AREVA

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Response to a Request for Additional Information Regarding BAW-2241(P), Appendix G, "Fluence and Uncertainty Methodologies"

Ref. 1: Letter, Jerald S. Holm (Framatome ANP) to Document Control Desk (NRC), "Request for Review and Approval of Revision 1 of Appendix G to BAW-2241(P), Revision 2, 'Fluence and Uncertainty Methodologies'," NRC:05:023, March 31, 2005.

Ref. 2: Letter, Ronnie L. Gardner (Framatome ANP) to Document Control Desk (NRC), "Response to a Request for Additional Information Regarding BAW-2241(P), Appendix G, 'Fluence and Uncertainty Methodologies'," NRC:05:044, July 18, 2005.

Framatome ANP requested the NRC's review and approval of Revision 1 of Appendix G to topical report BAW-2241(P), Revision 2, "Fluence and Uncertainty Methodologies" in Reference 1. A request for additional information was provided by the NRC in a facsimile on June 17, 2005. The questions and responses to the request were provided to the NRC in Reference 2.

A second request for additional information was provided by the NRC in an email on September 8, 2005. The questions and responses to this request are provided in Attachment A to this letter.

Framatome ANP considers some of the material contained in the attachment to be proprietary. The affidavit submitted with the original topical report satisfies the requirements of 10 CFR 2.390(b) to support withholding of the information from public disclosure.

Sincerely,

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Set 2, RAI 1

In view of the increased void fraction (relative to PWRs) in the regions above the core and in the upper core and downcomer, justify the use of the [ ] in BWR applications. Are the BWR models more likely to have negative fluxes (as experienced in Model C of the Davis Besse PWR evaluation) and, if so, how will this be treated? How are the void fractions in these upper regions determined and what fluence uncertainty is introduced by this determination?

Response

RAI 1, Set 2 has four parts: [ ] negative fluxes, void fractions, and the upper region uncertainty. While the additional information that is provided is somewhat related, each part also involves information that is separate and unique from the other parts. Consequently, each part is addressed separately in a paragraph specifically concerning that part.

[

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The negative fluxes encountered in the *r, z* Model C DORT calculation of the Davis Besse benchmark evaluation occurred in the upper nozzle and seal plate regions, above the top of the beltline, in the air cavity between the vessel and the concrete. These flux values were determined to be the result of computer-memory-related inabilities to specify a large enough angular quadrature and/or small enough spatial mesh intervals. If the mesh and quadrature intervals are too large in either the PWR or BWR models, then negative fluxes and other indications of inaccurate results would be expected. However, the computer technology continues to advance with greater and greater capabilities in processing calculations like those found in DORT models. Today, neither PWR nor BWR models would be expected to have negative fluxes. Calculations of the nozzle and seal plate regions following the Davis Besse benchmark evaluation have not had negative flux values and none would be expected in the future.

The void fractions in the upper regions, beyond the top of the fuel, are determined from the volumetric weighting of the void fractions exiting each fuel assembly. The model for the void fraction is a three-dimensional core-follow calculation. The key to the core-follow calculation is that it is a time-dependent benchmark of the actual operation of the core throughout each cycle. The inputs to the core-follow calculations are the measured core parameters, such as the control rod positions, system flow, etc. The outputs from the core-follow calculations are power distributions that may be directly compared to

measurements. The accuracy of the void fractions and power distributions is represented by a standard deviation of [ ] in the power.[H3] The void fraction – power uncertainty is discussed in the additional information provided in the response to RAI 1, Set 1.

The upper shroud, dome-head region contains the homogenized void fraction from the assembly weighted void fractions. In this region [

] Calculations of this region have shown no negative flux values.

The fluence uncertainty from the top of the active fuel to the shroud dome-head varies from a value of [ ]. The [ ] value is representative of the core-follow benchmark comparison, while the [ ] value is representative of dosimetry benchmark comparisons combined with analytic uncertainties. The development of the components of the [ ] value is presented in *Appendix G* on pages G - 49 and G - 50. The development is further discussed on page H - 18 in the additional information provided for RAI 6, Set 1. [

] the fluence uncertainty in the region above the top of the active fuel is defined as being representative of [ ].

Set 2, RAI 2

How will core/vessel/dosimetry configurations that do not have sufficient symmetry to allow a 45-degree sector representation be treated?

Response

The FANP modeling of the core, vessel and dosimetry is explicit. As discussed in *Appendix G*, Section G.3.2, "Neutron Transport Through Jet Pumps", if there is some complexity resulting from the geometrical shape of an object and the model coordinates, then [

] Consequently, if the core/vessel/dosimetry configurations do not have sufficient symmetry to allow a 45-degree sector representation, then the model will be expanded to a 90-degree treatment or whatever angular mesh representation would be appropriate.

