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**PROPRIETARY INFORMATION – WITHHOLD UNDER 10 CFR 2.390**

10 CFR 50.90

NMP1L3233

August 17, 2018

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Nine Mile Point Nuclear Station, Unit 1
Renewed Facility Operating License No. DPR-63
NRC Docket No. 50-220

Subject: Response to Request for Additional Information by the Office of Nuclear Reactor Regulation to Support Review of Nine Mile Point Nuclear Station, Unit 1, Removal of Boraflex Credit License Amendment Request (L-2018-LLA-0039)

References:

1. Letter from J. Barstow (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission, "License Amendment Request – Proposed Change to remove Boraflex Credit from Spent Fuel Racks," dated February 9, 2018

2. Email from M. Marshall (Senior Project Manager, U.S Nuclear Regulatory Commission) to R. Reynolds (Exelon)," Request for Additional Information for Nine Mile Point Unit 1 Removal of Boraflex Credit License Amendment Request (L-2018-LLA-0039)," dated July 3, 2018

By letter dated February 9, 2018, (Reference 1) Exelon Generation Company, LLC (Exelon) requested approval to change the Nine Mile Point Unit 1 (NMP1) Technical Specifications (TS). The proposed amendment request would remove the credit for Boraflex as a neutron absorber in the criticality analysis for the two remaining Boraflex storage racks in the spent fuel pool.

On July 3, 2018, (Reference 2) the U.S. Nuclear Regulatory Commission (NRC) identified questions where additional information was necessary to complete the NRC Staff review.

Attachment 1 to this letter contains the NRC's questions immediately followed by Exelon's Proprietary responses. Attachment 2 contains the NRC's questions immediately followed by Exelon's Non-Proprietary responses. Attachment 3 contains the affidavit for the proprietary information.

**Attachment 1 transmitted herewith contains Proprietary Information.
When separated from Attachment 1, this document is decontrolled.**

In addition to the attached responses, Exelon is requesting a revision to the implementation date for this proposed amendment request. The revised implementation is as follows:

> "Exelon requests approval of the proposed amendment by May 31, 2019. Once approved, the amendment shall be implemented by December 31, 2019."

Exelon has reviewed the information supporting a finding of no significant hazards consideration and the environmental consideration provided to the NRC in Reference 1. The additional information provided in these responses does not affect the basis for concluding that the proposed license amendment does not involve a significant hazards consideration. Furthermore, the additional information provided in these responses does not affect the basis for concluding that neither an environmental impact statement nor an environmental assessment needs to be prepared in connection with the proposed amendment.

There are no commitments contained in this response.

If you should have any questions regarding this submittal, please contact Ron Reynolds at 610-765-5247.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 17th day of August 2018.

D. P. Helker

David P. Helker
Manager - Licensing & Regulatory Affairs
Exelon Generation Company, LLC

Attachments:

1. Proprietary Response to Request for Additional Information
2. Non-Proprietary Response to Request for Additional Information
3. Affidavit

| | | |
|---|---|---|
| cc: | USNRC Region I Regional Administrator | w/attachments |
| | USNRC Senior Resident Inspector – NMP | " |
| | USNRC Project Manager, NRR – NMP | " |
| | A. L. Peterson, NYSERDA | " |

**ATTACHMENT 2**

Non-Proprietary Response to Request for Additional Information

**RAI 1:**

In Section 2.2.3 of Attachment 3 of the licensee's submittal, the licensee indicates that CASMO-5 version 2.08.00 is being used to perform depletion calculations for the determination of isotopic compositions. However, Nine Mile Point Unit 1 is a boiling water reactor (BWR), and CASMO-5 has only been reviewed and approval by the NRC for application to pressurized water reactors (PWRs). The NRC has not examined BWR fuel assembly validation data for CASMO-5 to assess the code's adequacy for performing depletion calculations. Accurate determination of isotopic compositions for spent fuel is necessary for the accurate prediction of assembly reactivity and assurance that the $k_{eff}$ of the spent fuel storage racks does not exceed 0.95 as per the regulatory requirement in 10 CFR 50.68(b)(4).

The limiting depletion conditions for the nuclear criticality safety (NCS) analysis presented in the licensee's submittal occur at conditions of relatively low burnup at 0 percent void in the presence of gadolinia, which is characteristic of PWR conditions. Given this, the NRC approval of CASMO-5 for PWRs may be applicable to Nine Mile Point Unit 1 for the narrow set of limiting depletion conditions, provided there is reasonable assurance the code will predict acceptable isotopic compositions for BWR fuel.

Provide information justifying that the CASMO-5 predictions of isotopic compositions for BWR fuel are consistent with NRC-approved applications of depletion codes for the narrow set of depletion conditions used in the present NCS analysis.

**Exelon Response to RAI 1**

The applicability of the NRC review and acceptance of CASMO-5 depletion isotopic compositions for use with PWR fuel is documented in the application package ML15355A280 [1]. The applicability of that acceptance review to BWR spent fuel is addressed by considering the following:

- Comparison/contrast of LWR BWR and PWR fuel design and operations characteristics based on ML15355A280.
- Comparison of CASMO-4 and CASMO-5 code depletion methodology.
- Comparison of Holtec's criticality methodology for BWR (and PWR) fuel using CASMO-4 and CASMO-5.
- Review of industry guidance NEI 12-16.
- Review of additional documentation of CASMO-5 for use with BWR fuel.

