Enclosure 4

DRAFT SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

FOR FRAMATOME, INC. TOPICAL REPORT ANP-10348,

"FLUENCE METHODOLOGIES FOR SLR"

PROJECT NO. 710; DOCKET NO. 99902041

ENTERPRISE PROJECT IDENTIFICATION NO. L-2020-TOP-0045

1.0 INTRODUCTION

By letter dated July 31, 2020, Framatome, Inc. submitted Topical Report (TR) ANP-10348,[1] Revision 0, "Fluence Methodologies for SLR [Subsequent License Renewal]," to the U.S. Nuclear Regulatory Commission (NRC) for review and approval for licensing applications (Agencywide Documents Access and Management System (ADAMS) Package Accession No. ML20223A019). ANP-10348 provides improved computational methods and qualification for application of Framatome's reactor vessel neutron fluence methods to SLR applications. Framatome supplemented the TR package, providing a response to an NRC staff request for additional information (RAI), by letter dated March 19, 2021 (ADAMS Package Accession No. ML21085A743).

2.0 REGULATORY EVALUATION

This regulatory evaluation provides a brief background discussion that explains the purpose and intent of ANP-10348. Following that background, the NRC staff identifies the applicable regulatory requirements and a discussion of regulatory guidance that was applied during the NRC staff review.

2.1 BACKGROUND

Over the lifetime of a nuclear power plant, reactor vessel exposure to neutron irradiation increases. This irradiation is a damage mechanism that must be taken into account when demonstrating the integrity of the reactor pressure vessel over the lifetime of a nuclear power plant. Thus, an accurate estimation of the reactor vessel neutron fluence is necessary, and the uncertainty associated with the fluence must also be estimated. The NRC Regulatory Guide (RG) 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence" (ADAMS Accession No. ML010890301), provides guidance on methods for calculating pressure vessel neutron fluence, including qualifying the methods and estimating the calculational uncertainty, that are acceptable to the NRC staff.

The methods described in ANP-10348 are intended for applications for extensions to nuclear power plant operating licenses from 60 years to 80 years, an action known as SLR. As nuclear utilities envision operating beyond 60 years, the region of the reactor vessel where neutronic exposure becomes significant enough to require consideration, i.e., greater than 10^{17} neutrons per square centimeter (n/cm^2) for neutrons with energies greater than 1 million electron volts

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[1] Framatome submitted both proprietary version of the TR, ANP-10348P, and a non-proprietary version, ANP-10348NP. This safety evaluation refers generally to both as ANP-10348 without a designator.

(MeV; E > 1 MeV), begins to expand axially. At the time RG 1.190 and the current RG 1.190-adherent methods were developed, the methods were used to estimate fluence in a beltline region that axially extended along the vessel from the bottom of the active fuel to the top of the active fuel. At 80 years of projected operation, it becomes commonplace for the axial region where the E > 1 MeV neutronic exposure exceeds the bottom-to-top of active fuel and extend for example to nozzles and welds above the core and welds, forgings, and plates below the core.

In ANP-10348, Framatome referred to this region as above-and-below-the-beltline (ABB). The primary purpose of ANP-10348 is to describe and qualify a hybrid deterministic/Monte-Carlo method that can be used to determine acceptably accurate fluence values for ABB regions of the reactor vessel. In this context, "acceptably accurate" means that the qualification provides evidence that the predictive capability of the method is shown to be consistent with an estimated uncertainty value. The qualifying evidence Framatome used is a combination of: (1) direct qualification using comparisons to irradiated materials, (2) comparison to a lower-order analytical method, and (3) statistical inference. This qualification drew heavily on the existing methods, benchmarking, and analysis described in BAW-2241, Revision 2, "Fluence and Uncertainty Methodologies" (ADAMS Package No. ML073310649), which has been reviewed and approved for use by the NRC staff.

2.2 APPLICABLE REGULATORY REQUIREMENTS

The NRC staff review was performed in consideration of three of the General Design Criteria (GDCs) set forth in Appendix A, "General Design Criteria for Nuclear Power Plants," to Title 10, "Energy," of the *Code of Federal Regulations* (CFR), Part 50, "Domestic Licensing of Production and Utilization Facilities" (10 CFR Part 50). Specifically, GDC 14, "Reactor Coolant Pressure Boundary," GDC 30, "Quality of Reactor Coolant Pressure Boundary," and GDC 31, "Fracture Prevention of Reactor Coolant Pressure Boundary," apply.