Set 2, RAI 3

Describe the differences between the BWR and PWR in-vessel and cavity dosimetry (dosimetry wires/foils, holder tubes, encapsulation, etc.) and how these differences will be accounted for in the BWR models. For example, how will the dosimetry perturbation and correction factors of Appendix B be determined in the case of BWRs? Is additional uncertainty introduced by these differences?

Response

There are various differences between the BWR and PWR in-vessel and cavity dosimetry, *i.e.*, dosimetry wires/foils, holder tubes, encapsulation, etc. Moreover, there are various differences between the various PWRs with respect to the in-vessel and cavity dosimetry. However, as discussed in the response to RAI 2, Set 2 above, FANP does not use modeling approximations to treat the dosimetry wires/foils, holder tubes, encapsulation, etc. FANP uses explicit modeling. Therefore, every different characteristic of the BWR in-vessel and cavity dosimetry is accounted for in FANP's modeling.

The *Appendix C* dosimetry perturbation factors are a good example of the explicit modeling that FANP uses in its fluence analysis. The Davis Besse benchmark evaluation included dosimetry measurements of the reactor support beams, the inlet and outlet nozzles, and the seal plate. (Pages 3 - 16 and 4 - 16 through 4 - 18 illustrate the locations of the beams and nozzles, and the dosimetry.) The modeling of the beltline dosimetry only required single channel synthesis. However, the modeling of the beams and nozzles required multi-channel

] As explained in Section 3.2, "DORT Perturbation Calculations", the beams and other cavity structures were explicitly modeled as were the dosimetry wires/foils, holder tubes, encapsulation, etc. The *Appendix C* dosimetry perturbation factors represented the ratio of the DORT results from the multi-channel synthesis model above the beltline to the DORT results from the single channel synthesis model below the beltline.

The *Appendix B* correction factors treat effects such as photofissions, impurities, dosimetry self absorption, etc. The treatment of these effects will be independent of whether the dosimetry is associated with PWRs or BWRs.

In general, the reason for using multi-channel [ ] synthesis rather than a single channel model is due to the non-separable complexities that are part of the fluence evaluation. In BWR models these complexities include channel voiding and control rods that result in a non-separable flux function. In the PWR model for Davis Besse these complexities included nozzles and support beams. As discussed in *Appendix G*, Section G.4.2, "Calculational Uncertainties", and specifically the "Analytical Sensitivity" in Section G.4.2.4, there are uncertainties introduced by differences in the BWR design

that are not part of PWR designs. Moreover, as noted in the additional information provided in the response to RAIs 1 and 6, Set 1, the benchmark to PWR dosimetry is represented by a standard deviation of [ ] while the benchmark to BWR dosimetry is represented by [ ]. Clearly differences between the BWR and PWR designs and operation result in additional uncertainties in the calculations of in-vessel and cavity dosimetry.

Set 2, RAI 4

Provide justification for any differences between the proposed dosimetry response methods and those described in the corresponding ASTM standards.

Response

The dosimeter measurements conform to the applicable ASTM standards. The discussion of the "Measurement Methodology" in Section 5.0, and the discussions of the "Measurement Techniques" for (1) fissionable and activation radiometric dosimeters in Sections 5.1.1 and 5.1.2 respectively, and (2) helium accumulation fluence monitors in Section 5.3.1, describe how the techniques and procedures comply with the ASTM standards. The ASTM standards include additional standards for "Spectrum Adjustment Methods", "Application for Reactor Vessel Surveillance", etc. These additional standards refer to techniques that differ from those explained in (a) the "Semi-Analytical (Calculational) Methodology", in Section 3.0, (b) the "Extension of Fluence Methods" for BWRs in Section G.3, (c) the "Uncertainty Methodology", in Section 7.0, and (d) the "Uncertainty Update" for BWRs in Section G.4. These additional ASTM standards refer to pseudo-measured fluence values, and to precision, bias, and uncertainty in terms that

are conflicting. ASTM standards that deviate from experimental practice as noted above are not used.

Set 2, RAI 5

Because of the strong exponential fluence attenuation, the calculation of the fluence is sensitive to both the distance separating the core and the vessel and the barrel thickness. What quality assurance procedures will be used to insure that these dimensions are accurate and within the uncertainty assumed in the Section G.4.2 fluence calculation uncertainty analysis?