The CASMO code package has been used domestically and internationally for reactor depletion calculations as well as spent fuel pool safety analysis both under the older CASMO-4 version and the newer CASMO-5 version. Holtec's spent fuel pool criticality methodology using CASMO-4 has long been accepted and approved and there exists a high degree of confidence in CASMO-4 to accurately predict spent fuel isotopic compositions for both PWR and BWR fuel.

Holtec's use of CASMO-5 for BWR spent fuel pool applications is now under consideration with this current application, and due to a lack of an application such as [1] for BWR fuel, additional justification is provided in this response that CASMO-5 is an improvement over CASMO-4 with respect to the depletion code's ability to predict spent fuel isotopic compositions for BWR fuel.

*Comparison of LWR BWR and PWR fuel design and depletion characteristics based on ML15355A280*

The important aspect of the comparison is related to the ability of the depletion code to accurately determine the spent fuel isotopic compositions under various reactor conditions. The non-proprietary details in [1] have been examined and there are some important points from [1] that are relevant to this discussion:

- Section 2, "CASMO5 Methodology":
  - Under this section is provided several items which are applicable to both PWR and BWR fuel with respect to depletion related characteristics:

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  - The above listed items are just a sample from [1], while greater details and explanations can be found elsewhere. Note that these items are all appropriate for depletion calculations for both PWR and BWR fuel, and, most importantly, there is nothing inherently different about the design of BWR fuel which would invalidate or negate the appropriateness of the CASMO-5 methodology items listed above. The differences between BWR fuel depletion and PWR fuel depletion which have an important impact on spent fuel isotopic compositions are operating parameters such as void fraction, control blade insertion, and the peak reactivity of BWR fuel due to use of Gadolinium (Gd). For the core operating parameters such as void fraction and control blade insertion, it was previously stated in RAI 1 that the limiting conditions of the BWR fuel in this application essentially mimic the depletion conditions of the PWR fuel in [1]. While that may already be sufficient justification to conclude that [1] is applicable to this application, additional support is provided below through a more detailed evaluation of the BWR core operating parameters:

- Void fraction does not imply the absence of moderator, just a change in density. Therefore, the difference between PWR moderation and BWR moderation is that BWR moderation covers a wider range of moderator density (much lower). The impact of the change in moderator density is a change in the neutron spectrum, and that impact is independent of fuel design. Furthermore, what is physically important to the fuel is not the moderator density but the energy of the neutrons that are transported through the system. Since CASMO-5 considers a large, 586 group microscopic cross section, 128 of which are in the fast group (see Section 2.2 of [1]), and neutron energies are covered in the range of $10^{-5}$ eV to 20 MeV, it is concluded that changes in moderator density from either temperature or void are all well covered by the CASMO-5 energy structure. And these, as reviewed and approved in [1] for PWR fuel, are also inherently applicable to BWR fuel as well. Note also that the range of neutron energies considered by CASMO-5 is greater than the range considered by CASMO-4.

- Control blade insertion is also of the same nature as moderator void fraction in that the impact which is relevant to the fuel is the energy of the neutron spectrum. The control blades harden the neutron spectrum by absorbing thermal neutrons. However, the fuel interacts physically with the neutrons, not the control blades. Therefore, the same discussion and conclusions as provided above for moderator void fraction are also applicable to control blades.

- The peak reactivity of BWR fuel due to the presence of Gd is an important design feature of BWR fuel. The inclusion of Gd in the BWR fuel is not unique to BWR fuel, however. PWR fuel can and does use Gd as an integral absorber. Thus, the inclusion of Gd was also considered in [1]. An excerpt from Section 3.2.2.2 of [1] is as follows:

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  This point in [1] provides the information needed to conclude that since the inclusion of Gd as an integral absorber is consistent between BWR fuel and PWR fuel, and its inclusion has been shown to be included in an acceptable way in CASMO-5 for PWR fuel, then its inclusion in BWR fuel in the same manner is also acceptable in CASMO-5. Note also that the CASMO-5 enhancements (see below) with respect to Gd depletion over CASMO-4 improve the code's ability to determine the spent fuel isotopic compositions for fuel with Gd and that while CASMO-4 was already considered acceptable, CASMO-5 is more so.

- In Section 2 we also find information related to the number of isotopes explicitly considered in CASMO-5. The change from CASMO-4 to CASMO-5 includes the explicit treatment of fission product isotopes in CASMO-5 whereas in CASMO-4 they were included in "lumped fission products (LFP)". The explicit treatment of isotopes is a code enhancement in CASMO-5 from CASMO-4 to more accurately predict the composition of spent fuel.

Based on the above discussion, and the additional discussions provided below, it is concluded that the validations provided in [1] can be considered to justify the use of CASMO-5 for BWR fuel for the full range of depletion parameters considered in the application, not just those present for the bounding condition.

*Comparison of CASMO-4 and CASMO-5 code depletion approach*

Studsvik provides a concise report which compares the migration of CASMO-4 to CASMO-5 in [2]. Within that report is found a summary in Section 1.6, "Burnup Calculation Differences Between CASMO-4 and CASMO-5/5M". This summary is directly applicable to this RAI response because it shows that the depletion methodology between the two code versions is very similar with the exception of some important enhancements in CASMO-5.

The following is an excerpt from Section 1.6 of [2]:

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Several important conclusions can be reached from this excerpt:

- *Improved Gd depletion models and options.* One of the main differences between BWR fuel and PWR fuel, as discussed above, is the peak reactivity. Improvements in the Gd depletion model is an important enhancement of CASMO-5 over CASMO-4 and improves the CASMO code's prediction of isotopic compositions in the presence of a strong integral absorber like Gd. As discussed above, the use of Gd with PWR fuel depletion for CASMO-5 in ML15355A280 [1] supports the review of CASMO-5 for BWR fuel depletion.

