of actual conditions.

The results of the VYNPS analysis of HPGe measurement data show exposure rate to dose equivalent rate conversion factors of:

0.60 μrem/μR [ISO field][16]

0.74 μrem/μR [ROT field][17]

The conversion factor of O'Brien and Sanna (0.561) is not too different than the results of the VYNPS direct measurement results. At DR-53 the radiation field is isotropic (ISO) and the conversion factor directly measured was 0.60 μrem/μR. For site boundary dose, this is the conversion factor that should be considered. As the detector (or point of interest) is moved further from the plant, the group dose equivalent conversion factors decrease as the source term is scattered downward (see Figure 1 above). As a result, if the same *in situ* gamma-ray spectrometry measurements were conducted radially further from the plant, the dose conversion factor would decrease. The origin of the direct gamma-radiation from N-16 production would down-scatter further, into a region of Figure 1 where the dose conversion factor continues to decrease.

References for Appendix B

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13 Duke Engineering Services, "Summary Report: *In Situ* Measurements Performed at the Vermont Yankee Nuclear Power Station," EL 018/02, (February 13, 2002).

14 AREVA, "FINAL Summary Report of *In Situ* Measurements Performed at the Vermont Yankee Nuclear Power Station," EL 145/05-FINAL, (January 26, 2006) *[Note: this report replaces Report EL 018/02 of February 13, 2002. EL 018/02 was the initial report of the measurements, edited and distributed as EL 145/05-FINAL, and then renumbered as EL 02-006. This document provides the spectrometric results for cosmic, terrestrial, and N-16 gamma radiation*

Revisions and updates include additional data incorporated to describe QA requirements and calibration]

15 Keefer D., Raimondi J., "Skyshine Fluence-to-Dose Methodology for HPGe Detector," AREVA report, (April 11, 2006). *[Note: this is the report that describes how the spectra collected in 2002 were reanalyzed according to ANSI/ANS 8.1 to yield a flux to dose conversion factor]*

16 Ibid.

17 Ibid.

Appendix C. Methods for Selecting Surrogate Background Subtraction Locations

Allowable exposure (or dose equivalent) limits for the public from the operation of a nuclear power plant are always provided in the context above natural background. Direct gamma dose from sources of natural background radiation can vary with the season, with weather conditions, day or night, and with varying conditions from the cosmos – solar flare or sun spot activity, charged-particle transport in the atmosphere, and cosmic-ray fluence rate variability from deep space. Background variability can be quite substantial from location to location, even when the two locations are relatively nearby. Background can vary by surface: over water, concrete, soil, igneous rock, granite, or chico formation. Direct gamma background radiation levels vary by surface material due to the relative concentrations of terrestrial radionuclides: ^{40}K, uranium, radium, and thorium as well as the photon scattering/absorption properties. Because of surface effects and ground variability, direct gamma dose varies with height from the surface: hence, it is customary to describe these quantities at 1-meter from the ground, just above waist-high for the average person. A slight additional contribution to direct gamma dose from natural sources arises from immersion in radon gas, just above the surface of the earth, or inside buildings.

Defining specifically what background is on any given day and any given location is a challenge, particularly in the presence of an external "man-made" source of radiation at rates "just above background." All of these factors significantly weigh-in at VYNPS when the measurement objective is to be able to detect a net 20 mrem per year in the presence of natural background which may vary from 70 to 100 mrem. Determination of the background subtraction location becomes of central importance in the compliance decision. It is for this reason, in fact, that VYNPS developed the MSLRM approach, for which the calibration function of MSLRM response to site boundary exposure rate accounts for background and directly reports net exposure rate at DR-53.

Looking to the federal regulation for some guidance about what natural background is and how to account for it, Appendix I to 10CFR50, for example, provides in detail that "background means radioactive materials in the environment and in the effluents from light-water-cooled power reactors not generated in, or attributable to, the reactors of which specific account is required in determining design objectives." This is an important definition because the USNRC has stated, implicitly, that enhanced dose from radon and decay products (which are naturally occurring) are not to be included in the contribution of net dose equivalent from operation of the plant. Other guidance documents imply that natural background radiation that results in a direct gamma dose to a member of the public be properly monitored and accounted for in the analysis for compliance with the regulatory limit.