Response

It was found that the ASME standards prescribed the acceptable tolerances when determining what was appropriate for the reactor pressure vessels and vessel internal structures, such as the shroud (barrel). Subsequently, it was found that manufacturing organizations met the prescribed ASME standards and that the tolerances were either noted on the respective drawings for the shroud, vessel, etc, or the drawings noted conformance with the ASME standards. When FANP develops a fluence model for a particular reactor, the drawings are reviewed and a quality assurance check performed. Not only are the nominal, best-estimate cylindrical dimensions obtained for each component's inside and outside diameter, but the tolerances are also obtained. These tolerances include inner and outer diameters as well as eccentricity, concentricity, ellipticity and parallelism. Thus, the complete three-dimensional tolerance characteristics of the diameters are known for each shroud and vessel. This ensures that the sensitivity of the fluence calculations to the strong exponential fluence attenuation in the distance

between the core and shroud (barrel), and shroud and vessel is appropriately treated with the fluence uncertainty.

Set 2, RAI 6

The PWR analysis included in Equation 7.25 provides an additional uncertainty for the temporal extrapolation to End-of-Life (EOL). Provide the corresponding EOL extrapolation uncertainty for the Appendix G BWR analysis.

Response

In 1961, when the ASTM established a standard for reactor vessel surveillance, ASTM E 185-61, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels", FANP (formerly Babcock & Wilcox) developed an integrated program to monitor vessel material test specimens (Reference 14, Section 8). Each of the 11 reactors would monitor the vessel only twice during the operational lifetime of 40 years. Vessel material characteristics at EOL would be determined through the combined characteristics of all test specimens.

When the NRC implemented 10 CFR 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements" in 1973, the monitoring of vessel materials continued to reflect combinations of multiple cycles with predictions of EOL vessel characteristics. However, there was the requirement to define an EOL fluence value and a comparable uncertainty (Reference 12, Section 8). It was evident that test specimen uncertainties and extrapolated vessel fluence uncertainties would not be the same.

The problem with defining an EOL fluence value was that there were no restrictions on reactor operation and core fuel management with respect to fluence – embrittlement damage to the vessel. Therefore, a hypothetical equilibrium cycle was defined to have the identical neutronic characteristics as the last group of monitored cycles. The uncertainty in the fluence extrapolated to EOL would then be the uncertainty in the last group of monitored cycles combined with the power uncertainties in the equilibrium cycle. Equation 7.25 includes the estimation of the power uncertainties in the equilibrium cycle combined with the standard vessel fluence uncertainty from DORT monitoring calculations based on core-follow measurements.

FANP supported 5 of the first 7 reactors that were granted a renewed license for 60 years of operation. Part of the 60 year licensing requirements was to update the EOL vessel fluence and to estimate the uncertainty in this extrapolated fluence. Based on more than 20 years of operation, in more than 50 reactors, it is obvious that the equilibrium cycle was just a hypothetical concept with no quantitative validity. However, to address license renewal RAIs, the propagation of uncertainties from "perturbed" equilibrium cycles was estimated. This uncertainty propagation provided the vessel fluence uncertainty in "un-monitored" cycles that were slightly perturbed from the monitored reference cycle. See Table D – 2, "Vessel Fluence Uncertainty Propagation" on page D - 20 of Reference H7.

FANP no longer supports the estimated uncertainty in un-monitored cycles. Consequently, the results from Equation 7.25 are not significant with respect to fluence uncertainties. FANP has developed a cycle-by-cycle monitoring program that bounds the estimate of the vessel fluence and has a precisely defined uncertainty. Nonetheless, to

address this RAI, the EOL uncertainty has been calculated using Equations G.18 and 7.25. The standard deviation is 18.99%.

Set 2, RAI 7

Operation with MELLLA+ can affect the conditions in the downcomer. How will these changes be accounted for in the fluence evaluation?

Response

The downcomer water properties are explicitly modeled in the core-follow simulation of each BWR's operating cycle. The downcomer water properties are the initial conditions for the water entering each fuel assembly. If the water properties are not accurate, the void – power relationship in each nodal volume will be inaccurate. Any such inaccuracies would be obvious in the core-follow benchmark comparison of calculated powers to measured values.

To replicate the core-follow benchmark to measurements and maintain the same degree of accuracy in the void – power relationship, the downcomer water properties as well as the nodal water properties are exactly duplicated in the DORT model for the fluence calculations. To exactly duplicate the core-follow downcomer water properties in the DORT model requires the integration of the core-follow time-steps. Thus, with each change in the conditions in the downcomer due to operation with MELLLA+, the water properties are explicitly represented. The time-step to time-step changes in the core-follow model are directly integrated over the time period to obtain time-averaged water properties for the DORT fluence model.

As the NRC noted in RAI 5, Set 2, there is a strong exponential attenuation in the fluence rate that is sensitive to the downcomer geometry between the shroud (barrel) and vessel. This strong exponential attenuation is also sensitive to the downcomer water properties. [

] This concept is further explained in the additional information provided in the response to RAI 9, Set 2.

