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January 30 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-09025

Subject: MHI's Responses to the NRC's Requests for Additional Information on Topical Report MUAP-07008-P(0) "Mitsubishi Fuel Design Criteria and Methodology"

References: 1) Letter from the NRC (ML082760492) to Y. Ogata (MHI), "Mitsubishi Heavy Industries, Inc.-Request for Additional Information on Topical Report MUAP-07008-P,Revision 0、"Mitsubishi Fuel Design Criteria and Methodology" dated on November 20, 2008

2) Letter MHI Ref: UAP-HF-08299 from Y. Ogata (MHI) to U.S.NRC, "MHI's Partial Responses to the NRC's Requests for Additional Information on Topical Report MUAP-07008-P(0) "Mitsubishi Fuel Design Criteria and Methodology" dated on December 19, 2008.

With this letter, Mitsubishi Heavy Industries, Ltd. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") responses entitled "MHI's Responses to the NRC's Requests for Additional Information on Topical Report MUAP-07008-P(0) Mitsubishi Fuel Design Criteria and Methodology". In the enclosed document, MHI provides the responses for RAI's item 19c, 33, 34 and 35 of those in Reference 1. All other responses to the RAI's have been already provided in Reference 2, dated December 19, 2008.

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 in this package (Enclosure 3). In the non-proprietary version, the proprietary information, bracketed in the proprietary version, is 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 this submittal. His contact information is provided below.

Sincerely,

D081
NRO

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

Enclosures:

1. Affidavit of Yoshiki Ogata

2. MHI's Responses to the NRC's Requests for Additional Information on Topical Report MUAP-07008-P(0) "Mitsubishi Fuel Design Criteria and Methodology" (proprietary)

3. MHI's Responses to the NRC's Requests for Additional Information on Topical Report MUAP-07008-P(0) "Mitsubishi Fuel Design Criteria and Methodology" (non-proprietary)

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

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ENCLOSURE 1

Docket No.52-021
MHI Ref: UAP-HF-09025

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 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 "MHI's Responses to the NRC's Requests for Additional Information on Topical Report MUAP-07008-P(0) "Mitsubishi Fuel Design Criteria and Methodology" dated January,2009, and have determined that portions of the report 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 technical report indicates that all information identified as "Proprietary" should be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a).

3. The information in the report 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 code and design methodology developed by MHI for the fuel of the US-APWR. These were developed at significant cost to MHI, since they required the performance of detailed calculations, analyses, and testing extending over several years. The referenced information is not available in public sources and could not be gathered readily from other publicly available information.

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 30th day of January, 2009.

Y. Ogata

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

Enclosure 3

UAP-HF-09025, Rev.0

MHI's Responses to the NRC's Requests for Additional Information on Topical Report MUAP-07008-P(0) “Mitsubishi Fuel Design Criteria and Methodology"

January 2009
(Non Proprietary)

INTRODUCTION

This report documents MHI's responses to the NRC's Request for Additional Information (RAI) dated November 20 2008 concerning the Topical Report MUAP-07008-P(0) "Mitsubishi Fuel Design Criteria and Methodology".

In this report, MHI provides the responses for RAI's 19c, 33, 34 and 35. All other responses to the RAI's have been already provided in "MHI's Partial Responses to the NRC's Requests for Additional Information on Topical Report MUAP-07008-P(0) "Mitsubishi Fuel Design Criteria and Methodology" dated on December 19, 2008.

RAI 19

The range of materials and irradiation conditions that are being requested for FINE need to be clearly stated.

c. The applicability range of linear heat generation rate for normal operation appears to be burn-up dependent. Discuss a revised applicability range versus burn-up based on fuel centerline temperature data (including added data comparisons from RAI number 33), FGR data (including added data comparisons from RAI number 34), and cladding strain data (including added data comparisons from RAI number 35).

RAI 19 RESPONSE

- RAI 19-c

As stated in the response to RAI 19a, MHI is requesting licensing of the FINE code up to a rod average burnup of 62GWd/MTU, corresponding to the US-APWR fuel design maximum rod burnup.

Figure 19c-1 shows the rod power versus rod average burnup for the FINE fuel rod performance database. This figure includes the database discussed in RAI 33, 34, and 35 in addition to the original data in Reference (19-1). The figure shows that this database includes

US-APWR maximum rod burnup. The design peak rod power for the US-APWR is 1.73 times the core average power of 4.65kW/ft.

Figure 19c-1 Rod power versus rod burnup for the FINE fuel rod database, including those data discussed in RAI 33,34 and 35.