Both VDH and VYNPS understand the difficulty in the matter of selecting a representative background radiation location. There are several specific issues that need to be addressed in the background assessment and monitoring plan, resulting from the fact that natural background radiation rates are extremely low over a quarterly interval and low, on an annual basis:

- Multiple methods should be used when feasible. MSLRM data, in real-time, provides a reasonable estimate of net background exposure rate at DR-53, and DR-52 from ^{16}N production in the reactor. This real-time measurement provides an accurate measure of at least 90% of the total net dose at the site boundary, and, it is calibrated to "self-correct" for natural background variation.

- Other detector responses, whether passive or active, placed in and around the VYNPS site to measure gross radiation exposure should be properly averaged over time, and methods should be used to detect spurious results and deal with the data in a statistical rejection test, whether for normality, an outlier, or data smoothing.

- The same detection methods used on site should be used off-site to estimate background. Similar statistical methods should be used to qualify the data for use: smoothing (time averaging), outlier detection, or identification of data that is non-Gaussian.

- Selecting the most representative background locations may take some time to determine because anomalous, non-representative features of the background distribution may take awhile to understand and evaluate. There is no reason to not start with the locations selected by VYNPS as representative of the site topology and geology at the west-side site boundary.

- Local topology effects must be accounted for, including the presence of berms, buildings constructed of naturally radioactive materials and similar other characteristics.

- Normalization to height above ground surface.

- Collocation of detectors/systems, if the State believes that it has the right to make the measurement.

Appendix D. Use of Environmental TLDs for Measurement of Low Level Radiation Dose

This appendix provides more detail on the work of G. Klemic and EML on attempts to develop ANSI standards for using TLDs for the measurement of ambient gamma radiation at or slightly above natural background levels. However, the performance criteria described below remain in draft form.

Twelve environmental inter-comparisons have been completed to date. ORAU speculates that there is a reluctance to put such an effort into this measurement modality because the expense to implement may not be worth the return. That is, to actually establish performance criteria on environmental dosimeters for measuring small environmental doses over 3 month and 12 month periods in order to produce satisfactory performance standards (e.g. B+s < 0.5) may be asking too much of the measurement modality, and beyond the existing intended use and deployment (e.g. accident analysis).

From the ANSI N13 status webpage (http://hps.org/hpssc/N13Status.html) the following current status exists for the N13.29 and N13.37 standards:

P/N13.29	ENV	**Criteria for Testing Environmental Dosimeter Performance** Status: Draft standard has been prepared.	Marko Moscovitch
N13.37	ENV	**Performance Testing and Procedural Specifications for Thermoluminescent Dosimeters** (new number was N545-1993) **Status:** Preparation of revised standard underway.	Gladys Klemic

Relevant references:

Klemic G., "Environmental Radiation Monitoring in the Context of Regulations on Dose Limits to the Public", *1996 International Congress on Radiation Protection* Volume 1, 321-328 (1996).

Klemic, G. et al. "Pilot Test of ANSI Draft Standard N13.29 Environmental Dosimetry – Performance Criteria for Testing," Environmental Measurements Laboratory, U.S. Department of Energy, EML-598 (September 1998).

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Stather J. W. et al. "Environmental Dosimetry," Proceedings of a Workshop, Radiation Protection Dosimetry, Vol. 92, No 1-3 2000.

Appendix E. Dealing with Measurement Uncertainty in Regulatory Compliance

Regulatory compliance involves decision making. A licensee is required to demonstrate by direct measurement that radiation exposures to personnel and to the public are below limits established in either the license condition with the USNRC or in this case, with the State of Vermont Department of Health. Fortunately, in most regulatory measurement cases, the instrument or method is physically capable of detecting the quantity of interest well below the limit. That is to say, measurements at 10% of the limit are extremely accurate, i.e., to within 10%. For example, consider an effluent release limit of 1 uCi/ml/yr. It would be desirable to install a radiation detector that is capable of detecting 0.1 uCi/ml/yr with a performance specification of 10% accuracy at that concentration. In many regulatory cases, this is true. And, as a result, if the regulatory compliance decision rule is applied to the mean of the measurement; total measurement uncertainty can be ignored. Regulating on the mean of the measurement has hence become the rule, and, measures are taken on the front end of the measurement process to ensure that the accuracy of the device/method is acceptably low for that particular decision rule.