Set 2, RAI 8

Recognizing that v/κ (ratio of neutron production rate to power) depends on fuel isotopics and burnup, how will this dependence be included in the BWR core neutron source? Describe how the effect of increased Pu in the high burnup fuel is included. Does this treatment allow for the cycle-specific variations? In view of the large variation in fuel burnup between fuel bundles and the dependence of the number of neutrons produced per fission (v) on fuel burnup, what uncertainty is introduced by neglecting this dependence in Equation (4.1)?

Response

In the explicit three-dimensional core-follow model of reactor operation each assembly is represented, generally with one-quarter core symmetry. Moreover, each assembly is divided into nodal sections along the axial length of the fuel. The axial length of each node is 6 inches or less. Thus, for a fuel stack height of 150 inches, there would be 25 nodes representing each assembly. The core-follow model explicitly treats the

neutron production rate and power production within each nodal volume. Consequently, the burnup of the fuel is explicitly modeled as are the resulting isotopic transmutations. [ ] the ratio of the neutron production rate to power production is explicitly modeled. This includes the nodal burnup and the related isotopic effects on the neutron and power production. As noted above in the response to RAI 7, Set 2, the explicit modeling of the ratio of the neutron production rate to power production in the core-follow model involves discrete time-steps. To replicate the explicit core-follow modeling in the DORT calculation, the neutron and power production rates are integrated over the core-follow time-steps. A time-averaged ratio of the neutron production rate to power production is thereby modeled in the DORT calculation.

The effect of the increased Pu on the neutron source calculation is a time-dependent effect that increases with higher and higher fuel burnups. The time dependence of the macroscopic cross sections (isotopics) and the neutron source eigenfunction (neutron emission rates by isotope) are treated with the integral of the macroscopic cross sections and source function over the time periods of interest.

The time dependence of the DORT isotopics, including the plutonium, is based on a quasi-static, core-follow calculation of the plant operation. The quasi-static calculational results, such as the Pu fission rates, are determined for each time-step. The results within each time step are considered static (independent of time), but the results, such as the Pu concentrations, vary from time step to time step. The Pu fission - neutron emission rates are determined within the three-dimensional nodal volume for each time step.

The integral over time is not specifically identified in Equation 3.2, but the process of determining the plutonium isotopic sources from the fission – emission rate includes

time-average weighting of the ratio of neutron emission rates to power production. Thus, the Pu isotopes as a function of burnup (time at power) are directly included in the calculations of the neutron source for each node. Since the Pu isotopics and reaction rates are determined as a quasi-static function of time, using discrete time steps which explicitly follow the core operation from cycle-to-cycle, the multi-cycle variations of Pu effects on the source are explicitly included in the calculations.

The variation in fuel burnup between nodes and thereby between assemblies is modeled explicitly. This modeling includes, core-follow calculations which match the measured operational data, quasi-static time steps to appropriately treat time dependent behavior, explicit representation of the isotopics within the each node, and three-dimensional representation of the fuel pin nodal segments. Thus, the dependence on the changing isotopics as a function of burnup, and the corresponding changes in the number of neutrons produced per fission in the fuel volume are not neglected. The burnup dependence of the neutrons produced per fission within a node is included in the neutron source calculation. However, Equations 4.1 and 4.2 (now Equations 3.1 and 3.2) on pages 3 - 11 and 3 - 12 respectively, have consolidated the expressions for the ratio of neutron production rates to fission power such that it is not clear how the neutron source in each node of the assembly is represented.

The ratio of neutron production rates to fission power is a weight applied to the nodal emission spectra. It is also combined with the normalized spatial power density and renormalized to represent a relative spatial source density. Finally, the absolute (not-normalized) source density for the core is determined from the integral of the spatial source density over the volume of the fuel.

While the burnup dependence of the number of neutrons produced per fission is "not neglected" in the calculations of the neutron source, there are combinations of source term components, expressed by Equation 4.1 (now Equation 3.1), that represent approximations. If the effect of neutron production per fission in the fuel is treated as an isolated component of the uncertainty, it would be directly related to the uncertainties in fuel isotopics.

The uncertainty in the neutron production per fission can be bounded by the isotopes producing the most neutrons per fission and the least neutrons per fission. This uncertainty was modeled in the analytic sensitivity evaluation and represents a relative deviation of nearly 20% with a 99% degree of confidence.

