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Subject: Draft Safety Evaluation for GE-Hitachi Nuclear Energy Americas LLC (GEH) Topical Report NEDE-33284P, Supplement 1, "Marathon-Ultra Control Rod Assembly" (TAC No. ME3524).

In Reference 1, the NRC provided the draft Safety Evaluation (SE) of the subject topical report and requested that GE-Hitachi Nuclear Energy Americas LLC (GEH) identify any information that it considers proprietary and provide comments on factual errors or clarity concerns.

GEH has examined the draft SE for NEDE-33284P, Supplement 1 and finds no factual errors or clarity concerns. Enclosure 1 contains the draft SE in a redacted form which identifies the GEH proprietary information.

If you have any questions, please contact me or Scott Nelson at 910-819-5829.

Sincerely,

James F. Harrison
Vice President, Fuel Licensing
Regulatory Affairs
GE-Hitachi Nuclear Energy Americas LLC

Project No. 710

References:

1. Letter from J.R. Jolicoeur (NRC) to J.G. Head (GEH), Subject: Draft Safety Evaluation for GE-Hitachi Nuclear Energy Americas LLC (GEH) Topical Report NEDE-33284P, Supplement 1, "Marathon-Ultra Control Rod Assembly" (TAC No. ME3524), February 6, 2012.

Enclosures:

1. Redacted Draft Safety Evaluation, Non-Proprietary Information-Class I (Public)

Commitments:

This letter and its enclosure contain no commitments.

cc: SS Philpott, NRC
JG Head, GEH Wilmington
PL Campbell, GEH Washington
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ENCLOSURE 1

MFN 12-011

Redacted Draft Safety Evaluation

Non-Proprietary Information - Class I (Public)

DRAFT SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TOPICAL REPORT NEDE-33284P, SUPPLEMENT 1, REVISION 0

"MARATHON-ULTRA CONTROL ROD ASSEMBLY"

GE-HITACHI NUCLEAR ENERGY AMERICAS, LLC

PROJECT NO. 712

1.0 INTRODUCTION

By letter dated January 29, 2010, GE-Hitachi Nuclear Energy Americas, LLC (GEH) submitted Topical Report (TR) NEDE-33284P, Supplement 1, Revision 0, "Marathon-Ultra Control Rod Assembly," to the U.S. Nuclear Regulatory Commission (NRC) for review and approval (Reference 1). This TR provides design specifications along with mechanical lifetime and nuclear lifetime calculations for the new Marathon-Ultra control blade design. The TR was supplemented with GEH nuclear and mechanical lifetime models and calculations and GEH responses to the NRC staff's request for additional information (RAI) in letters dated March 4, 2011 (Reference 2), March 28, 2011 (Reference 3), and November 15, 2011 (Reference 4) respectively.

2.0 REGULATORY EVALUATION

Regulatory guidance for the review of fuel rod cladding materials and fuel system designs and adherence to Title 10 of the *Code of Federal Regulations* Part 50, Appendix A, General Design Criteria (GDC) for Nuclear Power Plants, GDC-10 "Reactor Design," GDC-27 "Combined Reactivity Control Systems Capability," and GDC-35 "Emergency Core Cooling" is provided in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (SRP), Section 4.2, "Fuel System Design" (Reference 5). In accordance with SRP, Section 4.2, the objectives of the fuel system safety review are to provide assurance that:

- The fuel system is not damaged as a result of normal operation and anticipated operational occurrences,
- Fuel system damage is never so severe as to prevent control rod insertion when it is required,
- The number of fuel rod failures is not underestimated for postulated accidents, and
- Coolability is always maintained.

NEDE-33284P, Supplement 1, provides nuclear and mechanical design calculations for the Marathon-Ultra control blade design. The NRC staff's review of this TR is to ensure that the Marathon-Ultra control blade design adequately addresses the regulatory requirements identified in SRP, Section 4.2.

The Marathon-Ultra control blade design has been evaluated to ensure compliance with the same licensing criteria as the original Marathon and Marathon-5S designs. As such, the NRC staff's review of the Marathon-Ultra control blade design followed the same logic as was used in the reviews for those designs (References 6 and 7).

3.0 TECHNICAL EVALUATION

The NRC staff's review of NEDE-33284P, Supplement 1, is summarized below:

- Verify that the control blade design criteria are consistent with regulatory criteria identified in SRP, Section 4.2.
- Verify that the control blade design criteria are consistent with past reviews.
- Verify that the mechanical design methodology is capable of accurately or conservatively evaluating each component with respect to its applicable design criteria.
- Verify that the nuclear design methodology is capable of accurately or conservatively evaluating boron depletion and blade worth.
- Verify that the Marathon-Ultra control blade design satisfies regulatory requirements.
- Verify that GEH's experience database supports the mechanical lifetime and nuclear lifetime being requested. If necessary, implement a surveillance program to monitor in-reactor behavior and confirm design calculations.

In addition to reviewing the material presented in Reference 1 and responses to RAIs, the NRC staff performed independent nuclear lifetime and mechanical lifetime calculations. Pacific Northwest National Laboratory (PNNL) assisted the NRC staff in the review of the Marathon-Ultra control blade component structural evaluations. PNNL's review of the Marathon-Ultra structural design analyses, documented in the attachment to this safety evaluation (SE), builds from prior reviews of the Marathon-5S and the Economic Simplified Boiling Water Reactor (ESBWR) control blade finite element analysis (FEA) models and methods.

3.1 Marathon-Ultra Mechanical Design Evaluation

3.1.1 Design Specifications

As described in Section 2 of NEDE-33284P, Supplement 1(Reference 1), the Marathon-Ultra control blade design is a derivative of the Marathon-5S design approved in Reference 6. The only differences between the two control blade designs are the absorber tube neutron poison loading pattern and the use of thin wall boron carbide (B_4C) capsules. Where Marathon-5S uses an all B_4C capsule design, the Marathon-Ultra design incorporates full-length hafnium rods

in outer edge, high-depletion tube locations. The outer structure of the control rod, consisting of the handle, absorber tubes, tie rod, and velocity limiter, is identical to the Marathon-5S design. Similarly, the component materials and manufacturing processes, including welding, are exactly the same. Table 2-1 of Reference 1 provides direct comparisons of design specifications between the two control blade designs for the different boiling water reactor (BWR) lattice configurations (e.g., C-, D-, and S-lattices).

The NRC staff understands the need for manufacturing flexibility, especially for shop maintenance and improvements. However, changes in design specifications or materials (e.g., alloying elements, thermal processing) may alter the basis for the NRC staff's approval of the Marathon-Ultra control blade design. Therefore, the NRC staff's approval is restricted to the design specifications provided within Section 2 of Reference 1, except as allowed within the provisions of Section 10 of Reference 1, as amended by the changes submitted with GEH's response to RAI-7 (Reference 4) and in accordance with Sections 3.1.1.1 and 3.1.1.2 of this SE.

3.1.1.1 Alternate Absorber Loading Patters

A good example of design flexibility which directly impacts the NRC staff's approval was provided in Section 10 of the Marathon-5S TR (Reference 6). During its prior review for the Marathon-5S TR, the NRC staff was unwilling to accept the hafnium option since the TR lacked nuclear and mechanical lifetime calculations unique to the hafnium design. Similarly, Section 10 of Reference 1 requests approval for design flexibility which would allow alternate load patterns of B_4C capsules and hafnium rods within the Marathon-Ultra control blade design. Reference 1 states that prior to implementation of any alternate loading pattern, GEH would demonstrate that the new absorber loading patterns meets all safety, design, and operational acceptance criteria presented in the TR including, but not limited to:

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- Demonstration of clearance between the hafnium rod and the outer absorber tube at end-of-life.
- Demonstration of acceptable stresses due to control rod scram, measured against applicable acceptance criteria.
- Demonstration of conformance to nuclear evaluation design criteria.