There are several cases where ORAU can weigh in on this rationale:

- In the measurement and management of transuranic waste, the instrument must be capable of determining whether the transuranic alpha activity concentration is greater than 100 nCi/g to qualify as transuranic waste for disposal at WIPP. Measurement systems are installed and tested to ensure that the objectives are met. This is a very difficult measurement to make in many practical instances. As a result, discussions are held to decide whether to make the best measurement possible and then add the total measurement uncertainty (TMU) to the mean measurement result. This is a fairly straightforward negotiation to reach because the practice results in a conservative estimate of the transuranic materials that the repository and the waste generating site can manage. The risk from this approach is that the repository may fill with marginally transuranic waste instead of the "real" transuranic waste. WIPP is revisiting this practice now.

- In final status surveys it is sometimes found that the detection limit of the measurement method is above the regulatory limit. That is, the instrument is not capable of detecting a quantity of material that is to be decided upon. The industry deals with this by requiring more sampling and analysis, the development of new methods, or longer count/measurement time.

For the direct gamma dose at the VYNPS site boundary, this question is at the forefront for several reasons: the total measurement uncertainty at these net dose rate levels is large mostly because the quarterly and annual dose objectives are several times below natural background levels; the "compliance points" are established on a quarterly basis, adding that much more difficulty in making an accurate measurement using passive dosimeters; and the question of how to deal with the uncertainty in the calibration of flux (or exposure rate) to dose conversion. The technical details of these issues have been presented in this report, but, it is clear that one cannot accept simply adding in quadrature, the total measurement uncertainty terms to the mean of the measurement in the decision making process for whether 20 mrem per year has been exceeded. If this were the case, and one dutifully examined all the sources of uncertainty (at these low exposure rates, say 20% above background), then one would quickly find that a mean measurement result of 12 mrem per year above background could become 23 mrem at the 95% confidence level.

It is therefore ORAU's opinion, based on prior service to and experience with ANSI and ASTM subcommittees, that the following protocols be accepted between VYNS and VDH:

- Radiation measurement systems, for purpose of regulatory compliance, report the "mean" dose equivalent.
- Decision making should be made on the mean measurement.
- A combination of measurement methods should be used to evaluate this decision rule. Statistical tests should be considered to rule out measurement results that are statistical outliers.

Appendix F. About the Authors

Biography for Alex J. Boerner, CHP

Alex J. Boerner is currently the Health Physics and Training Manager at Oak Ridge Associated Universities (ORAU), Oak Ridge, Tennessee, serving in this role since February 2006.

Mr. Boerner previously was the Health Physics and Technical Projects Manager at ORAU from December 2002 until February 2006 and a senior health physicist at ORAU from December 2001 until January 2002. He was a senior health physicist at Integrated Environmental Management (IEM, Inc), Knoxville, Tennessee, from January 1997 through December 2001. Additional ORAU work experiences include health physics instructor from October 1986 until December 1997, health physics team leader from September 1983 until September 1986, and a health physics technician from September 1982 until September 1983. Prior to his graduate work, Mr. Boerner was a health physics technician at the Georgia Power Company Plant Hatch Nuclear Power Station, Baxley, Georgia, from September 1977 until August 1979.

Mr. Boerner received his B.S. in Biology with a minor in Chemistry from Augusta College, Augusta, Georgia, in 1977. He completed his M.S. in Radiation Biology from the University of Tennessee, Knoxville in 1982. He has been a certified health physicist (CHP) through the American Board of Health Physics since 1989.

