ENCLOSURE 2

MFN 12-018

NEDO-33284 Supplement 1-A

Non-Proprietary Information– Class I (Public)

IMPORTANT NOTICE

This is a non-proprietary version of NEDE-33284 Supplement 1P-A, 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 [[ ]].

Note the NRC's Final Safety Evaluations are enclosed in NEDO-33284, Supplement 1-A. Portions of the Safety Evaluations that have been removed are indicated with a single square bracket as shown here. [ ].

GE Hitachi Nuclear Energy

NEDO-33284 Supplement 1-A
Revision 1
eDRFSection 0000-0112-6090 R1
March 2012

Non-Proprietary Information – Class I (Public)

Licensing Topical Report

MARATHON-ULTRA CONTROL ROD ASSEMBLY

INFORMATION NOTICE

This is a non-proprietary version of the document NEDE-33284 Supplement 1P-A, Revision 1, which has the proprietary information removed. Portions of the document that have been removed are indicated by an open and closed bracket as shown here [[ ]]. Within the **US NRC Safety Evaluations,** the portions that have been removed are indicated by a single open and closed bracket as shown here [ ].

IMPORTANT NOTICE REGARDING THE CONTENTS OF THIS REPORT

Please Read Carefully

The information contained in this document is furnished **for the purpose of obtaining NRC approval for the use of the Marathon-Ultra control rod in Boiling Water Reactors**. The only undertakings of GE-Hitachi Nuclear Energy (GEH) with respect to information in this document are contained in contracts between GEH and any participating utilities, and nothing contained in this document shall be construed as changing those contracts. The use of this information by anyone other than that for which it is intended is not authorized; and with respect to **any unauthorized use**, GEH makes no representation or warranty, and assumes no liability as to the completeness, accuracy, or usefulness of the information contained in this document.

February 16, 2012

Mr. Jerald G. Head
Senior Vice President, Regulatory Affairs
GE-Hitachi Nuclear Energy Americas LLC
P.O. Box 780, M/C A-18
Wilmington, NC 28401-0780

SUBJECT: FINAL SAFETY EVALUATION FOR GE HITACHI NUCLEAR ENERGY AMERICAS TOPICAL REPORT NEDE-33284P, SUPPLEMENT 1, "MARATHON-ULTRA CONTROL ROD ASSEMBLY" (TAC NO. ME3524)

Dear Mr. Head:

By letter dated January 29, 2010, (Agencywide Documents Access and Management System (ADAMS) Package Accession No. ML100331610), GE Hitachi Nuclear Energy Americas, LLC (GEH) submitted Topical Report (TR) NEDE-33284P, Supplement 1, "Marathon-Ultra Control Rod Assembly," to the U.S. Nuclear Regulatory Commission (NRC) staff. By letter dated February 3, 2012, an NRC draft safety evaluation (SE) regarding our approval of NEDE-33284P, Supplement 1, was provided for your review and comment. By letter dated February 6, 2012, GEH responded, but found no factual errors or clarity concerns in the draft SE.

The NRC staff has found that NEDE-33284P, Supplement 1, is acceptable for referencing in licensing applications for GEH designed boiling water reactors to the extent specified and under the limitations delineated in the TR and in the enclosed final SE. The final SE defines the basis for our acceptance of the TR.

Our acceptance applies only to material provided in the subject TR. We do not intend to repeat our review of the acceptable material described in the TR. When the TR appears as a reference in license applications, our review will ensure that the material presented applies to the specific plant involved. License amendment requests that deviate from this TR will be subject to a plant-specific review in accordance with applicable review standards.

In accordance with the guidance provided on the NRC website, we request that GEH publish accepted proprietary and non-proprietary versions of this TR within three months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed final SE after the title page. Also, they must contain historical review information, including NRC requests for additional information and your responses. The accepted versions shall include an "-A" (designating accepted) following the TR identification symbol.

Enclosure 1 and its Attachment transmitted herewith contain proprietary information. When separated from Enclosure 1 and its Attachment, this document is decontrolled.

As an alternative to including the RAIs and RAI responses behind the title page, if changes to the TR were provided to the NRC staff to support the resolution of RAI responses, and the NRC staff reviewed and approved those changes as described in the RAI responses, there are two ways that the accepted version can capture the RAIs:

1. The RAIs and RAI responses can be included as an Appendix to the accepted version.
2. The RAIs and RAI responses can be captured in the form of a table (inserted after the final SE) which summarizes the changes as shown in the approved version of the TR. The table should reference the specific RAIs and RAI responses which resulted in any changes, as shown in the accepted version of the TR.

If future changes to the NRC's regulatory requirements affect the acceptability of this TR, GEH and/or licensees referencing it will be expected to revise the TR appropriately, or justify its continued applicability for subsequent referencing.

Sincerely,

/RA/

Robert A. Nelson, Deputy Director
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project No. 710

Enclosures:
1. Proprietary Final SE with Proprietary Attachment
2. Non-Proprietary Final SE with Non-Proprietary Attachment

cc w/encl 2 only: See next page

As an alternative to including the RAIs and RAI responses behind the title page, if changes to the TR were provided to the NRC staff to support the resolution of RAI responses, and the NRC staff reviewed and approved those changes as described in the RAI responses, there are two ways that the accepted version can capture the RAIs:

1. The RAIs and RAI responses can be included as an Appendix to the accepted version.
2. The RAIs and RAI responses can be captured in the form of a table (inserted after the final SE) which summarizes the changes as shown in the approved version of the TR. The table should reference the specific RAIs and RAI responses which resulted in any changes, as shown in the accepted version of the TR.

If future changes to the NRC's regulatory requirements affect the acceptability of this TR, GEH and/or licensees referencing it will be expected to revise the TR appropriately, or justify its continued applicability for subsequent referencing.

Sincerely,

/RA/

Robert A. Nelson, Deputy Director
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project No. 710

Enclosures:
1. Proprietary Final SE with Proprietary Attachment
2. Non-Proprietary Final SE with Non-Proprietary Attachment

cc w/encl 2 only: See next page

DISTRIBUTION:
PUBLIC (Public Documents Only)
NON-PUBLIC (Proprietary SE and Attachment)
PLPB R/F
RidsNrrDpr
RidsNrrDprPlpb
SPhilpott
RidsNrrLADBaxley
RidsOgcMailCenter
RidsAcrsAcnwMailCenter
RidsNrrDss
RidsNrrDssSnpb
RidsNroDsra
RidsNroDsraSrsb
PClifford
APatel, NRO
CVanWert, NRO

ADAMS Accession Nos.:
PUBLIC documents:
Package: ML120370633
Cover letter: ML120370620
Final SE (Non-Proprietary): ML120380180
Attachment (Non-Proprietary): ML120380234

NON-PUBLIC documents:
Final SE (Proprietary): ML120370627
Attachment (Proprietary): ML120370631

***From Draft SE - no significant changes** **NRR-043**

OFFICE	PLPB/PM	PLPB/LA	SNPB/BC	NRO/BC	PLPB/BC	DPR/DD
NAME	SPhilpott	DBaxley	AMendiola*	JDonoghue*	JJolicoeur	RNelson
DATE	2/13/2012	2/13/2012	1/31/2012	2/3/2012	2/14/2012	2/16/2012

OFFICIAL RECORD COPY

GE-Hitachi Nuclear Energy Americas Project No. 710
cc:

Mr. James F. Harrison
GE-Hitachi Nuclear Energy Americas, LLC
Vice President - Fuel Licensing
P.O. Box 780, M/C A-55
Wilmington, NC 28401-0780
james.harrison@ge.com

Ms. Patricia L. Campbell
Vice President, Washington Regulatory Affairs
GE-Hitachi Nuclear Energy Americas LLC
1299 Pennsylvania Avenue, NW
9th Floor
Washington, DC 20004
patriciaL.campbell@ge.com

Mr. Andrew A. Lingenfelter
Vice President, Fuel Engineering
Global Nuclear Fuel–Americas, LLC
P.O. Box 780, M/C A-55
Wilmington, NC 28401-0780
Andy.Lingenfelter@gnf.com

Edward D. Schrull
GE-Hitachi Nuclear Energy Americas LLC
Vice President - Services Licensing
P.O. Box 780, M/C A-51
Wilmington, NC 28401-0780
Edward.schrull@ge.com

FINAL 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 <u>Alternate Absorber Loading Patterns</u>

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:

- [

  ]
- 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. [

]

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

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.
- [

]

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. [

  ]

[ ] 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. [

]

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.

[

]

[

]

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

[

]

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

[

]

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

[

]

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

[

]

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% of end-of-life).
- For refueling outages in which the depletion of the lead Marathon-Ultra assemblies are greater than 75% 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% 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% 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% 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% 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% 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

Principal Contributors: Paul M. Clifford, NRR
Amrit Patel, NRO
Chris Van Wert, NRO

Date:

Pacific Northwest National Laboratory

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 stress.

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. [

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

Table of Contents

List of Tables

List of Figures

REVISION SUMMARY

Revision	Section	Description of Change
1	**Revision 1 is the accepted version of Revision 0, including the following revisions dictated by RAI responses and the Final Safety Evaluation.**	
	Figure 3-8	Replaced with a corrected figure in response to RAI-1.
	Figure 3-9	Replaced with a corrected figure in response to RAI-2.
	Table 4-4	Corrected the S-Lattice Mechanical 4 Segment Average EOL B-10 Equivalent Depletion.
	Table 4-11	Replaced with a new table.
	6.5.2	Added a final bullet in response to RAI-8.
	10	Replaced all of Section 10 in response to RAI-7. Amended new text in Section 10 in accordance with Final SE Sections 3.1.1.1 and 3.1.1.2.
	11	Deleted Section 11 because relevant text was incorporated into the revised Section 10.
	12	Deleted Reference 12. It is no longer referenced.
	12 and 13	Renumbered Sections 12 and 13 as Sections 11 and 12, respectively to accommodate the deletion of Section 11.
	All	Revision bars added to the right margin signifying additions/changes to the text compared to Revision 0.

ACRONYMS AND ABBREVIATIONS

Acronym / Abbreviation	Description
AOO	Anticipated Operational Occurrence
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient Without Scram
CFR	Code of Federal Regulations
CRB	Control Rod Blade
CRD	Control Rod Drive
CRDA	Control Rod Drop Accident
ECCS	Emergency Core Cooling System(s)
ECP	Engineering Computer Code
ESF	Engineered Safety Feature
FHA	Fuel Handling Accident
GEH	General Electric Hitachi Nuclear Energy
GNF	Global Nuclear Fuels
IASCC	Irradiation Assisted Stress Corrosion Cracking
LOCA	Loss of Coolant Accident
LTR	Licensing Topical Report
MCPR	Minimum Critical Power Ratio
MSLBA	Main Steamline Break Accident
NRC	U.S. Nuclear Regulatory Commission
OBE	Operating Basis Earthquake
QA	Quality Assurance
RAI	Request for Additional Information
SRSS	Square Root Sum of Squares
SSE	Safe Shutdown Earthquake
STS	Standard Technical Specifications
TS	Technical Specifications

EXECUTIVE SUMMARY

The GEH Marathon-Ultra control rod is a derivative of the Marathon-5S design approved by Reference 1. The only difference between the Marathon-Ultra and the Marathon-5S design is the absorber section load pattern. Where the Marathon-5S is an all-boron carbide capsule design, the Marathon-Ultra incorporates full-length hafnium rods in outer edge, high depletion tube locations. The geometry and composition of these hafnium rods is identical to those used in the Marathon design, approved in Reference 2. In addition, to maximize the neutron absorber mass, thin-wall capsules are used, with a similar wall thickness to the capsules in the Marathon design (Reference 2).

Like the Marathon-5S capsules, the outer diameter of the Marathon-Ultra capsules is sized [[

]]

A nuclear evaluation of the Marathon-Ultra control rod shows that the initial cold and hot reactivity worths are within ±5% of the original equipment control rod ("matched worth criteria"). Therefore, the Marathon-Ultra is a direct nuclear replacement for previous control rod designs, and no special nuclear calculation or BWR plant change is required.

The outer structure of the Marathon-Ultra control rod, which is identical to the Marathon-5S control rod, has been evaluated during all normal and upset conditions, and has been found to be mechanically acceptable. The fatigue usage of the control rod has also been found to be well below lifetime limits.

[[

]]

For all cases, the mechanical lifetime exceeds the nuclear lifetime. Therefore, the Marathon-Ultra control rod is nuclear lifetime limited.

The operational performance of the Marathon-Ultra is also evaluated. The scram time, no settle characteristics, and control rod drop speeds are all better than or equal to the original Marathon design. Installation of Marathon-Ultra control rods does not affect any item in the Standard Plant Technical Specifications, and no plant operational change is required. Further, there is no effect on plant safety analyses or on design basis analysis models.

The licensing acceptance criteria applied to the original Marathon and Marathon-5S designs in References 1 and 2 are re-evaluated and are judged to be sufficient and complete. Therefore, the Marathon-Ultra is evaluated against the licensing acceptance criteria in References 1 and 2, and is found to be acceptable. GEH concludes that the new absorber loading of the Marathon-Ultra control rod, combined with the same outer structure as the Marathon-5S control rod approved in Reference 1, is justified for use in Boiling Water Reactors. GEH therefore requests NRC approval for the use of Marathon-Ultra control rods in Boiling Water Reactors.

1. INTRODUCTION AND BACKGROUND

GEH currently manufactures Marathon and Marathon-5S Control Rods. The Nuclear Regulatory Commission's (NRC) acceptance of the Marathon-5S Control Rod is documented by a Licensing Topical Report (LTR), Reference 1. The Marathon-5S Control Rod consists of 'simplified' absorber tubes, edge welded together to form the control rod wings, and welded to a full-length tie rod to form the cruciform assembly shape. The absorber tubes are filled with a combination of boron carbide (B_4C) capsules, and empty capsules. The previously approved Marathon Control Rod design was approved by the NRC in Reference 2. This control rod consists of 'square' absorber tubes, edge welded together, and welded to individual tie rod segments to form the cruciform assembly shape.

The Marathon-Ultra is a derivative version of the Marathon-5S control rod in that it uses an identical outer structure. The only differences for the Marathon-Ultra is the inclusion of full-length hafnium rods in high-depletion absorber tubes, and the use of a thin-wall boron carbide capsule, similar in geometry to the Marathon control rod design.

Potential effects of the proposed change are evaluated to ensure

(i) the integrity of the reactor coolant pressure boundary;

(ii) the capability to shut down the reactor and maintain it in a safe shutdown condition; and

(iii) the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the applicable guideline exposures set forth in 10 CFR 50.34(a)(1) and 10 CFR 100.11.

The following sections address the potential effect of the proposed changes on fission product barriers (e.g., fuel cladding) and other involved structures, systems and components, safety functions, design basis events, special events and Standard Technical Specifications (STS) to ensure continued compliance with design and regulatory acceptance criteria.

GEH requests NRC approval for the use of Marathon-Ultra control rods in Boiling Water Reactors.

2. DESIGN CHANGE DESCRIPTION

The only difference between the Marathon-Ultra and the Marathon-5S approved by Reference 1, 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 (Figures 2-3 through 2-5). The component materials and manufacturing processes, including welding, are exactly the same. The simplified absorber tube and capsule configuration is shown in Figure 2-1. The absorber section shell structure is shown in Figure 2-2.

2.1 HAFNIUM APPLICATION

Like the Marathon control rod (Reference 2), the Marathon-Ultra control rod employs hafnium rods as a neutron absorber in high duty tube locations. The geometry of the hafnium rods, including diameter, is identical to those used in the Marathon design. As in tubes containing boron carbide capsules, a diametral gap is provided between the hafnium rods and the outer absorber tube to accommodate any possible expansion of the hafnium rod.

Section 3.6.5 provides further evaluation of hafnium for the Marathon-Ultra assembly. This includes a discussion of in-reactor tests that have shown insignificant hydriding of hafnium under BWR conditions, and an expansion calculation demonstrating clearance between the hafnium rod and the outer absorber tube at end of life. Further, the successful application of hafnium in the original Marathon design (Reference 2) is discussed.

2.2 CAPSULE GEOMETRY

The Marathon-Ultra control rod uses a thin-wall capsule, similar to the Marathon design approved in Reference 2. As shown in Table 2-1, the Marathon-Ultra uses a [[ ]] nominal wall-thickness capsule. This is comparable to the [[ ]] nominal wall-thickness capsules used in the original Marathon design (Reference 2).

The Marathon-Ultra capsules use the same crimped end cap connection as the Marathon and Marathon-5S designs. The capsule body tube and end cap materials are the same, as is the compacted boron carbide density.

Like the Marathon-5S design, the capsule of the Marathon-Ultra design is sized [[

]] See Section 3.6.2 for a more detailed analysis.

Table 2-1
Comparison of Typical Parameters of Marathon-5S and Marathon-Ultra CRBs

Parameter	BWR/2-4 D Lattice		BWR/4-5 C Lattice		BWR/6 S Lattice	
	M-5S CRB [1]	M-Ultra CRB	M-5S CRB [1]	M-Ultra CRB	M-5S CRB [1]	M-Ultra CRB
Control Rod Weight (lb) [2]	[[					
Absorber Tubes per Wing						
Nominal Wing Thickness (in)						]]
Absorber Tube						
Length (in)	[[					
Inside Diameter (in)						
Nominal Thin Section Wall Thickness (in)						]]
Material	304S	304S	304S	304S	304S	304S
Cross-sectional area (in^2)	[[					]]
B_4C Absorber Capsule						
Length (in)	[[					
Inside Diameter (in)						
Wall Thickness (in)						
Material						
B4C Density (g/cc)						
B4C Density (% theoretical)						]]
Hafnium Rods						
Length (in)	[[					
Diameter (in)						
Density (lb/in^3)						]]

1. Values from Table 2-1 of the Marathon-5S LTR (Reference 1).
2. For 'no settle' and scram considerations, the Marathon-Ultra CRB has been designed to have dry and wet weights that are within the range of previously supplied Marathon and Marathon-5S control rod designs.