In response to RAI-7 regarding the alternate absorber loading patterns (Reference 4), GEH provided further details about the applicability, fixed and variable design parameters, evaluation methodologies, and acceptance criteria. In addition, a notification process consisting of a Compliance Demonstration Report is described. [[

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The material of the capsule body tubing may be varied from that shown in Table 2-1, [[ ]], provided the acceptance criteria described in Section 10.5 below are met.

The NRC staff had concerns with some wording in Section 10 of Reference 1, specifically the text “methodology equivalent to that in Section 4.2.” The GEH response to RAI-7 clearly states that methodologies used to evaluate any future alternative absorber loading will be identical to the methodologies reviewed by the NRC staff and that the nuclear analysis methodology shall not be modified unless specifically reviewed and approved by the NRC staff.

As part of its prior review for Marathon-5S, the NRC staff performed independent calculations and audited the B10 depletion calculations and FEA mechanical calculations for the all-B_4C capsule configuration. During this review, the NRC staff performed independent calculations and audited the GEH calculations supporting the combined B_4C capsule and hafnium rod configuration. Based upon these reviews, the NRC staff finds the methodology and design criteria acceptable for developing and implementing alternate absorber loading configurations. As such, the optional absorber load patterns provision detailed in the amended version of Section 10 of Reference 1 (submitted with the RAI-7 response, Reference 4), as amended by [[ ]] is acceptable.

3.1.1.2 <u>Applicability of Marathon-Ultra Design to the Advanced Boiling Water Reactor (ABWR) and ESBWR</u>

Section 1 of NEDE-33284P, Supplement 1 (Reference 1), requests NRC approval for the use of Marathon-Ultra control rods in “Boiling Water Reactors.” Section 11 of Reference 1 requests approval for design flexibility which would allow an alternate blade design applicable to the advanced reactor designs ABWR and ESBWR. The primary differences in the control rod designs are the replacement of the velocity limiter with a connector for both the ABWR and the ESBWR (coupling with a motor driven control rod drive system), and a shorter absorber section for the ESBWR.

In response to RAI-7 (Reference 4), GEH has proposed a more detailed control blade design change process by merging the alternate absorber loading and ABWR/ESBWR design options into a revised Section 10 of Reference 1. Section 11 of Reference 1 would be deleted. During its review, the NRC staff identified several methodology differences employed for the ESBWR control blade design relative to the methodology detailed in the Marathon-Ultra TR. These differences introduce uncertainty in the design change process outlined in the revised Section 10 (RAI-7, Reference 4). Furthermore, no mechanical design calculations have been provided with this TR for NRC staff review of the ABWR or ESBWR versions of the Marathon-Ultra control blade. Based upon these differences in design methodology and uncertainty in the design change process, the NRC staff’s approval does not include the ABWR or ESBWR design change option for the Marathon-Ultra control blade design.

In the final, approved version of this TR, Section 10 should be modified to clearly state that the design change process is not applicable to ABWR and ESBWR. Conforming changes may also be necessary throughout the TR.

The NRC staff's SE includes a limitation defining the regulatory definition of Marathon-Ultra as the detailed description provided in Section 2 of NEDE-33284P, Supplement 1. Any deviations must be within the bounds of Section 10 of NEDE-33284P, Supplement 1, as amended to restrict applicability to BWR/2 through BWR/6.

3.1.2 Operating Experience

The original Marathon control blade design, with its unique square absorber tube geometry, has extensive operating experience in the U.S. BWR commercial fleet. As part of its approval of the original Marathon design in 1991 (Reference 7), the NRC staff imposed a surveillance program requirement for GEH to monitor and confirm the control rod performance. Attachments 2 and 3 of Reference 7 provide details of the Marathon surveillance program. The surveillance program includes the following action statement:

> "Should evidence of a problem with the material integrity arise; (1) arrangements will be made to inspect additional Marathon control rods to the extent necessary to identify the root cause and (2) if appropriate, GE shall recommend a revised lifetime limit to the NRC based on the inspections and other applicable information available."

One weakness in the Marathon surveillance plan was the lack of required periodic reporting to the NRC. This is evident from the first Marathon surveillance program status report transmitted to the NRC, which was dated February 2007. During the 15 years between its approval and introduction and the first surveillance status report, the Marathon control blade had experienced in-reactor material degradation. Specifically, cracking was observed in the control blade handles and square absorber tubes.

The latest surveillance report (Reference 8) details the results of [[ ]] visual examinations conducted on Marathon control blades, including the following observations:

- No crack indications have been observed on any absorber tubes containing hafnium rods.
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The Marathon-5S control blade includes features designed to address the in-reactor material degradation experienced by the older Marathon design. As part of its approval of the Marathon-5S design in 2009 (Reference 6), the NRC staff required a more rigorous surveillance program which included annual reporting requirements. Detailed visual inspections were chosen to ensure that the Marathon-5S design features were not susceptible to the same material degradation problems observed in the older Marathon control blade design. The surveillance program was designed to detect material degradation due to early-in-life failure mechanisms (e.g., stress corrosion cracking, weld degradation) and validate end-of-life mechanical design lifetime predictions (e.g., absorber tube failure). In addition, surveillance was required for control blades in each lattice type and different BWRs.

The primary difference between the Marathon-Ultra and Marathon-5S is the introduction of hafnium rods in high-duty absorber tube locations. The configuration of the Ultra hafnium rods, including the material requirements, diameter, and length, are identical to the hafnium rods used in the existing Marathon design. Based on past operating experience which has shown no indications of cracks in absorber tubes containing hafnium rods, there is reasonable assurance that the hafnium rods will behave in an acceptable manner.

Section 3.3 of this SE describes the surveillance requirements for the Marathon-Ultra control blade.

3.1.3 Mechanical Design Evaluation

The same licensing criteria used to judge the acceptability of the original Marathon (Reference 7) and Marathon-5S (Reference 6) control blade designs were used for the Marathon-Ultra design. Specifically,

1) The control rod stresses, strains, and cumulative fatigue shall be evaluated to not exceed the ultimate stress or strain of the material.
2) The control rod shall be evaluated to be capable of insertion into the core during all modes of plant operation within the limits assumed in the plant analyses.
3) The material of the control rod shall be shown to be compatible with the reactor environment.
4) The reactivity worth of the control rod shall be included in the plant core analyses.
5) Prior to the use of new design features on a production basis, lead surveillance control rods may be used.

The first three licensing criteria will be discussed in this section. Section 3.2 addresses the fourth licensing criterion, reactivity worth. Section 3.3 addresses the fifth licensing criterion, which was modified to build upon the Marathon-5S surveillance program requirements.

3.1.3.1 Stress, Strain, and Fatigue

Failure or deformation of control blade components may challenge control blade insertion or may result in a loss of reactivity worth (i.e., leaching of B_4C). GEH's licensing criterion is that stresses, strains, and cumulative fatigue shall not exceed the ultimate stress or strain of the material due to normal, abnormal, emergency, and faulted loads. The integrity of the welds under these loading conditions is also part of this criterion. This criterion is consistent with SRP, Section 4.2 and therefore acceptable.