GDC 14 states:

> The reactor coolant pressure boundary shall be designed fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

GDC 30 states:

> Components which are part of the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting, and to the extent practical, identifying the location of the source of reactor coolant leakage.

GDC 31 states:

> The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating maintenance, testing, and postulated accident conditions and the

uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state and transient stresses, and (4) size of flaws.

Conformance to these GDCs provides assurance that the reactor vessel maintains its structural integrity during conditions of normal operation, including anticipated operational occurrences, and under postulated accident conditions.

An explanation of the region of the reactor vessel requiring fracture toughness evaluations, and hence fluence estimates, is based on the requirements of 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements," and Appendix H, "Reactor Vessel Material Surveillance Program Requirements. According to NRC Regulatory Issue Summary (RIS) 2014-11, "Information on Licensing Applications for Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components," the requirements contained in these appendices establish that the beltline definition is applicable to all reactor vessel ferritic materials with projected neutron fluence values greater than 10^{17} n/cm^2 (newtons per square centimeter) E > 1 MeV (ML14149A165).

As described above, ANP-10348 makes a distinction between beltline and ABB regions of the reactor vessel. The NRC staff adopted a different convention from this point onward in this safety evaluation (SE), based on the discussion in RIS 2014-11. While the Framatome lexicon (i.e., beltline/ABB) is a geometrically succinct way to characterize the two regions, the NRC staff referred to these as "traditional beltline," meaning the vessel plates and welds axially adjacent to the core, and "extended beltline," meaning the nozzles, welds, plates, and forgings that are above or below the core. This terminology was adopted with an understanding that where an estimated fluence exceeds 10^{17} n/cm^2, that area would be considered "beltline" for regulatory purposes, consistent with RIS 2014-11, even if that region is what Framatome designates as ABB.

Framatome described additional regulatory requirements in Chapter 2, "Regulatory Requirements," of ANP-10348; however, in the NRC staff view, these additional requirements provide context for the downstream uses of fluence and uncertainty estimates and were as such not directly applicable to the present review.

2.3 APPLICABLE REGULATORY GUIDANCE

Based on GDCs 14, 30 and 31, RG 1.190 provides guidance on methods for calculating pressure vessel neutron fluence, including qualifying the methods and estimating the calculational uncertainty, that are acceptable to the NRC staff. As noted in Section 2.1, "Background," this guidance was developed in consideration of estimating fluence in the traditional beltline. In the extended beltline, the methods that RG 1.190 recommends for beltline fluence estimates may require refinement, because the geometry becomes more complex, the neutron transport distance is longer, and the neutron energy spectrum changes. This SE notes where Framatome exceeds the recommendations in RG 1.190 in order to provide an appropriate estimate of the fluence in the extended beltline.

3.0 TECHNICAL EVALUATION

This technical evaluation describes the NRC staff assessment of the technical adequacy of the Framatome SOLIDWORKS[2] – VICTORA, ADVANTG[3], MCNP [Monte Carlo N-Particle][4] (SVAM) fluence method described in ANP-10348. It addresses, in sequence, the applicable guidance contained in Regulatory Position 1, "Neutron Fluence Calculational Methods," of RG 1.190 and describes how SVAM meets, or in some cases, exceeds the recommendation. Several exceptions to RG 1.190 guidance are also noted and justified.

The SVAM method renders plant geometry developed in the commercially available SOLIDWORKS computer-aided design (CAD) tool, using a Framatome-proprietary plugin called VICTORIA that adapts the CAD model into a 3-D input for MCNP. The ADVANTG variance reduction tool is used to translate the MCNP input into a 3-D model for the Denovo discrete ordinates transport code to calculate variance reduction parameters that are then used to accelerate the MCNP solution.

3.1 MATERIALS AND GEOMETRY

Section 4.3.1, "MCNP Geometric Modeling," of ANP-10348 described the Framatome approach to model plant geometry in the SVAM methodology. Framatome rendered the SVAM plant model using a software plugin, VICTORIA, to adapt a plant model developed using the 3D SOLIDWORKS CAD tool. Framatome stated that the model is built from detailed plant drawings.