The uncertainty in neutron production due to the uncertainty in the nodal burnup of the fuel can be modeled with the uncertainty in the power distribution. The uncertainty in the power distribution is represented by a normal distribution when it is defined on the basis of an absolute deviation in the relative power distribution. Using an upper bounding deviation with a 95% confidence level in the analytic sensitivity indicates that the local uncertainty would be about 18% with a relative peripheral power of 0.50, and about 30% with a relative peripheral power of 0.30.

Set 2, RAI 9

The method of Section G.3 [

] assumes a simple correlation, based on core-follow calculations, [

]

provide planar comparisons of [

]. The comparisons should be made for typical planes in the upper region of the core where there is substantial voiding. Comparisons should be provided for a range of BWR conditions including (a) power distribution and (b) cycle fuel burnup. In addition to the bundle-wise comparisons, provide the percent mean and standard deviation [ ] for bundles in the outer three rows of the core.

Since the treatment of boundary conditions and core leakage has a substantial dependence on (a) the core boundary geometry (e.g., number of bundles with two faces to the reflector) and (b) the specific core-follow code used to determine the correlation, comparisons should also be provided for various core boundary shapes and the core-follow codes which will be used to determine the correlation

Response

RAIs 9, 10, 11 and 13 in Set 2 are all related. Moreover, as discussed in a telephone conference call with the NRC [

]. Consequently, there is a misunderstanding of the information presented in the topical. The NRC has subsequently requested additional information based on the misunderstanding. FANP however cannot provide the additional information requested in RAIs 9, 10, 11 and 13 from Set 2 because analyses were not performed as the NRC assumed. Therefore, the NRC agreed that it would be appropriate to send the additional information that clarified the misunderstanding without providing the specific data that was requested.

The explanation that follows therefore provides additional details concerning the explanation of the leakage function. It also explains how the [

] The explanations in response to this RAI (9, Set 2) are also part of the response to RAIs 10, 11 and 13 in Set 2. Thus, the response to RAIs 10, 11 and 13 will entail much less information and will use this RAI's explanations as a reference.

[

]

[ ] (G.7)

[

] Therefore, the NRC wanted data that would confirm that the Equation G.7 relationship was valid.

[

]

[ ] the water density in each nodal and downcomer region, at each time-step, is determined from core-follow calculations as suggested by the NRC and as explained in the response to RAI 7, Set 2: The water properties are explicitly modeled in the core-follow simulation of each BWR's operating cycle. Moreover, if the water properties were inaccurate, the void – power relationship in each nodal volume would be inaccurate. Inaccurate powers would be obvious in the core-follow benchmark comparison of calculations to measurements. [ ] As noted in Section G.3.3, if the DORT results with average time-weighted parameters are not the same as the time-averaged results from the core-follow calculations, then the approximations associated with the DORT models and procedures are insufficient.

[ ]

[ ] (G.5)

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Equation G.4, on page G - 14 of *Appendix G*, represents the "steady state" neutron transport equation. Its solution provides "steady state" simulations of reactor operation in that it provides a flux solution for time periods that are long compared to the neutron precursor half-life and short compared to isotopic depletion and thermal - hydraulic feedback effects. The time-eigenvalue for this quasi-static state is hypothetically included in the source eigenfunction. If fluence analyses were to be based on the quasi-static form of Equation G.4 there would be no issue [ ] Quasi-static - core-follow analyses based on Equation G.4, with the source eigenfunction appropriately expanded to include the time-eigenvalue, can determine the flux [ ] This flux value however is only valid for the appropriate time-step. To consider a complete operating cycle, or multiple cycles of operation, multiple time-steps are required. [

]

Utilizing multiple time-steps in a fluence analysis is not an effective utilization of resources. Moreover, with the appropriate neutron physics methods there is no technological benefit to employing multiple time-steps to develop the flux solution. By expanding the Equation G.4 solution to represent long time periods, the effects of isotopic depletion and thermal – hydraulic changes may be appropriately treated. [

]

Equation H.10 below is Equation G.4 expanded to include the time-dependent integral for multiple operational periods with variable isotopic concentrations.

$$\int_t \mathbf{\Omega} \bullet \nabla \phi(\mathbf{r}, \mathrm{E}, \mathbf{\Omega}, t)\, dt + \int_t \Sigma_{\mathrm{T}}(\mathbf{r}, \mathrm{E}, t)\, \phi(\mathbf{r}, \mathrm{E}, \mathbf{\Omega}, t)\, dt = \int_t S(\mathbf{r}, \mathrm{E}, \mathbf{\Omega}, t)\, dt \tag{H.10}$$

Average time-weighted terms may be defined to represent the integration of the second term on the left side of Equation H.10 and the term on the right side. In RAI 8 the NRC requested additional information to ensure that the time-averaged source term had included all the variables that are represented by the core-follow burnup calculations. FANP responded that every variable and parameter in the core-follow analysis is precisely represented in the DORT fluence analysis. Thus, a time-averaged source term, $S(\mathbf{r}, \mathrm{E}, \mathbf{\Omega}, \bar{t})$, may be defined to represent the integration over multiple operational periods.