The outer structure of the Marathon-Ultra control rod design, consisting of the handle, absorber tubes, tie rod, and velocity limiter, is identical to the Marathon-5S design. Similarly, the component materials and manufacturing processes, including welding, are exactly the same. As such, many of the Marathon-5S mechanical design analyses are directly applicable to the Marathon-Ultra design. Section 3 of NEDE-33284P, Supplement 1 details the structural evaluation for the Marathon-Ultra control blade components under various loading conditions. According to Table 3-24 of NEDE-33284P, Supplement 1, the following mechanical design analyses are unchanged from the Marathon-5S design:

- External Pressure and Channel Bow Lateral Load Analysis
- Internal Pressure Analysis
- Pressurization Stress on Absorber Tubes Analysis
- Combined Internal Pressure + Fuel Channel Bow Induced Bending Analysis

Due to slight design differences, the thermal analysis and lifting load analysis were reanalyzed for the Marathon-Ultra control rod design using the same methodology as the Marathon-5S design. PNNL's technical review of these two design analyses is documented in the attachment to this SE. In response to RAI-2 regarding the lifting load analysis (Reference 4), GEH provided an alternative lifting load evaluation including a weld quality factor. In response to RAI-7 (Reference 4), GEH confirmed that the alternate loading patterns would not exceed the maximum control blade weights listed in Table 2-1, so the alternate lifting load evaluations reported in RAI-2 cover the permissible range of the alternate absorber loads and demonstrate a positive design margin. Because the NRC staff's review relies upon the alternate lifting load evaluation provided in the RAI-2 response, rather than the methodology defined within the originally submitted Marathon-Ultra TR (Reference 1), approval of the Marathon-Ultra design and the optional design change process in Section 10 of NEDE-33284P, Supplement 1, is limited to the control blade weights listed in Table 2-1 of NEDE-33284P, Supplement 1.

The Marathon-5S control blade introduced new design features which were intended to avoid problems observed with prior control blade designs. These same features were maintained for the Marathon-Ultra control blade design and are summarized below:

- Field inspections of the existing Marathon control blades revealed cracking in the handle near the roller pin. The root cause was determined to be IASCC prompted by chemical remnants (from the manufacturing process) within the roller pin hole. Note that due to its design and geometry, it is believed that stagnant flow conditions existed in the pin hole.

This stagnant condition allows for the chemical interaction (along with mechanical loading) needed to produce IASCC. The Marathon-5S and Marathon-Ultra control blade designs eliminate the handle roller pins. Figures 2-3 and 2-4 of NEDE-33284P, Supplement 1, illustrate the spacer pad and plain extended handle design.

- Field inspections of the existing Marathon control blades revealed absorber tube cracking. These cracks may be the result of either (1) under prediction of swelling in B_4C with irradiation or (2) over prediction of strain capability in absorber tube material with irradiation. [[

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[[ ]] the limiting mechanical lifetime mechanism for the Marathon-5S and Marathon-Ultra designs is the pressurization of the absorber tubes due to the release of helium gas from the absorption of neutrons by the B_4C powder. Based upon an identical absorber tube design, the Marathon-5S internal pressurization analysis and confirmatory burst tests are applicable to the Marathon-Ultra control rod design.

The end of life ^{10}B depletion calculations demonstrate that the Marathon-Ultra design is nuclear lifetime limited for all lattice configurations. In other words, ^{10}B depletion leads to a loss of 10 percent cold worth prior to exceeding the allowable limit for internal pressure due to the associated helium release.

Based upon the applicability of previously approved Marathon-5S design analyses along with PNNL's review including its independent calculations, the NRC staff finds the Marathon-Ultra control rod mechanical design analyses acceptable.

3.1.3.2 Control Rod Insertion

Failure or deformation of control blade components may challenge control blade insertion. GEH's licensing criterion is that the control rod shall be evaluated to be capable of insertion into the core during all modes of plant operation within the limits assumed in the plant analyses. This criterion is consistent with SRP, Section 4.2 and therefore acceptable.

The thickness of the Marathon-Ultra wing (i.e., absorber tube cross section) is identical to the Marathon-5S and Marathon designs. Other envelope dimensions, including those for control rods with plain handles or with spacer pads, are also identical. Therefore, the fit and clearance of the Marathon-Ultra control blade in the fuel cell is identical to the Marathon-5S and Marathon which have significant operating experience.

As discussed in Section 3.1.3.1 above, mechanical design analyses demonstrate that the Marathon-Ultra design is capable of withstanding all normal, abnormal, emergency, and faulted loads without permanent deformation or failure, and therefore maintains the capability of

insertion.

As discussed in Section 3.4.4 of NEDE-33284P, Supplement 1, seismic scram tests of the Marathon-5S were performed. The test facility consisted of a simulated pressure vessel and reactor internals, and a control rod drive. Prototype Marathon-5S control blades were installed and the control rod drive was set to simulate D-, C-, and S-lattice operation. GEH's criteria for the seismic testing are (1) control rod insertion within scram time requirements at Operational Basis Earthquake conditions and (2) control rod insertion at Safe Shutdown Earthquake conditions. These criteria satisfy applicable SRP requirements and are therefore acceptable.

The parameters affecting seismic scram performance are the bending stiffness of the assembly, and the overall weight of the assembly. In general, a stiffer assembly and a heavier assembly will have slower seismic scram times. The test specimens used for the Marathon-5S seismic scram tests were purposefully made heavier than production Marathon-5S assemblies as a test conservatism. The weight of production Marathon-Ultra control rod assemblies is also conservatively bounded by the weight of the test assemblies. Because the outer structure of the Marathon-Ultra is identical to the Marathon-5S, the lateral bending stiffness will also be identical. Therefore, the Marathon-5S seismic scram tests apply equally to the Marathon-Ultra control blade design.

Based upon the applicability of previously approved Marathon-5S design analyses and seismic testing, the NRC staff finds that the Marathon-Ultra control blade design satisfies the control rod insertion licensing criterion.

3.1.3.3 Control Rod Material

GEH's licensing criterion is that the material of the control rod shall be shown to be compatible with the reactor environment. This criterion is consistent with SRP, Section 4.2 and therefore acceptable.

The Marathon-Ultra control blade design uses the same materials as the Marathon and Marathon-5S control rod designs. No new material has been introduced. The Marathon-Ultra and Marathon-5S share the same absorber tube design made from the same high-purity stabilized type 304 stainless steel as the Marathon absorber tubes. Material testing and the service history of the Marathon control rod blades confirm the compatibility of the materials with the reactor environment.

One of the top challenges facing operating BWRs is shadow corrosion induced channel bow and resulting control blade interference. Deep control blade insertion programs are sometimes used to hold down excess reactivity in order to achieve longer operating cycles. The close proximity of the type 304 stainless steel blades with the zircaloy channel boxes for extended duration could result in shadow corrosion. The industry has developed fuel management programs coupled with augmented surveillance programs to aid in managing channel bow. Changes in channel design and materials are also being introduced to limit control blade interference. At this time there does not appear to be an easy fix to this phenomenon besides channel replacement; however, there is no evidence that any features of the Marathon-Ultra

design will exacerbate the problem.

Based upon in-reactor service of these materials, the NRC staff finds that the Marathon-Ultra design has satisfied this licensing criterion.

3.2 Marathon-Ultra Nuclear Design Evaluation

3.2.1 Design Specifications

Section 4 of Reference 1 details the Marathon-Ultra nuclear evaluation design criteria and depletion methodology. Section 4.1 states that "a control rod's nuclear worth characteristics shall be compatible with reactor operation requirements." Using precedence from the approved Marathon-5S control blade design (Reference 6), GEH meets these compatibility limits by demonstrating that the initial hot and cold control blade reactivity worths are within ±5 percent $\Delta k/k$ (defined by $1\text{-}k_{con}/k_{unc}$) of the original equipment design worth.