Regulatory Position 1.1.1, "Materials and Geometry," and sections of Regulatory Position 1.3.2, "Monte Carlo Transport Calculation," provide guidance for modeling the plant configuration. As appropriate and to the greatest extent possible, the 3D CAD model is an [ ] representation of the plant geometry including [ ] and the material compositions of the support structure and concrete shielding, as recommended by RG 1.190. This modeling approach provides a [ ] representation of the problem geometry, whereas the guidance in RG 1.190 provides for simplified geometry. Thus, in its review, the NRC staff determined that the Framatome approach for modeling the materials and geometry exceeds the guidance provided in RG 1.190 and is hence acceptable. This more detailed approach is also necessary and appropriate to represent the larger solution domain that includes the reactor cavity and extended beltline areas.

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2 SOLIDWORKS is developed by Dassault Systemes. https://help.solidworks.com
3 ADVANTG is developed by Oak Ridge National Laboratory. https://ornl.gov/onramp/advantg
4 MCNP is developed by Los Alamos National Laboratory. https://mcnp.lanl.gov

Regulatory Position 1.3.2 indicates that the Monte Carlo model geometry input should be checked thoroughly against the model design data to validate the definition of the model geometry. In response to RAI 3, Framatome confirmed that the plant model, once developed and converted to an ADVANTG/MCNP geometric input, is rigorously checked within MCNP to assure that the conversion is correct. Framatome indicated that the input is currently reviewed line by line by two engineers, **[ ]** This approach of model validation is necessary for MCNP model geometries and is consistent with Regulatory Position 1.3.2.

Since the NRC staff determined that the Framatome approach for modeling materials and geometry is consistent with or exceeds the guidance provided in RG 1.190, the NRC staff determined that the modeling and geometry in SVAM is acceptable. This conclusion applies to the use of SOLIDWORKS – VICTORIA to develop the model, and with the verification process described in response to RAI 3. In Section 4.3.5, "SOLIDWORKS – VICTORIA," of the TR, Framatome stated that other software packages are available that could be used to develop model geometry, and in the response to RAI 3, Framatome stated:

> **[ ]**

The NRC staff notes that use of other software **[ ]** processes was not reviewed, and hence alternatives would require either plant-specific justification as described in the quote above, or a supplement or update to this TR, which the NRC staff would need to review and approve.

3.1.1 <u>Nuclear Data</u>

Section 4.3.2, "Cross-Sections & Materials," of ANP-10348 describes the treatment of nuclear data within the SVAM methodology. Framatome relied on **[ ]** acceptable and consistent with the regulatory position.

[ ]

RG 1.190 provides further guidance for collapsed cross section libraries in Regulatory Positions 1.1.2.1, "Multigroup Libraries," and 1.1.2.2, "Constructing a Multigroup Library," but since the vendor relied on [ ], this guidance is not applicable. Essentially, the use of [ ] to estimate the fluence can be considered a higher order methodological approach than RG 1.190 recommends, which would be expected to provide a more precise result than the use of a broad group, collapsed cross section library with discrete ordinates transport methods. Based on this consideration, the NRC staff determined that the treatment of nuclear data was acceptable, since it exceeds the recommendation provided in RG 1.190.

3.2 CORE NEUTRON SOURCE

Section 4.3.3, "Source Modeling," of ANP-10348 describes how the core neutron source is modeled within the SVAM methodology. [ ]

This approach is more exact than recommended in Regulatory Position 1.2, "Core Neutron Source," of RG 1.190, which recommends representing at least the peripheral fuel assemblies on a pin-by-pin basis. While the approach would be considered acceptable for the traditional beltline because it exceeds the RG 1.190 recommendation, the NRC staff also considered the fact that these methods are being applied to the extended beltline. In such distant areas from the core, the contribution of neutrons from upper elevations of interior fuel assemblies may contribute more significantly to the accumulated fluence, and therefore the NRC staff determined that the more explicit approach used by Framatome is appropriate for extended beltline fluence estimates.

The nuclear characteristics of the core are obtained from a three-dimensional core-follow computer code. Such codes are reviewed and approved by the NRC based on extensive experimental qualification and quantified uncertainty allowances. This is consistent with both BAW-2241, and with the guidance contained in Regulatory Position 1.2, "Core Neutron Source," of RG 1.190, which indicates that the variations in the key physics parameters may be obtained from standard lattice physics depletion calculations. Based on the consistency with Regulatory Position 1.2, the NRC staff determined that the core neutron source modeling within ANP-10348 was acceptable.