- *CASMO-5/5M does not incorporate a "lumped fission product" model as was previously done in CASMO-4.* CASMO-4's LFP model was based on historical

simplifications in modeling large numbers of isotopes. The CASMO-5 explicit treatment of every isotope is a welcome enhancement and provides a much smoother symmetry between the depletion code and Monte Carlo code used to determine keff. These changes to the depletion approach are improvements which enhance the code's ability to predict depletion isotopic compositions more accurately, mainly because individual isotopes are treated explicitly, rather than lumped together as a smeared composition with some weighted average cross sections. While the overall reactivity impact of the fission product (FP) isotopes is not very large compared with the major actinides and minor actinides (the FPs are expected to contribute about 1% $\Delta k$), explicit treatment provides a much more accurate composition of relative fission product isotopic abundances. As discussed above, the use of the explicit CASMO-5 depletion isotopes for all fission products for PWR fuel depletion in ML15355A280 [1] supports the use of CASMO-5 for BWR fuel depletion since isotopic composition is independent of fuel design.

*Comparison of Holtec's criticality methodology for BWR (and PWR) fuel using CASMO-4 and CASMO-5*

Holtec's criticality methodology is well documented. The change from CASMO-4 to CASMO-5 has not changed the approach used by Holtec other than to remove the use of LFP and explicitly consider the number densities for all nuclides previously considered as LFP. Further, as noted in Section 2.2.3 of the application, CASMO-5 also uses the same cross section data sets as MCNP5. One reason Holtec has moved from CASMO-4 to CASMO-5 for its criticality methodology is for the explicit determination of the fission product isotopes (discussed above) and the consistency of the cross section data sets with MCNP5.

Additionally, Holtec's BWR criticality methodology using CASMO-4 for the determination of depletion isotopic compositions has been previously approved by the NRC many times. The manner and method in which Holtec is using CASMO-5 is the same manner and method in which Holtec was using CASMO-4 with the only exception being that more isotopes are now provided with CASMO-5.

Review of industry guidance NEI 12-16 [5]

While it is not the specific purpose of [5] to provide detailed guidance on qualifying new codes or code versions, there is relevant information in [5] with respect to the topic. Specifically, Section 4.3.2 of [5] discusses the BWR depletion uncertainty where the text states: "The BWR lattice physics/depletion codes used for SFP criticality analyses are the same depletion codes used and validated for BWR core design and core monitoring applications." The intent is that for any given depletion code that has been approved for use in reactor core design, i.e. CASMO-4 or CASMO-5, then that code is also acceptable for the determination of the depletion isotopics for spent fuel pool criticality analysis. CASMO has a long history of use for BWR core design, including the most recent version of CASMO-5 used in this application. This point has been independently confirmed again by Holtec in conversations with Studsvik, for this RAI response. Finally, it should be noted that the given topic of Section 4.3.2 of [5] is actually the uncertainty

associated with the number densities of the spent fuel isotopics, not the isotopic number densities themselves. Thus, we conclude that the ability of the depletion code to predict number densities is not questioned because the code was already approved for the same use for reactor core design. Given that, there is good agreement with NEI 12-16 for the acceptable use of 5% for the depletion uncertainty and the use of reactor core design codes for the calculation of spent fuel pool depletion isotopics because the ability of the code to predict the number densities is not questioned.

*Review of additional documentation of CASMO-5 for use with BWR fuel*

There are several resources available from Studsvik and other sources that provide useful information to support the conclusion of the acceptability of using CASMO-5 for BWR fuel depletion calculations for spent fuel pool criticality, specifically related to code validation efforts:

- CASMO-4 validation can be found in [3] (for comparison purposes to CASMO-5 improvements), as well as related reactor critical validation for dry storage in [4] (PWR only, though BWR reactor criticals are also available).
- CASMO-5 BWR validation report can be found in [6], [7] and [8]

While there are alternative approaches to the validation of depletion codes in lieu of critical experiments and/or chemical assays which have been accepted for other applications and uses (see [3]), for example the use of reactor criticals (see [4]), or even code to code comparisons as has also been provided in [1] and [3], there is no current need for such approaches since CASMO-5 for PWR fuel has been accepted under [1] and the similarities between the two LWR fuel designs is sufficient justification for the purpose of this application when viewed in conjunction with the technical discussion provided in this response.

Summary of Response

The discussion provided in this response has provided sufficient technical justification to conclude that CASMO-5 is well equipped to determine the isotopic composition of spent fuel for BWR fuel depleted over the full range of core operating parameters experienced by BWR fuel. This technical justification is based on (1) information provided in ML15355A280 [1] as it relates to PWR fuel depletion with CASMO-5 and its relation to BWR fuel depletion, (2) a comparison of Studsvik's literature with respect to CASMO-4 and CASMO-5 code depletion methodology found in [2], (3) the similarity of Holtec's criticality methodology for BWR (and PWR) fuel using CASMO-4 and CASMO-5 and how Holtec's methodology which has been approved many times for BWR fuel with CASMO-4 depletion isotopic compositions is the same as Holtec's methodology using CASMO-5 (only more accurate in the inclusion of explicit fission product isotopes), (4) a review of industry guidance NEI 12-16 and the relevance of the confidence given to depletion codes which are approved for core design and therefore also acceptable for spent fuel pool criticality, and finally, (5) a brief review of additional documentation of CASMO-5 for use with BWR fuel. The comprehensive discussion provides a significant amount of technical arguments and additional supporting documentation to conclude that CASMO-5 is acceptable for use in this application over the full range of core operating parameters.