GEH defines the control blade nuclear lifetime as "the quarter-segment depletion at which the control rod cold worth ($\Delta k/k$) is 10 percent less than its zero-depletion cold worth." (Reference 1, Section 4.1). As discussed previously, a new design may have an initial cold worth that differs by up to ±5 percent of the initial cold worth of the original equipment control blade. The end of nuclear lifetime for the new control blade design is defined as the quarter-segment depletion at which the cold worth is the same as the end of nuclear lifetime cold worth of the original equipment control blade that it is replacing. The NRC staff agrees with this approach with the understanding that a new design is always compared with the original equipment nuclear design (e.g., Duralife) and not the control blade design that is being replaced if multiple control blade design replacements have occurred over the plant's lifetime.

3.2.2 Nuclear Design Evaluation

The goal of this review was to verify that the end of nuclear lifetime for the control blade is being calculated appropriately. Proper determination of the end of nuclear lifetime is important to ensure that a given control blade always satisfies the established reactivity worth criteria for safe operation of the blade with respect to reactivity control. This was done by verifying the underlying modeling assumptions, reviewing the calculational models, and performing independent confirmatory analyses.

3.2.2.1 Methodology

The nuclear lifetime for a particular control blade is calculated by the use of a two-dimensional Monte Carlo analysis applied in a step-wise fashion in order to account for ^{10}B depletion over time. For each time step, the poison reaction rates are assumed to be constant and the poison inventories are calculated in each discrete area of the blade. The poison number densities are then updated by averaging on a cell by cell basis and the process is repeated until the reduction in cold worth reaches the end of nuclear lifetime criterion. This process was used and approved previously for the Marathon-5S control blade design (Reference 6).

The main code used by GEH to calculate the amount of ^{10}B depletion and the various k-effective values used to determine the change in cold worth is MCNP4A. Use of this code was approved in the NRC staff's SE of the Marathon-5S TR (Reference 6). Important parameters in the MCNP4A input were verified such as model geometry, moderator densities (including verification of the stated void fraction), and nuclear data (including the proper temperature specification and physics models). The geometry was also checked for errors using a visualization program.

3.2.2.2 Nuclear Lifetime and Initial Control Blade Worth

The NRC staff reviewed the Marathon-Ultra control blade nuclear lifetime and initial blade worth results for the D-, C-, and S-lattice designs as calculated by the methodology described in Section 3.2.2.1 of this SE. The control blade designs and corresponding fuel bundle designs that they will control, are described in Section 4 of Reference 1. A sample of these designs, the S-lattice, was chosen for more in-depth review. The S-lattice design contains [[ ]] As was previously stated, the end-of-life criterion for the original equipment control blade is a 10 percent change in cold worth occurring in any quarter segment of the control blade. The end of nuclear lifetime for the new control blade design is reached when the cold worth is the same as the end of nuclear lifetime cold worth of the original equipment control blade. The amount of ^{10}B depletion calculated at this point (expressed as a percent of the initial loading) then becomes the quarter segment control blade depletion limit which defines the control blade end of nuclear lifetime.

The confirmatory analysis for the S-lattice design relied on the T-DEPL calculational sequence of the TRITON module within the SCALE 6 software suite (Reference 9). The model was built according to the S-lattice specifications given in Table 4-15 and fuel lattice information given in Figure 4-3 of Reference 1. The confirmatory model was also visually compared with the MCNP4A model for consistency. Figure 3.2-1 of this SE shows the two models side-by-side.

As documented in RAI-3 of Reference 4, the NRC staff calculated a different change in relative worth versus equivalent ^{10}B depletion curve compared to GEH's curve given in Figure 4-6 of Reference 1. The NRC staff noticed that GEH's curve showed a similar trend but appeared to be shifted by some amount. GEH indicated in its response that the curve was adjusted to match the reactivity worth of the zero-depletion original equipment in order to satisfy the mandatory matched-worth criterion. After accounting for the initial reactivity worth value for the original equipment blade (given in Table 4.8 of Reference 1) in the confirmatory analysis, the NRC staff reached the same end of nuclear lifetime result as GEH. The NRC staff consequently determined that the methodology described in Sections 4.1 and 4.2 of Reference 1 was correctly implemented.

GEH performed the control blade depletion calculations assuming fresh fuel throughout the period of irradiation. The NRC staff questioned the conservatism of the assumption in RAI-4 (Reference 4). GEH responded by stating that assuming fresh fuel throughout control blade depletion is conservative since the beginning of life fuel state gives the highest fission density. Consequently, the maximum neutron flux is being imposed on the surrounding blade throughout

the entire depletion calculation. The NRC staff confirmed this by performing two separate depletion calculations. One calculation was analogous to GEH's assuming fresh fuel throughout irradiation and the other depleted the fuel materials in addition to the control blade absorber materials. For the fresh fuel calculation, the maximum flux (averaged over all fuel pins containing 4.9 percent enriched UO_2) was [[

]] The NRC staff also looked at the neutron flux in the B_4C to observe the impact of the fresh fuel assumption throughout control blade depletion, and as stated by GEH and confirmed in the NRC staff's analysis (see Figure 3.2-3), this assumption does maximize the flux seen in the B_4C. Since the NRC staff observed that GEH's method is conservative, the NRC staff agrees with the presented approach.

The NRC staff also questioned the assumed 40 percent void fraction in the MCNP4A analysis. GEH uses a limiting axial profile shape corresponding to end-of-life to determine which quarter segment of the control blade is most limiting. Based on the shape provided and the results of GEH's analysis, the limiting segment occurs toward the bottom of the control blade relative to its positioning in the core. This indicates that a lower void fraction might actually be seen at this limiting quarter segment. Consequently, the NRC staff issued RAI-5 asking GEH to explain the basis for the 40 percent void fraction and whether or not this assumption is conservative (Reference 4). GEH explained that while the absorber depletion rate may be sensitive to the assumed void fraction, the depletion limit is not. The NRC staff performed a sensitivity study at a void fraction of 20 percent to verify this and the results show that using a lower void fraction gives the same result as the 40 percent void fraction case. Figure 3.2-4 shows the results of the NRC staff's sensitivity study. Since GEH's statement that the assumed void fraction is independent of the control blade depletion limit was confirmed, the NRC staff found the approach to be acceptable.

In RAI-6, the NRC staff questioned the treatment of the hafnium absorber during the depletion calculation since the end of nuclear lifetime is related only to ^{10}B depletion (Reference 4). The NRC staff also asked whether alternate absorber loading patterns would invalidate the claim that the Marathon-Ultra control blade design is nuclear lifetime limited. Based on the response provided by GEH, the hafnium absorptions are converted to equivalent ^{10}B absorptions and are included in the determination of the total amount of ^{10}B depletion as a function of the change in control blade cold worth. This is done by preserving the reaction rates which are calculated in MCNP4A. The NRC staff finds this treatment acceptable since it only serves to simplify the tracking of the absorber material under irradiation and does not affect the control blade depletion limit. GEH also referred to Section 10 of Reference 1 stating that the impact of alternate absorber loading patterns on nuclear and mechanical lifetime shall be evaluated on an as-needed basis further stating that a technical SE must demonstrate that all safety, design, and operational acceptance criteria will be met before any loading patterns are offered. The NRC staff finds that re-analysis of all future proposed loading patterns using the same stipulations used for the currently proposed pattern is acceptable to indicate whether the future pattern will be nuclear or mechanical lifetime limited.

3.2.2.3 Radial Peaking Profile

One important aspect of the end of mechanical lifetime calculation is determining the radial peaking profile across the absorber wing. The mechanical tube limit is based on the amount of helium pressurization as a result of nuclear interactions within the control blade tubes containing ^{10}B. The radial peaking profile factors into the control blade mechanical design since tubes with high radial peaking have a proportionally higher pressure due to an increased reaction rate which influences the allowable number of absorber capsules in a given absorber tube. This is important in determining the feasibility of a given absorber loading pattern for a given control blade design.