3.3 TRANSPORT MODELING

This section of the SE discusses the NRC staff review of the neutron transport modeling in SVAM. The methodology relies on ADVANTG variance reduction, a subroutine of the Oak Ridge SCALE code system. [ ] Therefore, the guidance in Regulatory Position 1.3.2, "Monte Carlo Transport Solution," of RG 1.190 applies generally, although the NRC staff also reviewed the methods used by Framatome to develop the variance reduction files. Since these are developed using [ ] and hence valid variance reduction files.

3.3.1 ADVANTG/Denovo

Framatome described its discrete ordinates modeling in the response to RAI 4. In Table 2-1 of the RAI response, Framatome provided the key characteristics of the Denovo modeling that can be used to assess whether the modeling is consistent with the recommendations in RG 1.190. [ ]

For transport modeling, RG 1.190 recommends the use of minimum S8 (level symmetric) quadrature. [ ] Although this meets the guidance for the traditional beltline, RG 1.190 indicates that higher order quadrature may be needed for reactor cavity fluence calculations, and similar considerations would apply to the extended beltline. However, the NRC staff notes that [ ]. Furthermore, the ADVANTG code documentation notes [ ] For other considerations addressed in the response to RAI 4, the [ ] is consistent with RG 1.190 recommendations.

As discussed above, the [ ] was generally consistent with RG 1.190 recommendations. [ ] Based on these considerations, the NRC staff determined that the ADVANTG modeling within SVAM was acceptable.

3.3.2 MCNP

The Framatome approach for verifying the results of the MCNP model is described in the response to RAI 6. Framatome stated that MCNP automatically provides, as part of its output, information to indicate the results of a series ten statistical tests that are commonly used to determine whether the results are statistically valid. In addition, Framatome stated [ ]

Regulatory Position 1.3.2, "Monte Carlo Transport Calculation," of RG 1.190 provides recommendations concerning the adequacy of Monte Carlo calculations such as MCNP. As noted in the response to RAI 6, the MCNP output with the results of the statistical tests is consistent with the recommended statistical tests listed in Regulatory Position 1.3.2. The

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[ ]

comparison to other calculated results or measurements is addressed in Section 3.4, "Qualification," of this SE.

The recommendations in Regulatory Position 1.3.2 are provided to ensure that Monte Carlo-based methods are applied in a way that produces reliable and valid results. Since Framatome ensures that all statistical tests provided by the code are satisfied or otherwise justified, the NRC staff determined that the MCNP portion of the SVAM methodology is consistent with RG 1.190 and hence acceptable.

3.4 QUALIFICATION

Framatome relied on an extensive database of in-plant dosimetry data to qualify its fluence methods. A significant portion of the database relied on a series of in-vessel capsule dosimetry, cavity dosimetry, and vertical chain dosimetry installed at the Davis-Besse Nuclear Power Plant. Framatome chose to model the Davis-Besse reactor using SVAM in order to qualify the modeling methods. The qualification was performed by estimating the activity of each dosimeter material and comparing the estimated, or calculated, value to the measured activity. This approach resulted in a direct comparison to a laboratory measurement.

Except for the extended beltline chain dosimetry, the measurement data are contained in BAW-2241, including the traditional beltline chain dosimetry. The measurements of the extended beltline chain dosimetry data were obtained using the same methods as described in BAW-2241. The same is true of both the measurement and calculational uncertainty estimates. Thus, the review described in this section of the SE was performed in recognition that this material is either contained in, or developed using the same methods as, those described in BAW-2241.

The uncertainty analysis, qualification, and estimation of the bias and uncertainty are closely related. Therefore, the staff review is described in each subsection following the structure of RG 1.190, and then an overall review and conclusion is presented.

3.4.1 Analytic Uncertainty Analysis

To estimate uncertainty associated with each of its analytic fluence methods, Framatome used an uncertainty methodology that is described in Chapter 7 and Appendix D of BAW-2241. The method provided a means to draw inferences using its entire dosimetry database and was based on a mathematical statistics analysis that incorporated specific elements of the overall uncertainty, along with covariances to account for dependencies among the uncertainty terms. The uncertainty estimated for the SVAM analysis is described in Section 3.5, "Uncertainty Methodology," and supported by Chapter 6, "Sensitivity," of ANP-10348.