**References for Response to RAI 1**

[1] ML15355A280, "Submittal of Generic Application of the Studsvik Scandpower Core Management System to Pressurized Water Reactors," December 30, 2015.

[2] SSP-06/466, Rev 2, "CASMO-4 to CASMO-5/5M Migration Notes", June 2009.

[3] ML003698495, "Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICROBURN-B2," EMF-2158(NP)(A), October 1999.

[4] Holtec International Report HI-951251, Safety Analysis Report HI-STAR 100 Cask System, USNRC Docket 71-9261, Revision 16.

[5] NEI 12-16, Revision 3, "Guidance for Performing Criticality Analyses of Fuel Storage at Light-Water Reactor Power Plants," Nuclear Energy Institute, ADAMS Accession No. ML18088B400.

[6] PHYSOR 2012 Advances in Reactor Physics Linking Research, Industry, and Education Knoxville, Tennessee, USA, April 15-20, 2012, on CD-ROM, American Nuclear Society, LaGrange Park, IL (2012), "EXPERIMENTAL VALIDATION OF CASMO-4E AND CASMO-5M FOR RADIAL FISSION RATE DISTRIBUTIONS IN A WESTINGHOUSE SVEA-96 OPTIMA2 BWR FUEL ASSEMBLY," Peter Grimm and Gregory Perret Paul Scherrer Institute CH-5232 Villigen PSI Switzerland Peter.Grimm@psi.ch; Gregory.Perret@psi.ch.

[7] PHYSOR 2016, Sun Valley, ID, May 1–5, 2016 "VALIDATION OF CASMO5 GADOLINIUM DEPLETION ANALYSIS," Yusuke Kuroda, Tsuyoshi Ama, and Takashi Yoshii TEPCO SYSTEMS CORPORATION, Tokyo, Japan kuroda-yuusuke@tepsys.co.jp.

[8] PHYSOR 2014 The Role of Reactor Physics toward a Sustainable Future, Kyoto, Japan, September 28-October 3, 2014, "Validation of CASMO5 Spent Fuel Isotopics with Decay and Fission Yield Uncertainties," Joshua Hykes and Joel Rhodes, Studsvik Scandpower, Inc, 1070 Riverwalk Drive, Suite 150, Idaho Falls ID.

**RAI 2:**

In Section 2.2.2 of Attachment 3 of the licensee's submittal, the licensee indicates trend analyses of the MCNP5-1.51 validation benchmarks were performed in order to determine the maximum bias and bias uncertainty associated with determining $k_{eff}$. Based on similar NCS analyses reviewed by the NRC (ADAMS Accession No. ML15343A126), an additional uncertainty may be necessary to account for the lack of validation for $k_{eff}$ calculations of burned fuel systems containing minor actinides and fission products.

While the licensee indicates in the Criticality Analysis Checklist table of Appendix C of Attachment 3 that a bias in fission product and minor actinide reactivity worth is considered, the magnitude of the bias and how it was determined does not appear to be discussed in the LAR or listed in Table 7.1. The uncertainty/bias associated with predicting the reactivity worth of fission products and minor actinides in spent fuel has a direct impact on the determination of assembly reactivity and on the assurance that the $k_{eff}$ of the spent fuel storage racks does not exceed 0.95 as per the regulatory requirement in 10 CFR 50.68(b)(4).

Provide a discussion on the uncertainty and bias in reactivity worth of fission products and minor actinides associated with MCNP5-1.51 $k_{eff}$ calculations. Specifically, address the magnitude of the uncertainty and bias (incl., how it was determined and how it is incorporated into the uncertainties and biases of the NCS analyses).

**Exelon Response to RAI 2**

The application documented in ML15342A126 was submitted early during the NEI 12-16 process, when there was some ongoing discussion related to the proper way to address the depletion related uncertainties and biases associated with spent fuel isotopic compositions and benchmarking. During the application process documented in ML15342A126, the Holtec methodology used a very conservative uncertainty factor to account for the depletion uncertainty associated with FP (in that case it was the FP and LFP because CASMO-4 was used). At that time, the 1.5% minor actinide and FP reactivity worth value provided in NUREG/CR-7109 was deemed a reasonable value to use as a bias to account for criticality code validation gaps; however, as discussed in ML15342A126, Holtec's FP/LFP uncertainty calculation was overly conservative and more than offset the inclusion of a small bias based on NUREG/CR-7109. The original FP/LFP 20% uncertainty applied in ML15342A126 is no longer required because CASMO-5 no longer uses LFP since each isotope is now tracked explicitly. Note that the version of NEI 12-16, ML130840163 (Revision 2, Draft B), which included the discussion relevant to NUREG/CR-7109 was actually abandoned in the current version, ML18088B400 (Revision 3). That is, Revision 3 does not reference or mention NUREG/CR-7109.

With respect to the validation gap for the FP's not explicitly validated in the MCNP benchmark (reference [10] of the application), the NUREG/CR-7109 recommendation of a bias of 1.5 % of the reactivity worth of the isotopes is applicable. However, it is not included for the following reasons:

- The actual bias value is very small. The calculation is performed as 1.5% of the difference between the case with all FP and the case with the unvalidated FP removed (except the FP isotopes considered allowable by NUREG/CR-7109). Using this approach, the potential bias is estimated for this application as approximately 0.00015 $\Delta k$. The reason the bias is so small is because FP (other than those considered validated and allowed by NUREG/CR-7109) contribute so little to the overall reactivity. Note that the calculation is performed using a small reactivity difference, not a large difference to a 0 burnup case like the depletion uncertainty, and therefore, 1.5% of a small reactivity difference yields a very small bias.