Furthermore, the radial peaking profile needs to be calculated correctly so that it can be accurately determined that the design is either nuclear lifetime or mechanical lifetime limited. The ^{10}B depletion is compared to the mechanical lifetime limit by using the axial and radial profiles to determine the amount of localized depletion occurring in each of the 24 nodes in GEH's model. Once the profiles have been applied to each node in a given absorber tube, the average ^{10}B depletion is calculated and compared to the tube mechanical limit which is determined as part of the mechanical analysis. [[

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Radial peaking for a given absorber tube is calculated by tallying the total reaction rate in the tube and normalizing by the average reaction rate among all tubes. The peaking factor calculated by GEH for the [[ ]] and is consistent with the NRC staff confirmatory case that calculated a value of [[ ]]. The radial peaking profile calculated by the NRC staff was also seen to be consistent with that calculated by GEH. Figure 3.2-5 shows both GEH and NRC staff calculated profiles. Based on the NRC staff's review and the result of the confirmatory calculation, there is reasonable assurance that the radial peaking profile is being correctly calculated and applied so that the absorber tubes are designed to be within the established mechanical limits.

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Figure 3.2-1: 2-D View of Modeled S-Lattice Fuel Bundle.
(Triton Model on Top, MCNP4A Model on Bottom)

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Figure 3.2-2: Averaged Neutron Flux for the 4.9 Percent Enriched UO2 Fuel Pins

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Figure 3.2-3: Neutron Flux in the Innermost B-10 Ring of the Innermost B-10 Cell

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Figure 3.2-4: S-Lattice CRB Cold Worth Reduction

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Figure 3.2-5: Radial Peaking Factors for the S-Lattice CRB Calculated with KENO-VI.

3.3 Marathon-Ultra Surveillance Program

Due to limited in-reactor service and no post-irradiation examinations for the Marathon-5S/Marathon-Ultra control blade designs, a surveillance program is necessary to confirm acceptable performance and lifetime calculations. The Marathon-5S surveillance program (Reference 6) was designed to detect material degradation due to early-in-life failure mechanisms (e.g., stress corrosion cracking, welding degradation) and validate end-of-life mechanical design lifetime predictions (e.g., absorber tube failure). In addition, surveillance is required for control blades in each lattice type and different BWR plants.

Section 6.5 of NEDE-33284P, Supplement 1 defines the proposed surveillance program for the Marathon-Ultra control blade design. This program is designed to complement the existing Marathon-5S surveillance program. In response to RAI-8 regarding the surveillance program (Reference 4), GEH provided further detail and proposed an additional inspection requirement. The amended surveillance program is listed below.

- A minimum of two (2) Marathon-Ultra control rods will be inserted in high-duty locations in a D-, C-, or S-lattice, domestic or international BWR.
- Additional Marathon-Ultra control rods may be inserted in other domestic BWRs, with the intent that they remain at a lower depletion than the two lead-depletion Marathon-Ultra control rods at the designated BWR. Should other control rods at a domestic or international BWR become the highest depletion in the BWR fleet, they will become the control rods inspected per this surveillance program.
- The two lead-depletion control rods will be irradiated, achieving as close to nuclear end-of-life as practical (target minimum 90 percent of end-of-life).
- For refueling outages in which the depletion of the lead Marathon-Ultra assemblies are greater than 75 percent of design nuclear life, the two (2) highest depletion Marathon-Ultra control rods will be moved to the spent fuel pool, with a visual inspection of all eight faces of each control rod performed. Lead Marathon-Ultra control rods may exceed 75 percent depletion prior to the eight-face inspections planned in the spent fuel pool as long as those inspections are performed before the control rods are utilized in another fuel cycle.
- For Marathon-Ultra control rods inserted in the opposite lattice type as the lead depletion units, two (2) highest depletion control rods shall be visually inspected during refueling outages in which the depletion of the control rods exceeds 90 percent of design nuclear life. These visual inspections shall consist of an inspection of all eight faces of the control rod. For the purpose of this surveillance program, D- and S- lattice applications are considered equivalent, since the geometry of the absorber tubes and capsules are identical. For example, if the lead depletion control rods are in a D- or S-lattice plant, inspections of the lead C-lattice Marathon-Ultra control rods shall be performed during outages for which the depletion exceeds 90 percent of the design nuclear life. Conversely, if the lead depletion Marathon-Ultra control rods are in a C-lattice plant, additional inspections of D- or S-lattice Marathon-Ultra control rods shall be performed during outages for which the depletion exceeds 90 percent of the design nuclear life.

- To confirm the end-of-life performance of the Marathon-Ultra control rod, the first twelve (12) control rods of each lattice type (D/S-lattice and C-lattice) shall be visually inspected upon discharge, for a total of 24 visual inspections, not to exceed four (4) control rods from any single plant. These visual inspections shall consist of an inspection of all eight faces of each control rod.
- Should a material integrity issue be observed, GEH will (1) arrange for additional inspections to determine a root cause and (2) if appropriate, recommend a revised lifetime limit to the NRC based on the inspections and other applicable information available.
- If, after the completion of the end-of-life visual inspection of the first twelve (12) control rods of each lattice type are complete, additional control rods reach a ¼ segment depletion that is 5 percent higher than the twelve inspected control rods, a minimum of four (4) of the additional control rods shall be visually inspected.
- GEH will report to the NRC the results of all Marathon-Ultra visual inspections at least annually.

The NRC staff finds that the proposed surveillance program provides reasonable assurance that material degradation mechanisms will be identified, evaluated, and reported in a timely fashion and therefore is acceptable.

4.0 LIMITATIONS AND CONDITIONS

Licensees referencing NEDE-33284P, Supplement 1 must ensure compliance with the following conditions and limitations:

1) In the approved (-A) version of NEDE-33284P, Supplement 1, Section 10 shall be revised according to the changes submitted with GEH's response to RAI-7 and the requirements of Sections 3.1.1.1 and 3.1.1.2 of this SE.

2) Except as allowed within the provisions of Section 10 of NEDE-33284P, Supplement 1, as amended by the changes submitted with GEH's response to RAI-7 and in accordance with Sections 3.1.1.1 and 3.1.1.2 of this SE, the Marathon-Ultra control blade design is restricted to the design specifications provided within Section 2 of NEDE-33284P, Supplement 1. Changes in component design, materials, or processing specifications may alter the in-reactor behavior of this design and the basis of the NRC staff's approval. Specifically:

   a) Approval of the Marathon-Ultra control rod design is limited to application in the BWR/2 through BWR/6 lattice configurations defined in Table 2-1 of NEDE-33284P, Supplement 1. The optional ABWR and ESBWR Marathon-Ultra control rod design is not part of this approval.

   b) Approval of the Marathon-Ultra control rod design is limited to the ranges in control rod weight listed in Table 2-1 of NEDE-33284P, Supplement 1.

   c) Approval of the Marathon-Ultra control rod design is limited to natural ^{10}B. Enriched B_4C powder (i.e., artificial increase in ^{10}B isotopic concentration) was not considered in the

NRC staff's review and therefore is not permitted.

d) Approval of the Marathon-Ultra control blade design is limited to 304L capsules.

3) The inspection and reporting requirements in the Marathon-Ultra surveillance program, detailed in Section 3.3 of this SE, must be fulfilled.

5.0 CONCLUSION

Based upon its review of NEDE-33284P, Supplement 1, and the required surveillance program, the NRC staff finds the Marathon-Ultra control blade design acceptable for licensing applications in BWR/2 through BWR/6 power plants. Licensees referencing this topical report must comply with the limitations and conditions listed in Section 4.0.

Section 7 of NEDE-33284P, Supplement 1 details the impact of the Marathon-Ultra control blade design on standard plant technical specifications and concludes that there is no effect from the introduction of the Marathon-Ultra design. Since the details of each plant's technical specifications may vary, it is up to each licensee to determine if the introduction of the Marathon-Ultra control rod design necessitates a license amendment.