Regulatory Position 1.4.1, "Analytic Uncertainty Analysis," provides recommendations concerning the typical contributors to analytic uncertainty and methods to estimate the uncertainty. In Chapter 6 of ANP-10348, Framatome identified a consistent set of contributors and discussed each individually. In its statistical analysis, Framatome converted the calculated fluence value at the location of each dosimeter into a measured activity using a response function. While the fluence values were obtained from the legacy database, the response functions incorporated the above-mentioned sources of uncertainty based on the SVAM method. This calculation resulted in a calculated activity that is compared to the activity measured for that dosimeter. The analytic uncertainty was estimated by the standard deviation in the ratio of the measured-to-calculated fluence in the dosimetry database. While RG 1.190 recommends that computational sensitivity studies be used to estimate uncertainty, the Framatome approach was slightly more rigorous, statistically valid, and was found acceptable in BAW-2241.

To update the analysis for SVAM within the traditional beltline, Framatome used the same approach, but rather than regenerate fluence estimates using its Monte Carlo methodology, Framatome used the existing fluence estimates obtained from its legacy, discrete ordinates methods. These fluence estimates were used to calculate dosimeter responses using the uncertainty attributes and covariances associated with the SVAM methodology (e.g., Monte Carlo statistics and nuclear data uncertainty). Thus, the uncertainty was quantified not by direct analysis, but rather by statistical inference.

This approach was acceptable for several reasons. First, the discrete ordinates fluence estimates would be expected to introduce more uncertainty (variability) into the estimated fluence than an analogous SVAM calculation because the discrete ordinates geometry is more simplified, the nuclear data are less exact, and MCNP is a higher order method than discrete ordinates. Second, the response functions used in the uncertainty analysis reflected the actual uncertainty attributes applicable to SVAM. Third, the guidance in RG 1.190 recommends an estimated uncertainty and not an exact quantification. Fourth, the overall uncertainty estimate was justified by additional benchmarking. Based on these considerations, the NRC staff determined that Framatome acceptably estimated the analytic uncertainty associated with SVAM in the traditional beltline.

To estimate the uncertainty in the extended beltline, Framatome extrapolated the traditional beltline uncertainty using a [ ] that correlated the predicted dosimeter activities in the extended beltline to those in the traditional beltline. This [ ] was based on the experimental observation that outside the traditional beltline, the dosimeters are less activated and have lower measured count rates and hence poorer count rate statistics. The cause of increased computational uncertainty in the extended beltline is analogous, but not the same. The uncertainty in the computational result arises from contributors such as the increased transport distance, the change in the incident neutron energy spectrum, and an increased contribution from neutron steaming in the reactor cavity. It is reasonable, as discussed in Section 3.4.2, "Benchmarking," below, to infer that both effects would be similar in trend and magnitude, and to estimate both using the same logarithmic function.

Framatome limited the generic model uncertainty obtained from this approach to [ ] Framatome then qualified this approach by comparing the estimated uncertainty to the experimental (dosimeter count rate) standard deviations, as well as the standard deviation in the ratio of the measured-to-calculated dosimeter activity. In Chapter 3, "Measurements," of ANP-10348, Framatome illustrated that the generic model uncertainty is significantly greater than the experimental and computational uncertainty, and on that basis, the NRC staff accepted the Framatome approach. Section 3.4.2 of this SE provides more detail on the comparison of the extended beltline uncertainty to the qualification data and calculational uncertainty.

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[7] As discussed in the response to RAI 2, the areas where the generic model uncertainty is capped at [ ] This concept is addressed in the final paragraph of Section 3.4.2.

Based on the SVAM uncertainty analysis described in Section 3.5, "Uncertainty Methodology," and Chapter 6, "Sensitivity," of ANP-10348, the NRC staff determined that Framatome identified a consistent set of contributors to the analytic uncertainty estimated for SVAM as with Regulatory Position 1.4.1 of RG 1.190. Although Framatome used a different method to estimate the computational uncertainty than RG 1.190, Regulatory Position 1.4.1 describes, the method is consistent with that approved for use in BAW-2241 and is statistically valid. Therefore, the NRC staff determined that the Framatome uncertainty analysis for ANP-10348 was acceptable.