Estimation of the Unvalidated FP Bias

| Description | Burnup (gwd) | $k_{calc}$ | $\Delta k$ |
|---|---|---|---|
| Design basis case (Table B.8 of the application) | 10 | 0.86771 | Reference |
| Same as reference but with no unvalidated FP | 10 | 0.87763 | 0.00992 |
| 1.5% Factor | n/a | n/a | 0.00015 |

- Holtec has considerable margin in the analysis to offset the small validation bias if needed. Table 7.1 of the application shows that the overall analysis margin is 0.0110, while in addition to that margin, the analysis counts as a bias a 1% administrative margin. These values are significantly larger than the estimated 0.00015 $\Delta k$ bias which has been estimated for the validation gap.

- The depletion code is benchmarked by Studsvik (see response to RAI 1 above) and is used for reactor core design. There is a high degree of confidence in the code's ability to predict spent fuel number densities.

- CASMO-5 and MCNP5 use the same cross section data set, and there is a high degree of confidence in the cross section data set.

- The lack of critical experiments for these spent fuel isotopes does create a validation gap, yet, as has been determined in NUREG/CR-7109, that gap is very small as evidenced by the discussion in NUREG/CR-7109 that the 1.5% extends to all FP since the benchmarking uncertainty for the validated isotopes is also 1.5%. The validation gap is actually due to a lack of critical experiments with the unvalidated isotopes. However, MCNP is well benchmarked with all the most recent critical experiments including the major actinides and some minor actinides, which simulate spent fuel. The sources of potential errors for the other minor actinides and FP

isotopes would come from either the depletion code or cross section data set errors. These are not considered high risk sources for errors that would require additional effort outside of that which has already been performed for validation (range of experiments, trending etc.).

- The 5% depletion uncertainty has been included in the application. In DSS-ISG-2010-01, it states that the way to interpret the depletion uncertainty is as "the uncertainty in the isotopic number densities generated during the depletion simulations." The uncertainty in the isotopic number densities can arise from uncertainty associated with the depletion code and the underlying nuclear data used by the depletion code – this also includes how the nuclear data is implemented by the depletion code. Therefore, the depletion uncertainty considered in the application also includes some consideration of the potential impact due to the unvalidated nuclides.

The above discussion provides an explanation of the magnitude of the unvalidated FP bias and an explanation for its exclusion from the application. As has been done for previous applications, the significant margin included in the criticality safety analysis can be used to offset the potential impact of excluding the small unvalidated FP bias if needed; however, sufficient technical justification has also been provided to allow neglecting the small bias as unnecessary.

**RAI 3:**

In Section 7.1 of Attachment 3 of the licensee's submittal, the licensee indicates that screening calculations were performed using CASMO-5 for all legacy and current fuel designs to determine the most reactive lattices. According to the discussion in Section 7.1, seven lattices were selected as having "reactivity higher than other lattices." However, the licensee does not indicate within the LAR what screening criterion was used to set these seven lattices apart. The NRC staff notes that several of the lattices screened out possess in-rack $k_{inf}$ reactivity predictions greater than two of the seven selected lattices. Even though conservatively biased super lattices are developed as part of the NCS methodology, screening in lattices that do not have the highest in-rack $k_{inf}$ reactivities for the design basis calculations may result in an under-estimation of the in-rack $k_{eff}$ and the amount of margin that may be available to the regulatory requirements. This would result in decreased assurance that the actual $k_{eff}$ of the spent fuel storage racks does not exceed 0.95 as per the regulatory requirement in 10 CFR 50.68(b)(4).

Provide a discussion on the screening criterion used to determine the most reactive lattices and its threshold. Within the discussion: a) justify that the screening criterion threshold used is sufficient to ensure the most reactive design basis lattice is included, and b) justify why several lattices with in-rack $k_{inf}$ reactivity predictions greater than two of the selected lattices were screened out.

**Exelon Response to RAI 3**

The purpose of the screening methodology is to determine from among a large set of all actual lattices which subset of lattices were appropriate to consider in the more rigorous design basis evaluations which follow in the criticality safety analysis. The approach correctly performs a specific calculation, in this case with CASMO-5, for every actual lattice; thus, it is rigorous and doesn't make any assumptions about lattice reactivity based on average enrichment or Gd rod number, etc. Further, it considers the range of core operating parameters (COP), thus ensuring that the screening calculations have sufficient rigor to reach the conclusions about which lattices are more reactive than the other lattices, and also allows for comparisons to be made between the same calculations using MCNP. The size of the subset of most reactive lattices which is considered sufficient is based solely upon engineering judgment, which is seasoned by the high degree of rigor required for the subsequent calculations needed to determine the design basis lattice. Thus, how many lattices to use and of which physical design is arbitrary. In the end, the results of the screening calculations were used to inform the decision. A table which provides the same results as those shown in Table A.2 of the application is provided below with reactivity increasing beginning with 10a84 through 10a63. The set of 7 most reactive lattices that were selected is highlighted in yellow. Note that the last digit in the lattice name is either a 1, 2, 3, or 4, depending on the location of the lattice along the bundle axial length (1 is the bottom enriched lattice with full length rods and 4 is the top enriched lattice with multiple part length rod locations). Note also that the results in the table show how reactivity is impacted by lattice design (physical design, i.e. location of part length rods). Initially, the selection criteria which falls out of these results is any lattice with a reactivity greater than 0.94 is appropriate to select. Secondly, it is clear that those 5 lattices cover lattice locations 1, 2 and 3, but not 4. Since the subsequent calculations require a high degree of rigor it was decided that it was not appropriate to exclude the lower reactivity "4" lattices, and therefore the 10a84 and 10a64 lattices were selected. Note that it is not a coincidence that the seven lattices all come from the same two bundles, since these two bundles are obviously designed to achieve a high reactivity.