The time-averaged source term is used in Equation G.4 to determine the time-averaged flux { fluence rate, $\phi(\mathbf{r}, \mathrm{E}, \mathbf{\Omega}, \bar{t})$ }. Moreover, the time-averaged collision rate, $\Sigma_{\mathrm{T}}(\mathbf{r}, \mathrm{E}, \bar{t})\ \phi(\mathbf{r}, \mathrm{E}, \mathbf{\Omega}, \bar{t})$, may be defined to represent the integration of the second term on the left side of Equation H.10. The total macroscopic cross section $\Sigma_{\mathrm{T}}(\mathbf{r}, \mathrm{E}, \bar{t})$ includes the integrated isotopic concentration changes due to depletion and thermal – hydraulic effects. In addition to the time-averaged source term, the time-averaged collision rate is used in Equation G.4 to determine the fluence rate.

The last time-integrated term that would provide the means of using Equation G.4 to determine the fluence rate is the leakage function. This is the first term on the left side of

Equation H.10. [

] Thus, the leakage rate is expressed by the function $\mathbf{\Omega} \bullet \nabla\phi\left(\mathbf{r}, \mathrm{E}, \mathbf{\Omega}, \bar{t}\right)$ when using Equation G.4 to solve for the fluence rate.

[

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[

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To resolve the issue of whether the time-averaged leakage function $\mathbf{\Omega} \bullet \nabla\phi\left(\mathbf{r}, \mathrm{E}, \mathbf{\Omega}, \bar{t}\right)$ is an appropriate expression to be used in Equation G.4 to represent the solution of the fluence rate $\phi\left(\mathbf{r}, \mathrm{E}, \mathbf{\Omega}, \bar{t}\right)$, [

]

[

]

[

[ ] (H.11)

[

.] There are two clear implications from the results of Equation H.11. The first is that the approximation of the time-averaged leakage function $\mathbf{\Omega} \bullet \nabla\phi\left(\mathbf{r}, \mathrm{E}, \mathbf{\Omega}, \bar{t}\right)$ would not be adequate if the [

'.] The second is that the right side of Equation H.11 is only an appropriate solution to the left side if some technique can be used to provide a solution to the integral. There is no mathematical technique for determining |

] can be determined, then the right side of Equation H.11 is an appropriate solution to the left side.

As the NRC noted in the second paragraph to RAI 9, - the treatment of the boundary conditions for the leakage function has a substantial dependence on (a) the core boundary geometry (*e.g.*, the number and geometry of the surfaces) and (b) the specific code used to determine the calculation. To solve the left side of Equation H.11, FANP uses the [

]

[

] for fluence rate analyses, it is not appropriate to represent the complete method that is applied to the DORT model. The expression for the leakage function needs to be expanded to include the coupled variables of space, energy and solid angle. Equation G.6 is the expansion of the [ ] leakage function to include the appropriate variables. Equation G.6 also shows that whether we are considering an exponential integral function or a Bickley-Naylor function for cylindrical coordinates, the solution will continue to be dependent on the [

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[

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[

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Set 2, RAI 10

In order to determine the effect of using [ ], provide a comparison of the DORT calculated fluence for (a) the case in which the [ ] determined by the core-follow code calculation are input and (b) the case in which the [ ] are input. The comparisons should be made for the azimuthal inner-wall > 1-MeV fluence for a typical plane in an upper region of the core where there is substantial voiding. Provide a comparison of the [ ] together with the fluence [ ] mean and standard deviation. [ ]

Response

[ ] The discussion from Section G.3.3, "Core Leakage Function" provides an explanation of the procedures used to ensure that the DORT calculational methods are accurate: The accuracy of the fluence evaluation process begins with the core-follow simulation of the measured fission rates for power production. The core-follow results match the measurements within the uncertainty criteria for the magnitude of the core power and nodal power distribution.

Assuming that there is no average time-weighted effect on the leakage function { $\Omega \bullet \nabla\phi(\mathbf{r}, E, \Omega)$ }, the collision density parameters and source parameters in DORT

will produce the same flux values as those from the average time-weighted core-follow calculations.