3.4.2 Benchmarking

Framatome benchmarked SVAM as described in Chapters 3, 5, "Benchmarks," and 6, "Sensitivity," of ANP-10348. The benchmarking consisted of direct comparisons to in-vessel and cavity dosimetry in the traditional beltline region, as well as to a series of ex-vessel vertical chains that extended through the traditional and extended beltline. Then, Framatome compared results obtained from SVAM to those obtained from the MERLIN methodology, which was based on BAW-2241. Finally, Framatome updated the predicted dosimeter responses from its BAW-2241 benchmarking database with new predicted values that include the uncertainty attributes of the SVAM methodology, which was described above in Section 3.4.1.

Regulatory Position 1.4.2, "Comparisons with Benchmark Measurements and Calculations," of RG 1.190 provides recommendations for appropriate validation of computational methods using measurement and calculational benchmarks. The guidance in RG 1.190 recommends comparison to operating reactor measurements, a pressure vessel simulator benchmark, and a calculational benchmark.

The benchmarking comparison using Davis-Besse measurements provided the direct, comparative qualification to operating reactor dosimetry data, consistent with Regulatory Position 1.4.2. In the beltline region, the calculated-to-measured results agreed over the benchmarking database within 7 percent, which is consistent with the measurement uncertainty set forth in BAW-2241, as well as the uncertainty estimated with the legacy, discrete ordinates methods. This is well within the recommended agreement set forth in RG 1.190, which is within 20 percent.

For the pressure vessel simulator benchmarks, Framatome relied on the existing dosimetry database, with estimated activities calculated using response functions with the SVAM uncertainty attributes described above. This database included data from the Oak Ridge National Laboratory Pool Critical Assembly benchmark, which is specifically recommended by RG 1.190. While the NRC staff acknowledges that the statistical approach is not the same as a direct comparison, using statistical inference is reasonable as described in Section 3.4.1. Framatome assessed the updated benchmarking database and concluded it remained well represented with a 7 percent standard deviation, consistent with both the legacy database contained in BAW-2241 and the direct qualification to Davis-Besse.

For the extended beltline, the deviation between measured and calculated activities increases exponentially as the distance from the active fuel increases. For the highest nozzle elevation where reliable measurement data were obtained, the experimental standard deviation was [

] Similar trending occurred below active fuel; the generic model standard deviation greatly exceeded both the experimental and calculational deviation. For several entries in Table 3-5, "Measurement Uncertainties," of ANP-10348, the NRC staff combined the laboratory and measurement model standard deviations in quadrature, as Framatome did with the traditional beltline benchmarking data. The results are consistent with the generic model standard deviation, which indicated that the generic model standard deviation in the extended beltline was an acceptable estimate.

For the upper and lower extents of the chain dosimetry, Framatome was unable to obtain reliable experimental count rate data. The response to RAI 2 explains that the bad data resulted from the very low activities of these dosimeters, indicative of low incident neutron flux. Framatome also illustrated that these locations were outside any region where the fluence would exceed 10^{17} n/cm^2 over an exposure period of 80 calendar years of operation. The low activation associated with the extreme high and low locations of these particular dosimeters is consistent with the expectation that materials in these locations would not receive significant exposures, and thus the NRC staff determined that their exclusion from the database was acceptable.

Regulatory Position 1.4.2 recommends additional benchmarking using two computational problems, which allow for evaluation of certain aspects of the uncertainty associated with the fluence estimate by fixing the geometry, materials, and space- and energy-dependent source. Framatome did not address this element of the benchmarking guidance. The NRC staff accepts this omission based on three considerations. The first is that the problem geometry and core neutron source specification are much more exact in the SVAM modeling approach than with the discrete ordinates methods that have become prevalent since publication of RG 1.190, and the second is neither of the computational problems represent the Babcock and Wilcox geometry that Framatome modeled. The third is that, while comparison to the computational benchmarks can help to estimate uncertainty associated with elements like the modeling geometry and materials, the Framatome uncertainty analysis accounts for those parameters explicitly. Based on these considerations, the NRC staff did not request that Framatome provide a computational benchmark.

Based on the comprehensive nature of the benchmarking described in ANP-10348, the NRC staff determined that Framatome has adequately benchmarked the SVAM methodology, consistent as appropriate with RG 1.190. Therefore, the NRC staff determined that the Framatome benchmarking was acceptable.