Comparison of lattice in-rack reactivity

| Lattice | In-rack $k_{calc}$ |
|---|---|
| 10a84 | 0.9287 |
| 10a64 | 0.9291 |
| 10a92 | 0.9297 |
| 10a42 | 0.9299 |
| 09b82 | 0.9307 |
| 09b83 | 0.9309 |
| 10a72 | 0.9314 |
| 10a71 | 0.9317 |
| 10a43 | 0.9336 |
| 10a93 | 0.9336 |
| 10a22 | 0.9348 |
| 10a73 | 0.9353 |
| 10a81 | 0.9370 |
| 10a23 | 0.9386 |
| 10a61 | 0.9431 |
| 10a82 | 0.9442 |
| 10a62 | 0.9453 |
| 10a83 | 0.9485 |
| 10a63 | 0.9490 |

The discussion provided above clarifies the approach used for the screening calculations and for the determination of the set of 7 most reactive lattices. As has been discussed above, the approach is very robust and appropriately includes lattices from each physical type present in the most reactive fuel design.

**RAI 4:**

In Section 2.3.2 of Attachment 3 of the licensee's submittal, the licensee considered the impact of core operating parameters on the design basis lattice selection by using four different sets of core operating parameters in the screening calculations. These sets of core operating parameters were selected because they are considered the most likely candidates for limiting conditions. However, once the bounding set of core operating parameters was selected, the licensee did not do specific sensitivity studies that vary the individual parameters to verify that the selected value for each would result in the most positive lattice reactivity. The reactivity impact of specific core operating parameters may vary for different fuel lattice designs, impacting the selection of the design basis lattice. Although conservatively biased super lattices are developed, selecting a design basis lattice that is not the most reactive may result in an under-estimation of the in-rack $k_{eff}$ and the amount of margin that may be available to the regulatory requirements. This would result in decreased assurance that the actual $k_{eff}$ of the

spent fuel storage racks does not exceed 0.95 as per the regulatory requirement in 10 CFR 50.68(b)(4).

Provide a discussion explaining why parameter-specific sensitivity studies are not needed to provide assurance that the in-rack $k_{eff}$ and the amount of margin available to the regulatory requirements have not been under-estimated. If parameter-specific sensitivity studies were conducted, provide a summary description of the studies.

**Exelon Response to RAI 4**

Please see Table B.16 of the application for the parameter specific sensitivity study that has been performed, it is entitled "MCNP In-rack with Cell Blocker (Empty Cell) Results for the Alternative COP Study". In Table B.16, variations of the bounding COP are used to demonstrate that the most reactive parameter case is considered in the bounding COP. Please note also the following historical discussion with respect to COP evaluations. For BWR fuel, the most important COP are control blade insertion and moderator void. Temperatures play a secondary role. For control blade insertion and moderator void, it is expected that insertion or non-insertion is bounding, as it is expected that 0 voids or maximum voids is bounding. However, the voids are physically related to the temperatures and therefore it is appropriate and acceptable to change the void fraction in a manner physically consistent with temperature. The study in Table B.16 for maximum void fraction with minimum temperatures and rodded conditions shows that the non-physical maximum void fraction reduces reactivity. Therefore, the most important COP is control blade insertion. In reality, the control blades are inserted and removed periodically for power maneuvers while the void fraction is a function of power and axial location, and some lattices rarely see large void fractions while others rarely see low void fractions even though those extremes can result in spent fuel isotopic compositions with the highest reactivity in the spent fuel pool. For any given bundle lattice, the actual control blade insertion and void fraction varies over burnup. These variations are considered to be bounded by using the condition which creates the maximum peak reactivity, although that condition is sometimes non-physical. Thus, while some intermediate void fraction could potentially create a slightly higher reactivity, the conservative consideration of the extremes over all burnups is considered sufficient to preclude the need to evaluate multiple intermediate steps between the extremes. With respect to the most important COP, control blade insertion, the rodded or un-rodded condition is treated in the most conservative manner possible. The COP study results in the application demonstrate that the impact of rod insertion is dramatic. For example, in Table B.2 the results for the bounding lattice 10a63 show that for minimum temperatures and void, the un-rodded case is 0.9205 and the rodded case is 0.9368 (0.0163 $\Delta k$). Similar results are shown for the maximum temperatures and void, where the un-rodded is 0.9170 and the rodded is 0.9294 (0.0124 $\Delta k$). For both temperature and void conditions, the rodded cases are more reactive. This is a result of the rodded configuration creating a harder neutron spectrum and therefore increasing the fissile Pu content of the spent fuel. Since these large conservatisms are built into the analysis by leaving the control blades in or out, this simplified approach makes it acceptable to not consider intermediate steps in individual COP values.