6.0 REFERENCES

1. Letter from GEH to NRC, MFN 10-034, "NEDE-33284P Supplement 1 and NEDO-33284 Supplement 1, 'Licensing Topical Report (LTR) Marathon-Ultra Control Rod Assembly,'" dated January 29, 2010. (ADAMS Package Accession No. ML100331610)

2. Letter from GEH to NRC, MFN 11-043, "Information Request Regarding the NRC Staff Review of NEDE-33284P, Supplement 1, 'Marathon-Ultra Control Rod Assembly' (TAC No. ME3524)," dated March 4, 2011. (ADAMS Accession No. ML110760290)

3. Letter from GEH to NRC, MFN 11-133, "Finite Element Analysis Information Request Regarding the NRC Staff Review of NEDE-33284P, Supplement 1, 'Marathon-Ultra Control Rod Assembly' (TAC No. ME3524)," dated March 28, 2011. (ADAMS Accession No. ML11104A0230)

4. Letter from GEH to NRC, MFN 11-245, "Response to Request for Additional Information Re: GE-Hitachi Nuclear Energy Americas Topical Report NEDE-33284P, Supplement 1, 'Marathon-Ultra Control Rod Assembly' (TAC No. ME3524)," dated November 15, 2011. (ADAMS Package Accession No. ML113200081)

5. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (SRP), Section 4.2, "Fuel System Design," Revision 3, dated March 2007. (ADAMS Accession No. ML070740002)

6. GEH TR NEDE-33284P-A, Revision 2, "Marathon-5S Control Rod Assembly," dated October 2009. (ADAMS Package Accession No. ML092950277)

7. GE Nuclear Energy TR NEDE-31758P-A, "GE Marathon Control Rod Assembly," dated October 1991. (ADAMS Legacy Library Accession No. 9107090009)

8. Letter from GEH to NRC, MFN 11-184, "Marathon Control Rod Assembly Surveillance Program Update," dated June 16, 2011. (ADAMS Accession No. ML1116714280)

9. SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation, ORNL/TM-2005/39, Version 6, dated January 2009.

Attachment: PNNL Evaluation

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Date:

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Technical Report: GEH Marathon-Ultra Control Blade Finite Element Analysis Calculations

Nick Klymyshyn

1.0 Introduction

This technical letter report details the PNNL evaluation of the finite element analyses (FEA) contained in "Marathon-Ultra Control Rod Assembly," NEDE-33284P, Supplement 1, Revision 0, January 2010. As defined in Section 2 of NEDE-33284P, Supplement 1, the only difference between the Marathon-Ultra (M-Ultra) and the Marathon-5S (M-5S) is the absorber section neutron absorber components. The outer structure of the control rod, consisting of the handle, absorber tubes, tie rod, and velocity limiter are identical. Further, the component materials and manufacturing processes, including welding, are exactly the same. Because of these limited changes, many of the FEA results reported in NEDE-33284P, Revision 2, October 2009 for the M-5S are still applicable to M-Ultra and the scope of this review was limited to the FEA models that were affected by the design changes.

The difference between the two designs is the internal configuration of the absorber tubes. The M-Ultra has thinner boron carbide capsule walls and some of the absorber tubes are filled with full length hafnium rods instead of boron carbide capsules. The changes to the boron carbide wall thickness affect the heat transfer scenario, which affects the peak temperature, the helium release rate, and the internal absorber tube pressure. The change to the internal components affects the total control blade weight, which affects the lifting load scenario and handle stresses. The FEA models that are not affected by these changes are noted in Table 3-24 of NEDE-33284P, Supplement 1 as being identical. The External Pressure + Channel Bow Lateral Load model is unaffected by the material or conditions inside the absorber tube because the model conservatively ignores internal pressure. The Internal Pressure model determines the maximum burst pressure based on the pressure required to cause the absorber tube to reach the material ultimate strength. One half of this burst pressure is considered to be the maximum allowable absorber tube pressure, which is used to determine the Pressurization Stresses in the absorber tube and the stresses occurring in the Combined Internal Pressure + Fuel Channel Bow Induced Bending analyses. This analysis strategy establishes a maximum internal pressure threshold, and evaluates stresses at that maximum allowable condition. Because the outside tube structure does not change, these models are applicable to both the M-5S and M-Ultra.

Section 2.0 describes the review process and history. Section 3.0 discusses the thermal model. Section 4.0 discusses the lifting load model. Section 5.0 summarized the conclusions of this review.

2.0 Review Process

GEH provided all of the FEA models associated with the M-Ultra in ANSYS input file format. The reviewer was able to run the models successfully and confirm that all results matched those reported in the topical report and to determine that the models were free of any fundamental modeling errors. Access to the model input files also allowed the reviewer to consider the effects of small modifications to the analysis methodology, such as modeling the lifting loads with three-dimensional structural models instead of two-dimensional, and the potential effects of GE's heat generation distribution assumptions on the thermal model peak temperature results.

Some points of the GEH analysis methodology were not clear from the models or available topical reports. The local heat generation rates specified in the thermal model is one example. Confirmatory calculations using the standard Jens-Lottes correlation were performed by the reviewer and matched the outside temperature of the absorber tube predicted by the GEH model, but it was not able to confirm the specific heat generation distribution applied to the B4C material. As many issues as possible were resolved before RAI questions were composed, but a number of issues still remained.

In preparation for an audit, open items were identified and transmitted to GEH in May 2011. During the audit at GEH – Wilmington in June 2011 (PNNL participated by phone), these open items were discussed. Information needed to support a safety finding was compiled and RAIs were issued. In a letter dated November 15, 2011 (MFN 11-245), GEH provided responses to the RAIs. These responses are discussed below.

3.0 Thermal Model

GEH uses a 2D ANSYS FEA model to calculate the peak temperature in the control rod. The geometry represents a horizontal section through a single vertical absorber tube. As illustrated in Figure 2-1 of NEDE-33284P Supplement 1, the components are: B4C powder region, capsule wall, helium gap, and absorber tube. The 2D model represents a slice out of the center and is taken to represent the full length of the absorber tube. In this thermal model the heat generation rates are the primary load. Heat is primarily generated within the central B4C powder zone and moves outward to the outside surface of the absorber tube. The amount of heat generated in the B4C is determined in the nuclear analysis code and applied to the ANSYS model as an input. The amount of heat allowed to pass out of the absorber tube into the coolant is derived from the Jens-Lottes correlation.

One important result of this model is the peak B4C temperature. This peak temperature is insensitive to many of the model parameters but is moderately sensitive to the distribution of heat generation in the B4C. The model divides the B4C into eight ring sections that each has its own heat generation rate specified, with a distribution that peaks in the outer ring. When this distribution was flattened to an average heat generation rate applied uniformly across the B4C, the peak temperature increased a notable amount. The results of this confirmatory analysis are

presented in Table 1. The “Baseline” case is the D/S Lattice worst case dimension results reported in Table 3-22 of NEDE-33284P, Supplement 1. The “Average” case uses the same model but sets all the B4C ring multiplication factors to one to explore the significance of the heat generation distribution. The result is an increase in centerline temperature [[ ]].

GEH’s method in this case was to calculate an average heat generation rate for the B4C in the nuclear analysis and assume a particular heat generation distribution (with a minimum at the center and a maximum at the outer radius) in the ANSYS thermal model. This assumption was questioned because the basis of the distribution was not clear, and the previously discussed confirmatory analyses showed the peak and average B4C temperature results were somewhat sensitive to the heat generation distribution. In the June telecon, GEH explained that the distribution was determined for a prior design and the average heat generation from the current Marathon Ultra nuclear analysis (which divided the B4C region differently) was scaled to the old distribution. They committed to justifying this method.