[[ ]]

Figure 2-1. Marathon-Ultra CRB Absorber Tube Geometry

[[

]]

Figure 2-2. Marathon-Ultra Absorber Wing Weld Locations

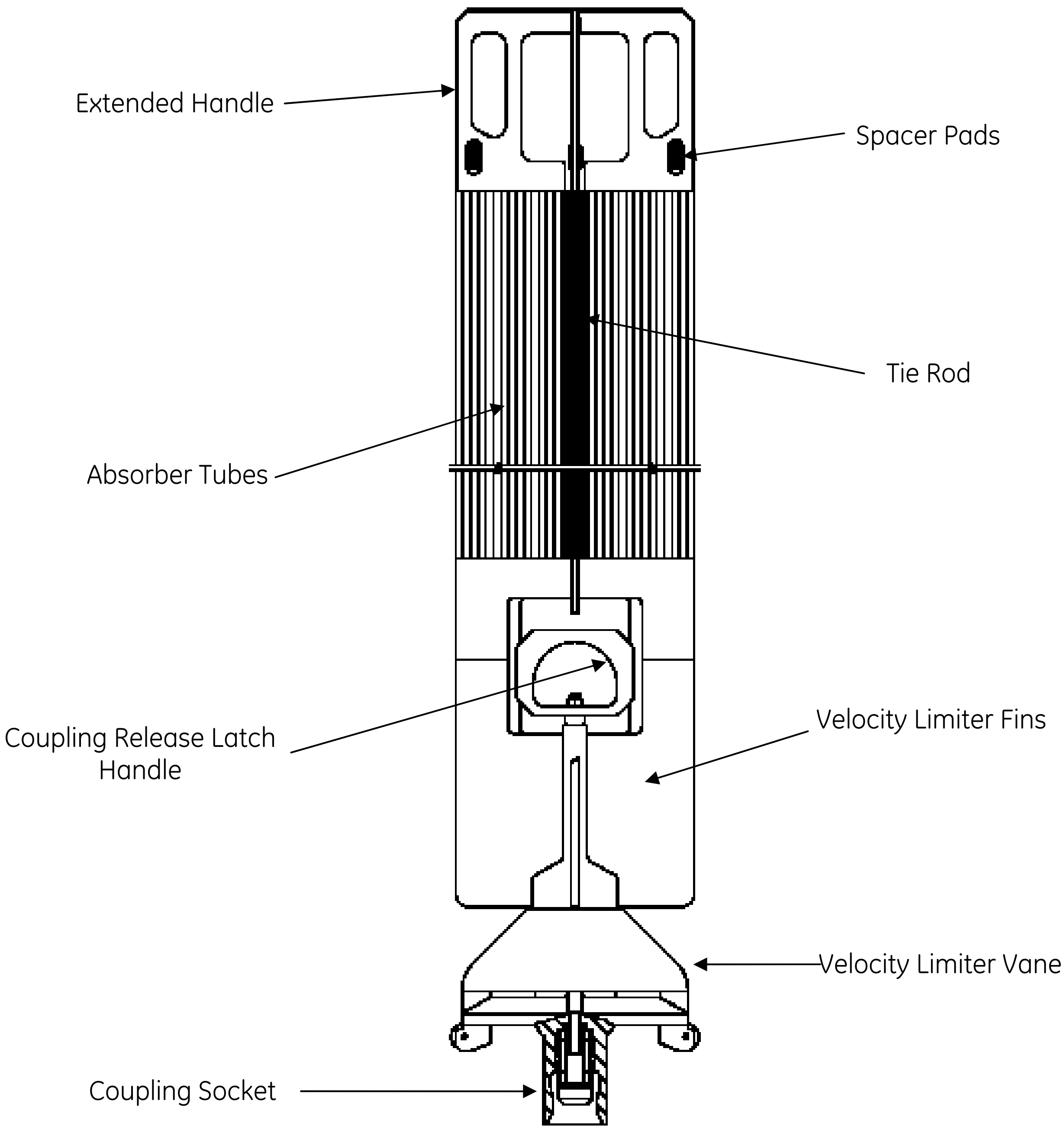


Figure 2-3. BWR/2-4 D Lattice Marathon-Ultra Control Rod

(Extended Handle Shown)

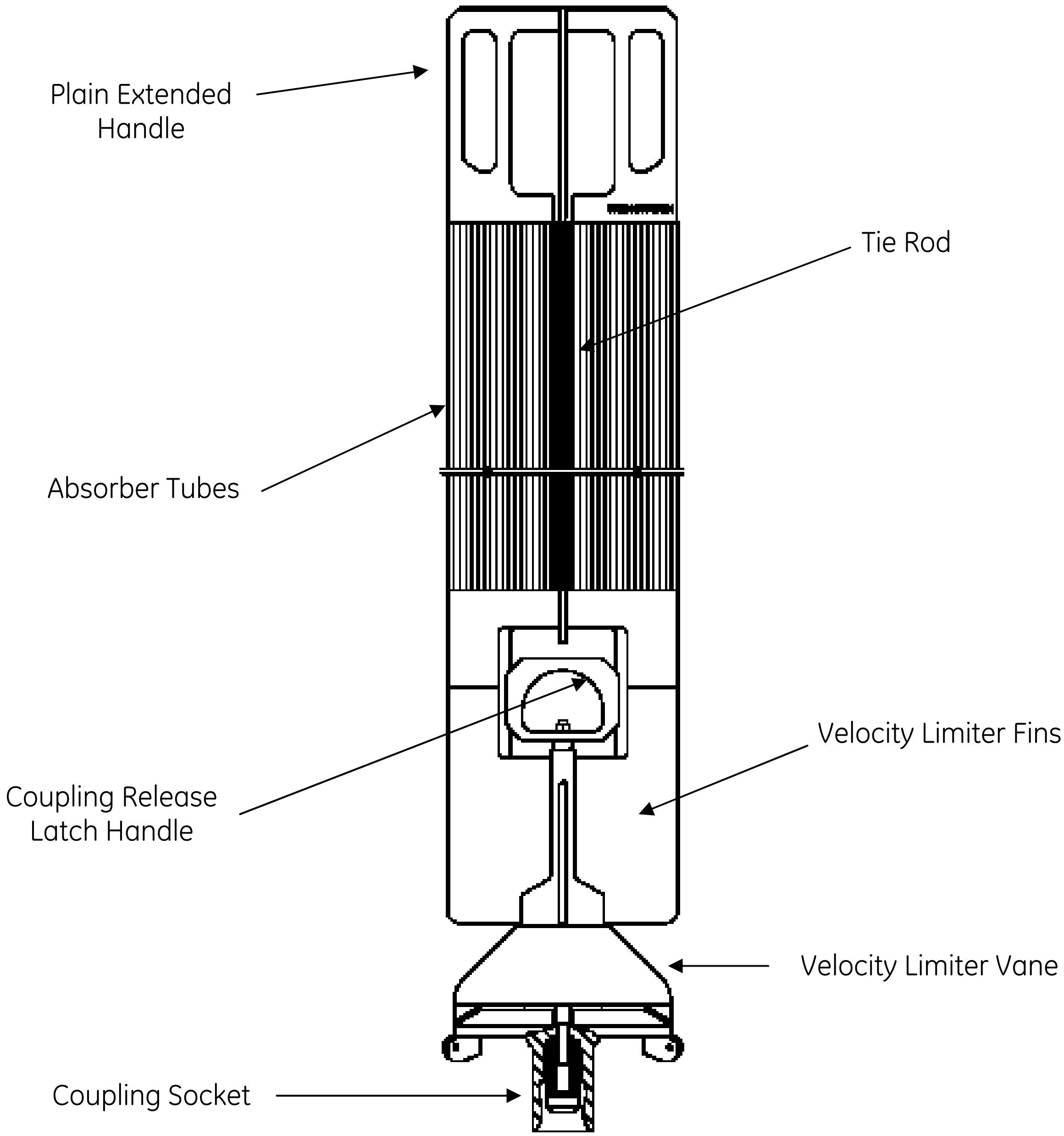


Figure 2-4. BWR/4,5 C Lattice Marathon-Ultra Control Rod

(Extended Handle Shown)

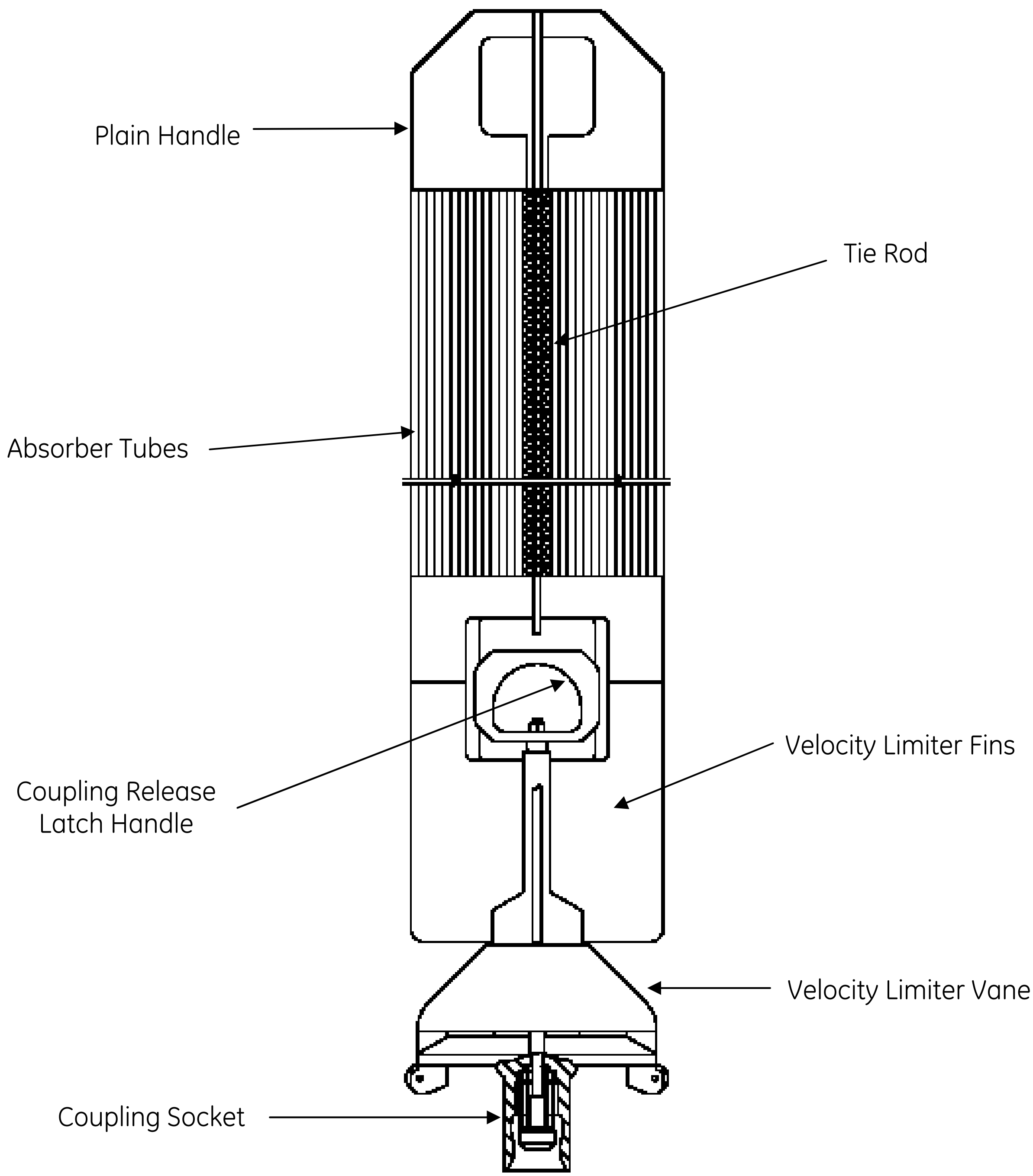


Figure 2-5. BWR/6 S Lattice Marathon-Ultra Control Rod

3. SYSTEM DESIGN

3.1 ANALYSIS METHOD

For each control rod load application, worst case or bounding loads are identified. Stresses are calculated using worst-case dimensions and limiting material properties. For analyses involving many tolerances, square root sum of squares (SRSS) or statistical tolerancing may be used. Corrosion, wear, and crud deposition are accounted for when appropriate.

It is noted that the analysis methodology for the Marathon-Ultra is identical to the methodology of the Marathon-5S, approved in Reference 1. Furthermore, since the outer structure of the control rods is identical, many of the structural analyses are identical. For example, the material property limits (Section 3.2), seismic and fuel channel bow induced bending (Section 3.4), and stuck rod compression (Section 3.5) analyses are identical to Reference 1. Because of the modified absorber section loading and capsule geometry, the scram loads in Section 3.3, and the absorber burn-up related loads in Section 3.6 are slightly different. However, the analysis methodology and acceptance criteria are identical to that approved in Reference 1.

It is also noted that, since the outer structure of the Marathon-5S and Marathon-Ultra control rods are identical, discussions related to material behavior, metallurgy, manufacturing processes and welding in Reference 1 are equally applicable to the Marathon-Ultra design.

Numerous finite element analyses are used in the design of the Marathon-Ultra control rod, as described in the following sections. Table 3-24 contains a summary of these analyses. As shown in this table, the methodology and finite element model for all of the analyses are identical to those used for the Marathon-5S control rod and approved in Reference 1. Two of the six analyses use slightly different geometry or load inputs applicable to the Marathon-Ultra control rod. The remaining four analyses pertain to the outer structure of the Marathon-Ultra / Marathon-5S control rod, and are completely unchanged.

3.1.1 Combined Loading

As in Reference 1, effective stresses and strains are determined using the distortion energy theory (Von Mises), and compared to allowable limits. Using the principal stresses: σ_1, σ_2, and σ_3, the equivalent Von Mises stress is calculated as:

$$\sigma_{VM} = \sqrt{1/2[(\sigma_1 - \sigma_2)^2 + (\sigma_2 - \sigma_3)^2 + (\sigma_3 - \sigma_1)^2]}$$

3.1.2 Unirradiated Versus Irradiated Material Properties

Each structural analysis is first evaluated to determine whether unirradiated or irradiated material properties are appropriate. In general, as stainless steel is irradiated, the yield and ultimate tensile strengths increase, while the ductility, or allowable strain decreases. In order to determine the correct technique, the analyses are broken into two categories:

1. Analyses with an applied load (i.e., scram). For these analyses, a maximum stress is calculated, and compared to the limiting unirradiated stress limit.
2. Analyses with an applied displacement (i.e., seismic bending). For these analyses, a maximum strain is calculated, and compared to the limiting irradiated strain limit.

3.2 MATERIAL PROPERTY LIMITS

The limiting unirradiated material strengths are first identified for the control rod structural materials, and shown in Table 3-1. For most materials, limiting values from the ASME Boiler and Pressure Vessel Code are used. In other cases, minimum material strengths are specified in GEH material specifications.

3.2.1 Stress Criteria

The licensing acceptance criteria of References 1 and 2 are used, in which the control rod stresses and strains and cumulative fatigue shall be evaluated to not exceed the ultimate stress or strain of the material.

The figure of merit employed for the stress-strain limit is the design ratio, where:

Design ratio = effective stress/stress limit, or, effective strain/strain limit.

The design ratio must be less than or equal to 1.0. Conservatism is included in the evaluation by limiting stresses for all primary loads to one-half of the ultimate tensile value.

Resulting allowable stresses for primary loads are shown in Table 3-2.

3.2.2 Welded Connections

For welded connections, a weld quality factor “q” is used to further reduce the allowable stress. Therefore, the allowable stress for a welded connection, S_m’ is:

$$S_m' = (q)\, S_m$$

Weld quality factors are determined based on the inspection type and frequency of the weld. Weld quality factors are shown in Table 3-3.

3.3 SCRAM

The methodology for determining maximum control rod loads during a scram event for the Marathon-Ultra control rod is identical to the Marathon-5S methodology approved in Reference 1.

The largest axial structural loads on a control rod blade are experienced during a control rod scram, due to the high terminal velocity. To be conservative, structural analyses of the control rod are performed assuming a 100% failed control rod drive buffer. A dynamic model of mass, spring and gap elements is used to simulate a detailed representation of the load bearing components of the assembly during a scram event. Simulations are run at atmospheric temperatures, pressures, speeds, and properties as well at operating temperatures, pressures, speeds, and properties. The resulting loads are shown in Table 3-4.

Structural stresses are determined from the scram loads shown in Table 3-4 using the limiting material properties, weld quality factors, and worst-case geometry for the area subject to the load. Figures 3-1 and 3-2 show the welds and cross-sections analyzed.

Resulting maximum stresses during a failed buffer scram are shown in Tables 3-5, 3-6 and 3-7 for D lattice BWR/2-4, C lattice BWR/4-5, and S lattice BWR/6 applications. These stresses are evaluated against the stress limits shown in Table 3-2. Specific details for each calculation are shown in Appendix A. As shown by the design ratios in Tables 3-5 through 3-7, sufficient margin exists against failure for all cross-sections and welds.

3.4 SEISMIC AND FUEL CHANNEL BOW INDUCED BENDING

Fuel channel deflections, which result from seismic events, impose lateral loads on the control rods. The Marathon-Ultra control rod is analyzed for Operating Basis Earthquake (OBE) events and Safe Shutdown Earthquake (SSE) events. It is noted that the contents of this section are the same as that approved for the Marathon-5S in Reference 1, as the outer structure of the control rods is identical.

3.4.1 Wing Outer Edge Bending

The OBE analysis is performed by evaluating the strain in the Marathon-5S absorber section with maximum OBE deflection. In addition, maximum control rod deflections due to fuel channel bulge and bow are conservatively added to the calculated seismic bending deflections. [[
]]

The limiting location for strain due to bending of the control rod cross-section occurs at the outer edge of the control rod wing. At this location, a combined strain due to simultaneous application of the following loads is calculated: (1) control rod bending due to an OBE seismic event, (2) control rod bending due to worst case channel bulge and bow, (3) axial absorber tube stress due to maximum internal pressure, and (4) a failed buffer scram. The results of these strain calculations are shown in Table 3-8. As shown, even under these combined worst-case conditions, the maximum strain is well below the limiting maximum allowable strain at irradiated conditions.

3.4.2 Absorber Tube to Tie Rod Weld

The combined effect of control rod bending due to OBE and channel bulge and bow deflection combined with maximum absorber tube internal pressure is also evaluated at the full-length tie rod to absorber tube weld. A finite element model is used, as shown in Figure 3-3. Resulting worst-case stresses are shown in Table 3-9. As shown, the resulting stresses are acceptable against the design criteria.

3.4.3 Absorber Tube Lateral Load

Finally, the lateral load imposed on the control rod absorber tube due to an excessively bowed channel is evaluated. The finite element model is shown in Figure 3-4. As shown, the entire lateral load is applied to a single square absorber tube, along with reactor internal pressure. For conservatism, no internal pressure is applied to the tube, which would offset the external pressure and reduce the stresses in the tube.

The resulting stress intensity plot is shown in Figure 3-5. The maximum stress intensity is calculated as [[ ]], which is less than the absorber tube allowable load of [[ ]] from Table 3-2.

3.4.4 Marathon-5S / Marathon-Ultra Seismic Scram Tests

For the SSE analysis, the control rod must be capable of full insertion during fuel channel deflections. Like the Marathon-5S (Reference 1), because the Marathon-Ultra control rod has a stiffness less than or equal to the Marathon assembly, and because the weight of the Marathon-Ultra control rod is less than previous designs, the Marathon-Ultra has seismic scram capability equal to or better than the Marathon control rod (See Section 5.2).

Seismic scram tests of the Marathon-5S control rod are discussed in Sections 3.4.4 and 5.2 of Reference 1. The parameters affecting seismic scram performance are the bending stiffness of the assembly, and the overall weight of the assembly. In general, a more stiff 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 bound 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 assemblies.

The test facility used consists of a simulated pressure vessel and reactor internals, and a control rod drive. Prototype control rods were installed, and the control rod drive was set to simulate D, C, and S lattice operation.

The prototypes used for the test incorporated plain, roller-less handles, as described in Appendix A of Reference 1. The acceptance criterion for the test was that scram time requirements were to be met up to fuel bundle oscillation consistent with an OBE (Operational Basis Earthquake) event, and that the control rods would successfully insert under an SSE (Safe Shutdown Earthquake) event. The results of the tests were very successful, in that scram time requirements were met through the much more severe SSE event for both the C lattice and S lattice applications. The D lattice application met scram time requirements with OBE fuel channel deflections, and successfully inserted under SSE conditions. Therefore, the acceptance criteria for the test were met. During the tests, the control rods received very little wear.

3.5 STUCK ROD COMPRESSION

Maximum compression loads from the control rod drive (CRD) are evaluated for a stuck control rod. Both buckling, and compressive yield are analyzed for the entire control rod cross-section (buckling mode A), and conservatively assuming that the entire compression load is applied to a single control rod wing (buckling mode B). Figure 3-6 shows the buckling modes. An additional axial load of 600 lb due to channel bulge and bow is also added to the compression load.

Results of the stuck rod compression loads are contained in Table 3-10 for the entire control rod cross-section (mode A), and in Table 3-11 for the single wing (mode B). As can be seen, neither compressive yielding nor buckling will occur for either buckling mode. Additionally, for both buckling modes, the compressive yield load is reached prior to the critical buckling load. This analysis for the Marathon-Ultra control rod is identical to that for the Marathon-5S control rod, approved in Reference 1.

3.6 ABSORBER BURN-UP RELATED LOADS

The methodology for evaluating absorber burn-up related loads for the Marathon-Ultra control rod is identical to the Marathon-5S methodology approved in Reference 1. This includes the use of the same irradiated boron carbide swelling design basis, clearance evaluation methodology, thermal analysis methodology, and absorber tube pressurization methodology. There are small differences in the results of the analyses, as a result of the use of the thin-wall capsule body tube. However, the conservative criteria [[

]]

The structure of a control rod must provide for positioning and containment of the neutron absorber material (boron carbide powder, hafnium, etc) throughout its nuclear and mechanical life and prohibit migration of the absorber out of its containment during normal, abnormal, emergency and faulted conditions. The Marathon-Ultra control rod, like the Marathon and Marathon-5S control rods, contains boron carbide (B_4C) powder within capsules contained within absorber tubes (capsule within a tube design).

The boron neutron absorption reaction releases helium atoms. Some of this helium gas is retained within the compacted boron carbide powder matrix, causing the powder column to swell. This swelling causes the B_4C capsule to expand. The remainder of the helium is released as a gas. Like previous Marathon designs, the capsule end caps for the Marathon-Ultra design are crimped to the capsule body tubes. This allows the helium gas to escape from the capsule and fill the absorber tube gap and any empty capsule plenum volume provided.

For the original Marathon capsule design, [[

]]

Like the Marathon-5S design, the Marathon-Ultra capsule tube dimensions are sized [[

]]

Using the pressurization capability of the absorber tube, limits are determined for each absorber tube configuration, in terms of B_4C column depletion.

These individual absorber tube depletion limits are then combined with radial depletion profiles and axial depletion profiles to determine the mechanical depletion limit for the control rod assembly (Section 4.6).

3.6.1 Irradiated Boron Carbide Swelling Design Basis

Mechanical test data of the irradiated behavior of boron carbide was obtained by irradiating test capsules as described in Reference 1.

As discussed in Reference 11, GEH completed a post-irradiation examination of a Marathon control rod in April 2009. Part of this examination was dimensional measurements related to boron carbide swelling rates. As discussed in Section 6 of Reference 11, these dimensional measurements strongly support the design basis boron carbide swelling rates used in Reference 1 for the Marathon-5S, as the new data very closely matches the existing data. Both sets of data are shown graphically in Figure 3-15. As shown, +3σ upper limit used for the Marathon-5S bounds both sets of data. Therefore, this same conservative design basis swelling used for the Marathon-Ultra control rod design.

3.6.2 Clearance Between Capsule and Absorber Tube

As a result of the welding process forming the control rod wings, the inside diameter of the absorber tubes shrink. Therefore, a minimum inside diameter is established, and is 100% inspected following the welding, before the absorber section is loaded with capsules.

The worst-case capsule dimensions are used, which result in the maximum outside diameter at 100% local depletion. These consist of the original maximum outside diameter, and minimum wall thickness, resulting in the maximum beginning boron carbide diameter.

The strain at the ID of the capsule is equal to the diametral strain of the boron carbide powder. The +3σ upper limit of boron carbide swelling data is used. Then, assuming constant volume deformation of the capsule, the strain on the outside diameter of the capsule is:

[[ ]]

Then, the capsule outside diameter at 100% local depletion is:

$OD_{100\%} = OD_0(1+ \varepsilon_{OD})$.

A summary of this calculation is shown in Table 3-18 for both the D/S lattice and C lattice absorber tube and capsule combinations. [[

]]

3.6.3 Thermal Analysis and Helium Release Fraction

The following methodology is identical to that approved for the Marathon-5S in Reference 1. The only difference is that the Marathon-Ultra capsule geometry and heat generation rates are incorporated.

Pressure in the absorber tube due to helium release is calculated accounting for worst-case capsule and absorber tube dimensions and B_4C helium release fraction. Because the amount of helium released from the B_4C powder increases with temperature, a finite element thermal analysis is performed to determine the peak B_4C temperature (see Figure 3-8). This thermal analysis is performed using worst-case dimensions, maximum end-of-life crud buildup, combined with maximum beginning-of-life heat generation.

For the thermal model, corrosion is modeled as the build-up of an insulating layer of crud. This crud may be corrosion products from the control rod absorber tube, or deposited from other reactor internals. For all thermal analyses, a crud layer corresponding to a 32-year residence time is used ([[ ]])

A temperature distribution is shown in Figure 3-8 for the D/S lattice case. The model used assumes that the tube is interior to the wing, in that there is another absorber tube to the left and right. The boundary on the left and right is conservatively assumed to be perfectly insulated (zero heat flux).

Results for both D/S lattice and C lattice are shown in Tables 3-22 and 3-23, and in Figures 3-10 and 3-11. The following conservatisms are applied to the thermal model:

- Peak beginning-of-life heat generation rates are used, these are combined with:
- End-of-life combined corrosion and crud build-up of [[ ]], twice that used in previous analyses.
- Peak heat generation rates are used from the highest heat generation tube, which is actually the outermost edge tube. In reality, this tube will have coolant on one side, rather than be insulated. Further, some heat transfer will occur from the peak heat generation tube to the adjacent tube, rather than be perfectly insulated.
- Maximum wall thickness dimensions are used.

Peak B_4C temperatures are shown in Table 3-12. The temperatures shown in this table are based on peak beginning-of-life boron carbide heat generation rates (see Section 4.5), and are from the peak heat generation absorber tube at the peak axial location. They are radially averaged only across the cross-section of an individual boron carbide capsule.

Helium release fractions are based on models developed using data from multiple sources. The data shows a significant dependence of helium release fraction on the irradiation temperature. The helium release fractions used for each lattice type are shown in Table 3-12. The helium release model is based on data from 500 °F to 1000 °F, which envelops the temperatures shown in Table 3-12.

3.6.4 Absorber Tube Pressurization Capability

As discussed in Section 2, the Marathon-Ultra control rod uses the same 'simplified' absorber tube as the Marathon-5S control rod. Therefore, the following analysis of the absorber tube pressurization capability is the same as that in Reference 1.

[[

]]

Finite element analyses are performed to determine the pressurization capability of the absorber tube. These analyses incorporate the use of worst-case dimensions, maximum expected wear, and the largest allowable surface defects (see Figure 3-7).

<u>Absorber Tube Defects</u>

The limiting case used for establishment of the absorber tube allowable pressure simultaneously combines worst-case absorber tube dimensions (thinnest wall per drawings), surface defects at the center of the flat portion of the tube, on the round portion of the tube, and a crack-like defect on the thinnest portion of the inside diameter of the tube.

The largest sized allowable surface defects are based on the manufacturing capability of the absorber tube. A collaborative effort was undertaken with the supplier of the absorber tubes to determine a maximum surface defect size that would maintain reasonable yield rates, but would not reduce the pressurization capability of the tube below acceptable values. A surface defect depth limit of [[ ]] in depth was determined, applied to the absorber tubing specification, and factored into the pressurization analysis.

At receipt inspection, the acceptance criteria for surface defects are based primarily on the depth of the defect. Additionally, matching sets of visual standards are used by both the supplier and by GEH to identify acceptable and unacceptable surface features.

The finite element analysis shows that smaller diameter defects result in larger stress concentrations around the defect. A survey was performed of surface defects, and the smallest area defect was found to be [[ ]] in diameter. Therefore, a diameter of [[ ]] was used for the finite element model surface defects.

After factoring in maximum allowable surface defects and worst-case (thinnest wall) absorber tube geometry, the finite element analysis is performed. An example stress distribution is shown in Figure 3-7. The surface defect geometry is also shown.

The burst pressure is defined as the internal pressure at which any point in the tube reaches a stress intensity equal to the true ultimate strength of the material. Then, to calculate an allowable pressure, a safety factor of 2.0 is applied to the differential pressure across the absorber tube wall such that:

$$P_{allow} = \frac{\left(P_{burst} - P_{external}\right)}{2} + P_{external}$$

The calculated burst and allowable pressures are shown in Table 3-19. The results at operating temperature are limiting, and are used as the design basis allowable pressure of the tubes.

Absorber Tube Wear and Corrosion

Corrosion and wear are significant to the pressurization capability analysis of the absorber tube. In the pressurization analysis, the peak stress concentrations occur on the 'flat' portion of the tube. Combined corrosion and wear on this surface are modeled as a removal of material.

The analysis shows that combined corrosion and wear, modeled as a removal of material for the pressurization analysis, can exceed [[ ]] without affecting the design basis allowable pressure of the outer absorber tube shown in Table 3-19. For the D/S lattice absorber tube, the upper limit for combined corrosion and wear that occurs after control rod installation is [[ ]] For the C lattice absorber tube, the upper limit is [[ ]] This amount of wear is considered sufficiently conservative.

Maximum Stress Components

Stress components at the point of maximum stress intensity were analyzed for the absorber tube with the maximum allowable internal pressure. The point of maximum stress intensity is found to be on the outer edge of the absorber tube, at the middle of the flat portion. Principle stress components are shown in Table 3-20. All stress values shown in Table 3-20 are within the allowable stress value for 304S tubing of [[ ]] shown in Table 3-2.

Effect of the Welded Connection Between Absorber Tubes

The effect of the welded connection between adjacent absorber tubes on the stresses in the tube due to internal pressure was evaluated using a multiple tube finite element model. In this model, three adjacent absorber tubes were pressurized. A stress intensity distribution is shown in Figure 3-12. As shown, the maximum stress is at the flat portion of the tube exposed to the coolant. The effect of the adjacent pressurized tubes is to produce compressive rather than tensile stresses in the flat portions of the tube that are welded together. In this way, the opposing pressures from opposite sides of this welded ligament is actually beneficial in terms of the pressurization capability of the tubes.

A comparison of this multiple tube model to the single tube model showed that the single tube model predicts lower burst pressures. Therefore, the single tube model is used to determine design basis allowable pressures, and there is no degrading effect due to the lack of gaps between the absorber tubes in the Marathon-Ultra design.

The Marathon-5S and Marathon-Ultra Control Rod Blades (CRB) are manufactured using very low heat input laser weld processes. The resulting regions of microstructural change including the associated heat affected zones (HAZ) are very small (see Section 3.2). Based on general understanding, the fine HAZ microstructure will have mechanical properties that are equivalent to, or exceed, those of the wrought base material. Therefore, the HAZ will have mechanical properties that exceed the required minimum properties of the associated wrought material.

Two potential issues arise from welding of the absorber section: (1) sensitization and (2) residual stress. These issues are addressed below:

Sensitization: The low heat input laser welding processes have minimal impact on the wrought tube material, in that they typically do not result in sensitized material. To confirm this conclusion, the processes are continually evaluated metallographically to confirm the acceptability of the weld region (i.e., lack of sensitization). [[

]] Note also from Section 3.6.2 that these contact hoop stresses (and associated strains) have been eliminated for the Marathon-Ultra control rod.

Residual stress: One major effect of the welding process is that it will introduce tensile residual stresses in the narrow weld/HAZ region. These stresses are not a significant concern for two reasons: (1) The field cracking has not been associated with the weld HAZ and (2) the irradiation experienced by the CRB over the initial time of operation can significantly reduce these stresses by 60% or more through radiation creep processes (Reference 8). At this level of reduced stress, there is little concern for any effect on stress corrosion cracking (SCC) initiation or their applied stresses and strains. In that the major concern are strains from swelling, this level of stress is well below those levels required to even produce yielding (see also Section 3.2).

Effect of Irradiated Material

The pressurization finite element model uses unirradiated material properties. To test the assertion that the use of unirradiated properties in the pressurization finite element model is conservative, a test case is performed. The D lattice, 550 °F case is chosen for the test, with worst-case dimensions and maximum allowable surface defects. An internal pressure of [[ ]] is applied, which is the burst pressure found using unirradiated materials, as shown in Table 3-19. At this internal pressure, the maximum stress intensity using irradiated materials is [[ ]], which is less than the true ultimate strength of the irradiated material, [[ ]] Therefore, since the test case using irradiated material properties does not reach the ultimate strength of the irradiated material, the burst pressure analysis using unirradiated material properties is conservative. Further, the maximum strain intensity in the tube for the irradiated property test is low, at [[ ]]

Burst Pressure Tests

As discussed above, the allowable pressure for the absorber tube for the Marathon-Ultra is based on a finite element model incorporating worst-case dimensions, along with maximum specification permitted surface defects and expected wear. The finite element analysis shows that the worst-case burst pressure, on which the allowable pressure of the Marathon-Ultra tube is based, is [[ ]] lower than the burst pressure using nominal dimensions and no surface defects. See Table 3-21.

To confirm the finite element results, burst pressure tests were performed on two test specimens consisting of a short panel of welded absorber tubes, in which all tubes are pressurized, see Figures 3-13 and 3-14. The resulting tested burst pressures are compared to the finite element calculated burst pressures in Table 3-21.

As shown, the test results exceed the nominal predicted burst pressure by approximately [[ ]], and exceed the worst-case burst pressure (worst-case dimensions and surface defects) by a wide margin ([[ ]]). Since the design basis allowable pressure for the absorber tube is based on the worst-case burst pressure combined with a safety factor of 2.0, the design is conservative.

Conclusions

The analysis is conservative because it considers the combined effects of: (1) worst case tube dimensions (thinnest wall), (2) maximum allowable surface defects, (3) a large amount of combined corrosion and wear, and (4) unirradiated material properties. The true ultimate strength of the material will increase with irradiation. Burst pressure tests further validate the design basis allowable pressures.

3.6.5 Hafnium Application

As discussed in Section 2.1, the Marathon-Ultra incorporates hafnium rods as a neutron absorber in high-duty absorber tube locations. The configuration of the hafnium rods, including hafnium material requirements, diameter and length, are identical to the hafnium rods currently used in the original Marathon design (Reference 2).

GEH has a long, successful history of using hafnium as a neutron absorber in both DuraLife and Marathon control rods (Reference 2). In the Marathon design in particular, the hafnium rods are sealed from reactor coolant within the outer absorber tube. [[ ]] The inspection history of the application of hafnium rods to the Marathon design is very good, in that for all inspections of irradiated Marathon control rods contained in Reference 10, no material failures have been observed in any absorber tubes containing hafnium rods.

The diameters of the hafnium rods, the maximum hafnium rod diameter after thermal expansion, and the minimum absorber tube inside diameters are shown in Table 3-17. As shown, there is a large diametral gap between the hafnium and the absorber tube that allows for any expansion of the hafnium rod, ensuring that no strain is placed on the outer absorber tube.

Hydrogen Hydriding

Issues with the hydriding of hafnium have been observed in PWR applications (Reference 9). Hydriding involves hydrogen from the reactor coolant permeating the outer stainless steel tubing, and reacting with the hafnium to form hafnium hydride. Since hafnium hydride has a higher specific volume than hafnium, the hafnium rod may swell. The effect of the hydriding in PWRs has typically been observed as localized blisters or bulges on the surface of the hafnium rods, which place a strain on the outer cladding of the control assembly.