**RAI 5:**

Section 2.3 of Attachment 3 of the licensee's submittal Indicates that the standard cold core geometry (SCCG) methodology shows that future lattices are bounded by lattices used in the peak reactivity methodology and therefore all future lattices "within the current fuel design" are qualified for storage in the Nine Mile Point Unit 1 spent fuel pool (SFP). However, Section 7.1 of Attachment 3 of the licensee's submittal states that "the SCCG methodology is not an exact methodology since it cannot be concluded that all lattices with a SCCG $k_{inf}$ reactivity of 1.31 correspondingly have the same reactivity in the SFP. NRC staff examination of Figure B.1 and Figure B.2, which show MCNP5-1.51 design-basis reactivity predictions for lattices in both the SCCG and in-rack geometry, confirms this statement; relative to each other, some lattices are less reactive in the in-rack geometry than the SCCG while others are more reactive. In light of this, it may be possible that future lattices for which NCS analyses have not been performed (e.g, as a result of a new number of Gd rods or new Gd rod locations) may not be bounded by the lattices developed in the peak reactivity method, even if the future lattices are within the current fuel design. This would result in decreased assurance that, with these future lattices in-place, the actual $k_{eff}$ of the spent fuel storage racks does not exceed 0.95 as per the regulatory requirement in 10 CFR 50.68(b)(4).

Provide a discussion of the various aspects of the design basis lattice developed in the peak reactivity methodology that provide assurance it represents the highest reactivity lattice in the in-rack configuration for lattices with a SCCG $k_{inf}$ of 1.31.

**Exelon Response to RAI 5**

Section 7.1 of Attachment 3 of the submittal states that "the SCCG methodology is not an exact methodology since it cannot be concluded that all lattices with a SCCG kinf reactivity of 1.31 correspondingly have the same reactivity in the SFP." Since this is how BWR lattices behave in the spent fuel pool, each lattice must be individually evaluated. Furthermore, to provide sufficient conservatism, super lattices which are much higher in reactivity than any actual lattice are used in the analysis. Otherwise, while the most reactive actual lattice could be used for the design basis analysis, a future lattice which could meet the TS requirements may in fact be more reactive than the design basis lattice, even though they have the same SCCG. Thus, super lattices, which are actually limiting in terms of reactivity because they have a SCCG slightly higher than the TS limit and meet the other TS requirement of 4.6% U-235, are considered. The reactivity difference in the storage rack geometry seen among the super lattices, which all have a SCCG reactivity of about 1.31, is dominated by the physical differences in lattice geometry (i.e. number of fuel rods and part length rods), not Gd rod loading or Gd rod number and location. Therefore, the statement in Section 2.3 of Attachment 3 that future lattices are bounded by lattices used in the peak reactivity methodology and therefore all future lattices "within the current fuel design" are qualified for storage in the Nine Mile Point Unit 1 SFP is valid. However, while the results presented in the application, specifically Table B.4

and Table B.5, provide evidence to support these statements, additional information is provided in this response.

The super lattice calculations in Table B.4 and Table B.5 were constructed as described in Section 2.3.2 of Attachment 3 of the submittal. There are two "types" of super lattices, that is the 10a61 - 10a64 and 10a82 - 10a84. Within each type of super lattice, the lattices have the same material composition with respect to fuel rod enrichment (4.6 wt% U-235) and Gd rod location. The difference between the 10a6* series and 10a8* series is the location of the Gd rods (specifically, 3 Gd rod locations are different). Between the two lattice series types, i.e. the 10a6* and 10a8*, the "1", "2" etc designation indicates the same lattice type with respect to number of fuel rods, number of missing rods, and number of plenum region rods. These in-rack reactivity variations among the super lattices as well as a comparison of their reactivity are summarized below:

- 10a61: 92 fuel rods, 0 part length rod locations, 0 plenum region rods, $k_{calc}$ = 0.8557 (no equivalent 10a8* series super lattice)
- 10a62: 84 fuel rods, 0 part length rod locations, 6 plenum region rods, $k_{calc}$ = 0.8516 (10a82 $k_{calc}$ = 0.8515)
- 10a63: 84 fuel rods, 6 part length rod locations, 0 plenum region rods, $k_{calc}$ = 0.8617 (10a83 $k_{calc}$ = 0.8612)
- 10a64: 76 fuel rods, 6 part length rod locations, 8 plenum region rods, $k_{calc}$ = 0.8493 (10a84 $k_{calc}$ = 0.8498)

As it can be seen from the comparison summary above, the reactivity difference between super lattices that have essentially the same SCCG reactivity but different storage rack reactivity is due solely to the number of fuel rods. From the results above we can see the following:

- The addition of plenum regions (i.e. no fuel but no increase in moderator) has a small negative reactivity impact.
- The reduction in fuel rods but the introduction of moderator (addition of part length rod locations) increases reactivity.
- The reactivity effect of the variation in Gd rod location has a negligible reactivity effect (within a few sigma of the uncertainty of the calculation).
- The source of the variation in reactivity between the super lattices which have the same SCCG reactivity by a different in-rack reactivity is dominated by the physical impact of the fuel to moderator ratio.

With respect to part length rod locations within fuel bundle designs, these locations are fixed (i.e. if a change is made to the part length rod location then it becomes a new fuel bundle design). Therefore, since that physical design is inherent in the fuel bundle design considered, the super lattices bound the fuel bundle design.

Finally, further clarification is provided to support the conclusion that the peak reactivity methodology employed with super lattices bounds all actual lattices, current and future.