3.4.3 Bias and Uncertainty Estimates

Framatome drew on the qualification database described in BAW-2241 to apply a multiplicative bias correction that is based on neutron energy, and to estimate the uncertainty associated with SVAM estimates. For the extended beltline uncertainty, Framatome estimated the value by extrapolating the beltline uncertainty in a statistical function that was correlated to the chain dosimetry activation. Because the dosimetry specimen further from the core have lower activities, their measured count rate uncertainty is higher. Analogously, the increase in mean free paths, complex material geometry, and spectral changes in incident neutron energy over the increased mean free paths, render the projected fluence in extended regions less certain than in the beltline.

3.4.3.1 <u>Bias Function</u>

Framatome applied the same function described in BAW-2241 to eliminate bias in the SVAM fluence estimate. The vendor asserted that the bias is associated with the core physics, which are an input and not a direct result of the transport modeling. In the present review, the NRC staff reviewed the SVAM-MERLIN (discrete ordinates) comparison and observed that, while comparison was only made to four dosimeter responses, the comparison suggested the presence of an energy-dependent variation in the results, which indicated that the bias function employed within MERLIN (i.e., BAW-2241) may not have been directly applicable to SVAM. To investigate the issue, Framatome computed results both with and without the bias removal and concluded that the SVAM results calculated with the bias removed would lead to a higher estimated fluence value, and hence that the bias removal remained somewhat conservative.

The reason that the bias removal function resulted in a slight overprediction of the total fluence is because the bias function has the effect of applying a net positive multiplicative factor to the fluence on the lower end of the energy spectrum, which accounts for an outsize portion of the total neutron fluence. In the response to RAI 5, Framatome performed a sensitivity study to compare the unbiased, predicted vessel inner surface flux, to unbiased values obtained using the BAW-2241 multigroup bias factor, and to unbiased values obtained using the comparative data from the Davis-Besse qualification. The table showed that the bias function from BAW-2241 continued to introduce a slightly conservative effect relative to the other two approaches and indicated that the BAW-2241 function would continue to be used within SVAM. Since this approach introduced a slight conservatism in the estimated fluence, the NRC staff determined it was acceptable.

3.4.3.2 <u>Uncertainty Estimate</u>

The methods Framatome used to estimate both the experimental and calculational uncertainties are described in the preceding sections of this SE. Framatome uses a standard value of **[** **]** as an estimate for both the experimental and computational components, although the uncertainty analysis and qualification database suggest a slightly lower value would be applicable to both. Framatome combines both sources of uncertainty for a total estimated uncertainty of **[** **]** for the traditional beltline. Framatome applied the generic model uncertainty in the extended beltline as described in Sections 3.4.1 and 3.4.2. Based on the data that Framatome provided, the NRC staff determined that the analysis and qualification support these uncertainty values.

Regulatory Position 1.4.3, "Estimate of Fluence Calculational Bias and Uncertainty," permits the application of a multiplicative bias function and recommends that a fluence uncertainty of 20 percent (one standard deviation) is acceptable for application within the formulas to determine reference temperatures for nil-ductility transition and pressurized thermal shock.[8] The fluence uncertainty and bias functions in the traditional beltline were fully consistent with these recommendations and hence acceptable. Although RG 1.190 does not apply to the

[8] These parameters, known as RT_{NDT} and RT_{PTS}, are the most typical uses for reactor vessel fluence estimates.

extended beltline, the estimated uncertainty was technically acceptable as described above. Therefore, the NRC staff determined that the bias and uncertainty for ANP-10348 is acceptable.

3.4.4 Qualification Conclusions

As described in the preceding sections, the NRC staff reviewed the Framatome uncertainty analysis and benchmarking. The qualification was based on rigorous statistical uncertainty analysis that was described in previously approved BAW-2241 and updated to reflect the uncertainty attributes associated with SVAM. Framatome showed, using direct comparative qualification for Davis-Besse, as well as statistical inference concerning the remainder of its dosimetry database, that SVAM performed consistently with MERLIN and other, legacy methods within BAW-2241. For the reactor vessel traditional beltline, the benchmarking and uncertainty demonstrate that the SVAM predictive capability was well within the 20 percent recommended by RG 1.190.