[

]

Using the PWR models and procedures developed in Section 3 of this topical to compute BWR leakage rates from the core periphery indicates that the approximations in the modeling and procedures must be updated. The average time-weighted "fixed" source eigenfunctions and collision density parameters do not produce accurate peripheral flux values. The average time-weighted effect of the leakage function $\{\mathbf{\Omega} \bullet \nabla\phi\left(\mathbf{r}, \mathrm{E}, \mathbf{\Omega}, \bar{t}\right)\}$ needs to be modified [ ]

Set 2, RAI 11

Recognizing the complex dependence of the fluence (φ), source (S) and total cross section (Σ_T) [

]?

Response

A unique and real value of the water density from the core-follow simulation of reactor operation is used [

]

Set 2, RAI 12

Provide the energy for which the FANP/BNL-6115 flux comparisons of Section G.4.2.2 have been made.

Response

The flux comparisons between FANP and BNL-6115 in Section G.4.2.2 were for neutron energies greater than 1.0 MeV (million electron volts, megavolts). This includes Figures G-3 through G-8 and the table, "$FANP/BNL-6115$ Comparison of Key Deviations".

Set 2, RAI 13

Describe the application of the Section G.3.2 jet-pump/riser modeling procedure and the [ ] in the Browns Ferry-2 (BF-2) dosimetry analysis. Provide a comparison of the BF-2 bundle-wise [ ]. What is the effect of this difference on the calculated fluence?

Response

This request contains five parts, describing the application of the procedures in Sections G.3.2, G.3.3 and G.3.4, providing a comparison of the differences between core-follow data and Equations G.7 and G.10, and discussing the effect of Equations G.7 and G.10 on the calculated fluence. To help ensure clarity, each response will be discussed independently in a paragraph.

The Section G.3.2 procedure, "Neutron Transport Through Jet Pumps" addresses one of the issues that the NRC has discussed previously. Other analysts have frequently found

greater inaccuracies in the calculations of reactions that are shadowed by the jet pumps than those calculations that have no shadowing effect from neutrons leaking from the core. In BNL-6115, "PWR and BWR Pressure Vessel Fluence Calculation Benchmark Problems and Solutions" Carew of Brookhaven and the NRC compare the results of MCNP and DORT calculations in Figure 5.4.6. One of the issues that Figure 5.4.6 addresses is the explicit modeling of the jet pumps with the MCNP geometry and the approximation required in DORT. Comparing Figure 5.4.6 and Figure G-4 in the FANP topical, some of the same biased deviations are evident. Based on the MCNP results and those in the FANP DORT analyses, the BNL-6115 DORT could probably be improved with a finer jet pump mesh. The resolution to the issue of the jet pumps possibly causing biased deviations in their shadow is addressed by the Section G.3.2 procedure. To accurately model the jet pumps, the non-uniform attenuation that they cause must be appropriately treated. Equations G.1 through G.3 provide a means of demonstrating that the DORT modeling represents the proper non-uniform attenuation. The Browns Ferry-2 DORT model that FANP developed included the appropriate treatment of the jet pumps. The criteria represented by Equations G.1 through G.3 were satisfied.

The application of the Section G.3.3 modeling procedure is discussed in the response to RAI 9, Set 2. [

] The Browns Ferry-2 DORT model that FANP developed included the appropriate treatment of [

]

The application of the Section G.3.4 modeling procedure is discussed in the response to RAI 2, Set 1. The issue that is addressed is the appropriate treatment of [

$$\Sigma_T\left(\overline{\mathbf{r}'}, g\right)$$

] The Browns Ferry-2 DORT model that FANP developed included the appropriate treatment of [

]

As noted in the response to RAI 2, Set 1, the FANP evaluation of axially homogenizing several nodes concluded that, - even with the appropriate treatment of [

] there would be too much detail lost. For example the shroud cracking seems to be around the jet pump supports. [

]

Unfortunately, there is no data to show the comparison of the core-follow code with Section G.3.3. The Browns Ferry-2 DORT model that FANP developed was judged to be sufficiently accurate [

]

The effect of [

] is to remove biases from

the methods for calculating the fluence rate. Thus, the difference in the calculated fluence rate between analyses [

]

Set 2, RAI 14

What is the location of the BF-2 dosimetry capsule relative to the jet pumps and riser?

Response

The Browns Ferry Unit 2 dosimetry capsule that was referenced in this topical is located 30 degrees from the major axis. Considering a radial vector from the center of the core, this places the capsule directly behind the riser piping for one set of jet pumps.

Set 2, RAI 15

The [ ] bias removal function of Appendix D can result in a (non- conservative) reduction in the vessel fluence prediction. Is this bias removal function applied in BWR applications? If so, provide justification.