Super lattice construction is described in Section 2.3.2 of Attachment 3. As part of the development of the super lattice methodology, the following points are made:

- The most reactive actual lattices were shown in Table B.3 to have a maximum in-rack kcalc = 0.8200.
- Since it is necessary to show compliance with the TS limits, these low reactivity actual lattices were modified with respect to enrichment (all rods have a reactivity of 4.6% for super lattice calculations) and various Gd rod loadings.
- The regulatory requirements of 0.95 have been shown to be met using these super lattices at the TS limits (4.6% and 1.31).
- These super lattices represent the upper limit of the fuel design possibilities with respect to enrichment and are not actual lattices.
- The evaluations using the super lattices show that for the variation in geometry related to the number of fuel rods, the reactivity effect of the increase in moderation from 92 to 84 fuel rods with 6 part length rod locations and no plenum rods is about 0.006 $\Delta k$ in reactivity.
- The reactivity difference between the bounding super lattice and the most reactive actual lattices is about 0.04 $\Delta k$.
- Thus, the reactivity differences seen in the in-rack reactivity for the super lattices that have the SCCG kinf of 1.31 (0.006 $\Delta k$) is much smaller than the reactivity difference between the super lattice and the most reactive actual lattice (0.04 $\Delta k$).
- Furthermore, as it has been shown above, the reactivity effect of variations in Gd rod location is very small, with respect to changes in reactivity in the storage racks for lattices with the same SCCG reactivity.
- Thus, the only means to significantly increase the reactivity of a future actual lattice is to either increase its enrichment and/or reduce the Gd rod loading.
- The super lattice method already considers the maximum allowed enrichment.

Therefore, while the lattice designs can change fuel rod enrichment and Gd rod number, loading and location, these lattice design changes within the bounding fuel design type are clearly bounded by the large difference in reactivity between as built lattices and the conservative lattices considered in the analysis. Thus, the analysis super lattices evaluations bound all current and future lattice designs within the fuel bundle design constraints.

**ATTACHMENT 3**

Affidavit

**AFFIDAVIT PURSUANT TO 10 CFR 2.390**

I, Kimberly Manzione, being duly sworn, depose and state as follows:

(1) I have reviewed the information described in paragraph (2) which is sought to be withheld, and I am authorized to apply for its withholding.

(2) The information sought to be withheld is provided in HI-2178001R0, which contains Holtec Proprietary Information.

(3) In making this application for withholding of proprietary information of which it is the owner, Holtec International relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC Sec. 552(b)(4) and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10CFR Part 9.17(a)(4), 2.390(a)(4), and 2.390(b)(1) for "trade secrets and commercial or financial information obtained from a person and privileged or confidential" (Exemption 4). The material for which exemption from disclosure is here sought is all "confidential commercial information", and some portions also qualify under the narrower definition of "trade secret", within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975F2d871 (DC Cir. 1992), and Public Citizen Health Research Group v. FDA, 704F2d1280 (DC Cir. 1983).

(4) Some examples of categories of information which fit into the definition of proprietary information are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by Holtec's competitors without license from Holtec International constitutes a competitive economic advantage over other companies;

b. Information which, if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product.

c. Information which reveals cost or price information, production, capacities, budget levels, or commercial strategies of Holtec International, its customers, or its suppliers;

d. Information which reveals aspects of past, present, or future Holtec International customer-funded development plans and programs of potential commercial value to Holtec International;

e. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection.

The information sought to be withheld is considered to be proprietary for the reasons set forth in paragraphs 4.c above.

(5) The information sought to be withheld is being submitted to the NRC in confidence. The information (including that compiled from many sources) is of a sort customarily held in confidence by Holtec International, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by Holtec International. No public disclosure has been made, and it is not available in public sources. All disclosures to third parties, including any required transmittals to the NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary agreements which provide for

maintenance of the information in confidence. Its initial designation as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in paragraphs (6) and (7) following.

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge. Access to such documents within Holtec International is limited on a "need to know" basis.

(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist or other equivalent authority, by the manager of the cognizant marketing function (or his designee), and by the Legal Operation, for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside Holtec International are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary agreements.

(8) The information classified as proprietary was developed and compiled by Holtec International at a significant cost to Holtec International. This information is classified as proprietary because it contains detailed descriptions of analytical approaches and methodologies not available elsewhere. This information would provide other parties, including competitors, with information from Holtec International's technical database and the results of evaluations performed by Holtec International. A substantial effort has been expended by Holtec International to develop this information. Release of this information would improve a competitor's position because it would enable Holtec's competitor to copy our technology and offer it for sale in competition with our company, causing us financial injury.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to Holtec International's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of Holtec International's comprehensive spent fuel storage technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology, and includes development of the expertise to determine and apply the appropriate evaluation process.

The research, development, engineering, and analytical costs comprise a substantial investment of time and money by Holtec International.

The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial.

Holtec International's competitive advantage will be lost if its competitors are able to use the results of the Holtec International experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

The value of this information to Holtec International would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive Holtec International of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing these very valuable analytical tools.

**AFFIDAVIT PURSUANT TO 10 CFR 2.390**

STATE OF NEW JERSEY )
) ss:
COUNTY OF CAMDEN )

Kimberly Manzione, being duly sworn, deposes and says:

That she has read the foregoing affidavit and the matters stated therein are true and correct to the best of her knowledge, information, and belief.

Executed at Camden, New Jersey, this 14th day of August 2018.

Kimberly Manzione
Licensing Manager
Holtec International

Subscribed and sworn before me this 14th day of August, 2018.

Erika Grandrimo
NOTARY PUBLIC
STATE OF NEW JERSEY
MY COMMISSION EXPIRES January 17, 2022