GEH resolved the heat generation distribution issue by showing that when a uniform average heat generation is applied to the nominal D/S lattice configuration the B4C capsule average temperature increases by [[ ]] This relatively small increase corresponds to an increased helium release fraction of [[ ]]. GEH references the boron carbide temperature to helium release relationship plotted in NEDE-33284P-A Revision 2, Appendix C, Figure 1, in stating that the amount of potential error is acceptable. The plot compares the temperature/release relationship to two test cases and it appears that the relationship is vastly conservative compared to the actual test data. The difference between the test case data and the modeled release fraction is on the order of 10% release fraction while the potential error due to heat generation distribution is only [[ ]] release fraction. This is a reasonable argument that there is no safety concern regarding heat generation distribution, but it may be worth noting that some facets of their standard modeling approach are highly conservative.

For future analyses it is recommended that the heat generation rates applied to the ANSYS thermal model be more clearly documented and more directly tied to their source (the nuclear analysis). The accuracy of the assumed heat generation distribution was not verified in this review. Instead, it was established that the potential error from making the heat generation distribution assumption was small compared to the expected degree of conservatism.

Other thermal model features were investigated. The thermal contact resistance value chosen by GEH to model energy transfer from the B4C to the capsule wall has a long history of use, but it is not based on specific experimental data. The argument that the thermal contact resistance value is appropriate is based on the fact that the pressurization methodology as a whole has been demonstrated to be highly conservative in tests reported in NEDE-33284P-A Revision 2, Appendix C. Confirmatory analyses show it takes a factor of two (or ½) applied to the contact resistance value to make a significant change in the Marathon-Ultra thermal model results.

The helium gap between the cladding and the absorber tube was also investigated. This is another model feature that is conservative rather than precise. It is modeled as a solid material with conduction as the only heat transfer mechanism. In reality, convection and radiation would add additional heat transfer capacity, but under-representing the transfer capacity leads to

conservatively higher temperatures in the B4C, which in turn leads to higher internal pressure and helium release which is conservative in the design analysis.

A confirmatory analysis using the Jens-Lottes (JL) correlation was performed to confirm the thermal model's prediction of the outside absorber tube surface temperature. The FEA model methodology applies a surface conduction coefficient to the outside absorber surface that is based on the JL correlation, which relates total heat flux to outer cladding surface temperature. A comparison between thermal FEA results and temperature estimates based on the JL correlation are listed in Table 2 for the D/S Lattice worst case geometry and Table 3 for the C Lattice nominal geometry. The comparison shows that the FEA model is close to the expected JL correlation value and consistently higher, which is conservative.

Table 1: Boron Carbide Heat Generation Distribution (D/S-Lattice Worst Case)

Distribution	Centerline (°F)	Ring4 OD (°F)	Ring8 OD (°F)
Baseline	[[		
Average			]]

Table 2: D/S-Lattice Worst Case Dimension Comparison to Jens-Lottes

	Crud Surface (°F)	Tube Surface (°F)
Thermal FEA	[[	
Jens-Lottes		]]

Table 3: C-Lattice Nominal Dimension Comparison to Jens-Lottes

	Crud Surface (°F)	Tube Surface (°F)
Thermal FEA	[[	
Jens-Lottes		]]

3.1 Thermal Model RAI Resolution (RAI-1)

The RAI issued to resolve the thermal model issues was in the format of a bulleted four-part question (RAI-01). Each bullet is listed here with a brief summary of the response. All issues raised by this RAI were satisfactorily resolved.

• Explain how the heat generation rates were determined for the thermal model. The B4C material was split into a number of rings, each with a particular heat generation rate. What is the basis for the diameters of the rings and the separate heat generation zones? How do these compare to the Marathon 5-S design, which has a different B4C capsule geometry?

Resolution: The average heat generation was scaled to fit an assumed radial distribution that

was originally determined for the Marathon 5-S. The division of the B4C in that case was based on the divisions in the nuclear analysis code. The division of the B4C in the Marathon Ultra nuclear analysis (4 uniform divisions) did not match the division in the existing thermal model (8 non-uniform divisions) so GEH assumed the distribution would be the same in both cases. The distribution affects the peak and average temperature calculated by the model, but GEH was able to show that assuming a conservative, uniform heat generation distribution would not affect the results enough to raise a safety concern.

• Explain how the convection coefficient that defines heat transfer between the B4C material and the capsule wall was determined. How well does this convection coefficient match experimental data? What physical conditions (such as temperature, diameter, amount of void space, etc.) affect this convection coefficient? Was the same convection value used in the M-5S and ESBWR? Is this convection coefficient intended to represent conduction and radiation heat transfer as well?

Resolution: The convection coefficient represents a thermal contact resistance that is intended to represent all heat transfer mechanisms. This is a constant approximated value that has been used in many design evaluations, including the Marathon, Marathon 5-S, and ESBWR. The justification for this value is that the model results using this value have been demonstrated to be conservative compared to experiments.

• Discuss the representation of the helium gap as a conductive material. With the change in gap size, is it necessary to include convection or radiation for correct heat transfer across the gap?

Resolution: Representing the helium gap as a conductive material is conservative because it neglects the other potential heat transfer mechanisms and thus provides more thermal resistance.

• Explain how the convection heat transfer coefficient between the crud layer and the coolant is calculated. This appears to be based on a Jens-Lottes correlation and modeled as a function of pressure, total heat generation, and exterior surface area. Was this same function used in the M-5S and ESBWR to define the convection coefficient? How well does this function match experimental convection data under similar conditions (temperatures, geometry, flow rates, etc.)?

Resolution: The convection coefficient to the coolant is a direct implementation of the Jens-Lottes correlation in the FEA model and independent calculations agree that it is correctly implemented. The identical method was used in the M-5S and ESBWR. GEH does not have direct comparison data for the Marathon-Ultra, but test data shows the methodology as a whole is conservative.

3.2 Thermal Model Review Conclusions

The review found no FEA modeling errors in the ANSYS thermal model, and confirms

that the models were behaving as intended. Confirmatory calculations using the Jens-Lottes correlation show that the ANSYS thermal model predicts the expected exterior temperatures. Some of the model's heat transfer parameters were not confirmable from test data or other means, but sufficient evidence was presented that the methodology as a whole leads to highly conservative internal pressure estimates. As this is the purpose of the ANSYS thermal model, to calculate internal pressure for comparison against a maximum pressure threshold, this model and methodology are found to be adequate.

The RAI responses provided information to support the conservatism of their methodology and to explain certain features of their model that were not clearly documented. The method of applying heat generation to the thermal model was not transparent, and an alternate evaluation using a conservative uniform distribution was used to show that the potential error in their assumptions was not significant compared to the degree of conservatism in their method as a whole.

Conservatism is a common theme in the ANSYS thermal model and in the RAI responses. The helium gap is treated conservatively as a conduction-only heat transfer path. Neglecting convection and radiation causes less heat to leave the central boron carbide and contributes to higher calculated temperatures. The contact resistance value used between the boron carbide and the capsule wall has been used for many designs, including the ESBWR and the original Marathon design, and its continued use is supported by the conservatism shown in NEDE-33284P-A Revision 2, Appendix C. The crud to coolant heat transfer behavior was similarly justified for its long use and contribution to conservative pressure results.

It should be noted that the model and methodology were demonstrated to be conservative as a whole. The FEA model assumes the maximum heat generation rate with the maximum peaking factor is applied to the entire length of an absorber tube and the only path for heat to leave is through the surfaces that directly contact the coolant. With these extreme assumptions, calculation of the B4C temperature, helium release fraction, and internal pressure should be conservative compared to actual in-reactor behavior. This makes it difficult to determine the contribution of individual model parameters, such as thermal contact resistance, to the overall conservatism of the analysis.

The method of assuming a heat generation profile in the boron carbide based on prior evaluations is a potentially non-conservative feature of the Marathon Ultra ANSYS thermal model. While this approach is found to be acceptable in this case due to the expected degree of overall conservatism, it is recommended that this not be continued in future analyses.

PNNL finds the ANSYS thermal FEA model and analysis methodology as a whole to be conservative, based on a review of the models, confirmatory analyses, the alternate uniform heat generation calculation, and the comparison to pressure test data reported in NEDE-33284P-A Revision 2, Appendix C.

4.0 Handling Load Model

The handling load model investigates the structural load on the handle due to a controlled

lift. The load is assumed to be twice the control rod weight (2*g*), which was also used in the Marathon 5-S. The ESBWR lifting load was analyzed at a higher load (3*g*) but it used a substantially different three-dimensional (3D) model and GEH staff explained at the time that 3*g* was known to be excessive. The actual physical lifting load is not well established, but from the slow speed is expected to be close to 1*g*. There is a small amount of positive buoyancy force in the submerged control blades, so the blades could be lifted with less than 1*g* lifting load applied to the handle. NRC has accepted 2*g* in the past for GEH and other vendors as an adequate representation of the lifting load.

GEH used a two-dimensional (2D) ANSYS FEA model of the handle plates for each of the handle design evaluations. All handle designs except the D Lattice Standard Handle are double bail configurations, comprised of two perpendicular interlocking plates joined by fillet welds. The 3D nature of the double bail designs requires some adjustment for analysis in a 2D model. The geometry of one handle plate is modeled with a thickness that represents the fillet welds instead of the constitutive plates. This method essentially focuses the load on the fillet welds as a conservative and computationally inexpensive simplification and alternative to 3D modeling. PNNL performed a confirmatory analysis of the lifting load on the D Lattice BWR/4 Extended Handle case by creating a 3D model from the 2D geometry of the GEH model. The peak stress intensity predicted in the weld regions of the 3D model was significantly lower than the stress predicted by the 2D model, supporting the conservatism of the GEH analysis method. Table 4 compares the 2D and 3D stress intensity results.

Table 4: Lifting Load Comparison

	Model Type	Model Results (maximum stress intensity)
GEH	2D	[[
PNNL	3D	]]

There were two initial issues of concern, the choice of temperature (70F) and the particular control rod weights used in the analyses. The necessity of a weld quality factor in the calculations was a third issue, raised during the June conference call/audit. The answer to most of these questions was straightforward. While the temperature was chosen to be 70F, the same ultimate tensile strength is valid up to 200F, and this covers the temperature range of other vendor evaluations. Maximum control rod weights and an appropriate weld quality factor were applied in a set of alternate calculations that showed the lifting stresses still remained below the design limits.

4.1 RAI Resolution (RAI-2)

The RAI issued to resolve the lifting load model issues was in the format of a bulleted two-part question (RAI-02). Each bullet is listed here with a brief summary of the response. All

issues raised by this RAI were satisfactorily resolved. GEH also satisfactorily addressed the issue of weld quality factor in their response to the second part of this RAI.

• Discuss the choice of analyzing the lifting load at a material temperature of 70F. Since yield strength and ultimate strength of the handle material decreases with temperature, is this a conservative temperature assumption?

Resolution: The allowable stress comes from the ASME Boiler and Pressure Vessel Code, and that value remains unchanged up to 200F. See: 2010 ASME Boiler and Pressure Vessel Code, Section II, Part D, "Properties (Customary)", Table U, pp. 486-487, line 46, SA-240, type 316, UNS S31600.

• The 2g lifting loads are based on control rod weights that are less than the maximum control rod weights listed in Table 2-1. Discuss the conservatism of these loads and the choice of control rod weight.

Resolution: GEH performed an alternate set of lifting load calculations assuming a weld quality factor of [[ ]] and maximum control rod weights from Table 2-1. The results of these more conservative evaluations all remained within the design allowable

4.2 RAI Resolution (RAI-7)

The handle lifting model is also related to RAI-7, which discusses the procedure for employing alternate absorber loads. GEH confirmed that the alternate loading patterns would not exceed the maximum control blade weights listed in Table 2-1, so the alternate lifting load evaluations reported in RAI-2 cover the permissible range of the alternate absorber loads and demonstrate a positive design margin.

4.3 Handling Load Model Conclusions

The handling load ANSYS FEA models were reviewed and found to be free of modeling errors. The choice to model the handles using a 2D method was explored using a 3D confirmatory model and found to be conservative. RAI questions were asked regarding the choice of temperature and the basis of the control rod weight used for the lifting load. The RAI responses explained that the material strength at the evaluated temperature was also correct for the full temperature range of interest. The response to the issue regarding control rod weight included an alternate lifting load study that included the maximum allowable control rod weights and a weld quality factor applied to the results.

The standard lifting load analysis methodology does not include a weld quality factor on the handle stress evaluation. Instead of justifying the lack of a weld quality factor in the

calculations, GEH performed alternate lifting load evaluations with a weld quality factor to show that a positive design margin existed for all the handle and lattice designs at the maximum weight listed in Table 2-1 of NEDE-33284P, Supplement 1. When alternate absorber tube loading configurations are considered, the total weight of the control rod may change and necessitate additional lifting load calculations at a new 2*g* lifting load. It is recommended that if such lifting load analyses are necessary, that they include consideration of weld quality. The existing alternate evaluations for the M-Ultra cover control rod weights up to the maximums listed in Table 2-1, so this will only be an issue if the alternate loading increases the weight beyond those values.

Based on the original and alternate lifting load ANSYS FEA models, PNNL finds that the Marathon Ultra handle designs are sufficient to withstand the handling loads. GEH's established analysis methodology does not include a weld quality factor, but it is recommended that weld quality be considered in future handling load evaluations of welded double-bail handle designs.

5.0 Conclusions

The thermal and handle lifting finite element models of the M-Ultra control rod assembly were reviewed and found to be reasonable and in-line with previous GEH models. The review process involved the sharing of GEH ANSYS input files to facilitate a fast and thorough evaluation of the models. PNNL staff were able to quickly confirm the results reported in the topical report supplement, check the models for errors, and perform confirmatory analyses and parameter variation studies prior to drafting RAI questions. The final RAI responses satisfactorily resolved all open technical questions.

Based on the review of the Marathon Ultra and Marathon-5S it can be concluded that the GEH methodology, inputs, assumptions and design criteria are acceptable and applicable to similar control rod designs. Two recommendations are made for future applications of this methodology. First, the heat generation distribution of the thermal model should be more clearly documented and more directly tied to the nuclear analysis model. Second, the handle lifting load analysis should consider the weld quality in welded double bail designs.

GEH has requested the freedom to alter the absorber material configuration, and these alterations may necessitate a full or partial re-evaluation of the control blade to reflect the changes in composition.

The current evaluation of the Marathon Ultra has considered a maximum control rod weight up to the value reported in Table 2-1 of NEDE-33284P, Supplement 1. If an alternate absorber tube loading configuration exceeds the Table 2-1 weight value it should be re-evaluated for handle lifting load and SCRAM. It is recommended that weld quality be considered in the handle lifting load evaluation, as it was done in the alternate lifting load evaluation performed for RAI-2.

The current evaluation of the Marathon Ultra has only considered 304L stainless steel as the boron carbide capsule material. Changes in capsule material will affect the ANSYS thermal model results and may require adjustments to the analysis methodology to maintain conservatism. [[

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The capsule that encapsulates the B4C is limited to stainless steel since this was the only ANSYS analysis provided.