To investigate the occurrence of the hydriding phenomenon under lower pressure BWR conditions, a test was performed. In this test, two D lattice square absorber tube sections, with 6” long hafnium rods sealed inside, were loaded into a ‘dummy’ neutron source holder irradiation capsule, and irradiated for two, twelve month cycles in a BWR. The accumulated fast fluence was 1.6 to 2.4 x 10^{21} n/cm^2 (E > 1MeV).

After the test, hydrogen content of the hafnium test specimens was found to be [[ ]] for both specimens. Archive samples of the same material found initial hydrogen content to be [[ ]]. Therefore, [[ ]] The conclusions of this test apply equally to the simplified absorber tube, since the geometry of the tube is not expected to have any effect on the ability of the hydrogen to permeate the stainless steel tube and migrate to the hafnium. This is conservative, as the simplified absorber tube has a larger minimum wall thickness than the square tube. Therefore, if anything, the simplified tube should be less permeable to hydrogen transport.

Irradiated Hafnium Rod Measurements

GEH completed a Post-Irradiation Examination (PIE) of a highly irradiated Marathon control rod in April 2009 (Reference 11). As part of the on-going investigation, hafnium rods from this control rod were examined. [[ ]]

Diameter data from the irradiated hafnium is shown in Figure 3-16. By specification, the hafnium rods used in this control rod were to be between [[ ]] in diameter. It is noted that this is larger than the rods currently used for Marathon control rod and for the Marathon-Ultra control rod ([[ ]]). Figure 3-16 plots the diameter measurements versus the distance from the top of the absorber section. Data on the actual initial diameter of the hafnium rods is not available. However, any irradiation-related expansion phenomenon should be apparent by comparing the diameter at the top of the rod to the diameter at the bottom. This is because the top of the hafnium rod receives significantly more irradiation than the bottom of the rod.

As shown in Figure 3-16, [[ ]], it may be concluded that there were no absorber burn-up related stresses or strains placed on the outer absorber tubes containing hafnium rods.

Conclusions

GEH's experience with the application of hafnium to the Marathon design is very good, with no observed material failures for absorber tubes containing hafnium rods. [[ ]] Therefore, it may be concluded that no hafnium irradiation related stresses or strains will be placed on the outer absorber tube for the Marathon-Ultra design.

3.6.6 Irradiation Assisted Stress Corrosion Cracking Resistance

In order for the stress corrosion cracking mechanism to activate it requires a material that is susceptible, a conducive environment and a sustained tensile stress. If one of these three mechanisms is not present to a sufficient degree, the likelihood of a stress corrosion crack to form is significantly reduced. These three areas are addressed for the Marathon-Ultra design as follows.

Susceptible Material: The Marathon absorber tube is made from a GEH proprietary stainless steel, "Rad Resist 304S," which is optimized to be resistant to Irradiation Assisted Stress Corrosion Cracking (IASCC). The Marathon-Ultra absorber tubes are also fabricated from this material, and thus, are expected to have the same crack resistant properties. The tubes are delivered by the tubing supplier in a fully annealed condition, minimizing residual stress from the drawing process. Finally, the tubes are welded together using a low heat input laser weld process, resulting in low residual plastic strains and a very small heat affected zone (Section 3.6.4).

Sustained Tensile Stress: The Marathon-Ultra is designed such that [[
]]
(see Section 3.6.2). This significantly reduces the amount of stress/strain present in the absorber tubes at the end of life, and significantly reduces the likelihood of stress-corrosion cracking.

Conducive Environment: Like the Marathon-5S, the Marathon-Ultra is a completely crevice-free design for the absorber section and handle. All absorber tubes are sealed at the top and bottom, and full-length welds joining the tubes ensure that no crevice condition exists between the tubes. The elimination of handle rollers also ensures that the upper handle is crevice-free.

3.7 HANDLING LOADS

As for the Marathon-5S control rod (Reference 1), the Marathon-Ultra control rod is designed to accommodate twice the weight of the control rod during handling, to account for dynamic loads. The handle is analyzed using a finite element model, using worst-case geometry (see Figure 3-9). Table 3-13 shows the results of the handle loads analysis.

3.8 LOAD COMBINATIONS AND FATIGUE

The Marathon-Ultra control rod is designed to withstand load combinations including anticipated operational occurrences (AOOs) and fatigue loads associated with those combinations. The fatigue analysis is identical to that approved in Reference 1 for the Marathon-5S, and is based on the following assumed lifetime, which is consistent with previous analyses:

- [[ ]] and
- [[ ]]

For scram, each cycle represents a single scram insertion. Scram simulations show that the oscillations in the control rod structure damp out quickly. Further, it is extremely conservative to assume [[ ]] scrams with a 100% inoperative control rod drive buffer, as the loads experienced by the control rod in a normal buffered scram are much less severe.

For the Operational Basis Earthquake (OBE), with a total of [[ ]] seismic events, each event consists of [[ ]] cycles of control rod lateral bending. The assumption of [[ ]] lifetime OBE events is also considered very conservative.

Based on the reactor cycles, the combined loads are then evaluated for the cumulative effect of maximum cyclic loadings. The fatigue usage is evaluated against a limit of 1.0. The maximum cyclic stress is determined using a conservative stress concentration factor of 3.0. Table 3-14 shows the fatigue usage due to control rod scram at three limiting weld locations. In this analysis, it is assumed that each scram occurs with a 100% failed CRD buffer.

Table 3-15 shows the fatigue usage at the control rod outer edge due to bending from OBE seismic events and severe channel bow, control rod scram, and maximum absorber tube internal pressure. As can be seen, the combined fatigue usage is much less than 1.0.

Table 3-16 shows the fatigue usage at the tie rod to first absorber tube weld. The combined loading due to failed buffer scram, maximum absorber tube internal pressure, OBE seismic events and severe channel bow is considered. As shown, the combined fatigue usage is much less than 1.0.

It is well known that the cycles for fatigue initiation are dependent on the stress or strain range. The number of loading cycles that the control rod blade experience are limited to 100 for all of the different designs. The stress amplitudes are all in the elastic range. As shown in Table 3-14 through Table 3-16, based upon the ASME Section III fatigue design curve for un-irradiated austenitic material (Reference 6), the low number of cycles represents only a small amount of cumulative damage, well below the design limit. The ½ ultimate tensile stress value represents the ASME design limit for ~30,000 cycles. It has been established that an increase in the strength level, consistent with the effect of irradiation, would only increase the margin. This is supported by data on high strength materials, which confirm that the endurance limit is close to ½ ultimate tensile stress (Reference 7).

The last consideration with regard to fatigue is an evaluation of whether there is any flow-induced vibration that could in turn provide the potential for fatigue initiation. An assessment was performed to evaluate the loads induced by transverse loading. The evaluation that treated the control blade as a cantilever beam, found that the loads were very small and would not be sufficient to even close the gap between the blade and the fuel assembly. This load is considered so small as to be negligible, and would not lead to any risk of fatigue.

Table 3-1
Marathon-Ultra Material Properties

Material Type	Control Rod Components	Ultimate Tensile Strength, S_U (ksi)		Yield Strength, S_Y (ksi)		Modulus of Elasticity, E (x 10^6 psi)		Poisson's Ratio, ν	
		70 °F	550 °F	70 °F	550 °F	70 °F	550 °F	70 °F	550 °F
316 Plate	Handles and pads; VL fins, VL Hardware	[[							
316 Bar	Handle pads; VL hardware								
XM-19 Bar	VL socket								
CF3 Casting	VL vane casting, latch handle casting								
ER 308L	Capsule end caps, absorber tube end plugs, weld filler metal								
304S Bar	Tie rods								
304S Tubing	Absorber Tubes								
Hardened 304L Tubing	Capsule body tubes								]]

Table 3-2
Design Allowable Stresses for Primary Loads

Material Type	CR Components	½ Ultimate Tensile Stress S_m (ksi)	
		70 °F	550 °F
316 Plate	Handles and pads; VL fins, VL Hardware	[[	
316 Bar	Handle pads; VL hardware		
XM-19 Bar	VL transition socket		
CF3 Casting	VL vane casting, latch handle casting		
ER 308L	Capsule end caps, absorber tube end plugs, weld filler metal		
304S Bar	Tie rods		
304S Tubing	Absorber Tubes		
Hardened 304L Tubing	Capsule body tubes		]]

Table 3-3
Weld Quality Factors

Weld	Weld Inspection	Weld Quality Factor, q
Transition Socket to Fin	[[	
Fin to Absorber Section		
Handle to Absorber Section		
End Plug to Absorber Tube		
Vane to Transition Piece		]]

Table 3-4
Maximum Control Rod Failed Buffer Dynamic Loads

Components	Maximum Equivalent Loads in Kips (10^3 lbs) (Tension Listed as Negative)					
	D Lattice		C Lattice		S Lattice	
	70 °F	550 °F	70 °F	550 °F	70 °F	550 °F
Coupling	[[					
Velocity Limiter (VL)						
VL/Absorber Section Interface						
Absorber Section						
Handle/Absorber Section Interface						
Handle						]]

Table 3-5
D Lattice BWR/2-4 Failed Buffer Scram Stresses

Location	Room Temperature (70 °F)			Operating Temperature (550 °F)		
	Maximum Stress	Allowable Limit	Design Ratio	Maximum Stress	Allowable Limit	Design Ratio
Socket Minimum Cross-Sectional Area	[[					
VL Transition Socket to Fin Weld						
VL Fin Minimum Cross-Sectional Area						
Velocity Limiter to Absorber Section Weld						
Absorber Section						
Handle to Absorber Section Weld						
Handle Minimum Cross-Sectional Area						]]

Table 3-6
C Lattice BWR/4-5 Failed Buffer Scram Stresses

Location	Room Temperature (70 °F)			Operating Temperature (550 °F)		
	Maximum Stress	Allowable Limit	Design Ratio	Maximum Stress	Allowable Limit	Design Ratio
Socket Minimum Cross-Sectional Area	[[					
VL Transition Socket to Fin Weld						
VL Fin Minimum Cross-Sectional Area						
Velocity Limiter to Absorber Section Weld						
Absorber Section						
Handle to Absorber Section Weld						
Handle Minimum Cross-Sectional Area						]]

Table 3-7
S Lattice BWR/6 Failed Buffer Scram Stresses

Location	**Room Temperature (70 °F)**			**Operating Temperature (550 °F)**		
	Maximum Stress	**Allowable Limit**	**Design Ratio**	**Maximum Stress**	**Allowable Limit**	**Design Ratio**
Socket Minimum Cross-Sectional Area	[[					
VL Transition Socket to Fin Weld						
VL Fin Minimum Cross-Sectional Area						
Velocity Limiter to Absorber Section Weld						
Absorber Section						
Handle to Absorber Section Weld						
Handle Minimum Cross-Sectional Area						]]

Table 3-8
Outer Edge Bending Strain due to Seismic and Channel Bow Bending, Internal Absorber Tube Pressure and Failed Buffer Scram

Description	**D Lattice**	**C Lattice**	**S Lattice**
	550 °F	**550 °F**	**550 °F**
Outer Edge Bending Strain, Seismic (%)	[[		
Outer Edge Bending Strain, Seismic + Channel Bow (%)			
Max Internal Pressure Axial Stress (ksi)			
Max Failed Buffer Scram Stress (ksi)			
Total Outer Edge Strain, Seismic + Failed Buffer Scram + Absorber Tube Internal Pressure (%)			
Total Outer Edge Strain, Seismic + Channel Bow + Failed Buffer Scram + Absorber Tube Internal Pressure (%)			
Allowable Strain (%) ½ Ultimate, Irradiated			
Design Ratio			]]

Table 3-9
Absorber Tube to Tie Rod Weld Stress

Description	**D Lattice 550 °F**	**C Lattice 550 °F**	**S Lattice 550 °F**
Seismic + Internal Pressure, Max S_{INT} (ksi)	[[		
Seismic + Channel Bow + Internal Pressure, Max S_{INT} (ksi)			
Ultimate Tensile Stress (ksi)			
Design Ratio			]]

Table 3-10
Stuck Rod Compression Buckling – Entire Control Rod (Mode A)

Description	D Lattice		C Lattice		S Lattice	
	70 °F	550 °F	70 °F	550 °F	70 °F	550 °F
Critical Buckling Load, P_{cr} (lb)	[[					
Compressive Yield Load (lb)						
Maximum Stuck Rod Compression Load (lb)						
Added Compression Load due to Channel Bow (lb)						
Total Compressive Load (lb)						
Design Ratio, Buckling						
Design Ratio, Compressive Yield						]]

Table 3-11
Stuck Rod Compression Buckling – Control Rod Wing (Mode B)

Description	D Lattice		C Lattice		S Lattice	
	70 °F	550 °F	70 °F	550 °F	70 °F	550 °F
Critical Buckling Load, P_{cr} (lb)	[[					
Compressive Yield Load (lb)						
Total Compressive Load (lb)						
Design Ratio, Buckling						
Design Ratio, Compressive Yield						]]

Table 3-12
Boron Carbide Peak Temperatures

Parameter	Nominal Dimensions		Worst Case Dimensions	
	D/S Lattice	C Lattice	D/S Lattice	C Lattice
B_4C Centerline Temperature (°F)	[[			
Average B_4C Temperature (°F)				
Helium Release Fraction (%)				]]

Table 3-13
Handle Lifting Load Stress

Lattice Type	Handle Type	Maximum Stress Intensity (ksi)	Design Ratio, ½ Ultimate Stress
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		]]

Table 3-14
Fatigue Usage due to Failed Buffer Scram

Location	D Lattice				C Lattice				S Lattice			
	Stress Amp. (ksi)	Allow Cycles (N)	Actual Cycles	Usage	Stress Amp. (ksi)	Allow Cycles (N)	Actual Cycles	Usage	Stress Amp. (ksi)	Allow Cycles (N)	Actual Cycles	Usage
Transition Piece to Fin Weld	[[											
VL Fin to Absorber Section Weld												]]

Table 3-15
Fatigue Usage at Absorber Section Outer Edge

Stress Type	D Lattice				C Lattice				S Lattice			
	Stress Amp. (ksi)	Allow Cycles (N)	Actual Cycles	Usage	Stress Amp. (ksi)	Allow Cycles (N)	Actual Cycles	Usage	Stress Amp. (ksi)	Allow Cycles (N)	Actual Cycles	Usage
Absorber Section Outer Edge - Scram + Internal Pressure	[[											
Absorber Section Outer Edge – Seismic + Channel Bow												]]
	Total Usage =		[[ ]]		Total Usage =		[[ ]]		Total Usage =		[[ ]]	

Table 3-16
Fatigue Usage at Absorber Tube to Tie Rod Weld

Stress Type	D Lattice				C Lattice				S Lattice			
	Stress Amp. (ksi)	Allow Cycles (N)	Actual Cycles	Usage	Stress Amp. (ksi)	Allow Cycles (N)	Actual Cycles	Usage	Stress Amp. (ksi)	Allow Cycles (N)	Actual Cycles	Usage
Absorber Tube to Tie Rod Weld - Scram	[[											
Absorber Tube to Tie Rod Weld – Seismic + Channel Bow + Internal Pressure												]]
	Total Usage =		[[ ]]		Total Usage =		[[ ]]		Total Usage =		[[ ]]	

Table 3-17
Hafnium Rod Dimensions

Parameter	D/S Lattice	C Lattice
DIA_{70}, Maximum Hafnium Rod Diameter (in)	[[	
DIA_{550}, Maximum Hafnium Diameter, Thermal Expansion (in)		
Minimum Absorber Tube Inside Diameter After Welding (in)		]]

Table 3-18
Irradiated Boron Carbide Capsule Swelling Calculation

Parameter	**D/S Lattice**	**C Lattice**
Absorber Tube ID Before Welding (in)	[[	
Minimum Absorber Tube ID After Welding (in)		
Capsule OD (in)		
Capsule Wall Thickness (in)		
Maximum Capsule OD_0 (in)		
Maximum Capsule ID_0 (in)		
Capsule ID strain (in/in)		
Capsule OD strain (in/in)		
Capsule OD at 100% local depletion (in)		]]

Table 3-19
Absorber Tube Pressurization Results: Minimum Material Condition with OD and ID Surface Defects

Lattice	Temp (ºF)	External Pressure (psi)	FEA Burst Pressure (psi)	**Allowable Pressure (psi)**
C	70	14.7	[[	
C	550	1050		
D	70	14.7		
D	550	1050		]]

Table 3-20
Absorber Tube Pressurization Results: Principle Stress Results at Operating Temperature and Pressure and Maximum Allowable Pressure

Stress Component	D/S Lattice	C Lattice
S1 (Hoop)	[[	
S2 (Axial)		
S3 (Radial)		
Stress Intensity		
Equivalent Stress		]]

Table 3-21
D/S Lattice Burst Pressure Results from FEA and Testing

Parameter (D/S Lattice)	Burst Pressure (psia)
Nominal Dimensions (FEA)	[[
Worst-Case Dimensions and Maximum Surface Defects (Design Basis) (FEA)	
Specimen 1 Tested Burst Pressure	
Specimen 2 Tested Burst Pressure	]]

Table 3-22
D/S Lattice Thermal Analysis Results

Location	Nodal Temp (°F)	
	Nominal Dimensions	Worst Case Dimensions
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		
Avg He Void		]]

Table 3-23
C Lattice Thermal Analysis Results

Location	Nodal Temp (°F)	
	Nominal Dimensions	Nominal Dimensions
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		
Avg He Void		]]

Table 3-24
Finite Element Analysis Summary

Analysis Description	Section	Geometry Inputs	Applied Loads	Material Properties	Acceptance Criteria	Marathon-Ultra Comparison to Marathon-5S
Thermal Analysis: Determines the temperature of boron carbide during operation. Uses heat generation due to neutron capture.	3.6.3	Absorber tube and capsule geometries. Worst-case geometries (largest helium gap) are used. Internal wing absorber tube modeled by assuming zero heat flux to adjacent tubes. Conservative crud build-up used.	Peak boron carbide heat generation rates from nuclear analyses.	Thermal conductivities from various sources.	Thermal stress less than allowable stress.	Identical methodology. Only difference is incorporating Marathon-Ultra capsule geometry and heat generation rates.
Lifting Load: Determines stresses in the handle while lifting the control rod.	3.7	Worst-case geometry from handle drawings.	2x control rod weight	Unirradiated linear-elastic material properties.	Maximum stress intensity is compared to material allowable stress.	Identical methodology. Only difference is slightly heavier Marathon-Ultra weights, resulting is slightly larger loads. Analysis shows linear relationship between control rod weight and peak stress intensity.
External Pressure + Channel Bow Lateral Load: Determines stresses in the absorber tube due to lateral loads imposed by bowed fuel channels combined with RPV operating pressure.	3.4.3	One-quarter affected tube with ¼-symmetry boundary conditions. Worst-case dimensions.	Maximum lateral loads from fuel channel bow studies.	Unirradiated linear-elastic material properties. Also checked using unirradiated elastic-plastic true stress-strain curves.	Maximum stress intensity compared to material allowable stress.	Identical
Internal Pressure: Determines maximum allowable absorber tube internal pressure.	3.6.4	Uses worst-case tube dimensions and allowable surface defects. Also checked first tube attached to the tie rod (tie rod modeled as an empty tube).	Reactor pressure vessel internal pressure to exterior of tubes for 'hot' cases. Unirradiated property analyses determine maximum allowable internal pressure. 'Check' analyses apply this pressure as appropriate.	Unirradiated elastic-plastic true stress-strain curves. Also checked using irradiated material properties.	Burst pressure defined to be internal pressure at which the stress intensity at any location in the tube first reaches the true ultimate strength. Then, a factor of safety of 2.0 is used to determine an allowable pressure.	Identical
Pressurization Stress on Absorber Tubes: Finite element analysis is used to determine the radial, hoop, and axial stress in the absorber tube at allowable internal pressure.	3.6.4	Worst-case absorber tube dimensions.	Maximum allowable pressure determined in internal pressure analysis.	Unirradiated elastic-plastic true stress-strain curves.	Combined stresses less than material allowable stresses.	Identical
Combined Internal Pressure + Fuel Channel Bow Induced Bending: Determines maximum stresses in the absorber tube to tie rod weld.	3.4.2	Worst-case absorber tube dimensions. Model consists of tie rod and entire wing of absorber tubes.	Lateral loads from channel bow studies and seismic event limits.	Unirradiated elastic-plastic true stress-strain curves.	Maximum stress intensity less than material allowable stress.	Identical

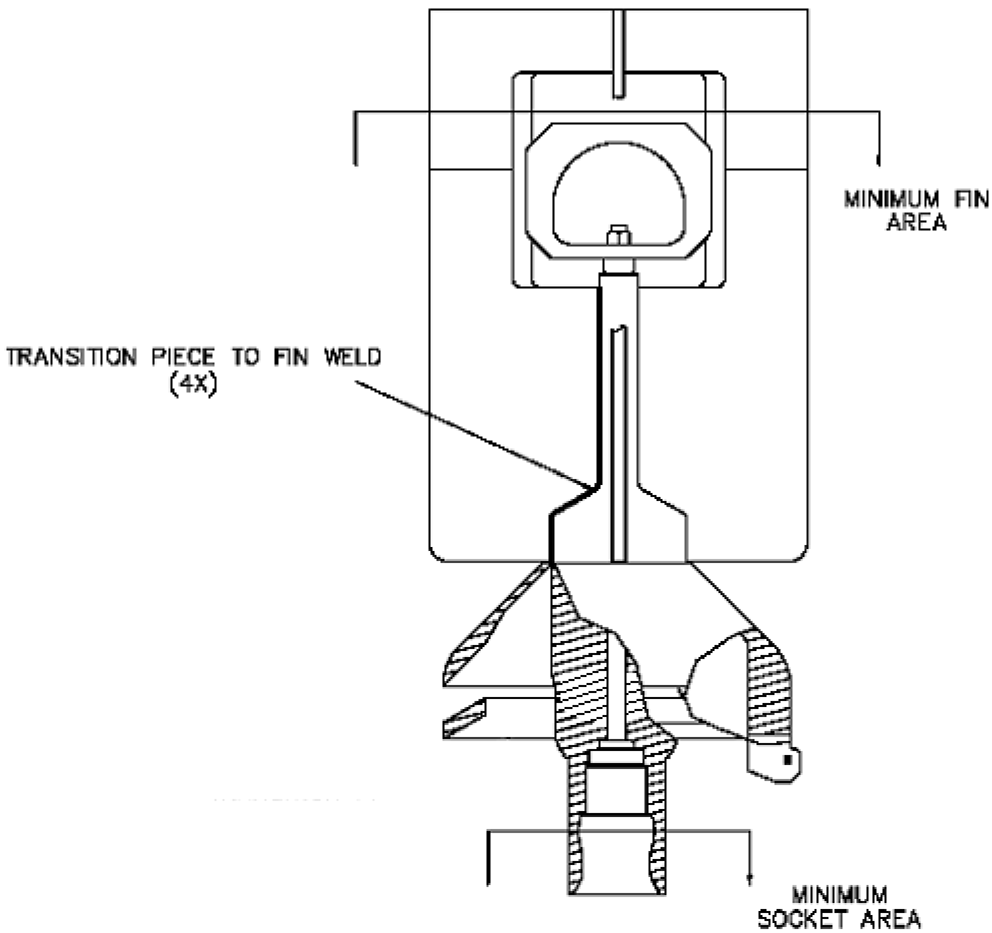


Figure 3-1. Velocity Limiter Welds and Cross-Sections Analyzed

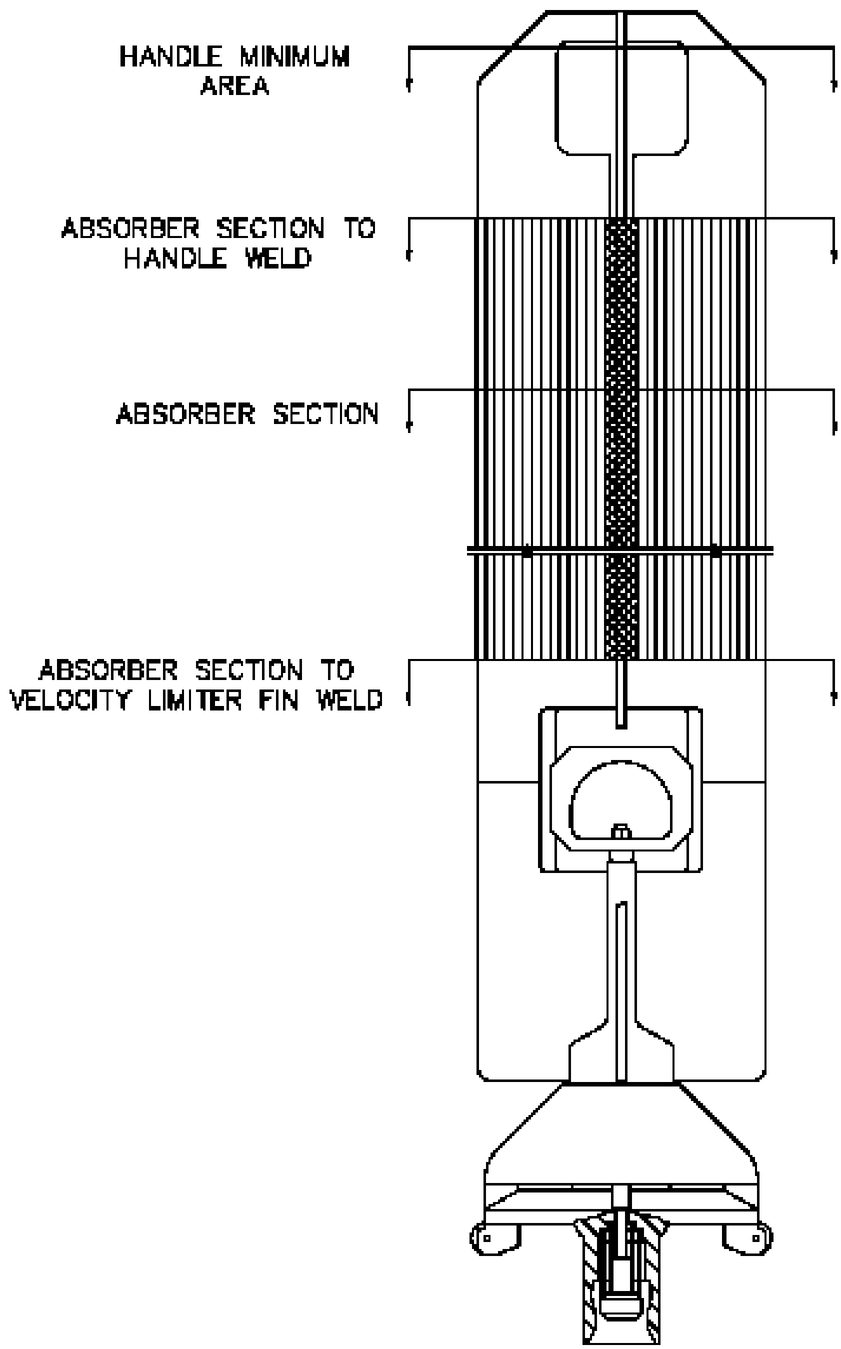


Figure 3-2. Control Rod Assembly Welds and Cross-Sections Analyzed

[[ ]]

Figure 3-3. Absorber Tube to Tie Rod Finite Element Model

[[ ]]

Figure 3-4. Lateral Load Finite Element Model

[[ ]]

Figure 3-5. Lateral Load Finite Element Results (C Lattice)

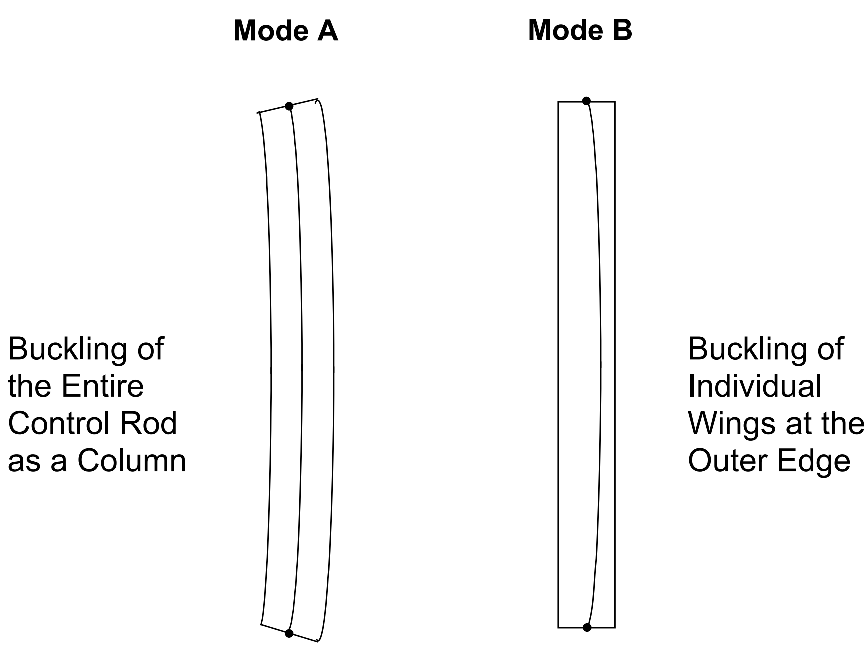


Figure 3-6. Control Rod Buckling Modes

[[ ]]

Figure 3-7. Absorber Tube Pressurization Finite Element Model

[[ ]]

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

[[ ]]

Figure 3-9. Handle Lifting Loads Finite Element Model

[[

]]

Figure 3-10. D/S Lattice Thermal Analysis Results

[[

]]

Figure 3-11. C Lattice Thermal Analysis Results

[[ ]]

Figure 3-12. Stress Intensity Distribution for Multiple Tube Pressurization Finite Element Model, All Tubes Pressurized

[[

]]

Figure 3-13. Absorber Tube Burst Pressure Test Specimen – After Test

[[ ]]

Figure 3-14. Absorber Tube Burst Pressure Test Specimen Rupture

[[

]]

Figure 3-15. Irradiated Boron Carbide Diametral Swelling Data

[[ ]]

Figure 3-16. Irradiated Hafnium Diameter Data

4. NUCLEAR EVALUATIONS

4.1 DESIGN CRITERIA

A control rod's nuclear worth characteristics shall be compatible with reactor operation requirements. As approved in References 1 and 2, a replacement control rod can meet these requirements by demonstrating that the initial hot and cold CRB reactivity worths are within ± 5% $\Delta k/k$ (where $\Delta k/k$ is $1-k_{con}/k_{unc}$) of the original equipment control rod blade design worth. Replacement rods with reactivity worth outside this tolerance require, as a minimum, evaluations on cold shutdown margin, AOO CPR, control rod drop accident, fuel cycle economics, nuclear methods, and control rod lifetime.

For GEH original equipment control rods, the nuclear lifetime is defined as the quarter-segment depletion at which the control rod cold worth ($\Delta k/k$) is 10% less than its zero-depletion cold worth. The original equipment (DuraLife 100) control rods consist of thin sheaths enclosing boron carbide filled tubes. The sheaths are welded to a central tie rod to form the cruciform shape of the control rods. The original equipment control rods are shown in Figure 4-7.

As discussed above, a retrofit design may have an initial cold worth that differs from the original equipment control rod that it is replacing, within ±5% of the initial worth of that control rod (the "matched worth" criterion). The nuclear lifetime for such a retrofit control rod is defined as the quarter-segment depletion at which the cold worth is the same as the end-of-nuclear-life cold worth of the original equipment control rod that it is replacing.

4.2 METHODOLOGY

The methodology applied to the Marathon-Ultra control rod is identical to the methodology of the Marathon-5S analysis approved in Reference 1. This includes the use of the same computer codes, described below, and in the Reference 1 report.

The nuclear lifetime for a particular control blade design is determined with a two-dimensional step-wise depletion of the control blade poisons. This is done by computing the eigenvalue for hot, voided conditions with a Monte Carlo neutron transport code. The poison reaction rates from the analysis are then assumed to be constant for a fixed period of time (Δt) to obtain the number of absorptions for each discrete area of the blade. The poison number densities are then updated in the Monte Carlo code input and another eigenvalue calculation is performed. This process continues until the reduction in cold worth – as computed by companion cold Monte Carlo eigenvalue calculations – reaches the end-of-nuclear-life criterion.

For locations within the blade that use boron carbide as a poison, the change in the number of absorber atoms is computed as:

$$\frac{dN_{B-10}}{dt} = -(N \cdot \sigma)_{B-10}$$

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:

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

Reactivity worth calculations for the Marathon-5S are performed using a GEH controlled version of MCNP4A, which was developed by the Los Alamos National Laboratory (Reference 3). MCNP is a Monte Carlo code for solving the neutral-particle transport equation as a fixed source or an eigenvalue problem in three dimensions. Continuous energy cross-section data is used in the calculation, thus making creation of multi-group cross-sections unnecessary. The use of MCNP is the only process change from the original Marathon nuclear analysis, which used MERIT. Otherwise, depletion calculations remain unchanged.

Two additional utility codes are used in conjunction with MCNP. The GEH utility code "MODL" is used to set up the MCNP input deck, based on lattice design data and control rod design data. The GEH utility code "HO" is coupled to MCNP for the depletion calculation. It reads the MCNP tallies (cell fluxes and absorber cross-sections) and then performs the control blade depletion calculation. The depleted absorber atom densities are then used to update the MCNP inputs for the next time step. MCNP input data for cold case are also generated with "HO" by modifying the input data from the hot inputs.

For the depletion calculations that are performed for each fuel lattice, the time step used is 100 days. In order to reach the 10% cold worth reduction for the nuclear lifetime evaluation, a total of 21 time steps are used for the re-calculation of DuraLife 100 (original equipment), and a total of 30 time steps are used for the calculation of Marathon-Ultra lifetime. Tables 4-13 through 4-15 contain input parameters used to model the original equipment and Marathon-Ultra control rods.

The self-shielding characteristics of B-10, defined as the faster depletion of B-10 on the outer edge of B_4C column than the average pin due to spatial self-shielding of B-10, is accounted for in the MCNP calculations. The calculations use a ring model that divides each B_4C column into four concentric rings of equal cross-sectional area. The radii of the boron carbide rings used in the updated analysis are shown in Table 4-12.

4.3 CONTROL ROD NUCLEAR LIFETIME

A description of the fuel bundles used for the D, C, and S lattice control rod nuclear lifetime calculations are shown in Figures 4-1 through 4-3. Both the hot and cold calculation results for the peak ¼ segment are shown in Tables 4-1 through 4-3. The cold calculation results, on which the nuclear lifetime is based, are shown graphically in Figures 4-4 through 4-6. The nuclear lifetimes, based on a cold worth equal to a cold worth reduction of 10% for an original equipment control rod are summarized in Table 4-4.

4.4 INITIAL CONTROL ROD WORTH

As discussed above, a control rod with an initial (non-depleted) reactivity worth within ±5% of the original equipment control rod is considered "matched worth" and therefore, does not require any special treatment in plant core analyses. The initial cold and hot worths (0% depletion) of the Marathon-Ultra control rod designs are found in Tables 4-1 through 4-3. These values of $\Delta k/k$ are then compared to the worths of the original equipment control rods in Tables 4-6 through 4-8. All cold and hot initial control rod worths are within ±5% of the original equipment, and can be considered to be direct nuclear replacements of the original equipment.

4.5 HEAT GENERATION RATES

The capture of neutrons by B-10 atoms results in the release of energy, or heat generation. As discussed in Section 3.6, a thermal model of the absorber tube and capsule is used to calculate boron carbide temperatures within the capsules, which affects the rate of helium release. The heat generation rates for the Marathon-Ultra designs are calculated assuming 2.79 MeV per neutron capture in B-10. Then, a radial peaking factor is employed to determine the heat generation rate in the highest fluence absorber tube, which is the outermost tube.

Both average and peak heat generation rates are shown in Table 4-5. The peak heat generation rates are used in the thermal model discussed in Section 3.6 to determine the capsule boron carbide temperatures shown in Table 3-12.

4.6 CONTROL ROD MECHANICAL LIFETIME

The control rod mechanical lifetime methodology is identical to the Marathon-5S methodology approved in Reference 1. As discussed in Section 3.6, the lifetime limiting mechanism for the Marathon-Ultra control rod is the pressurization of the absorber tubes due to the helium release from the irradiated boron carbide. An absorber tube mechanical limit as a function of average B-10 per cent depletion is calculated based on peak heat generation, temperatures and helium release fractions, combined with worst-case component geometries. As discussed in Section 3.6, the method for evaluating the swelling phenomenon of irradiated boron carbide is very conservative, using worst-case capsule and absorber tube dimensions, along with a +3σ upper limit swelling rate assumption. Using these conservatisms, the Marathon-Ultra capsule is designed [[

]]

The table used to calculate the control rod mechanical lifetime limit, in terms of a four-segment average B-10 depletion, is shown in Tables 4-9, 4-10 and 4-11 for D, C, and S lattice applications. Along the top of the table is the absorber tube number, where tube 1 is the first absorber tube, welded to the cruciform tie rod. Also shown are the span-wise radial peaking factors, which show the relative absorption rate of each absorber tube. A limiting axial depletion profile is used to calculate the B-10 depletion for each absorber tube and axial node. At the bottom of the table, the average depletion for each tube is shown, along with the depletion limit for that tube, which varies depending on the number of empty capsule plenums employed at the bottom of the absorber column. Through an iterative process, the peak ¼ segment depletion is raised until the limiting absorber tube reaches its mechanical limit. The 4-segment mechanical lifetime of the control rod is then the average of the four ¼ segments. The 4 segment mechanical lifetime limits are summarized in Table 4-4, along with the peak ¼ segment nuclear lifetime limits.

Tables 4-9 through 4-11 calculate the average depletion in all absorber tubes, at nuclear end of life. To accomplish this, the ¼-segment nuclear limit is entered into the peak ¼-segment. As shown along the bottom of the tables, the average depletion for each tubes is well below each tubes' limit. Therefore, the nuclear lifetime of the Marathon-Ultra control rod is limiting, in that the mechanical lifetime exceeds the nuclear lifetime for all cases.

Table 4-1
D Lattice Depletion Calculation Results

Irradiation Time (days)	Equivalent B-10 Depletion (%)	Hot, Voided Eigenvalue	Hot Worth (Δk/k)	Hot Change in Worth (%)	Cold Eigenvalue	Cold Worth (Δk/k)	Cold Change in Worth (%)
[[							]]

Table 4-2
C Lattice Depletion Calculation Results

Irradiation Time (days)	Equivalent B-10 Depletion (%)	Hot, Voided Eigenvalue	Hot Worth ($\Delta k/k$)	Hot Change in Worth (%)	Cold Eigenvalue	Cold Worth ($\Delta k/k$)	Cold Change in Worth (%)
[[							]]

Table 4-3
S Lattice Depletion Calculation Results

Irradiation Time (days)	Equivalent B-10 Depletion (%)	Hot, Voided Eigenvalue	Hot Worth (Δk/k)	Hot Change in Worth (%)	Cold Eigenvalue	Cold Worth (Δk/k)	Cold Change in Worth (%)
[[							]]

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 4-5
Heat Generation Rates

Application	Average Heat Generation Rate (Watts/gram B_4C)	Radial Peaking Factor	Peak Tube Heat Generation Rate (Watts/gram B_4C)
D Lattice, BWR/2-4	[[		
C Lattice, BWR/4,5			
S Lattice, BWR/6			]]

Table 4-6
Initial Reactivity Worth, D Lattice (BWR/2-4) Original Equipment and Marathon-Ultra CRBs

Condition	Original Equipment Δk/k	Marathon-Ultra Δk/k	Marathon-Ultra Change from Original Equipment
Cold	[[		
Hot (40% Void)			]]

Table 4-7
Initial Reactivity Worth, C Lattice (BWR/4,5) Original Equipment and Marathon-Ultra CRBs

Condition	Original Equipment Δk/k	Marathon-Ultra Δk/k	Marathon-Ultra Change from Original Equipment
Cold	[[		
Hot (40% Void)			]]

Table 4-8
Initial Reactivity Worth, S Lattice (BWR/6) Original Equipment and Marathon-Ultra CRBs

Condition	Original Equipment Δk/k	Marathon-Ultra Δk/k	Marathon-Ultra Change from Original Equipment
Cold	[[		
Hot (40% Void)			]]

Table 4-9
D Lattice Mechanical Lifetime Calculation

[[

]]

Table 4-10
C Lattice Mechanical Lifetime Calculation

[[

]]

Table 4-11
S Lattice Mechanical Lifetime Calculation

[[

]]

Table 4-12
Boron Carbide Ring Radii in MCNP Model

Ring Number	Ring Radial Thickness (cm)	
	Marathon-Ultra, D and S Lattice	Marathon-Ultra, C Lattice
1 (inner)	[[	
2		
3		
4 (outer)		]]

Table 4-13
D Lattice Original Equipment and Marathon-Ultra Dimensions

Description		**DuraLife 100 D**		**Marathon-Ultra D**	
		(inches)	(cm)	(inches)	(cm)
Span		[[			
Half Span	SBL				
Wing Thickness (Square Tube Width)					
Half Wing Thickness	TBL				
Tie Rod Half Thickness	TTR				
Radius of Central Support Filet	RBLF				
Radius of Blade Tip	RBLT				
Span of Central Support (Tie Rod)					
Half Span of Central Support	SCS				
Thickness of Sheath	TSH				
Inner Diameter of Tube (Capsule)	TID				
Outer Diameter of Tube	TOD				
Wall Thickness of Tube					
Diameter of Hafnium Rod					
Type	IBLADE				
Number of B_4C Tubes (Capsules)	NOPT				
Number of Hafnium Rods	NOHFT				
Number of Empty Tubes	NOBT				]]

Table 4-14
C Lattice Original Equipment and Marathon-Ultra Dimensions

Description		**DuraLife 100 C**		**Marathon-Ultra C**	
		(inches)	(cm)	(inches)	(cm)
Span		[[			
Half Span	SBL				
Blade Thickness (Square Tube Width)					
Half Blade Thickness	TBL				
Tie Rod Half Thickness	TTR				
Radius of Central Support Filet	RBLF				
Radius of Blade Tip	RBLT				
Span of Central Support (Tie Rod)					
Half Span of Central Support	SCS				
Thickness of Sheath	TSH				
Inner Diameter of Tube (Capsule)	TID				
Outer Diameter of Tube	TOD				
Wall Thickness of Tube					
Diameter of Hafnium Rod					
Type	IBLADE				
Number of B4C Tubes (Capsules)	NOPT				
Number of Hafnium Rods	NOHFT				
Number of Empty Tubes	NOBT				]]

Table 4-15
S Lattice Original Equipment and Marathon-Ultra Dimensions

Description		**DuraLife 100 S**		**Marathon-Ultra S**	
		(inches)	(cm)	(inches)	(cm)
Span		[[			
Half Span	SBL				
Wing Thickness (Square Tube Width)					
Half Wing Thickness	TBL				
Tie Rod Half Thickness	TTR				
Radius of Central Support Filet	RBLF				
Radius of Blade Tip	RBLT				
Span of Central Support (Tie Rod)					
Half Span of Central Support	SCS				
Thickness of Sheath	TSH				
Inner Diameter of Tube (Capsule)	TID				
Outer Diameter of Tube	TOD				
Wall Thickness of Tube					
Diameter of Hafnium Rod					
Type	IBLADE				
Number of B4C Tubes (Capsules)	NOPT				
Number of Hafnium Rods	NOHFT				
Number of Empty Tubes	NOBT				]]

[[

]]

Figure 4-1. D Lattice Fuel Bundle Rod Position and Enrichment

[[

]]

Figure 4-2. C Lattice Fuel Bundle Rod Position and Enrichment

[[

]]

Figure 4-3. S Lattice Fuel Bundle Rod Position and Enrichment

[[

]]

Figure 4-4. D Lattice Control Rod Cold Worth Reduction with Average Depletion

[[

]]

Figure 4-5. C Lattice Control Rod Cold Worth Reduction with Average Depletion

[[

]]

Figure 4-6. S Lattice Control Rod Cold Worth Reduction with Average Depletion

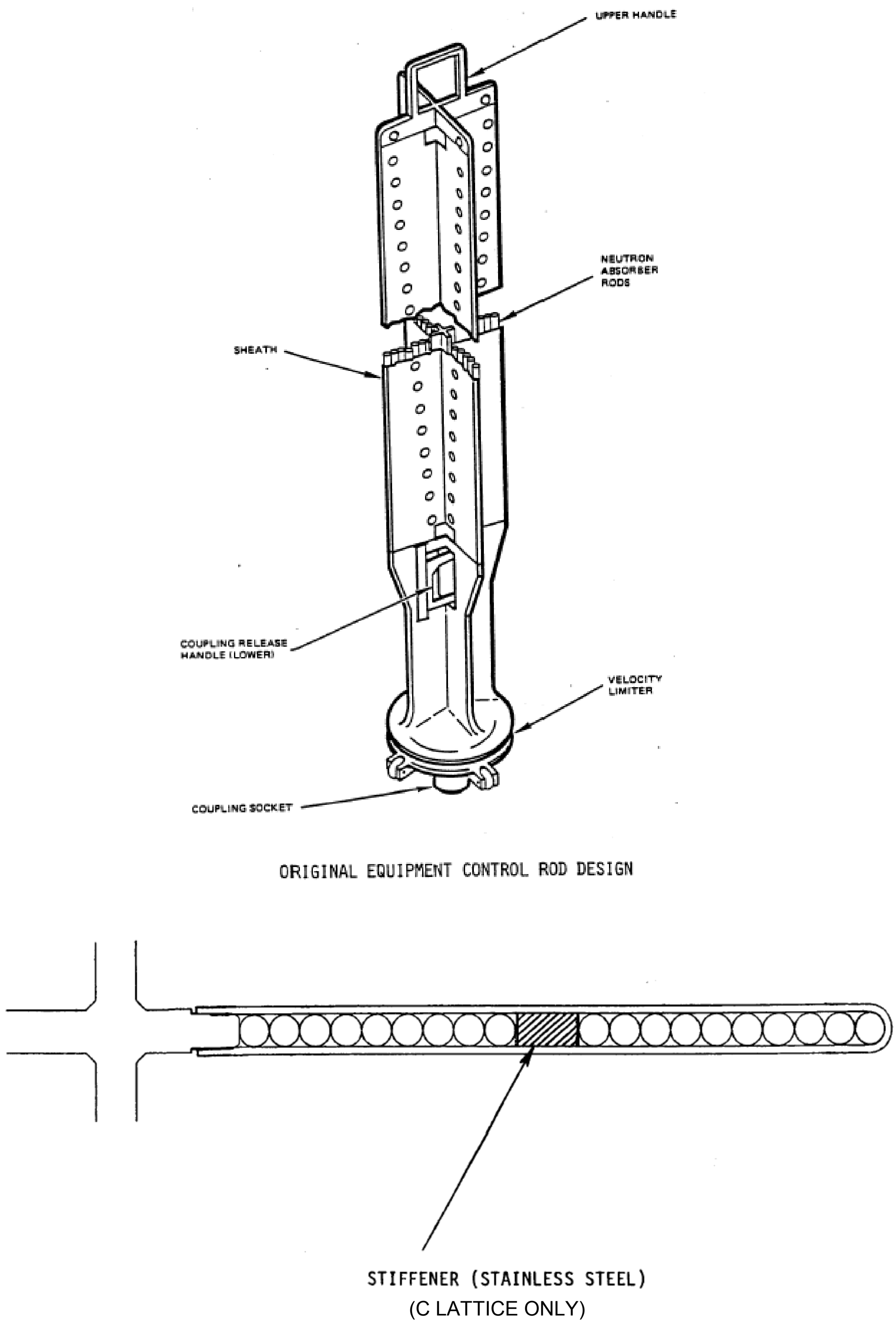


Figure 4-7. BWR/2-6 Original Equipment

5. OPERATIONAL EVALUATIONS

5.1 DIMENSIONAL COMPATIBILITY

As discussed in Section 2, the outer structure of the Marathon-Ultra is identical to the Marathon-5S approved in Reference 1. The width of the absorber tube and the width of the control rod wing of the Marathon-Ultra control rod are also identical to the original Marathon control rod. Plus, all other envelope dimensions, including tie rod, handle, and velocity limiter are identical. Therefore, the fit and clearance of the Marathon-Ultra control rods in the fuel cell is identical to the Marathon and Marathon-5S control rods.

Reference 10 provides a summary of the inspection history of the Marathon control rod. For all of these inspections, no issues have been identified with respect to the lack of dimensional stability of the Marathon control rod assembly. The inspections have not shown signs of excessive wear on the control rod due to any distortion of the control rod assembly.

Therefore, the inspection history of the Marathon control rod demonstrates that the Marathon design is dimensionally stable, even with significant amounts of irradiation and residence time.

5.2 SCRAM TIMES

An OBE or SSE earthquake condition could cause the fuel channels to temporarily bow or bend. In addition, as fuel channels age, they tend to both bulge and bow, which can negatively affect the insertion capability of the control rod blade.

Previous Marathon prototype scram testing shows that the insertion capability of the CRB is affected by the stiffness of the assembly. The stiffer (less flexible) the control rod assembly, the longer the scram times. The stiffness of the Marathon-5S and Marathon-Ultra control rods have been evaluated to be equal to or less stiff than the Marathon CRB, in terms of the assembly cross-sectional area moment of inertia. Therefore, the Marathon-Ultra control rod will have a scram insertion capability equal to or better than the Marathon CRB, in the event of temporary or permanent channel deformation.

The overall assembly weight of the Marathon-Ultra CRB is not greater than the maximum weights of Marathon control rod designs produced. This, combined with the bending stiffness characteristics, ensure that the Marathon-Ultra CRB design will not have an adverse effect on scram times.

The results of seismic scram tests applicable to the Marathon-Ultra design are discussed in Section 3.4.4. As discussed, for all lattice types, the control rods met the acceptance criteria of successful insertion within scram time requirements under OBE fuel channel deflection conditions, and successful insertion under SSE fuel channel deflection conditions.

5.3 'NO SETTLE' CHARACTERISTICS

A 'no settle' condition may occur in the event of excessive friction between the control rod and the fuel channels. If this additional friction does not allow the weight of the CRB to settle the assembly into a control rod drive (CRD) positional notch, a 'no settle' condition occurs. As previously discussed, the envelope dimensions for the Marathon-Ultra CRB are identical to the Marathon and Marathon-5S control rods. Further, the wet (buoyant) weight of the Marathon-Ultra assembly is within the range of weights of previous Marathon and Marathon-5S control rod designs. Therefore, the ability of the Marathon-Ultra assembly to settle into a CRD notch is equal to that of the Marathon or Marathon-5S control rod.

5.4 DROP SPEEDS

The parameters that affect the drop speed of the control rod in the event of a rod drop accident are the weight of the control rod assembly, and the geometry of the "bell" of the velocity limiter. The Marathon-Ultra control rod uses the same cast or FabriCast (hybrid cast/fabricated) velocity limiters as those on the Duralife, Marathon and Marathon-5S control rods. Since the weight of the Marathon-Ultra control rod is less than the weight of the Duralife control rods used for the original drop tests, the Marathon-Ultra control rod will have drop speeds less than the [[ ]] required. Therefore, the Marathon-Ultra CRB will limit the reactivity insertion rate during a CRDA within the existing safety analysis parameters.

5.5 FUEL CELL THERMAL HYDRAULICS

The surface geometry of the Marathon-Ultra and Marathon-5S control rods are different than the Marathon control rod due to the different outer absorber tube geometry. In order to evaluate the effect on the thermal hydraulics of the fuel cell, the total displaced volume of the Marathon-Ultra or Marathon-5S control rod is compared to the Marathon control rod, approved in Reference 1. The S lattice, BWR/6 version of these control rods are chosen for this comparison.

The total displaced volume for the Marathon control rod is [[ ]] the total displaced volume of the Marathon-Ultra control rod is [[ ]], for a difference of [[ ]] from the Marathon control rod. This small difference is judged to be negligible in its effect on the thermal hydraulics of the fuel cell.

The topographic differences between the Marathon-Ultra and the Marathon control rods is less significant than the differences between the Marathon control rods and DuraLife type control rods and control rods from other vendors. These small topographic changes will have no significant effect on the thermal hydraulics of the fuel cell.

6. LICENSING CRITERIA

The NRC Safety Evaluation Report for the Marathon and Marathon-5S Control Rod Blades (within References 1 and 2) identify five criteria for the licensing and evaluation of BWR control rods. These same five criteria are used for the Marathon-Ultra control rod.

6.1 STRESS, STRAIN, AND FATIGUE

6.1.1 Criteria

The control rod stresses, strains, and cumulative fatigue shall be evaluated to not exceed the ultimate stress or strain of the material.

6.1.2 Conformance

As discussed in Section 3, the design changes for the Marathon-Ultra CRB have been evaluated using the same or more conservative design bases and methodology than the Marathon and Marathon-5S control rods. All components of the Marathon-Ultra control rod are found to be acceptable when analyzed for stresses due to normal, abnormal, emergency, and faulted loads. The design ratio, which is the effective stress divided by the stress limit or the effective strain divided by the strain limit, is found to be less than or equal to 1.0 for all components. Conservatism is included in the evaluation by limiting stresses for all primary loads to one-half of the ultimate strength (i.e., a safety factor of two is employed).

The fatigue usage of the Marathon-Ultra control rod is calculated using the same methodology as the Marathon and Marathon-5S control rods. The fatigue analysis assumes [[ ]]. It is found that the calculated fatigue usage is less than the material fatigue capability (the fatigue usage factor is much less than 1.0).

6.2 CONTROL ROD INSERTION

6.2.1 Criteria

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.

6.2.2 Conformance

The thickness of the wing of the Marathon-Ultra CRB, [[ ]], is identical to the Marathon and Marathon-5S control rods. 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 rod in the fuel cell is identical to the Marathon and Marathon-5S control rods.

An OBE or SSE earthquake condition potentially could cause the fuel channels to temporarily bow or bend. In addition, as fuel channels age, they tend to both bulge and bow, which can negatively affect the insertion capability of the control rod blade.

Previous Duralife and Marathon prototype seismic scram testing has shown that the insertion capability of the CRB is affected by the stiffness of the assembly and by the assembly weight. If the control rod assembly is stiffer (less flexible), then the scram times are longer. The stiffness of the Marathon-5S and Marathon-Ultra control rods has been evaluated to be equal to or less stiff than the Marathon control rod, in terms of the assembly cross-sectional area moment of inertia. This, combined with the fact that the Marathon-Ultra assembly is lighter than previous control rod designs shows that the Marathon-Ultra CRB has a scram insertion capability equal to or better than the Marathon CRB in the event of temporary or permanent channel deformation.

The results of seismic scram tests, applicable to Marathon-Ultra control rods, are discussed in Section 3.4.4. As discussed, for all lattice types, the control rods successfully inserted within scram time requirements under OBE fuel channel deflection conditions, and successfully inserted under SSE fuel channel deflection conditions. This meets all acceptance criteria for the test.

6.3 CONTROL ROD MATERIAL

6.3.1 Criteria

The material of the control rod shall be shown to be compatible with the reactor environment.

6.3.2 Conformance

The Marathon-Ultra CRB uses the same materials as the Marathon and Marathon-5S control rods. No new material has been introduced. The new design absorber tubes are made from the same high purity stabilized type 304 stainless steel (Radiation Resist 304S) as the Marathon absorber tubes. Material testing and the service history of the Marathon control rod blades confirm the resistance to IASCC.

6.4 REACTIVITY

6.4.1 Criteria

The reactivity worth of the control rod shall be included in the plant core analyses.

6.4.2 Conformance

The compatibility of the Marathon-Ultra control rod is evaluated using the matched worth criterion approved in the Marathon control rod LTR (Reference 2); that is, replacement control rods whose initial reactivity worth is ± 5 % $\Delta k/k$ with respect to the original equipment do not need special treatment in plant core analyses. The nuclear design of the Marathon-Ultra control rod meets this criterion as discussed in Section 4. Therefore, Marathon-Ultra control rods can be used without change to current GEH lattice physics codes and design procedures.

6.5 SURVEILLANCE

6.5.1 Criteria

Prior to the use of new design features on a production basis, lead surveillance control rods may be used.

6.5.2 Conformance

Section 3.3 of Reference 1 Safety Evaluation requires a visual inspection program for the Marathon-5S control rod. The visual inspection program is designed to detect both (1) early-in-life failure mechanisms such as stress corrosion cracking and weld degradation, and (2) end-of-mechanical life predictions such as absorber tube failure. Since the outer structure of the Marathon-Ultra control rod is identical to the Marathon-5S, visual inspections performed for Marathon-5S apply equally to the Marathon-Ultra, satisfying the early-in-life inspection requirements. However, since the Marathon-Ultra has a longer nuclear lifetime than the Marathon-5S, the stainless steel structure of the Marathon-Ultra will achieve a higher irradiation at end of life than the Marathon-5S. Therefore, only an end-of-life surveillance should be required.

A comparison of ¼-segment nuclear lifetimes between Marathon-5S and Marathon-Ultra control rods is shown in Table 6-1. As shown, the Marathon-5S ¼-segment nuclear lifetime exceeds [[ ]] of the Marathon-Ultra nuclear lifetime.

As of the date of this report, a total of six (6) Marathon-5S control rods have been inserted in high duty locations at one domestic, and one international BWR. These six assemblies will be visually inspected after each two-year cycle. This exceeds the requirement in Section 3.3 of the Safety Evaluation in Reference 1 to visually inspect the two (2) lead use assemblies. For this surveillance program proposal, it is assumed that at least two of these Marathon-5S lead control rod assemblies will remain at a higher depletion than any Marathon-Ultra lead use assemblies until the Marathon-5S lead use assemblies have reached the end of their inspection campaign. In the unlikely event that lead Marathon-Ultra lead use assemblies pass the Marathon-5S lead use assemblies in terms of ¼-segment depletion, it is proposed that the Marathon-5S surveillance program described in Reference 1 be transferred to the Marathon-Ultra. Otherwise the following surveillance program is proposed.

- 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% of end-of-life).

- For refueling outages in which the depletion of the lead Marathon-Ultra assemblies are greater than 75% 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% 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% 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 tube and capsule 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% 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-5S control rods shall be performed during outages for which the depletion exceeds 90% 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 the 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.

- GEH will report to NRC the results of all Marathon-Ultra visual inspections at least annually.

- 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.

Table 6-1
Marathon-5S / Marathon-Ultra Nuclear Lifetime Comparison

Application	End of Life B-10 ¼-Segment Equivalent Depletion (%)		Ratio: Marathon-5S / Marathon-Ultra
	Marathon-5S	Marathon-Ultra	
D Lattice, BWR/2-4	[[		
C Lattice, BWR/4,5			
S Lattice, BWR/6			]]

7. EFFECT ON STANDARD PLANT TECHNICAL SPECIFICATIONS

The purpose and function of control rods are discussed in the Bases sections of the BWR/4 and BWR/6 Standard Technical Specifications (STS), References 4 and 5. Section B3.1.3, of both states:

"…*the CRD System provides the means for the reliable control of reactivity changes to ensure under conditions of normal operation, including anticipated operational occurrences, that specified fuel design limits are not exceeded. In addition, the control rods provide the capability to hold the reactor core subcritical under all conditions and to limit the potential amount and rate of reactivity increase caused by a malfunction in the CRD System*."

The nuclear worth characteristics of the Marathon-Ultra CRB are compatible with the core cold shutdown requirements and hot operational requirements of the original equipment control rods. This is achieved by meeting the matched worth criteria, described in the Marathon LTR (Reference 1), as a reactivity worth within ± 5 % Δk/k of the reactivity worth of the original equipment CRB. Therefore, the Marathon-Ultra CRB provides the means for the reliable control of reactivity changes to ensure that under conditions of normal operation, including AOOs, specified fuel design limits are not exceeded. Furthermore, the Marathon-Ultra CRB provides the capability to hold the reactor core subcritical under all conditions, while meeting current Technical Specification shutdown margin requirements. The overall Marathon-Ultra assembly weight and velocity limiter design will limit the amount and rate of reactivity increase caused by a malfunction of the CRD system, i.e.) a Control Rod Drop Accident (CRDA).

Therefore, there is no effect on the STS from introduction of the Marathon-Ultra control rod blade.

8. PLANT OPERATIONAL CHANGES

The fit, form and function of the Marathon-Ultra CRB are equivalent to the existing Duralife, Marathon, and Marathon-5S CRB designs. The Marathon-Ultra CRB meets all scram insertion criteria, reactivity control criteria, and CRDA.

No changes to the STS or their Bases (References 4 and 5) are needed. Therefore, it is expected that no plant-specific Technical Specifications (TS) or their Bases will require a change to implement the Marathon-Ultra control rod. Thus, no plant operating procedure change is expected, except for CRB replacement schedules. Therefore, the introduction of the Marathon-Ultra CRB has no effect on plant operations.

9. EFFECTS ON SAFETY ANALYSES AND DESIGN BASIS ANALYSIS MODELS

9.1 ANTICIPATED OPERATIONAL OCCURRENCES AND OTHER MALFUNCTIONS

As previously discussed, the reactivity worth of the Marathon-Ultra CRB is an equivalent replacement for previous control rod designs. Furthermore, the Marathon-Ultra CRB meets all scram time criteria. Therefore, use of the Marathon-Ultra CRB does not adversely affect the mitigating response function (i.e., scram) for AOOs.

Introduction of the Marathon-Ultra CRB is unrelated to the initiating events of the analyzed AOOs, and thus, the probabilities of the different AOOs occurring are unaffected.

Because the Marathon-Ultra CRB meets the existing design and licensing requirements for Marathon CRBs, the probability of any CRB-related malfunction or of causing a malfunction is not increased, and no new malfunction scenario is created.

The introduction of the Marathon-Ultra CRB does not (1) introduce a new failure mode or sequence of events that could result in the MCPR safety limit being challenged, (2) cause a 10 CFR 50.2 design bases criterion or limit to be changed or exceeded (such that a safety-related function is adversely affected), (3) create a possibility of a new safety-related component interaction. Therefore, the change does not create a possibility for a malfunction of equipment important to safety different than previously evaluated.

In the safety analyses, the equipment modeled or assumed to function for mitigating the radiological consequences of all design basis abnormal events is not affected by the use of Marathon-Ultra CRBs. Therefore, the analyzed consequences of the malfunctions in plant Safety Analysis Reports are not affected.

9.2 ACCIDENTS

The ECCS-LOCA performance, LOCA radiological, containment performance, and Main Steamline Break Accident (MSLBA) analyses all assume reactor scram within Technical Specifications requirements, and these are met by Marathon-Ultra CRBs. The Engineered Safety Feature (ESF) functions, which are modeled/assumed in the accident radiological consequence analyses, are also not affected by the use of Marathon-Ultra CRBs. Therefore, these analyses' models, scenarios, and the final radiological consequences are not affected.

The failures assumed in the initiating events for the LOCA and MSLBA are not related to the CRBs, and thus, the probabilities of these accidents occurring are not affected.

Other than the event evaluation assumption that the CRBs maintain structural integrity, the Fuel Handling Accident (FHA) initiating event and its related mitigation functions do not involve the CRBs. Therefore, the probability and consequences of a FHA are unaffected.

There is no additional friction between the Marathon-Ultra CRB relative to the Marathon CRB, and the CRD coupling mechanism is unchanged. Therefore, the probability of a stuck and decoupled control rod occurring does not change, and thus, the probability of a CRDA cannot significantly increase.

The reactivity insertion rate during a CRDA is controlled by the weight of the control rod and by the shape of the velocity limiter. The Marathon-Ultra CRB remains within all rod drop parameters assumed or modeled in the safety analysis. Therefore, the analysis and consequences of a CRDA are unchanged.

The change to Marathon-Ultra CRBs does not create a new fission product release path, result in a new fission product barrier failure mode, or create a new sequence of events that results in significant fuel cladding failures. Therefore, the use of Marathon-Ultra CRBs cannot create an accident of a different type.

9.3 SPECIAL EVENTS

The ATWS event assumes a failure to scram (without a specific cause) and that the Standby Liquid Control System is used for reactor shutdown. Therefore, the ATWS analysis scenario and results are independent of control rod blade design, and thus, the ATWS analysis is unaffected.

The station blackout, shutdown from outside control room, and safe shutdown fire analyses all assume reactor scram within TS requirements, which are not affected by the use of Marathon-Ultra CRBs. The other safe shutdown functions, which are modeled/assumed in the analyses, are also not related to or affected by the use of Marathon-Ultra CRBs. Therefore, these analyses' models, scenarios, and the final results are not affected.

9.4 FISSION PRODUCT BARRIER DESIGN BASIS LIMITS

During all design basis events, Marathon-Ultra CRB performance is equal to or better than existing CRBs. The margins to the thermal limits on fuel cladding, Minimum Critical Power Ration (MCPR) Safety Limit, Reactor Coolant Pressure Boundary stress limits (e.g., temperature and pressure), and containment structural stress limits are unaffected by the use of Marathon-Ultra CRBs. Therefore, the fission product barrier design basis limits are not affected.

9.5 SAFETY AND DESIGN BASIS ANALYSIS MODELS

Marathon-Ultra CRB implementation does not change any safety analysis input, model, or result. No design analysis methodology change is used or needed in the design of the Marathon-Ultra CRB. Therefore, this change does not involve a departure from a method of evaluation used in establishing a design basis or in a safety analysis

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; however, the design change process outlined in this section is not applicable to ABWR and ESBWR.

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 BWR/2 through BWR/6 Boiling Water Reactors (BWRs).

10.2 FIXED PARAMETERS

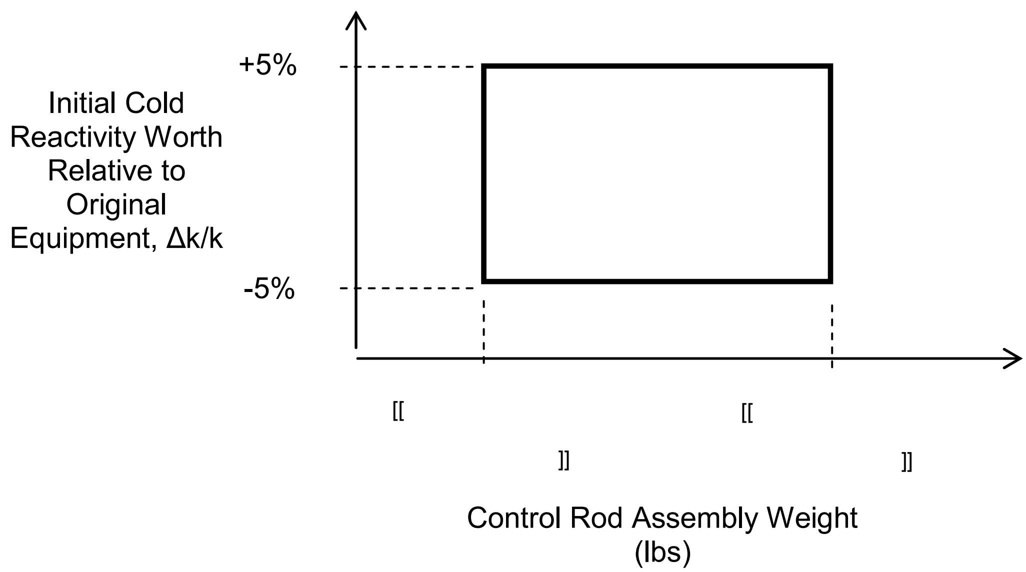
- The outer absorber tube geometry as defined in Table 2-1 shall not be changed.
- The outer absorber tube material, Type 304S, as defined in Table 2-1 and Section 3.2.4 of Reference 1 shall not be changed.
- Absorber materials may only consist of vibratory compacted boron carbide with naturally occurring 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.
- 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 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, [[

]]

- 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.

11. SUMMARY AND CONCLUSIONS

The Marathon-Ultra control rod blade is designed as an acceptable direct replacement control rod for BWR/2-6. Conservative mechanical evaluations show acceptability of the control rod structure. Conservative nuclear analyses show that the Marathon-Ultra is a 'matched worth' control rod and is interchangeable with the original equipment.

Operational evaluations show no adverse effect on plant operations, including control rod scram, 'no settle' characteristics, and control rod drop.

The Marathon-Ultra control rod, which is a derivative of the Marathon design, meets all licensing acceptance criteria of the Marathon and Marathon-5S designs (References 1 and 2).

The introduction of the Marathon-Ultra CRB does not affect the Standard Technical Specifications (References 4 and 5) or their Bases, any plant safety analysis, or any plant design basis. In addition, no adverse effect is found when examining safety analyses and design basis analysis models. The Marathon-Ultra CRB meets all applicable design and regulatory requirements. Therefore, the use of the Marathon-Ultra CRB is judged to be acceptable.

12. REFERENCES

1. GE Hitachi Nuclear Energy, "Marathon-5S Control Rod Assembly," NEDE-33284P-A Revision 2, October 2009 and NEDO-33284-A, Revision 2, October 2009.

2. GE Nuclear Energy, "GE Marathon Control Rod Assembly," NEDE-31758P-A, October 1991 and NEDO-31758-A, October 1991.

3. MNCP4A – A General Monte Carlo N-Particle Radiation Transport Code, Version 4A, March 1994.

4. USNRC, "Standard Technical Specifications General Electric Plants, BWR/4," NUREG-1433 Vol. 1 & 2, Revision 3.0, June 2004.

5. USNRC, "Standard Technical Specifications General Electric Plants, BWR/6," NUREG-1434 Vol. 1 & 2, Revision 3.0, June 2004.

6. 1989 ASME Section III, Division 1, Appendix I, Figure I-9.2.1.

7. JA Bannantine, JJ Comer and JL Handrock, "Fundamentals of Metal Fatigue Analysis," Prentice Hall, 1990.

8. BWRVIP-99, "Boiling Water Reactor Vessel and Internal Project: Crack Growth Rates in Irradiated Stainless Steels in BWR Internal Components," EPRI TR 1003018 Final Report, December 2001.

9. USNRC Information Notice 89-31, "Swelling and Cracking of Hafnium Control Rods," March 1989.

10. GE Hitachi Nuclear Energy, "GEH Marathon Control Rod Lifetime Surveillance Update," GENE-0000-0071-8269-R1-P, April 2009 and GENE-0000-0071-8269-R1-NP, April 2009.

11. GE Hitachi Nuclear Energy, "Marathon Control Rod Post-Irradiation Examination (PIE)," GENE-0000-0095-4020-R1-P, April 2009 and GENE-0000-0095-4020-R1-NP, April 2009.

APPENDIX A – FAILED BUFFER SCRAM STRESS EVALUATION

Failed buffer scram stress calculations for all cross-sections shown in Figures 3-1 and 3-2 are shown in Table 3-5 through 3-7. During a control rod scram, large axial loads are imparted on the control rod. These axial loads are determined using a dynamic spring and mass model, the results of which are presented in Table 3-4. For this analysis, the scram loads are determined assuming a 100% inoperative control rod drive buffer. The following cross-sections are analyzed.

A-1 SOCKET MINIMUM CROSS-SECTIONAL AREA (FIGURE 3-1)

The minimum cross-sectional area of the socket is calculated from the drawing to be [[ ]] Actual and allowable stress calculations are shown in Table A-1. As shown, all design ratios are less than 1.0. Therefore, the structure is acceptable.

A-2 VELOCITY LIMITER TRANSITION SOCKET TO FIN WELD (FIGURE 3-1)

The transition piece to fin welds are double fillet welds, joining the type 316 transition piece and fins, with ER 308L filler metal required.

For the calculation of the area of these welds, only the vertical portions of the welds are considered. The angled portions of the welds are conservatively neglected (Figure 3-1). Also, since the welds are in shear, the resulting area is multiplied by $(1/\sqrt{3})$ to calculate an equivalent normal area. The minimum equivalent normal weld area is calculated to be [[ ]]

Table A-2 shows the actual and allowable stresses for this weld. As shown, all design ratios are less than 1.0. Therefore, the weld is acceptable.

A-3 VELOCITY LIMITER FIN MINIMUM CROSS-SECTIONAL AREA (FIGURE 3-1)

The minimum cross-sectional area of the fins is calculated from the drawing to be [[ ]] Actual and allowable stress calculations are shown in Table A-3. As shown, all design ratios are less than 1.0. Therefore, the structure is acceptable.

A-4 VELOCITY LIMITER TO ABSORBER SECTION WELD (FIGURE 3-2)

The weld connecting the absorber section to the velocity limiter is analyzed using the combined loading of the scram loads and axial loads due to the maximum allowable internal pressure of the absorber tubes.

Since both the scram loads and the load due to the internal pressure of the absorber tubes is considered, a combined weld area of the absorber section to handle weld, and the end plug to absorber tube weld is calculated. Since the end plug weld is in shear for this loading, the weld area is multiplied by $(1/\sqrt{3})$ to calculate an effective normal weld area. This is added to the minimum absorber section to velocity limiter weld area, which is determined using CAD software:

$$A_{normal} = (\text{\# Pressurized Tubes})(1\sqrt{3})(\pi)OD_{plug,min}(\text{weld penetration})$$
$$+ (\text{\# Tubes})(\text{absorber section to handle/VL area per tube}).$$

The weld area per tube is then multiplied by the number of tubes. The weld area calculation is summarized in Table A-4.

Once the effective normal weld area is known, the combined maximum stresses due to scram and internal pressure are calculated as described in Table A-5. As shown, all design ratios are less than 1.0. Therefore, the weld is acceptable.

A-5 ABSORBER SECTION (FIGURE 3-2)

The minimum cross-sectional area of the absorber section is calculated in Table A-6. Actual and allowable stresses are shown in Table A-7. As shown, all design ratios are less than 1.0. Therefore, the structure is acceptable.

A-6 ABSORBER SECTION TO HANDLE WELD (FIGURE 3-2)

The weld connecting the absorber section to the handle is analyzed using the combined loading of the scram loads and axial loads due to the maximum allowable internal pressure of the absorber tubes.

Since both the scram loads and the load due to the internal pressure of the absorber tubes is considered, a combined weld area of the absorber section to handle weld, and the end plug to absorber tube weld is calculated. Since the end plug weld is in shear for this loading, the weld area is multiplied by $(1/\sqrt{3})$ to calculate an effective normal weld area. This is added to the minimum absorber section to handle weld area, which is determined using CAD software:

$$A_{normal} = (\#\text{ Pressurized Tubes})(1\sqrt{3})(\pi)OD_{plug,min}(\text{weld penetration}) \\ + (\#\text{ Tubes})(\text{absorber section to handle/VL area per tube}).$$

The weld area per tube is then multiplied by the number of tubes. The weld area calculation is summarized in Table A-8. Once the effective normal weld area is known, the combined maximum stresses due to scram and internal pressure are calculated as described in Table A-9. As shown, all design ratios are less than 1.0. Therefore, the structure is acceptable.

A-7 HANDLE MINIMUM CROSS-SECTIONAL AREA (FIGURE 3-2)

The minimum cross-sectional areas of the handle, and actual and allowable stresses, are shown in the Table A-10. As shown, all design ratios are less than 1.0. Therefore, the structure is acceptable.

Table A-1 Socket Axial Stress Calculations

Description	Source	D Lattice		C Lattice		S Lattice	
		70 °F	550 °F	70 °F	550 °F	70 °F	550 °F
Max Failed Buffer Scram Load (kips)	Table 3-4	[[					
Max Failed Buffer Scram Stress (ksi)	=P/1.919 in^2						
Allowable Stress (ksi)	Table 3-2 (XM-19)						
Design Ratio	=stress/allow						]]

Table A-2 Transition Socket to Fin Weld Stress Calculations

Description	Source	D Lattice		C Lattice		S Lattice	
		70 °F	550 °F	70 °F	550 °F	70 °F	550 °F
Max Failed Buffer Scram Load (kips)	Table 3-4	[[					
Max Failed Buffer Scram Stress (ksi)	=P/A						
Allowable Stress (ksi)	Table 3-2 (ER 308L)						
Weld Quality Factor	Table 3-3						
Allowable Weld Stress (ksi)	=S_m*q						
Design Ratio	=stress/Allow						]]

Table A-3 Minimum Fin Area Stress Calculations

Description	Source	D Lattice		C Lattice		S Lattice	
		70 °F	550 °F	70 °F	550 °F	70 °F	550 °F
Max Failed Buffer Scram Load (kips)	Table 3-4	[[					
Max Failed Buffer Scram Stress (ksi)	=P/A						
Allowable Stress (ksi)	Table 3-2 (316 plate)						
Design Ratio	=stress/allow						]]

Table A-4 Velocity Limiter to Absorber Section Weld Geometry

Description	Reference	D Lattice	C Lattice	S Lattice
Absorber Tube to VL Weld Area (in^2)	CAD analysis	[[		
Min End Plug OD (in)	Drawing			
Max End Plug OD (in)	Drawing			
Min End Plug Weld Penetration (in)	Assembly Drawing			
Number of Absorber Tubes per Assembly	Assembly Drawing			
Number of Pressurized Absorber Tubes per Assembly	Assembly Drawing			
Total Weld Area (in^2)	Equation in Section A-4			]]

Table A-5 Velocity Limiter to Absorber Section Weld Stress Calculations

Description	Source	D Lattice		C Lattice		S Lattice	
		70 °F	550 °F	70 °F	550 °F	70 °F	550 °F
Max Failed Buffer Scram Load (kips)	Table 3-4	[[					
Maximum Allowable Internal Pressure (ksi)	Finite Element Analysis						
End Plug Pressure Area (in^2)	$=\pi/4*(OD_{plug})^2$						
Number of Pressurized Tubes	Assembly Drawing						
Total Axial Load (kips)	=Scram Load + (press)(area)(# tubes)						
Total Weld Area (in^2)	Table A-4						
Max Failed Buffer Scram + Internal Pressure Stress (ksi)	$=P_{tot}/A$						
Allowable Stress (ksi)	Table 3-2 (304S Tubes)						
Weld Quality Factor	Table 3-3						
Allowable Weld Stress (ksi)	$=S_m*q$						
Design Ratio	=Stress/Allow						]]

Table A-6 Absorber Section Geometry Calculation

Description	Source	D Lattice	C Lattice	S Lattice
Min Absorber Tube Area (in^2)	CAD Analysis	[[		
Min Tie Rod Area (in^2)	CAD Analysis			
Number of Absorber Tubes	Assembly Drawing			
Total Minimum Absorber Section Cross-sectional Area (in^2)	=(# tubes)(tube area) + tie rod area			]]

Table A-7 Absorber Section Stress Calculation

Description	Source	D Lattice		C Lattice		S Lattice	
		70 °F	550 °F	70 °F	550 °F	70 °F	550 °F
Max Failed Buffer Scram Load (kips)	Table 3-4	[[					
Max Failed Buffer Scram Stress (ksi)	=P/A						
Allowable Stress (ksi)	Table 3-2 (304S Tubes)						
Design Ratio	=stress/allow						]]

Table A-8 Absorber Section to Handle Weld Area Calculation

Description	Source	D Lattice	C Lattice	S Lattice
Absorber Tube to Handle Weld Area (in^2)	From CAD analysis	[[		
Min End Plug OD (in)	From drawing			
Max End Plug OD (in)	From drawing			
Min End Plug Weld Penetration (in)	From assembly drawing			
Number of Absorber Tubes per Assembly	Assembly Drawing			
Number of Pressurized Absorber Tubes per Assembly	Assembly Drawing			
Total Weld Area (in^2)	=(# tubes)(area)			]]

Table A-9 Absorber Section to Handle Weld Stress Calculations

Description	Source	D Lattice		C Lattice		S Lattice	
		70 °F	550 °F	70 °F	550 °F	70 °F	550 °F
Max Failed Buffer Scram Load (kips)	Table 3-4	[[					
Maximum Allowable Internal Pressure (ksi)	Finite Element Analysis						
End Plug Pressure Area (in^2)	$=\pi/4*(OD_{plug})^2$						
Number of Pressurized Tubes	From assembly drawing						
Total Axial Load (kips)	=Scram Load + (press)(area)(# tubes)						
Total Weld Area (in^2)	Table A-8						
Max Failed Buffer Scram + Internal Pressure Stress (ksi)	$=P_{tot}/A$						
Allowable Stress (ksi)	Table 3-2 (304S Tubes)						
Weld Quality Factor	Table 3-3						
Allowable Weld Stress (ksi)	$=S_m*q$						
Design Ratio	=Stress/Allow						]]

Table A-10 Handle Scram Stress Calculations

Description	Reference	D Lattice		C Lattice		S Lattice	
		70 °F	550 °F	70 °F	550 °F	70 °F	550 °F
Max Failed Buffer Scram Load (kips)	Table 3-4	[[					
Max Failed Buffer Scram Stress (ksi)	=P/A						
Allowable Stress (ksi)	Table 3-2 (316 plate)						
Design Ratio	=stress/allow						]]

Appendix B

GEH Responses to NRC RAIs

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.

[[

]]

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

[[

]]

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.

[[

]]

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.

GEH Response:

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.

[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.

[[

]]

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.

[[

]]

<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?

<u>GEH Response</u>:

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.

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

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.

GEH Response:

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:

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

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

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

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

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

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

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.

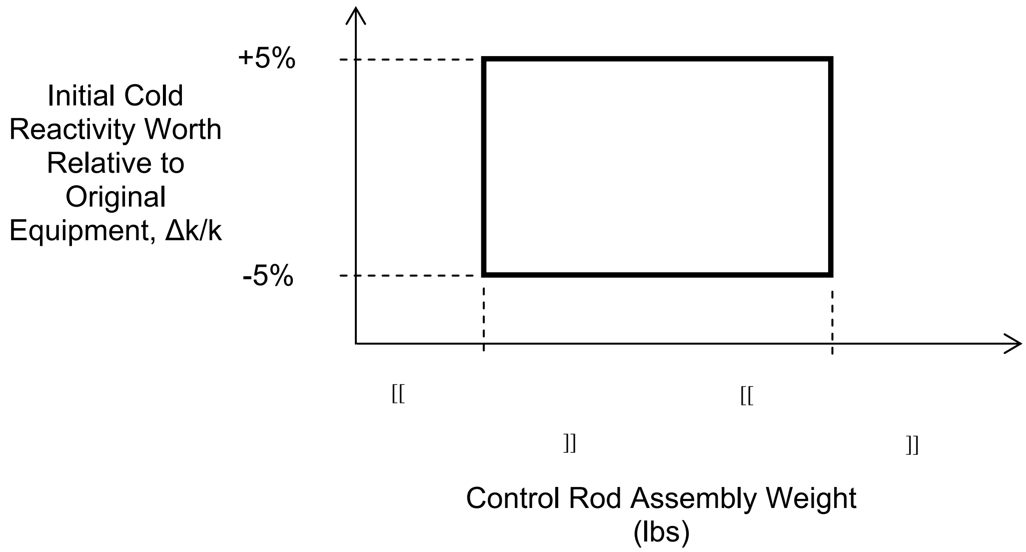
GEH Response:

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.
    - 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.
    - Fixed Parameters:
        - 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.
- <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.



- Methodologies:
  - 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.
- Acceptance Criteria:
  - 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, [[

]]
- 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 B-25.

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)?

GEH Response:

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			]]

Changes to NEDE-33284P Supplement 1 Revision 0:

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

[[

]]

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.

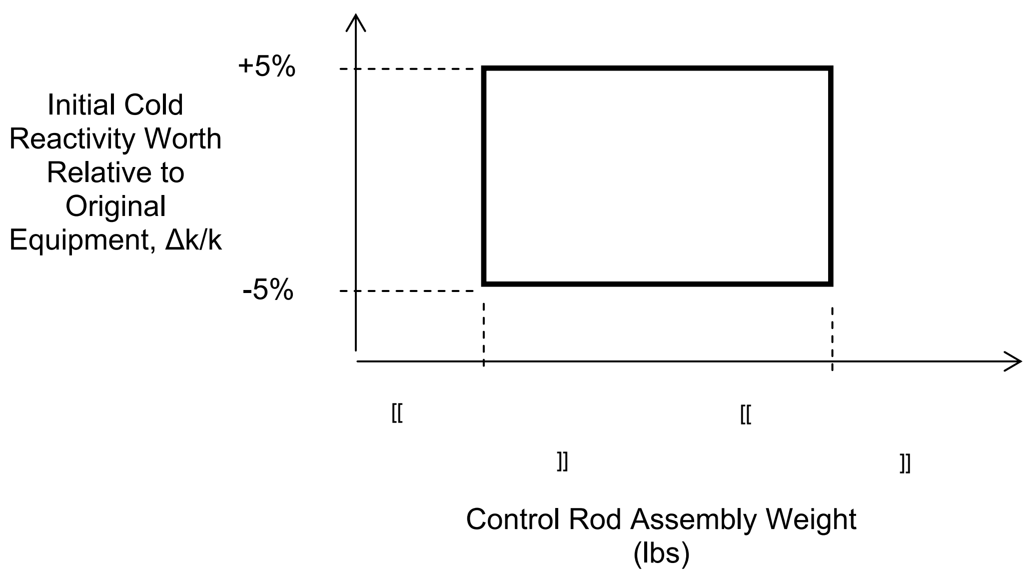
10.2 Fixed Parameters

- 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, [[

  ]]
- 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.

Appendix C

Correspondence Provided to NRC Containing Supplemental Information for Staff Independent Calculations

HITACHI

GE Hitachi Nuclear Energy

Proprietary Notice
This letter transmits proprietary information in accordance with 10CFR2.390. Upon the removal of Enclosure 1, the balance of the letter may be considered non-proprietary.

James F. Harrison

GE-Hitachi Nuclear Energy Americas LLC
Vice President, Fuel Licensing, Regulatory Affairs
P.O. Box 780, M/C A-55
Wilmington, NC 28401 USA

T 910.819.6604
james.harrison@ge.com

MFN 11-043

March 4, 2011

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555-0001

Subject: Information Request Regarding the NRC Staff Review of NEDE-33284P, Supplement 1, "Marathon-Ultra Control Rod Assembly" (TAC No. ME3524)

In response to the NRC Staff's e-mail request, this letter transmits input information that can be used for Staff independent calculations regarding the subject submittal. The CD-ROM included as Enclosure 1 contains the requested information.

Please note that Enclosure 1 contains proprietary information of the type that GEH maintains in confidence and withholds from public disclosure. The information has been handled and classified as proprietary to GEH as indicated in its affidavit. The affidavit contained in Enclosure 2 identifies that the information contained in Enclosure 1 has been handled and classified as proprietary to GEH. GEH hereby requests that the information in Enclosure 1 be withheld from public disclosure in accordance with the provisions of 10 CFR 2.390 and 9.17.

Enclosure 1 contains detailed fuel assembly information and modeling input which is deemed proprietary in its entirety. Thus a non-proprietary version of this enclosure has not been provided in accordance with NRC Information Notice 2009-07, Requirements for Submittals, (2), which states: "In instances in which a nonproprietary version would be of no value to the public because of the extent of the proprietary information, the agency does not expect a nonproprietary version to be submitted."

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

Sincerely,

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

Project No. 710

Enclosures:

1. Input Information – CD-ROM – GEH Proprietary Information – Class III (Confidential)
2. Affidavit

cc: SS Philpott, NRC
JG Head, GEH Wilmington
PL Campbell, GEH Washington
AA Lingenfelter, GNF Wilmington
eDRF Section 0000-0130-4552

HITACHI

GE Hitachi Nuclear Energy

James F. Harrison

GE-Hitachi Nuclear Energy Americas LLC
Vice President, Fuel Licensing, Regulatory Affairs
P.O. Box 780, M/C A-55
Wilmington, NC 28401 USA

T 910.819.6604
james.harrison@ge.com

Proprietary Notice
This letter transmits proprietary information in accordance with 10CFR2.390. Upon the removal of Enclosure 1, the balance of the letter may be considered non-proprietary.

MFN 11-133
March 28, 2011

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555-0001

Subject: Finite Element Analysis Information Request Regarding the NRC Staff Review of NEDE-33284P, Supplement 1, "Marathon-Ultra Control Rod Assembly" (TAC No. ME3524)

In response to the NRC Staff's e-mail request, this letter transmits input information that can be used for Staff independent calculations regarding the subject submittal. The CD-ROM included as Enclosure 1 contains the requested information.

Please note that Enclosure 1 contains proprietary information of the type that GEH maintains in confidence and withholds from public disclosure. The information has been handled and classified as proprietary to GEH as indicated in its affidavit. The affidavit contained in Enclosure 2 identifies that the information contained in Enclosure 1 has been handled and classified as proprietary to GEH. GEH hereby requests that the information in Enclosure 1 be withheld from public disclosure in accordance with the provisions of 10 CFR 2.390 and 9.17.

Enclosure 1 contains detailed control blade information and modeling input which is deemed proprietary in its entirety. Thus a non-proprietary version of this enclosure has not been provided in accordance with NRC Information Notice 2009-07, Requirements for Submittals, (2), which states: "In instances in which a nonproprietary version would be of no value to the public because of the extent of the proprietary information, the agency does not expect a nonproprietary version to be submitted."

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

Sincerely,

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

Project No. 710

Enclosures:

1. Marathon Ultra Control Rod Finite Element Analysis Input Information – CD-ROM – GEH Proprietary Information – Class III (Confidential)
2. Affidavit

cc: SS Philpott, NRC
JG Head, GEH Wilmington
PL Campbell, GEH Washington
AA Lingenfelter, GNF Wilmington
DRF Section 0000-0131-1693 R0