MITSUBISHI HEAVY INDUSTRIES, LTD.
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TOKYO, JAPAN

June 25, 2009

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-09336

Subject: **MHI's Responses to US-APWR DCD RAI No. 377-2629 Revision 1**

Reference: 1) "Request for Additional Information No. 377-2629 Revision 1, SRP Section: 04.04 – Thermal and Hydraulic Design, Application Section: 4.4.2.5", dated May 29, 2009

With this letter, Mitsubishi Heavy Industries, Ltd. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") a document entitled "MHI's Responses to RAI No. 377-2629 Revision 1".

Enclosed are the responses to 6 RAIs contained within Reference 1.

As indicated in the enclosed materials, this document 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. A non-proprietary version of the document is also being submitted with the information identified as proprietary redacted and replaced by the designation "[ ]".

This letter includes a copy of the proprietary version (Enclosure 2), a copy of 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 submittals. His contact information is below.

Sincerely,

Y. Ogata

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

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NRO

Enclosures:

1. Affidavit of Yoshiki Ogata

2. MHI's Responses to RAI No. 377-2629 Revision 1 (proprietary version)

3. MHI's Responses to RAI No. 377-2629 Revision 1 (non-proprietary version)

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

Contact Information
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Telephone: (412) 373 – 6466

ENCLOSURE 1

Docket No.52-021
MHI Ref: UAP-HF-09336

MITSUBISHI HEAVY INDUSTRIES, LTD.

AFFIDAVIT

I, Yoshiki Ogata, 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 document entitled "MHI's Responses to RAI No. 377-2629 Revision 1" and have determined that portions of the document contain 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 identified as proprietary in the enclosed document 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 thermal and hydraulic design developed by MHI and not being used in the exact form by any MHI's competitors. This information was developed at significant cost to MHI, since it required the performance of research and development and detailed design for its software and hardware extending over several years.

5. The referenced information is being furnished to the Nuclear Regulatory Commission ("NRC") in confidence and solely for the purpose of information to the NRC staff.

6. The referenced information is not available in public sources and could not be gathered readily from other publicly available information. Other than through the provisions in paragraph 3 above, MHI knows of no way the information could be lawfully acquired by organizations or individuals outside of MHI.

7. Public disclosure of the referenced information would assist competitors of MHI in their design of new nuclear power plants without incurring the costs or risks associated with the design of the subject systems. Therefore, disclosure of the information contained in the

referenced document would have the following negative impacts on the competitive position of MHI in the U.S. nuclear plant market:

A. Loss of competitive advantage due to the costs associated with the development of the thermal and hydraulic design. Providing public access to such information permits competitors to duplicate or mimic the methodology without incurring the associated costs.

B. Loss of competitive advantage of the US-APWR created by benefits of enhanced plant safety, and reduced operation and maintenance costs associated with the thermal and hydraulic design.

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 25th day of June, 2009.

Y. Ogata

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

Enclosure 3

UAP-HF-09336, Rev.0

MHI's Responses to RAI No. 377-2629 Revision 1

June 2009
(Non Proprietary)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

6/25/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 377-2629 REVISION 1**
SRP SECTION:	**04.04 – THERMAL AND HYDRAULIC DESIGN**
APPLICATION SECTION:	**4.4.2.5**
DATE OF RAI ISSUE:	**5/29/2009**

QUESTION NO.: 04.04-1

In MUAP-07009-P, the Thermal Design Topical Report, Section 4.6, there is a statement that the inlet mass velocity of the hot assembly could be approximately 5 to 10% lower than the core inlet mass velocity average. In MUAP-07022-P, the 1/7 scale flow testing model, in Section 1.3.1.item b., there is a statement that DNBR(design margin) will not be evaluated for flow rates 0 to 20% below the core average value. Explain the apparent difference in these two reports. If an allowed 20% lower flow rate is MHI's criteria for evaluation, please provide quantitative results, for a range of power distributions, that a 20% decrease in assembly inlet flow rate has a negligible effect on minimum DBNR. If not, what inlet flow rate maldistribution is assumed in the DNBR analysis?

ANSWER:

In MUAP-07009-P, it is stated that the inlet mass velocity of the hot assembly could be approximately 5 to 10% lower than the average core inlet mass velocity, a provision of the standard procedure of MHI's thermal hydraulic design analysis applied to the US-APWR design. In the past, as a conservative engineering practice, MHI had assumed that the hot assembly inlet flow is 5% lower than the core average in DNB analysis for the conventional type PWR reactor in Japan. For the recent plant design including the US-APWR, 10% reduction is assumed for the practice as described in DCD Subsection 4.4.4.2. This reduced inlet flow condition is, however, not physically a bounding value for the actual core, but just as a conventional practice with the idea trying to be conservative in analysis. As a matter of fact, abundant test data and engineering analyses all indicated that the minimum DNBR analysis is really insensitive to such inlet flow distribution setting. Sensitivity studies in Appendix A of MUAP-07009-P show that the effect of the hot assembly inlet flow rate on the minimum DNBR is negligibly small even if it is of 20% lower than the core average flow rate, because the flow re-distribution in the lower core swiftly flattens out the channel inlet mass flux disparity.

In Subsection 1.3.1 of MUAP-07022-P, the statement that DNBR (design margin) will not be evaluated for flow rates 0 to 20% below the core average value has nothing to do with the design criteria for a safety design. The value of 80% is set just as the reference target for the Reactor Vessel Internals (RVI) design. As long as this RVI design target is met, the DNB design is not affected as shown in Appendix A of MUAP-07009-P. The test result showed a maximum [ ] reduction in inlet flow rate at the hot assembly entrance, which is comparable with the proposed

DNBR analysis condition.

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

6/25/2009

US-APWR Design Certification

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Docket No. 52-021

RAI NO.:	**NO. 377-2629 REVISION 1**
SRP SECTION:	**04.04 – THERMAL AND HYDRAULIC DESIGN**
APPLICATION SECTION:	**4.4.2.5**
DATE OF RAI ISSUE:	**5/29/2009**

QUESTION NO.: 04.04-2

In Design Control Document, Section 4.4.2.2.4, 'Effects of Rod Bow on DNBR', it is stated that "the maximum DNBR rod bow penalty for the US-APWR core is less than 1 percent."

Provide the basis for the 1 percent value. Include reference to relevant test data for 14-foot US-APWR fuel.

ANSWER:

The DNBR rod bow penalty for US-APWR less than 1 percent is based on the correlation developed in Ref. 04.04.2-1, which has been approved by the NRC. The database for the correlation contains various bundle test data including the MHI 14-ft bundles (Ref. 04.04.2-2 and 04.04.2-3).

The correlation reflects DNB heat flux suppression due to rod bow at 85 and 100 % of rod-to-rod gap closure. No DNB heat flux suppression is assumed at less than 50% gap closure for the reason that any gap closure less than 50% would provide negligible DNB effect. The DNB heat flux suppression rate between the rod-to-rod gap closure of 50, 85 and 100% is given by linear interpolation of the values at 50, 85, and 100% closures as shown in Fig. 04.04.2-1.

The δ_{con} and δ_{85} evaluated for the US-APWR core condition are of [ ] and [ ] % , respectively.

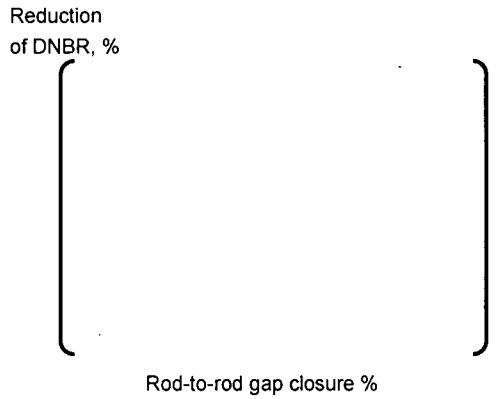

Fig. 04.04.2-1 Correlation between DNBR reduction rate and rod to rod gap closure rate

The rod-to-rod gap closure used for the US-APWR design is conservatively determined from the correlation between the assembly average burn-up and the standard deviation of the gap closure, which is based on visual inspection data of the irradiated fuel assemblies. The detailed descriptions of the data and methodology are discussed in Ref. 04.04.2-4 and Ref. 04.04.2-5. In general, the rod-to-rod gap closure becomes greater with increasing burn-up, whereas such high burn-up fuel assemblies can not be DNBR-limiting in the core. The standard deviation of [ ] at the assembly average burn-up of [ ] has been conservatively applied to the US-APWR design.

The rod-to-rod gap closure rate is assumed to be normally distributed with the above mentioned standard deviation. Each random value of rod-to-rod gap closure from the distribution is correlated to a DNB suppression value by using Fig. 04.04.2-1. Since there is a gap on each side of a fuel rod, the greater 'random rod-to-rod gap closure value' of the two sides is selected as the random value corresponding to that specific rod.

Finally, the rod bow penalty (δ_{bow}) to the minimum DNBR is defined from the two DNBR design limit values of the Revised Thermal Design Procedure (RTDP), DL_{bow} and DL_{unbow}, which does and does not include the effect of the rod bow uncertainty, respectively.

$$\delta_{bow} = 1 - \frac{DL_{unbow}}{DL_{bow}}.$$

The DL_{bow} is obtained by a Monte Carlo calculation which combines the random values of DNBR suppression and random z (= DNBR x M/P) values of RTDP.

By using the above methodology, the DNBR rod bow penalty for US-APWR has been evaluated as [ ] percent.

References

04.04.2-1 Skaritka, J., Ed, "Fuel Rod Bow Evaluation, WCAP-8691, Revision 1," July 1979.

04.04.2-2 Nagino, Y., et al., "Rod Bowed to Contact Departure from Nucleate Boiling Tests in Coldwall Thimble Cell Geometry," Journal of Nuclear Science and Technology, 15[8], pp. 568-573, August 1978.

04.04.2-3 Nagino, Y., et al., "Effect of Partially Bowed Heated Rod on DNB in Coldwall Thimble Cell Geometry," Journal of Nuclear Science and Technology, 15[12], pp. 943-945,

December 1978.

04.04.2-4 "Mitsubishi Fuel Design Criteria and Methodology," MUAP-07008-P, May 2007.

04.04.2-5 "MHI's Responses to the NRC's Requests for Additional Information on Topical Report MUAP-07008-P(O) "Mitsubishi Fuel Design Criteria and Methodology"," UAP-HF-09025, January, 2009 (ML090340614)

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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US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 377-2629 REVISION 1**
SRP SECTION:	**04.04 – THERMAL AND HYDRAULIC DESIGN**
APPLICATION SECTION:	**4.4.2.5**
DATE OF RAI ISSUE:	**5/29/2009**

QUESTION NO.: 04.04-3

Design Control Document, Section 4.4.2.6, 'Core Pressure Drops and Hydraulic Loads', does not contain sufficient description of the hydraulic loads calculation method or results for staff evaluation. Provide a discussion of the calculation method, assumptions, and results. A reference to the structural evaluation of the fuel and internal vessel components should be included.

ANSWER:

The hydraulic loads on each component is calculated by the following equation

$HL = \Delta P \times A$

where,

HL: hydraulic load (lb_f)

ΔP: unrecoverable pressure loss across the component (psi)

A: cross sectional area for which the pressure loss works (in^2)

The hydraulic loads are evaluated to ensure the mechanical design integrity of (1) fuel assemblies, and (2) core support structures.

(1) The fuel assembly is designed not to be lifted off the lower core support plate during normal operations and AOOs except at the pump-over-speed condition during which the fuel assembly lift-off is permitted but the holddown spring needs to retain its elastic deformation characteristics. The details and results of the holddown spring design are discussed in Section 4.6 of Ref. 04.04.3-1.

During normal operations and AOOs except at the pump-over-speed, the holddown spring should provide the downward force greater than upward force including the hydraulic load. The forces relevant to this design are determined both at hot full power and at cold startup conditions. Mechanical Design Flow (MDF) is basically used, while it is re-evaluated for the cold startup condition assuming cold coolant properties.

The pump-over-speed condition is defined as the hot full power coolant condition with 120% MDF. The hydraulic load is evaluated at this condition to ensure that no additional plastic deformation of the holddown spring would occur.

(2) For core support structures, the stress due to the mechanical loads including hydraulic loads

should satisfy the design criteria. The hydraulic loads are determined at the pump-over-speed condition described above. Details of the design criteria and the stress analyses for the core support structures are presented in Reference 04.04.3-2.

References

04.04.3-1 "US-APWR Fuel System Design Evaluation," MUAP-07016-P, February 2008.

04.04.3-2 "Summary of Stress Analysis Results for the US-APWR Core Support Structures," MUAP-09004-P, March, 2009.

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.

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APPLICATION SECTION:	**4.4.2.5**
DATE OF RAI ISSUE:	**5/29/2009**

QUESTION NO.: 04.04-4

Provide a tabulation of all numerical uncertainty values considered in the statistical evaluation of DNBR for the US-APWR fuel. These uncertainties are discussed qualitatively in Design Control Document, Section 4.4.2.9.1 for DNB analyses.

ANSWER:

The Revised Thermal Design Procedure (RTDP, Ref. 04.04.4-1) is applied to obtain the design limit of the minimum DNBR in the US-APWR core. As required by applying the RTDP methodology, sensitivity factors are evaluated for the US-APWR based on the uncertainties of key parameters involving plant operating parameters, heat flux distribution, computer codes, and the DNB correlation. Tables 04.04.4-1 and 04.04.4-2 show the nominal and uncertainty values of those parameters in generic sense and the sensitivity factors for typical cell and thimble cell as well.

Since some of those uncertainties are dependent on the measurement error allowances of the plant-specific instrumentation and the fabrication tolerances of the manufactured fuel pellets, the design limit of minimum DNBR will be re-evaluated after the plant-specific uncertainties become available. The design limit and the safety analysis limit of minimum DNBR will be confirmed/ updated accordingly.

Table.04.04.4-1 US-APWR RTDP sensitivity factors for low flow rate condition (typical cell)

Parameters (x_i)	μ_i	σ_i	σ_i/μ_i	S_i	$S_i^2(\sigma_i/\mu_i)^2$
Power (fraction)					
Tin (°F)					
Pressure (psia)					
Flow (fraction)					
Effective Core Flow (fraction)					
$F^N_{\Delta H}$					
$F^E_{\Delta H,1}$					
Subchannel Code					
Transient Code					

-Uncertainties of input parameters and code predictions:

$(\sigma_y/\mu_y)^2 = \Sigma S_i^2(\sigma_i/\mu_i)^2 = [\quad\quad]$

-Uncertainty of DNB correlation prediction (Refer to Table B.3-6 in Appendix B of Ref. 04.04-4-2):

$\mu_{M/P} = [\quad\quad]$

$\sigma_{M/P} = [\quad\quad]$

-Design Limit of minimum DNBR (DL)

$$\frac{\sigma_Z}{\mu_Z} = \left[\left(\frac{\sigma_y}{\mu_y}\right)^2 + \left(\frac{\sigma_{M/P}}{\mu_{M/P}}\right)^2\right]^{\frac{1}{2}} = \left[\quad\quad\right]$$

$$DL = \frac{1}{\mu_{M/P}(1-1.645\sigma_Z/\mu_Z)} = \left[\quad\quad\right]$$

where,

μ: mean or nominal value

σ: standard deviation

S_i: sensitivity factor associated with i-th parameter, x_i $(=\partial(\ln y)/\partial(\ln x_i))$

y: DNBR(variable)/DNBR(nominal)

M/P: measured to predicted ratio of DNB heat flux

z: (M/P) x (y)

Table.04.04.4-2 US-APWR RTDP sensitivity factors for low flow rate condition (thimble cell)

Parameters (x_i)	μ_i	σ_i	σ_i/μ_i	S_i	$S_i^2(\sigma_i/\mu_i)^2$
Power (fraction)					
Tin (°F)					
Pressure (psia)					
Flow (fraction)					
Effective Core Flow (fraction)					
$F^N_{\Delta H}$					
$F^E_{\Delta H,1}$					
Subchannel Code					
Transient Code					

-Uncertainties of input parameters and code predictions:

$(\sigma_y/\mu_y)^2 = \Sigma S_i^2(\sigma_i/\mu_i)^2 = [\qquad\qquad]$

-Uncertainty of DNB correlation prediction (Refer to Table B.3-6 in Appendix B of Ref. 04.04-4-2):

$\mu_{M/P} = [\qquad\qquad]$

$\sigma_{M/P} = [\qquad\qquad\qquad\qquad\qquad\qquad]$

-Design Limit of minimum DNBR (DL)

$$\frac{\sigma_Z}{\mu_Z} = \left[\left(\frac{\sigma_y}{\mu_y}\right)^2 + \left(\frac{\sigma_{M/P}}{\mu_{M/P}}\right)^2\right]^{\frac{1}{2}} = \left[\qquad\qquad\qquad\qquad\qquad\qquad\qquad\qquad\right]$$

$$DL = \frac{1}{\mu_{M/P}(1 - 1.645\sigma_Z/\mu_Z)} = \left[\qquad\qquad\qquad\qquad\qquad\qquad\qquad\qquad\right]$$

where,

- μ: mean or nominal value
- σ: standard deviation
- S_i: sensitivity factor associated with i-th parameter, x_i $(=\partial(\ln y)/\partial(\ln x_i))$
- y: DNBR(variable)/DNBR(nominal)
- M/P: measured to predicted ratio of DNB heat flux
- z: (M/P) x (y)

References

04.04.4-1 Friedland, A. J. and Ray, S., "Revised Thermal Design Procedure," WCAP-11397-P-A, April 1989.

04.04.4-2 "Thermal Design Methodology," MUAP-07009-P, May 2007.

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

6/25/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 377-2629 REVISION 1**
SRP SECTION:	**04.04 – THERMAL AND HYDRAULIC DESIGN**
APPLICATION SECTION:	**4.4.2.5**
DATE OF RAI ISSUE:	**5/29/2009**

QUESTION NO.: 04.04-5

Provide a tabulation of all numerical uncertainty values considered in the statistical evaluation of fuel and cladding temperatures for the US-APWR fuel. These uncertainties are discussed qualitatively in Design Control Document, Section 4.4.2.9.2 for fuel and cladding temperatures.

ANSWER:

Uncertainties considered in the evaluation of fuel pellet center-line temperature are summarized in Table 04.04.5-1. As shown in the table, causes of uncertainties are categorized into analytical models and manufacturing tolerances.

The fuel temperature model uncertainty is [ ], which is quantified based on the comparison of the models to the measurements at a 95-percent probability and a 95-percent confidence level, as described in MUAP-07008-P (Ref. 04.04.5-1) Subsection 4.3.1.1. The temperature uncertainty due to the cladding creep model uncertainty is evaluated with a fuel center-line temperature rise using the lower bound of creep model uncertainty [ ], as described in MUAP-07008-P Subsection 4.3.8.

The other temperature uncertainties are given as the differences between the temperature calculation results with and without considering the uncertainty such as manufacturing tolerances for each parameter value.
All the uncertainties are considered to be statistically independent of each other, and therefore, the total uncertainty is calculated by the SRSS (Square Root of the Sum of the Squares) method.

Regarding the fuel cladding temperature, additional cladding surface temperature rise can occur due to the uncertainties of crud and oxide thickness. However, crud and oxide are usually very thin, and therefore, their effects are negligible compared with the margin in determining the limit fuel temperature. At the Beginning of Life (BOL), when the fuel temperature is most limiting, the oxide thickness is approximately [ ] and crud thickness is assumed to be [ ], and the impact on the fuel temperature is less than 10°F. Therefore, the uncertainties of those parameters are not explicitly considered in cladding temperature determinations.

Table 04.04.5-1 Fuel Center-line Temperature Uncertainty Evaluation (UO_2 fuel)

unit: °F (°C)

Items	Categories	Model/Manufacturing Uncertainties	Fuel Temperature Uncertainties

References

04.04.5-1 "Mitsubishi Fuel Design Criteria and Methodology," MUAP-07008-P, May 2007.

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.

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APPLICATION SECTION:	**4.4.2.5**
DATE OF RAI ISSUE:	**5/29/2009**

QUESTION NO.: 04.04-6

Provide a tabulation of all numerical uncertainty values considered in the statistical evaluation of core hydraulics for the US-APWR core design. These uncertainties are discussed qualitatively in Design Control Document, Section 4.4.2.9.3 for core hydraulics.

ANSWER:

The pressure drop across the core and the Reactor Vessel (RV) are determined from the experimental results and empirical formulas from the cumulative hydraulic experiments at MHI for the similar plant designs to the US-APWR. The uncertainty of the pressure drop is generally assumed to be 10 percent as described in the Table 4.4-1 of DCD. This level of assumed uncertainty is more conservative than that usually considered for the measurement uncertainty of pressure drops.

For more conservatism, an additional uncertainty of [ ] percent was considered in the evaluation of the Reactor Coolant System (RCS) flow, which is the most important part of RV pressure drops. Thermal Design Flow (TDF) and Mechanical Design Flow (MDF) are determined from the Best Estimate Flow (BEF), both with a total of [ ] percent uncertainties in RV pressure drops, Confirming tests of the RCS flow rate will be conducted prior to the initial criticality and during start-up as described in DCD Subsection 4.4.2.9.3.

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.