ENCLOSURE 2

MFN 11-245

Response to NRC RAIs - NEDE-33284P, Supplement 1, Marathon-Ultra Control Rod Assembly

Non-Proprietary Information – Class I (Public)

INFORMATION NOTICE

This is a non-proprietary version of Enclosure 1 to MFN 11-245, from which the proprietary information has been removed. Portions of the enclosure that have been removed are indicated by an open and closed bracket as shown here. [[ ]]

RAI-1: Section 3.6.3 of Topical Report (TR) NEDE-33284P, Supplement 1, describes a thermal finite element analysis.

- Explain how the heat generation rates were determined for the thermal model. The boron carbide (B_4C) 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-5S design, which has a different B_4C capsule geometry?
- Explain how the convection coefficient that defines heat transfer between the B_4C 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 Marathon-5S and Economic Simplified Boiling Water Reactor (ESBWR)? Is this convection coefficient intended to represent conduction and radiation heat transfer as well?
- 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?
- 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 Marathon-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.)?

<u>GEH Response</u>:

With reference to Section 3.6.3 of NEDE-33284P Supplement 1:

- The boron carbide heat generation rates are determined as part of the nuclear analysis, as described in Section 4.5 of NEDE-33284P Supplement 1. The boron carbide column is split into eight concentric rings, such that the cross-sectional area of each ring is similar. The following table shows ratio of the outside ring radius (R) to the outer radius (Ro) of the boron carbide column for each ring. As shown, the same values of R/Ro have been used for both lattice types for both Marathon-5S and Marathon-Ultra.

Ring #	R/Ro	Capsule Ring Outside Radius (in)			
		Marathon-5S		Marathon-Ultra	
		D/S Lattice	C Lattice	D/S Lattice	C Lattice
8	[[				
7					
6					
5					
4					
3					
2					
1					]]

Table 1-1: Boron Carbide Ring Radius Values

- The boron carbide to capsule interface is modeled as a contact resistance of [[ ]]. This same contact resistance value has been used for Marathon, Marathon-5S, Marathon-Ultra and ESBWR Marathon designs, as well as previous control rod designs. It is meant to model the thermal resistance at the boron carbide to capsule interface, incorporating all modes of heat transfer. While there is no experimental data on the thermal resistance at the boron carbide to capsule interface, there is experimental data measuring the conservatism of the Marathon pressurization methodology, of which the thermal analysis is part. As discussed in Section 3.6.3 and 3.6.4 of NEDE-33284P Supplement 1, the primary purpose of the thermal model is to determine the temperature of the boron carbide, which affects the helium release fraction of this irradiated neutron absorber. As discussed in Appendix C of NEDE-33284P-A Rev. 2, these helium release fractions are used in the prediction of absorber tube pressurization, which is the mechanical life limiting mechanism for the Marathon-Ultra control rod. As shown in Appendix C of NEDE-33284P-A Rev. 2, the measured pressures are significantly less than the predicted values, demonstrating significant conservatism in the pressurization methodology, of which the thermal model is a part.
- The helium gap is conservatively modeled as a conduction layer. The additive effects of conduction and radiative heat transfer will tend to improve the heat transfer across this insulating layer, resulting in lower boron carbide temperatures. Therefore, ignoring convection and radiative effects is conservative, results in higher predicted boron carbide temperatures, helium release fractions, and absorber tube pressures, which are all conservative results.

- The heat transfer from the surface of the crud layer to the coolant is modeled using the Jens-Lottes heat transfer correlation for boiling heat transfer. The Jens-Lottes correlation is a function of pressure, local temperature and heat flux, and has been coded into the finite element input file. The identical methodology is used for the Marathon-5S, ESBWR Marathon, and Marathon-Ultra. Although there is no experimental data for the Marathon-Ultra scenario, experimental data on irradiated control rod absorber tube pressures demonstrate the conservatism of the pressurization methodology, of which the thermal model is a part.

An additional topic is raised regarding the radial heat generation distribution within the boron carbide powder cross-section. The 2-D finite element model currently used was developed for the Marathon-5S control rod project (NEDE-33284P-A Rev. 2). The input file for the finite element model is written such that the user inputs the average heat generation rate for the boron carbide cross-section, and the input file automatically calculates the relative heat generation rates (HGR) for each of the eight concentric rings that make up the finite element model.

At the time the model was created, the relative heat generation of each ring was established, based on nuclear analyses of Marathon control rods (NEDE-31758P-A). A plot of non-dimensionalized heat generation rate, HGR/(average HGR), versus the non-dimensionalized radius (radius/outer radius) is shown below. The green average values we used in the finite element input file to convert the single input average heat generation rate, into separate heat generation rates for each ring.

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Figure 1-1: Marathon Analyses Boron Carbide Heat Generation Profiles

In order to evaluate the effect of the heat generation distribution to the final results, an extreme case of a uniform heat generation distribution is evaluated. For comparison purposes only, the D/S lattice, nominal dimension case shown in Table 3-22 and Figure 3-10 of NEDE-33284P Supplement 1 is used as the baseline. Then, the finite element input file is modified, using a uniform heat generation rate for all boron carbide rings. A comparison of results is shown in the following table and graph.

Location	**Nodal Temperature (°F)**	
	Nominal Dimensions (Table 3-22 of NEDE-33284P Supplement 1)	**Nominal Dimensions, Uniform B_4C Heat Generation Profile**
Centerline	[[	
Ring1 OD		
Ring2 OD		
Ring3 OD		
Ring4 OD		
Ring5 OD		
Ring6 OD		
Ring7 OD		
Ring8 OD		
Capsule ID		
Capsule OD		
Abs Tube ID		
Abs Tube OD		
Crud Surface		
Avg B_4C		]]

Table 1-2: Thermal Analysis Results – Comparison to Uniform Heat Generation Case

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Figure 1-2: Thermal Analysis Results – Comparison to Uniform Heat Generation Case

As discussed in Section 3.6.3 of NEDE-33284P Supplement 1, the primary purpose of the thermal analysis is to determine the average boron carbide temperature in order to determine the boron carbide helium release fraction. As shown in the table, the use of an extreme case, uniform heat generation profile results in less than a [[ ]] increase in the average boron carbide temperature. The dependence of helium release fraction on average boron carbide temperature is shown in Figure 1 of Appendix C of NEDE-33284P-A. Based on this figure, an increase in boron carbide temperature of [[ ]] will cause a change in helium release fraction of less than [[ ]]. This is judged to be insignificant, as:

- The uniform heat generation profile is an extreme, unrealistic case
- The pressure prediction methodology, of which the thermal model is a part, shows significant conservatism relative to measured pressures, as described in Appendix C of NEDE-33284P-A Rev. 2.

Changes to NEDE-33284P Supplement 1 Revision 0:

In the course of this review, it is noted that Figure 3-8 of NEDE-33284P Supplement 1 Revision 0 is in error, as it shows results from the Marathon-5S analysis (NEDE-33284P-A Revision 2). This Figure will be updated with the following for the Acceptance version of NEDE-33284P Supplement 1.

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Figure 3-8: Absorber Tube and Capsule Thermal Finite Element Model

RAI-2: Section 3.7 of TR NEDE-33284P, Supplement 1, describes a handling load structural finite element analysis.

- Discuss the choice of analyzing the lifting load at a material temperature of 70 degrees Fahrenheit. Since yield strength and ultimate strength of the handle material decreases with temperature, is this a conservative temperature assumption?
- The 2*g* lifting loads are based on control rod weights that are less than the maximum control rod weights listed in Table 2-1 TR NEDE-33284P, Supplement 1. Discuss the conservatism of these loads and the choice of control rod weight.

<u>GEH Response</u>:

Concerning the handle lifting load of Section 3.7 of NEDE-33284P Supplement 1:

- The handle lifting load is analyzed only at room temperature (70 °F) as this load is only applied when moving the control rod when the reactor is shut down. Also note that in Table 3-13 of NEDE-33284P Supplement 1, the design ratio for the handle lifting load analysis is calculated using ½ of the material ultimate tensile stress as the allowable stress. Per the ASME Boiler and Pressure Vessel code[1], the ultimate tensile strength for the type 316 stainless steel handle material is constant at 75.0 ksi through 200 °F. Therefore, the handle lifting load stress calculation shown in Table 3-13 is applicable up to a temperature of 200 °F.
- Table 2-1 copies the same range of weights for each lattice type as was used in the original Marathon SE (NEDE-31758P-A) and the Marathon-5S SE (NEDE-33284P-A Rev. 2). As in the Marathon-5S SE (NEDE-33284P-A Rev. 2), the handle lift analysis is based on the actual weight of the proposed Marathon-Ultra assemblies. Conservatism in this analysis arises from:
  - The use of twice (2x) the actual control rod dry weight.
  - Ignoring the upward buoyant force on the submerged control rod, which is approximately [[ ]] at room temperature conditions.
  - The use of minimum material dimensions.

All of the handle configurations shown in Table 3-13 of NEDE-33284P Supplement 1, except for the D lattice "Standard Handle" are double bail configurations, with two interlocking plates joined at the top by fillet welds. Since the fillet welds are not full penetration welds, the fact that the strength of the welds is less than that of the full thickness plate must be addressed. For this 2-D analysis, this is done by conservatively setting the entire handle thickness to twice the minimum thickness of the fillet weld throat. Since the fillet welds have a minimum leg length of [[ ]], the handle thickness is set to [[ ]]. The exception is the D lattice single bail "Standard Handle", whose thickness is set to the minimum handle plate thickness.

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[1] 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 upper handle fillet weld is qualified, and is visually inspected on all production control rods. Consistent with Table 3-3 of NEDE-33284P Supplement 1, if a weld quality factor of [[ ]] were applied to this weld, the allowable strength of the weld will not be challenged, as evidenced by design ratios of approximately [[ ]] in Table 3-13 of NEDE-33284P Supplement 1.

To confirm these conclusions, an alternate calculation to that in Table 3-13 of NEDE-33284P Supplement 1 is performed. Two changes are made:

- Twice the maximum control rod weights from Table 2-1 of NEDE-33284P Supplement 1 are used as the applied load.
- A weld quality factor of [[ ]] is used for the double bail handle designs.

The results of this alternate analysis are shown in the following table.

Lattice Type	Handle Type	Control Rod Weight (lbs)	Peak Handle Stress (ksi)	Allowable Stress (ksi)	Design Ratio
D Lattice BWR/2-4	BWR/4 Extended Handle	[[			
	BWR/3 Extended Handle				
	Standard Handle				
C Lattice BWR/4,5	Extended Handle				
	Standard Handle				
S Lattice BWR/6	Standard Handle				]]

As shown in the table, the use of a weld quality factor and the maximum weights from Table 2-1 leaves ample margin in the handle lifting load calculation. Therefore, the handle structures are sufficient to withstand all expected loading during the handling of control rods during refueling outages.

Changes to NEDE-33284P Supplement 1 Revision 0:

In the course of this review, it is noted that Figure 3-9 of NEDE-33284P Supplement 1 Revision 0 is in error, as it shows results from the Marathon-5S analysis (NEDE-33284P-A Revision 2). This Figure will be updated with the following for the Acceptance version of NEDE-33284P Supplement 1.

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Figure 3-9: Handle Lifting Loads Finite Element Model

RAI-3: Tables 4-1, 4-2, and 4-3 of TR NEDE-33284P, Supplement 1 contain the depletion calculation results for the D, C, and S Lattice designs, respectively. Specifically, they list the calculated changes in hot and cold worth as a function of irradiation time with respect to an unirradiated blade. Why is there a change in hot and cold worth listed for the 0-day irradiated case? The U.S. Nuclear Regulatory Commission (NRC) staff's confirmatory calculation below depicts this change, which is potentially being propagated throughout the entire calculation. Address this apparent bias over the entire irradiation domain.

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<u>GEH Response</u>:

Note that these values signify blade worth as a function of the Marathon-Ultra worth (Δk/k) relative to the initial, zero-depletion reactivity worth of an *original equipment (OE) DuraLife-100 blade*. The initial reactivity worth values for OE blades are listed in Tables 4-6, 4-7, and 4-8 of NEDE-33284P Supplement 1.

The purpose of calculating these worth values relative to initial reactivity worth values for OE blades is to demonstrate that the replacement control blade satisfies the mandatory matched-worth criterion and to determine the equivalent B-10 depletion that yields a 10% worth reduction compared to the OE design. Discussion of the matched worth criterion is provided in Section 6.4.2 of NEDE-33284P Supplement 1.

Changes to NEDE-33284P Supplement 1 Revision 0:

No changes to the subject LTR will be made in response to this RAI.

RAI-4: In the nuclear design analysis presented in Chapter 4 of TR NEDE-33284P, Supplement 1, the depletion of the B_4C absorber material in the blades is tracked with Monte Carlo calculations. The NRC staff noted that while the depletion of the blades is tracked from time step to time step, the fuel assembly is assumed to be fresh throughout the analysis. Provide an explanation as to how this is conservative for calculating the limiting quarter-segment depletion.

GEH Response:

A beginning of life (BOL) fuel lattice is assumed as a conservative input to the depletion model since the fuel lattice will exhibit its highest fission density at BOL, thus maximum neutron flux impact on the blade throughout its life is conservatively assumed.

Changes to NEDE-33284P Supplement 1 Revision 0:

No changes to the subject LTR will be made in response to this RAI.

RAI-5: The B-10 depletion calculations described in Section 4 of TR NEDE-33284P are performed with a constant 40 percent void fraction. What is the basis for the 40 percent void fraction assumption and is this conservative for the expected limiting conditions?

GEH Response:

40% is a typical representative core average void fraction value for a BWR plant, and is the calculated average of all core average void fraction values listed in GEH internal operating plant parameter documentation. The range of core average void fraction values across the BWR fleet as listed in GEH internal operating plant parameter documentation varies from 16% to 44%.

The 40% void fraction value is used as a generalized constant in nuclear calculations, and has no bearing—in terms of conservatism—on the calculated nuclear lifetime for a control blade. Only the absorption-to-fission rate may be impacted by changes in the void fraction value. So while the depletion rate may change due to change in void fraction, the depletion limit will be unaffected by void fraction.

Heat generation rates may be impacted by changes in void fraction and resultant changes in absorption-to-fission rate. However, as specified in NEDE-33284P Supplement 1, only the peak boron carbide heat generation rates from nuclear analyses are used as input to downstream mechanical analyses. This assumption is inherently conservative.

It is additionally noted that void-dependent absorption to fission correlation (μ) values relating the "absorption rate" in the control blade poison to the fission rate in the adjacent fuel are provided to fleet customers of GEH control blades and are available for NRC review in NEDE-30931P Rev. 13. These μ values for 0%, 40% and 52% void conditions may be implemented in the GEH/GNF recommended variable void depletion model for core tracking. The use of the void dependent depletion rate model provides realistic poison depletion calculations that account for the axial changes in fast and thermal neutron spectra that accompany the changing void condition axially in the core.

Changes to NEDE-33284P Supplement 1 Revision 0:

No changes to the subject LTR will be made in response to this RAI.

RAI-6: The Executive Summary of TR NEDE-33284P, Supplement 1, states that the Marathon-Ultra control blade design is nuclear lifetime limited. Describe how the hafnium depletion is tracked in the nuclear lifetime calculations and whether alternate absorber loading patterns (described in Section 10 of TR NEDE-33284P, Supplement 1) would invalidate this statement.

<u>GEH Response</u>:

Since hafnium has multiple neutron absorbing isotopes that form a chain, as compared to the single high-neutron-capture cross section isotope in boron carbide, depletion of control blades utilizing hafnium is expressed in terms of B^{10}- equivalent depletion. This allows the current plant computer tracking models to be used with control blade designs using multiple absorber types.

For locations that incorporate hafnium, the chain absorber characteristics of that material are considered:

$$\frac{dN_{174}}{dt} = -\left(N \cdot \sigma\right)_{174}$$

$$\frac{dN_{176}}{dt} = -\left(N \cdot \sigma\right)_{176}$$

$$\frac{dN_{177}}{dt} = -\left(N \cdot \sigma\right)_{177} + (N \cdot \sigma)_{176}$$

$$\frac{dN_{178}}{dt} = -\left(N \cdot \sigma\right)_{178} + (N \cdot \sigma)_{177}$$

$$\frac{dN_{179}}{dt} = -\left(N \cdot \sigma\right)_{179} + (N \cdot \sigma)_{178}$$

$$\frac{dN_{180}}{dt} = -\left(N \cdot \sigma\right)_{180} + (N \cdot \sigma)_{179}$$

Here, σ is the reaction rate for B^{10} from the Monte Carlo code.

The number of absorptions from each of the regions is summed to obtain the total number of absorptions (A) for the time interval. This total number of absorptions is normalized by the total number of B^{10} atoms if the design would have incorporated only boron carbide as an absorber. The resulting value is the B^{10}-equivalent depletion:

$$\%_{depletion} = \frac{A}{N_{B-10}}$$

The lifetime in B^{10}-equivalent depletion contains embedded in it the total number of absorptions in a control blade, and the chain depleting characteristics of hafnium are treated correctly. The effect of including hafnium in a design is to increase the B^{10}-equivalent depletion limit.

The impact of alternate absorber loading patterns (as described in Section 10 of NEDE-33284P Supplement 1) on nuclear lifetime and mechanical lifetime shall be evaluated on an as-needed basis, per the statement issued in the first paragraph of Section 10: “Before any alternate load patterns are offered, a technical safety evaluation shall demonstrate that the control blades employing the alternate load patterns meet all the safety, design, and operational acceptance criteria presented within this report.”

Changes to NEDE-33284P Supplement 1 Revision 0:

No changes to the subject LTR will be made in response to this RAI.

RAI-7: Section 10 of TR NEDE-33284P, Supplement 1, describes a process whereby alternate absorber loading patterns may be developed and implemented without NRC involvement or notification.

- Confirm that the alternate absorber loading patterns are limited to interchanging B_4C capsules (and optional empty capsules) with a full length hafnium rod. In other words, a partial length hafnium rod will not reside within the same absorber tube as B_4C capsules.
- Confirm that the potential impact of weight differences between alternate absorber loading patterns is being addressed in the mechanical design calculations and identify the limiting loading pattern. Discuss the maximum possible control rod weights and how they compare to the loads used in the current lifting load finite element models.
- The NRC staff is considering imposing a letter notification requirement, similar to the GESTAR-II process, on any Marathon-Ultra control blade design with an alternative absorber loading pattern. The notification would provide detailed specifications of the alternate absorber loading pattern for each lattice configuration, document the acceptance criteria used to judge its performance, and confirm compliance with these criteria. Discuss the use of a notification process for future design alterations.

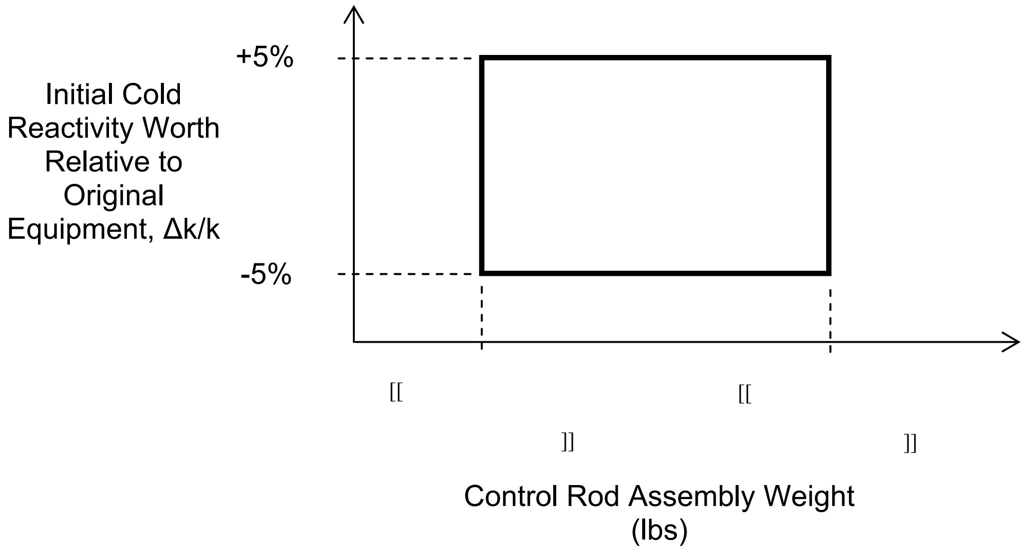
<u>GEH Response</u>:

With regards to Section 10 of NEDE-33284P Supplement 1.

- GEH confirms that alternate load patterns employed under Section 10 of NEDE-33284P Supplement 1 will not include partial length hafnium rods, but will instead only include full-length hafnium rods.
- For any control rod design employing optional load patterns, complete nuclear and mechanical analyses will be performed to ensure conformance to the licensing requirements contained in NEDE-33284P Supplement 1. The mechanical design will include the effects of any increased weight in both the scram loads, and the handle lifting loads analysis.
- GEH proposes the following notification process for alternate absorber loadings for Marathon-5S (NEDE-33284P-A) and Marathon-Ultra (NEDE-33284P Supplement 1) control rods.
    - <u>Application</u>:
        - Marathon-5S or Marathon-Ultra control rods with alternate absorber loadings may be applied to all Boiling Water Reactors (BWR), including BWR/2 through BWR/6, ABWR and ESBWR.
    - <u>Fixed Parameters</u>:
        - The outer absorber tube geometry as defined in Table 2-1 of NEDE-33284P Supplement 1 shall not be changed.
        - The outer absorber tube material, type 304S, as defined in Table 2-1 of NEDE-33284P Supplement 1 and Section 3.2.4 of NEDE-33284P-A shall not be changed.

- Absorber materials may only consist of vibratory compacted boron carbide with naturally occurring Boron-10 isotopic content, and hafnium.
- The vibratory compacted boron carbide must be contained within capsules within the outer absorber tube, providing a diametral gap between the inner capsule and the outer absorber tube.
- The outer absorber section structure, with absorber tube and a tie rod laser welded together as described in Section 3.2.4 of NEDE-33284P-A, must be maintained.

o <u>Varied Parameters</u>:
  - The length and location of boron carbide capsules, empty capsules, spacers, and hafnium rods may be varied for alternate absorber loading configurations. The use of hafnium rods is restricted to the use of full-length hafnium rods.
  - The diameter and wall thickness of the capsule tubing may be varied, such that the methodologies and acceptance criteria described below are met.
  - The material of the capsule body tubing may be varied from that shown in Table 2-1 of NEDE-33284P Supplement 1, [[ ]], provided the acceptance criteria described below are met.
  - Manufacturability changes to the velocity limiter and handle are permissible such that there is no negative affect on the fit, form, or function of these sub-components, and such that the acceptance criteria described below are met.
  - The overall weight of the control rod assembly may vary due to the alternate absorber load patterns, such that the weight of the control rod remains within the range of weights in Table 2-1, and such that there is no negative effect on the control rod insertion, withdrawal, or SCRAM performance.
  - The overall length of the absorber section may be reduced to accommodate ESBWR.
  - Control rods for ABWR or ESBWR application employ a connector rather than a velocity limiter, as described in Section 11 of NEDE-33284P Supplement 1.
  - In summary, the following figure summarizes permissible design space. The vertical axis of the design space is the primary nuclear requirement of matched initial reactivity worth as discussed in Section 4.4 of NEDE-33284P Supplement 1. The horizontal axis of the design space is the mechanical requirement of overall control rod weight, defined by the weight limits shown in Table 2-1 of NEDE-33284P Supplement 1.



- o <u>Methodologies</u>:
  - ▪ The mechanical and nuclear evaluation methodologies shall be identical to those described in NEDE-33284P-A and NEDE-33284P Supplement 1. The following are emphasized:
    - For the boron carbide swelling evaluation in Section 3.6, the evaluation shall use worst-case absorber tube and capsule dimensions, as well as a +3σ upper bound B_4C swelling limit based on available data.
    - The thermal and helium release fraction methodologies discussed in Section 3.6.3 shall remain unchanged.
    - The absorber tube pressurization methodologies in Section 3.6.4 shall remain unchanged, including the consideration of worst-case absorber tube dimensions, absorber tube surface defects and wear, and a factor of safety of 2.0.
    - Should any alternate absorber loading patterns change the control rod assembly weight, the SCRAM and handling loads shall be re-evaluated using the methodology of Sections 3.3 and 3.7.
    - The nuclear analysis methodology described in Section 4.2 shall not be modified unless specifically reviewed and approved by the NRC.
- o <u>Acceptance Criteria</u>:
  - ▪ The mechanical and nuclear evaluation acceptance criteria shall be identical to those described in NEDE-33284P-A and NEDE-33284P Supplement 1. The following are emphasized:
    - For the boron carbide swelling evaluation of Section 3.6, using worst-case absorber tube and capsule dimensions and +3σ upper bound B_4C swelling limits, [[

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- Using worst-case dimensions, a clearance between any hafnium absorber and the outer absorber tube at end of life shall be demonstrated.
- The allowable material stresses of Table 3-2 shall not change.
- The adoption of alternate load patterns may result in control rods with a longer nuclear lifetime than the Marathon-Ultra control rod. Should this occur, the surveillance program of Section 6.5 of NEDE-33284P Supplement 1 will continue to apply for the range of irradiation above the Marathon-Ultra lifetime limit.
- The nuclear design criteria of Section 4.1 shall not be changed.
- The licensing acceptance criteria of Section 6 shall not be changed.

o Notification:
  - Before any alternate loading pattern Marathon-5S or Marathon-Ultra control rods are delivered, GEH will provide NRC with a Compliance Demonstration Report.
  - The Compliance Demonstration Report will have content and format similar to NEDE-33284P-A and NEDE-33284P Supplement 1, shall confirm that the fixed parameters listed above are unchanged, and fully describe the changes and acceptability of all changed parameters.
  - The Compliance Demonstration Report will also be provided to BWR licensees to support 10 CFR 50.59 evaluations.

Changes to NEDE-33284P Supplement 1 Revision 0:

GEH will incorporate the notification process described above into Section 10 of the Acceptance version of NEDE-33284P Supplement 1, and will delete Section 11 The revised Section 10 is attached to the back of this enclosure beginning on Page 24.

RAI-8: Section 6.5 of TR NEDE-33284P, Supplement 1, describes the surveillance program to confirm in-reactor performance. The proposed surveillance requirements are based upon those established for the Marathon-5S design. The fourth and fifth bullet were designed to confirm the mechanical performance as the blade approaches the nuclear lifetime and the breakpoint (percent of design nuclear lifetime) was originally selected based on in-reactor degradation experienced with the Marathon-5S design.

- Discuss the logic used to alter the breakpoint in the fifth bullet (90 percent of design nuclear lifetime) relative to the requirement for the Marathon-5S design ([[ ]] of design nuclear lifetime).
- Discuss the extension of this breakpoint to 90 percent of design nuclear lifetime (in the fifth bullet) and the potential for this surveillance to merge with the end-of-life surveillance requirement (sixth bullet). In other words, will utilities elect to retire a blade once it exceeds 90 percent of design nuclear lifetime (based on concerns that the end of life would be exceeded during a subsequent operating cycle)?

<u>GEH Response</u>:

In Section 6.5 of NEDE-33284P Supplement 1, GEH asserts that visual inspections performed on irradiated Marathon-5S apply equally to Marathon-Ultra assemblies. The basis for this assertion is (1) the absorber tube and tie rod structures are identical, and (2) the capsule clearance requirements are identical. However, the nuclear lifetime of the Marathon-Ultra exceeds that of the Marathon-5S. Therefore, the Marathon-Ultra inspection program should cover the region of Marathon-Ultra lifetime beyond the Marathon-5S lifetime.

There are then two break points in the proposed surveillance program. The first breakpoint is the nuclear depletion above which inspections must begin. Based on the far-right column of Table 6-1 of NEDE-33284P Supplement 1, this is when the Marathon-Ultra control rods have exceeded [[ ]] of their nuclear lifetime limit. This is the point at which the Marathon-Ultra control rods have gone beyond the Marathon-5S lifetime limits for the same lattice type.

The second break point is the minimum depletion at which when inspections end. Using the same approach as the Marathon-5S surveillance program, this is set as 90% of the stated nuclear lifetime. The reason for this is that when plants plan a cycle, they typically allow a buffer between each control rod's projected end of cycle depletion and each control rod's depletion limit. This is to allow flexibility to respond to unforeseen events during the cycle, such as the need to insert control rods to suppress a leaking fuel bundle. Therefore, control rods very rarely reach 100% of their stated nuclear lifetime before being discharged. Many plants use a 10% buffer to the control rod's stated nuclear lifetime as effective end of life for the surveillance program.

The inspection requirements for each lattice type are more specifically stated in the following table.

Lattice Type	Marathon-Ultra Equivalent B-10 ¼ Segment Percent Depletion Limit	Inspections Start ([[ ]] of Nuclear Life)	Minimum Required Inspections Until: (90% of Nuclear Life)
D Lattice	[[		
C Lattice			
S Lattice			]]

<u>Changes to NEDE-33284P Supplement 1 Revision 0</u>:

The following bullet will be added to the Acceptance version of NEDE-33284P Supplement 1.

- 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% higher than the twelve inspected control rods, a minimum of four (4) of the additional control rods shall be visually inspected.

Additional Change:

A minor error has been detected in the calculation of absorber tube peaking factors for the S lattice case shown in Table 4-11 of NEDE-33284P Supplement 1. The figure on the following page shows the updated peaking factors, which will be updated in the Acceptance version of NEDE-33284P Supplement 1. As a result, the mechanical lifetime for the S lattice case shown in Table 4-4 of NEDE-33284P Supplement 1 also changes slightly; from [[ ]] to [[ ]]. Table 4-4 will also be updated as shown below in the Acceptance version of NEDE-33284P Supplement 1.

Table 4-4

Marathon-Ultra Control Rod Nuclear and Mechanical Depletion Limits

Application	**End of Life B-10 Equivalent Depletion (%)**	
	Nuclear Peak Quarter Segment	**Mechanical Four Segment Average**
D Lattice, BWR/2-4	[[	
C Lattice, BWR/4,5		
S Lattice, BWR/6		]]

Table 4-11

S Lattice Mechanical Lifetime Calculation

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10. ABSORBER LOADING OPTIONS, ABWR AND ESBWR DESIGNS

In the future, GEH may offer alternate loading patterns of boron carbide capsules and hafnium rods, within the Marathon-5S / Marathon-Ultra outer structure. For example, GEH may choose to offer an all-boron carbide capsule design, employing the Marathon-Ultra capsule or to vary the number and location of boron carbide capsules and hafnium rods to produce control rods of varying nuclear lifetime. In addition, the Marathon-5S and Marathon-Ultra designs may also be adapted to ABWR and ESBWR applications.

The following evaluation and reporting process will be used for alternate absorber loadings for Marathon-5S (NEDE-33284P-A, Reference 1) and Marathon-Ultra (NEDE-33284P Supplement 1) control rods.

10.1 Application

Marathon-5S or Marathon-Ultra control rods with alternate absorber loadings may be applied to all Boiling Water Reactors (BWR), including BWR/2 through BWR/6, ABWR and ESBWR.

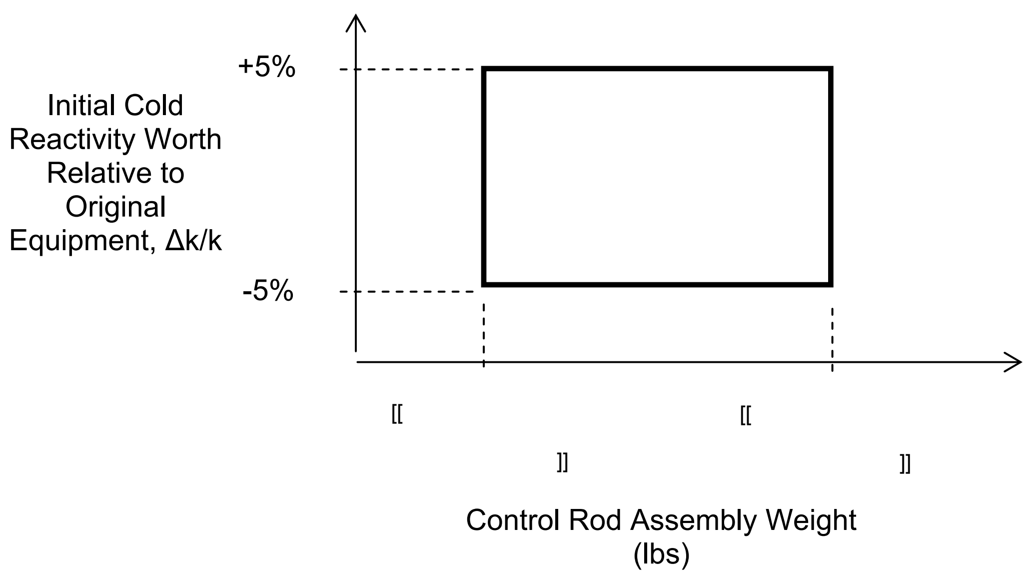
10.2 Fixed Parameters

- The outer absorber tube geometry as defined in Table 2-1 shall not be changed.
- The outer absorber tube material, type 304S, as defined in Table 2-1 and Section 3.2.4 of Reference 1 shall not be changed.
- Absorber materials may only consist of vibratory compacted boron carbide with naturally occurring Boron-10 isotopic content, and hafnium.
- The vibratory compacted boron carbide must be contained within capsules within the outer absorber tube, providing a diametral gap between the inner capsule and the outer absorber tube.
- The outer absorber section structure, with absorber tube and a tie rod laser welded together as described in Section 3.2.4of Reference 1, must be maintained.

10.3 Variable Parameters

- The length and location of boron carbide capsules, empty capsules, spacers, and hafnium rods may be varied for alternate absorber loading configurations. The use of hafnium rods is restricted to the use of full-length hafnium rods.
- The diameter and wall thickness of the capsule tubing may be varied, such that the methodologies and acceptance criteria described in Section 10.5 below are met.
- 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.

- Manufacturability changes to the velocity limiter and handle are permissible such that there is no negative affect on the fit, form, or function of these sub-components, and such that the acceptance criteria described in Section 10.5 below are met.
- The overall weight of the control rod assembly may vary due to the alternate absorber load patterns, such that the weight of the control rod remains within the range of weights in Table 2-1, and such that there is no negative effect on the control rod insertion, withdrawal, or SCRAM performance.
- The overall length of the absorber section may be reduced to accommodate ESBWR.
- Control rods for ABWR or ESBWR application employ a connector rather than a velocity limiter.
- The following figure summarizes the permissible design space. The vertical axis of the design space is the primary nuclear requirement of matched initial reactivity worth as discussed in Section 4.4. The horizontal axis of the design space is the mechanical requirement of overall control rod weight, defined by the weight limits shown in Table 2-1.



10.4 Methodologies

The mechanical and nuclear evaluation methodologies shall be identical to those described in this report and Reference 1. The following are emphasized:

- For the boron carbide swelling evaluation in Section 3.6, the evaluation shall use worst-case absorber tube and capsule dimensions, as well as a +3σ upper bound B_4C swelling limit based on available data.

- The thermal and helium release fraction methodologies discussed in Section 3.6.3 shall remain unchanged.
- The absorber tube pressurization methodologies in Section 3.6.4 shall remain unchanged, including the consideration of worst-case absorber tube dimensions, absorber tube surface defects and wear, and a factor of safety of 2.0.
- Should any alternate absorber loading patterns change the control rod assembly weight, the SCRAM and handling loads shall be re-evaluated using the methodology of Sections 3.3 and 3.7.
- The nuclear analysis methodology described in Section 4.2 shall not be modified unless specifically reviewed and approved by the NRC.

10.5 Acceptance Criteria

The mechanical and nuclear evaluation acceptance criteria shall be identical to those described in this report and Reference 1. The following are emphasized:

- For the boron carbide swelling evaluation of Section 3.6, using worst-case absorber tube and capsule dimensions and +3σ upper bound B_4C swelling limits, [[

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- Using worst-case dimensions, a clearance between any hafnium absorber and the outer absorber tube at end of life shall be demonstrated.
- The allowable material stresses of Table 3-2 shall not change.
- The adoption of alternate load patterns may result in control rods with a longer nuclear lifetime than the Marathon-Ultra control rod. Should this occur, the surveillance program of Section 6.5 will continue to apply for the range of irradiation above the Marathon-Ultra lifetime limit.
- The nuclear design criteria of Section 4.1 shall not be changed.
- The licensing acceptance criteria of Section 6 shall not be changed.

10.6 Notification

Before alternate loading pattern Marathon-5S or Marathon-Ultra control rods are delivered, GEH will provide the NRC with a Compliance Demonstration Report.

The Compliance Demonstration Report will have content and format similar to this report and Reference 1. The report shall confirm that the fixed parameters listed above are unchanged, and fully describe the changes and acceptability of all changed parameters.

The Compliance Demonstration Report will also be provided to BWR licensees to support 10 CFR 50.59 evaluations.