In the extended beltline, Framatome extrapolated a generic uncertainty that bounded both the experimental and computational uncertainties predicted for the region. Since the estimated uncertainty exceeds both values and was consistent with the quadrature sum of those values, the NRC staff determined that the uncertainty was acceptable. RG 1.190 does not provide a quantitative acceptance criterion for methodological uncertainty in the extended beltline, but the NRC staff determined the generic model standard deviation was an acceptable estimate.

The NRC staff recognizes two practical shortcomings in the Framatome qualification. The first is that the SVAM model was only demonstrated for Davis-Besse, whereas its capabilities were estimated using statistical inference for other plant designs. In the response to RAI 2, Framatome concluded that no further applicant / licensee actions are required to broaden the applicability of ANP-10348 to other reactor types. While the NRC staff agrees that the statistical-based qualification database is a reasonable means to infer the calculational uncertainty as described in Section 3.4.1, a direct comparison to dosimetry data remains necessary for an appropriately qualified fluence methodology. Therefore, applying the model to other plants requires a direct comparison to available dosimetry data for that plant design. This information is routinely included in licensing applications, in which licensees submit fluence estimates to the NRC staff for review and approval, and such information is expected to be included in requests to apply SVAM for other reactor designs. This TR addresses the acceptability of application of SVAM to Babcock and Wilcox reactor designs, like Davis-Besse.

The second shortcoming is that the uncertainty estimated in the extended beltline was performed using experimental data that are not widely available; such data would not necessarily be expected to be available for other reactor designs. However, the current NRC staff practice when considering extended beltline fluence evaluations is to evaluate such applications based on additional considerations specific to each application. Therefore, the SVAM methodology can be considered acceptable in the extended beltline using similar uncertainty analysis methods as described in Chapters 3 and 5 of ANP-10348 without further, direct qualification, provided the traditional beltline qualification described above continues to show statistical consistency with the existing, BAW-2241 database. Such further traditional beltline qualification can provide reasonable evidence that the uncertainty estimated for the extended beltline is similarly valid.

Based on the uncertainty analysis, benchmarking, and bias an uncertainty estimates described in ANP-10348, the NRC determined that the Framatome qualification was acceptable, in that it was largely consistent with the guidance and recommendations of Regulatory Position 1.4, "Methodology Qualification and Uncertainty Estimates," of RG 1.190, noting the exceptions described and justified above. Therefore, the NRC staff determined that Framatome has acceptably qualified the methods described in ANP-10348.

4.0 LIMITATIONS AND CONDITIONS

The NRC staff reviewed the methodology and uncertainty associated with application of SVAM in the traditional and extended beltline regions of the core, as described in ANP-10348. The NRC staff determined that the method is acceptable for application as follows:

- The model was demonstrated to be applicable to traditional and extended beltline regions for Babcock and Wilcox reactor vessel geometries. The application of SVAM to model other reactor vessel geometries must be justified with direct comparative qualification data and a corresponding update to the applicable portion of the dosimetry database to confirm the statistical consistency between SVAM and the legacy fluence methods described in BAW-2241. This consideration is addressed in further detail in Section 3.4, above.

- The review was based on the SVAM methodology, which specifically includes the SOLIDWORKS – VICTORIA approach for developing MCNP model geometry. The use of other approaches to develop model geometry must be described and justified with qualification data. This consideration is addressed in further detail in Section 3.1, above.

- Framatome stated that a continuous energy bias removal function could be developed and implemented within SVAM. Such an approach was not considered in the present review and is not considered approved for use by the NRC staff. The use of SVAM with a continuous energy bias removal function must be justified in any such application. This consideration is consistent with the Framatome response to RAI 5, as evaluated in Section 3.4.3, above.

5.0 CONCLUSION

Based on the considerations discussed in Section 3 of this SE, and subject to the limitations and conditions identified in Section 4, the NRC staff determined that the SVAM methodology described in ANP-10348 is acceptable. The methodology is generally consistent with the applicable regulatory positions in RG 1.190, and as such, provides acceptable fluence estimates for the traditional beltline, in partial fulfillment of the requirements contained in GDCs 14, 30, and 31. While RG 1.190 does not provide detailed guidance for fluence estimates specifically in the extended beltline, the NRC staff review determined that the Framatome fluence prediction and associated, estimated uncertainties are technically appropriate and acceptable for use for downstream evaluations, such as for those supporting SLR applications.

Principal Contributor: Benjamin Parks

Date: