MITSUBISHI HEAVY INDUSTRIES, LTD.
16-5, KONAN 2-CHOME, MINATO-KU
TOKYO, JAPAN

April 2, 2010

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

Attention: Mr. Jeffrey A. Ciocco

Docket No. 52-021
MHI Ref: UAP-HF- 10093

Subject: MHI's Response to US-APWR DCD RAI No. 545-4290 Rev. 1

References: 1) "Request for Additional Information No. 545-4290 Revision 2, SRP Section: 04.03 – Nuclear Design, Application Section: 4.3," dated March 3, 2010

With this letter, Mitsubishi Heavy Industries, Ltd. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") a document entitled "Response to Request for Additional Information No.524-4020 Revision 1".

Enclosed are the responses to questions 50, 54 through 56, 58 through 61, 63 and 64 of the RAI contained within Reference 1. The responses to questions 48, 49, 51 through 53, 57, 62, 65 and 66 will be issued at a later date (ie-60 days) by a separate transmittal.

As indicated in the enclosed materials, this document (Enclosure 2) contains information that MHI considers proprietary, and therefore should be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4) as trade secrets and commercial or financial information which is privileged or confidential. Non-proprietary versions of the documents are also being submitted in this package (Enclosure 3). In the non-proprietary versions, the proprietary information, bracketed in the proprietary versions, is replaced by the designation "[ ]".

This letter includes a copy of the proprietary version (Enclosure 2), a copy of the non-proprietary version (Enclosure 3) and the Affidavit of Yoshiki Ogata (Enclosure 1) which identifies the reasons MHI respectfully requests that all materials designated as "Proprietary" in Enclosure 2 be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4).

Please contact Dr. C. Keith Paulson, Senior Technical Manager, Mitsubishi Nuclear Energy Systems, Inc. if the NRC has questions concerning any aspect of the submittal. His contact information is below.

Sincerely,

Y. Ogata

Yoshiki Ogata
General Manager- APWR Promoting Department
Mitsubishi Heavy Industries, LTD.

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NRO

Enclosure:

1. Affidavit of Yoshiki Ogata

2. Response to Request for Additional Information No.545-4290 Revision 2 (Proprietary version)

3. Response to Request for Additional Information No.545-4290 Revision 2 (Non-proprietary version)

CC: J. A. Ciocco
C. K. Paulson

Contact Information
C. Keith Paulson, Senior Technical Manager
Mitsubishi Nuclear Energy Systems, Inc.
300 Oxford Drive, Suite 301
Monroeville, PA 15146
E-mail: ck_paulson@mnes-us.com
Telephone: (412) 373-6466

Enclosure 1

Docket No.52-021
MHI Ref: UAP-HF-10093

MITSUBISHI HEAVY INDUSTRIES, LTD.

AFFIDAVIT

I, Yoshiki Ogata, being duly sworn according to law, depose and state as follows:

1. I am General Manager, APWR Promoting Department, of Mitsubishi Heavy Industries, Ltd ("MHI"), and have been delegated the function of reviewing MHI's US-APWR documentation to determine whether it contains information that should be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4) as trade secrets and commercial or financial information which is privileged or confidential.

2. In accordance with my responsibilities, I have reviewed the enclosed "Response to Request for Additional Information No.545-4290 Rev.2" and have determined that the document contains proprietary information that should be withheld from public disclosure. Those pages containing proprietary information are identified with the label "Proprietary" on the top of the page and the proprietary information has been bracketed with an open and closed bracket as shown here "[ ]". The first page of the document indicates that all information identified as "Proprietary" should be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4).

3. The information in the document identified as proprietary by MHI has in the past been, and will continue to be, held in confidence by MHI and its disclosure outside the company is limited to regulatory bodies, customers and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and is always subject to suitable measures to protect it from unauthorized use or disclosure.

4. The basis for holding the referenced information confidential is that it describes the unique methodology for evaluation to comply with Requlatory Guide 1.190, developed by MHI. This methodology was developed to significant cost to MHI, and with knowledge and know-how about using the DORT code.

5. The referenced information is being furnished to the Nuclear Regulatory Commission ("NRC") in confidence and solely for the purpose of supporting the NRC staff's review of MHI's Application for certification of its US-APWR Standard Plant Design.

6. Public disclosure of the referenced information would assist competitors of MHI in their design of new nuclear power plants without the costs or risks associated with the design of new fuel systems and components. Disclosure of the information identified as proprietary would therefore have negative impacts on the competitive position of MHI in the U.S. nuclear plant market.

I declare under penalty of perjury that the foregoing affidavit and the matters stated therein are true and correct to the best of my knowledge, information and belief.

Executed on this 2th day of April, 2010.

Y. Ogata

Yoshiki Ogata,
General Manager- APWR Promoting Department
Mitsubishi Heavy Industries, LTD.

Enclosure 3

UAP-HF-10093
Docket No. 52-021

Response to Request for Additional Information
No. 545-4290 Revision 2

April, 2010
(Non Proprietary)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

4/2/2010

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 545-4290 REVISION 2**
SRP SECTION:	**04.03 – Nuclear Design**
APPLICATION SECTION:	**4.3**
DATE OF RAI ISSUE:	**3/3/2010**

QUESTION NO.: 04.03-50

As stated in Regulatory Guide 1.190, comparison of fluence predictions with: (a) fluence benchmark calculations (b) operating reactor fluence measurements and (c) vessel fluence simulators should be used for the qualification of the vessel fluence methodology.
Because of the substantial uncertainties in fluence calculations and measurements, the single operating reactor calculation-to-measurement comparison for the H. B. Robinson-2 Cycle-9 fluence measurement given in Section-A.1 of MUAP-09018 together with the comparison for the VENUS-1 vessel simulator experiment (with scaled-down and simplified geometry) given in Section-A.1.3 do not provide a reliable estimate of the calculational bias. Therefore provide additional calculation/benchmark comparisons as qualification for the MUAP methodology.

ANSWER:
NUREG/CR-6115, "PWR and BWR Pressure Vessel Fluence Calculation Benchmark Problems and Solutions" provides detailed specifications and corresponding numerical solutions for a set of PWR and BWR pressure vessel fluence benchmark problems. In accordance with the recommendation of Regulatory Guide 1.190, Mitsubishi performed a comparison between the fluence predictions of their calculation methodology and the benchmark solutions provided in NUREG/CR-6115. The outline and results of the benchmark problem analysis are shown below. The low-leakage core loading problem was chosen from among the PWR benchmark problems described in NUREG/CR-6115 since the US-APWR adopts the low-leakage core loading pattern.

The neutron transport calculation was carried out using the DORT code. The calculation was performed using the cylindrical coordinates (r, θ) geometric model. The geometry is shown in Figure 04.03-50.1. The calculation was performed using nuclear data from the BUGLE-96 cross-section library [ ] The cycle averaged pin-wise power distribution given in NUREG/CR-6115 was used as the source distribution. The cycle averaged assembly-wise power distribution is shown in Figure 04.03-50.2. [ ] The calculation conditions are summarized in Table 04.03-50.1.

Figures 4.03-50.3 through 4.03-50.5 compare the results of the calculated neutron flux (E>1 MeV,

E>0.1 MeV) and DPAs at the inner surface of the reactor vessel with the data described in NUREG/CR-6115. The DPA cross sections described in NUREG/CR-6115 are used to calculate DPAs to facilitate comparison of the DPA results with those of NUREG/CR-6115.
Table 04.03-50.2 compares the results of the calculated reaction rates in the surveillance capsules with the data described in NUREG/CR-6115. Dosimeter cross sections described in NUREG/CR-6115 were used to calculate the reaction rates to facilitate comparison of the calculated reaction rate results with those of NUREG/CR-6115.

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] The good agreement between the Mitsubishi fluence evaluation methodology and the benchmark problem demonstrates additional qualification of the MHI vessel fluence methodology, as recommended by the reviewer.

Impact on DCD

There is no impact on the DCD.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

Table 04.03-50.1 Input Conditions for the Benchmark Analysis of NUREG/CR-6115

Term	Contents	Information
Benchmark problem	NUREG/CR-6115	
Computer code	DORT	DOORS-3.2
Cross section library Activation cross section	BUGLE-96	[ ]
Material composition	Based on NUREG/CR-6115	NUREG/CR-6115
Pℓ, Sn	[ ]	-
Geometry	(r, θ) : Figure 04.03-50.1	NUREG/CR-6115
Source distribution	Radial distribution : Figure 04.03-50.2	NUREG/CR-6115
Fission rate		
Fission spectra	Table 4 of MUAP-09018	
Neutron production rate per fission		
Core thermal power	2527.73 MWt	NUREG/CR-6115
Boundary conditions	(r, θ) Left, Top, Bottom : reflection Right : vacuum	
Convergence criteria	≤0.001 (≤0.1%)	

Table 04.03-50.2 Surveillance Capsule Reaction Rate Results

Figure 04.03-50.1 Core Center Cross Section ((r, θ) Geometry) of NUREG/CR-6115

1.09							
1.20	1.21						
1.24	1.04	1.05					
1.13	1.29	1.10	1.24				
1.25	1.07	1.29	1.27	0.77			
1.28	1.31	1.39	1.13	0.48			
1.23	1.22	1.06	0.53	0.17			
0.35	0.45	0.36					

Figure 04.03-50.2 Relative Cycle Averaged Assembly-wise Power Distribution for Fuel Assemblies for the NUREG/CR-6115 Benchmark Analysis

Figure 04.03-50.3 Azimuthal Distribution of the Neutron Flux at the Inner Surface of RV(E>1 MeV)

Figure 04.03-50.4 Azimuthal Distribution of the Neutron Flux at the Inner Surface of RV(E>0.1 MeV)

Figure 04.03-50.5 Azimuthal Distribution of DPAs at the Inner Surface of RV

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SRP SECTION:	**04.03 – Nuclear Design**
APPLICATION SECTION:	**4.3**
DATE OF RAI ISSUE:	**3/3/2010**

QUESTION NO.: 04.03-54

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References
2. H. Soodak, Reactor Handbook Second Edition, Vol. III Part A, "Physics," 1962

ANSWER:

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Impact on DCD

There is no impact on the DCD.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

4/2/2010

US-APWR Design Certification

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QUESTION NO.: 04.03-55

Technical Report MUAP-09018P provides documentation of the design calculations for the pressure vessel fluence. However, there is a significant inconsistency in the application of the methodology used to represent the cartesian boundary of the core. The modeling of the core boundary is important since the vessel fluence is highly sensitive to the location of the core boundary which determines the separation-distance between the core and the vessel.

Discuss the modeling of the core boundary in the DORT calculations (Section 2.3, p. 4, ¶-3). Specifically, explain the core-boundary differences between the Figure-3 (r, θ) geometry (p. 21) and the Figure-4 (x, y) geometry (p. 22). If this inconsistency affects the calculations provide the effect on the vessel fluence predictions

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ANSWER:

Geometry is chosen from the viewpoint of whether the evaluated parts are modeled more accurately in the neutron fluence evaluation of the US-APWR. For evaluation of the reactor vessel, cylindrical (r, θ) geometry is adopted to precisely model the cylindrical configuration of a reactor vessel. For evaluation of the core boundary, rectangular (x, y) geometry is adopted to precisely model the linear configuration (the cartesian boundary) of the core. When the reactor core boundary is modeled by cylindrical (r, θ) geometry, it is necessary to use the appropriate (fine) mesh, in order to closely approximate the (x, y) geometry case.

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Impact on DCD

There is no impact on the DCD.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

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Docket No. 52-021

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APPLICATION SECTION:	**4.3**
DATE OF RAI ISSUE:	**3/3/2010**

QUESTION NO.: 04.03-56

The former-plates between the baffle and core-barrel of Figure-A-2 of MUAP-09018 can have a significant effect on the vessel fluence prediction. Provide an axial and radial description of the former plates (if they are present in the US-APWR) and any other structures/supports located in the region between the core and the vessel. Have the former-plates and all supports/structures between the core and vessel been included in the calculational models? If not, update the models to include these effects and provide the resulting changes in the calculated vessel fluence. The uncertainty and bias introduced by any approximations in representing these structures should also be determined and incorporated in the fluence analysis.

ANSWER:

The design of the US-APWR core internals incorporates the neutron reflector but not the former-plates. As shown in Figure 04.03-56.1, the design includes tie rod and alignment pins that are modeled as the same stainless steel as the neutron reflector.

The surveillance capsules located outside the core barrel are not modeled. This has no effect on the maximum fluence since the installation position of the surveillance capsules is not aligned with the direction (45°) which gives the maximum fluence on the vessel, as shown in Figure 04.03-56.2 (US-APWR DCD Figure 5.3-1).

Impact on DCD

There is no impact on the DCD.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

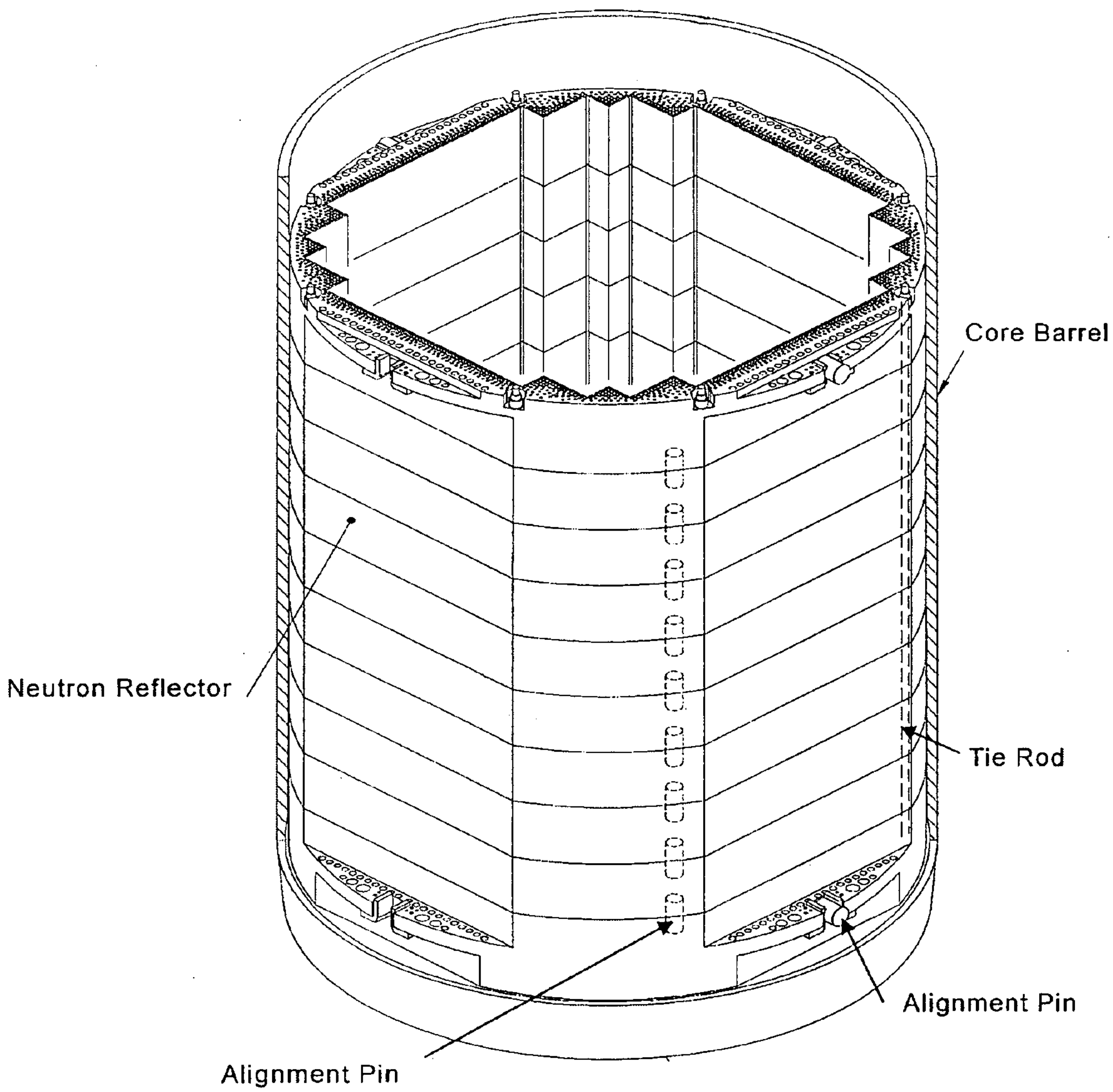

Figure 04.03-56.1 Structure of Neutron Reflector and Core Barrel

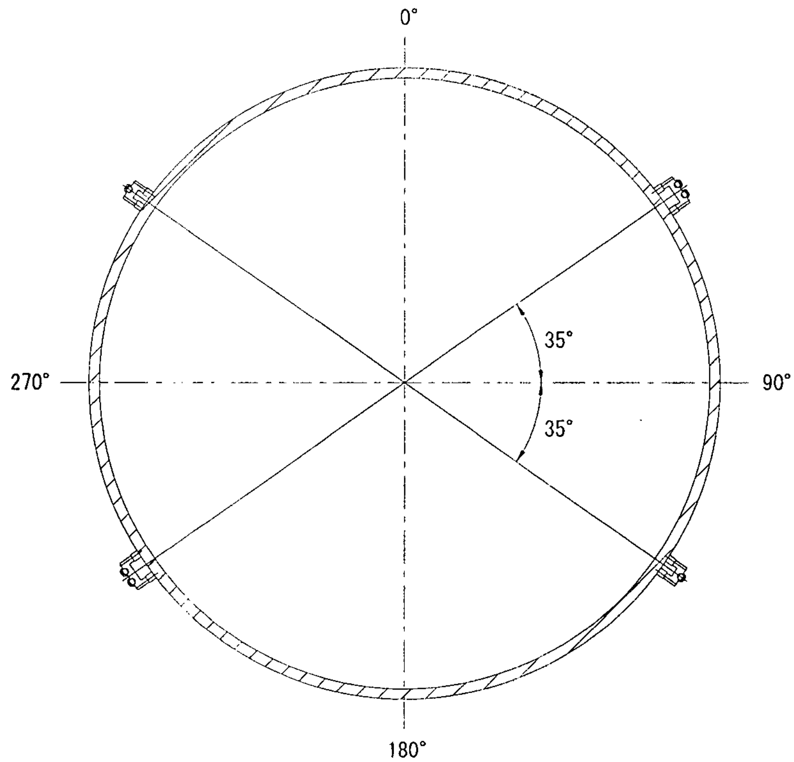

Figure 04.03-56.2 Orientation of Surveillance Capsules

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APPLICATION SECTION: **4.3**

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QUESTION NO.: 04.03-58

The pressure vessel surveillance capsule introduces a significant perturbation to the local neutron flux. The capsule fluence measurement is applied as a direct multiplier to the calculated fluence when a bias is determined. It is therefore important to account for the measurement perturbation by accurately modeling the surveillance capsule in the DORT calculation.

Describe in detail the modeling of the measurement capsule used in the Section-A.1 DORT calculation of the H. B. Robinson-2 Cycle-9 fluence measurement (MUAP-09018 Section A.1.1, p. A-1, ¶-2). Identify all approximations (e.g., geometric distortion and spatial mesh) and determine the uncertainty and bias introduced in the capsule fluence prediction.

ANSWER:

The surveillance capsule and the guide tube of the DORT calculation of the H.B.Robinson-2 fluence are modeled using cylindrical (r, θ) geometry as shown in Figure 04.03-58.1. The fine mesh intervals indicated in the figure are used in order to model the actual dimension of the surveillance capsule and guide tube. Therefore, there is no approximation, and the uncertainty and bias are not considered.

Impact on DCD

There is no impact on the DCD.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

Figure 04.03-58.1 DORT Mesh of Surveillance Capsule and Guide Tube ((r, θ) geometry)

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DATE OF RAI ISSUE: 3/3/2010

QUESTION NO.: 04.03-59

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ANSWER:

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Impact on DCD

There is no impact on the DCD.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

4/2/2010

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SRP SECTION: 04.03 – Nuclear Design

APPLICATION SECTION: 4.3

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QUESTION NO.: 04.03-60

Fluence benchmark measurements are frequently standardized using an equivalent fission flux. However, there are several definitions of the equivalent fission-flux that are used. In order to allow evaluation of the benchmark comparisons, describe the definition of equivalent fission-flux used in determining the Venus-1 M/C comparisons (MUAP- 09018 Section A.1.3, p. A-2, ¶-8) and how the resulting M/Cs compare with those determined based on reaction rates. What dosimeter cross sections were used in the Venus-1 benchmark calculations?

ANSWER:

The definition of equivalent fission flux in the benchmark calculation of Venus-1 is shown below.

$$equivalent\ fission\ flux = \frac{\sum_{i=1}^{47} \phi_i \times \sigma_i}{\sum_{i=1}^{47} f_i \times \sigma_i}$$

ϕ_i : neutron flux (n/cm^2/s)

σ_i : dosimeter cross section (barn)

f_i : fission spectrum

The equivalent fission flux by measurement and the equivalent fission flux by calculation have the same definition shown by the above equation.

BUGLE-96 dosimeter cross sections are used for the analysis, except for Rh-103. The IRDF-2002 cross section is used for Rh-103.

Impact on DCD

There is no impact on the DCD.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

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QUESTION NO.: 04.03-61

Pressure vessel simulator measurement configurations are typically much smaller than operating reactors both radially and axially. This is an important distinction since the core leakage, which determines the vessel fluence, is determined by the core dimensions. As a result, determining the axial leakage is a major difficulty in calculating vessel fluence for a pressure vessel simulator.

Describe the method and the effect of any approximations used to include the axial effects in the Venus-1 benchmark calculations (MUAP-09018 Section A.1.3, p. A-2, ¶-2). How does this compare with the method used for operating reactors?

ANSWER:

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Impact on DCD

There is no impact on the DCD.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

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APPLICATION SECTION:	**4.3**
DATE OF RAI ISSUE:	**3/3/2010**

QUESTION NO.: 04.03-63

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ANSWER:

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Impact on DCD

There is no impact on the DCD.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

4/2/2010

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.: NO. 545-4290 REVISION 2

SRP SECTION: 04.03 – Nuclear Design

APPLICATION SECTION: 4.3

DATE OF RAI ISSUE: 3/3/2010

QUESTION NO.: 04.03-64

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ANSWER:

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Impact on DCD

There is no impact on the DCD.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.