His health physics interests include environmental monitoring, biological effects of radiation, decontamination and decommissioning of nuclear facilities, training individuals in the radiological sciences, and program management.

Biography for Jeffrey A. Chapman, CHP, PE

Jeffrey A. Chapman is a health physicist in the Health Physics and Training group at Oak Ridge Associated Universities (ORAU).

Mr. Chapman has been a practicing nuclear engineer and health physicist since 1982. He has worked with and for a number of institutes, universities, laboratories, and reactor facilities: Oak Ridge National Laboratory, Central Research Institute for Physics Budapest Hungary, Rocketdyne Space Systems California, Los Alamos National Laboratory, University of Tennessee, Yale University, and Texas A&M University. He also spent 4 years with Canberra Industries as the manager of the nondestructive assay laboratories at ORNL, Y-12, K-25, and the Paducah and Portsmouth gaseous diffusion plants. His interests are radiation shielding and transport; radiation detection measurement, and analysis; dosimetry; nondestructive assay systems engineering; environmental monitoring; homeland defense instrumentation; and nuclear nonproliferation. Mr. Chapman has calibrated, deployed, and tested radiation monitoring systems in Japan, Vienna, Russia, New York, Boston, Kirtland Air Force Base, Baltimore, and several national laboratories. He has lectured in graduate level courses at Yale University and the University of Tennessee.

Mr. Chapman received his B.S. in Physics in 1982 from Birmingham Southern College. In 1983 he completed his M.S. in nuclear engineering from Texas A&M University, and is currently a Ph.D.-ABD in nuclear engineering at the University of Tennessee. He has been a certified health physicist (CHP) through the American Board of Health Physics since 1991, and a licensed professional engineer (PE) through the State of California since 1988. He has served on several ASTM and ANSI committees for radiation detection and measurement.

Appendix G. Useful Conversion Units and Formulas for the Project

1. **<u>Micro</u>roentgen per hour to <u>milli</u>roentgen per year (μR/h to mR/y)**

   This is a straightforward conversion achieved by multiplying the number of hours in a year and converting the scale from micro to milli, a factor of 1000. The multiplier is 8.5.

   1 μR/h = 8.76 mR/y

   This conversion is particularly helpful for intercomparing TLD results between VDH and VYNPS.

2. **Exposure to Absorbed Dose (R to rad)**

   This conversion is complicated because the derived measurement or calculational concepts depend on the energy and angular dependency of the gamma-ray flux striking the object of interest. Furthermore the mass energy absorption coefficients describing the physical radiation absorption properties of the object are energy and composition dependent. Some common conversions are provided:

   1 R = 1 rad (a practical, yet conservative assumption-no technical merit)
   1 R = 0.87 rad (exposure to absorbed dose <u>in air at STP</u>)
   1 R = 0.71 rad (VYNPS calculations, 1999)
   1 R = 0.60 rad (VYNPS/ORAU calculations, 2006, per ANSI-ANS 6.1)

3. **Absorbed Dose to Dose Equivalent (rad to rem)**

   For <u>photons</u>, this conversion is by definition, unity.

   1 rad = 1 rem

4. **μC/kg/y to mR/quarter**

   Literature values of environmental TLD results are reported in the SI-based units of μC/kg/y (microcoulombs per kilogram of air, per year). This unit conveniently converts to mR/quarter (one to one).

5. **Derived Formulas for Direct Gamma Exposure as a function of distance from turbine building**

   In 1976, after a shield wall was installed around the turbine, a PIC detector was used to measure net exposure rate at radial distances from turbine building in a westerly direction, toward DR53 and DR52. ORAU performed a regression analysis on the data points, to establish an equation for the net exposure rate (μR/h) as a function of

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the distance, d (ft), from the turbine building. The fit results in a power function base 10:

$$X(net) = 100 * 10^{\left(-d/492\right)} \qquad \left(\mu R\ h^{-1}\right)$$

If the reactor were operated at 100% capacity and 100% power for one year, the annual exposure would be 20 mR at 800 feet from the turbine building, the location of the fenceline between DR52 and DR53.

Report Distribution

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