Response

The [ ] bias removal function is applied to BWR calculations of the "best-estimate" fluence values. The cause of the [ ] bias in the DORT calculated fluence values is most probably the method of treating the source

eigenfunction. While the reactor core is operating with [ ] the fluence rate – time-integrated form of Equation G.4 is not represented. Consequently, there is no mathematical function for producing [ ] The bias caused by this approximation in the methods is not related to a PWR, BWR or any other core model. Thus, the bias removal function should be applied to the results of all DORT models that lack the mathematical function [ ]

The NRC request for additional information includes the statement that the [ ] bias removal function can result in a non-conservative reduction in the vessel fluence. While the bias removal function contains [ ] In fact as FANP noted and the NRC confirmed in *Appendix D*, pages D - 57, Set 2 – Question 16, and D - 61 through D - 68, in the section on the "Statistical Processing of Table A-1 Data", FANP has no statistically significant bias in the greater than 1.0 MeV fluence rate. That is, calculating the M/C ratio before the application of the bias removal function, the NRC obtained a value of .9940. This shows a statistically insignificant bias. When FANP applied the bias removal function to the calculations, the bias continued to be statistically insignificant. The [ ] in the DORT methods and results in the best-estimate of the fluence throughout the internal structures, within the vessel, and throughout the reactor cavity structure.

Set 2, RAI 16

Were the calculation/modeling and measurement methods described in the topical report used in the analysis of the PCA dosimetry experiment, the Browns Ferry Unit 2 (BF-2) capsule measurement and the BNL-6115 benchmark? If not, describe any differences and their effect on the comparisons. For example, were the jet pump [ ] procedures of Section G.3 used in the BF-2 analysis?

Response

The calculational modeling and methods described in this topical report were used in the analysis of the PCA dosimetry experiment, the Browns Ferry Unit 2 capsule, and the BNL-6115 benchmark. Thus, these benchmark comparisons appropriately represent samples from the benchmark database. Moreover, they provide additional confirmation of the uncertainty values noted in Section G.4 of the topical. As noted in the response to RAI 13, Set 2 above, [

]

The measurement methods described in this topical report were not used in the development of the data from the PCA dosimetry experiment, the Browns Ferry Unit 2 capsule, and the BNL-6115 benchmark. The PCA experiment contained measurements from the Oak Ridge National Laboratory (ORNL). While ORNL and other national laboratory contributors to the PCA experiment described methods that were consistent with FANP standards, which are consistent with ASTM standards, there are measurements of reactor power and power distributions that have different methods from

those associated with power reactors. This is discussed further in the response to RAI 4, Set 1. In addition, the measurement uncertainties associated with the BNL-6115 benchmark are those from the PCA experiment. Since the uncertainties associated with the BNL-6115 benchmark are equivalent to the PCA experiment, these uncertainties are not included in the benchmark database. Including them would be equivalent to weighting the benchmark to the PCA experiment twice.

The Browns Ferry Unit 2 capsule measurement was performed by GE Nuclear Energy. While GE discusses measurements methods for the iron, copper, and nickel dosimeters that are consistent with FANP standards, which are consistent with ASTM standards, there is no evidence that the GE laboratory has been benchmarked to a reference field. Thus, it would be expected that the GE laboratory results lack the confirmation that is required by Regulatory Guide 1.190 [H4] and that is part of the FANP quality. Nonetheless, the calculated benchmark comparison to the Browns Ferry Unit 2 capsule measurement indicates that the benchmark uncertainty is consistent with the FANP database.

References

H1. M.C. Honcharik (NRC), "Request for Additional Information, Issues Related to BAW-2241, *Appendix G*, 'Fluence and Uncertainty Methodologies,' Framatome ANP, Project Number 728," facsimile to G.F. Elliott (Framatome ANP), June 17, 2005.

H2. H.A. Hassan, *et alia*, "Power Peaking Nuclear Reliability Factors", FANP Document # BAW-10119P-A, February, 1979.

H3. H. Moon, "Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4 / MICROBURN-B2", FANP Document # [ ], December, 1998.

H4. Office of Nuclear Regulatory Research, "Calculational And Dosimetry Methods For Determining Pressure Vessel Neutron Fluence", Regulatory Guide 1.190, U.S. Nuclear Regulatory Commission, March, 2001

H5. Office of Nuclear Regulatory Research, "Radiation Embrittlement Of Reactor Vessel Materials", Regulatory Guide 1.99, Revision 2, U.S. Nuclear Regulatory Commission, May, 1988.

H6. M.C. Honcharik (NRC), "Request (Round Two) for Additional Information on BAW-2241, *Appendix G*, 'Fluence and Uncertainty Methodologies'," e-mail to G.F. Elliott (Framatome ANP), September 8, 2005.

H7. M.A. Rinckel, J.R. Worsham III, *et alia*, "Demonstration of the Management of Aging Effects for the Reactor Vessel", BAW-2251-A, August, 1999.