(19-1) "Mitsubishi Fuel Design Criteria and Methodology", MUAP-07008-P (Proprietary) and MUAP-07008-NP (Non-Proprietary), May 2007

RAI 33

The code verification and validation data for fuel temperatures utilizes a significant number of rods that were irradiated between 30 and 40 years ago. These were state of the art irradiations and temperature measurements 30 years ago but there are much better characterized Halden irradiations available at this time. In addition, the other rods used for verification, other than the two rods from IFA-562, are all unknown to PNNL and NRC and appear to be between 10 to 20 years old. Also, the data set used to verify the fuel temperatures for UO2- Gd2O3 fuel is limited to only 3 rods. The following request would extend the verification and validation database for FINE using additional Halden rods irradiated at high power (lower burnups), at moderate powers up to high burnups and additional gadolinia rods with measured fuel temperatures. This additional data will also provide a better estimate of code calculational uncertainties in temperature as a function of burnup and power.

a. In relation to verification and validation at high powers and low burnup provide FINE temperature predictions of the IFA-677.1 Rods 2 and 6 (most recent data provided in HWR-819) and the IFA-681 Rods 1 and 5 (most recent data provided in HWR-832). Fuel centerline temperature predictions and data should be plotted versus time or burnup.

b. In relation to verification and validation at high burn-up levels, provide FINE temperature predictions of IFA-597.3 Rod 8 that was irradiated up to moderate rod powers at a burn-up of ~ 70 GWd/MTU and IFA-515 Rods A1 and B1 that were irradiated to low powers but burn-ups extended up to 80 GWd/MTU. The IFA-597.3 Rod 8 test also measured FGR by rod puncture so this rod can be used to verify both fuel temperatures and FGR at relatively high local burn-ups; provide predicted FGR for this rod. Fuel centerline temperature predictions and data should be plotted versus time or burn-up.

c. In relation to verifying and validating predicted fuel temperatures for UO_2-Gd_2O_3 fuel provide FINE temperature predictions of IFA-636 Rods 2 and 4 that were irradiated up to 25 GWd/MTU (data provided in HWR-817) and of IFA-681 Rods 2, 4 and 6 (data provided in HWR-832). Fuel centerline predictions and data should be plotted versus time or burnup.

d. The uncertainty in the FINE prediction appears to increase with increasing linear heat generation rate (Figure 4.3.1.1-2) such that the constant upper-bound value used by FINE may not bound the higher heat rate data with a high degree of confidence. This is of particular concern because most licensing analyses are performed at the higher linear heat generation rates. Provide measured-minus-predicted fuel temperature versus linear heat generation rate and burnup for each set of rods in a., b. and c. above, similar to the plots in Figures 4.3.1.1-1 and 4.3.1.1-2 (separating out UO_2 and UO_2-Gd_2O_3 rods) and include the FINE assumed upper-bound uncertainty in these figures. Recalculate the standard deviation, mean and upper-bound for all of the fuel centerline temperature data including those in a., b., and c. above.

e. Calculate the standard deviation, mean and upper-bound for only those measured temperature data above 1300°C and plot predicted- minus- measured versus fuel burnup for this high temperature data

RAI 33 RESPONSE

- RAI 33a

MHI has the TFDB data for IFA-677.1 through September 15th 2007 and for IFA-681.1 through September 20th 2008. The TFDB data are normally recorded every 15 minutes, and

The measured data and the FINE calculation results for IFA-677.1 Rod 2 and Rod 6 are compared in Figures 33a-1, -2, -3, and -4.

as can be seen from the comparisons between the calculated and measured rod internal pressures shown in Figures 33a-5 and -6 for IFA-677 Rod 2 and Rod 6, respectively.

Figures 33a-7 and -8 compare the measured and the calculated temperatures for IFA-681 Rod 1 and Rod 5, respectively.

A comparison between the measured and predicted pressure data for Rod 1 is shown in Figure 33a-9.

Reference

(1) Radmir Josek, "The high initial rating test IFA-677.1 : Final report on in-pile results", HWR-872, April 2008

Figure 33a-1 Temperature history of TF2, Rod 2, IFA-677.1

Figure 33a-2 Temperature history of TF8, Rod 2, IFA-677.1

Figure 33a-3 Temperature history of TF6, Rod 6, IFA-677.1

Figure 33a-4 Temperature history of TF12, Rod 6, IFA-677.1

Figure 33a-5 Pressure history of Rod 2, IFA-677.1

Figure 33a-6 Pressure history of Rod 6, IFA-677.1

Figure 33a-7 Temperature history of TF1, Rod 1, IFA-681.1

Figure 33a-8 Temperature history of ET14, Rod 5, IFA-681.1

Figure 33a-9 Pressure history of Rod 1, IFA-681.1

- RAI 33b

The Halden TFDB data for IFA-515 and IFA-597.3 have been condensed in the same way as described in the response to RAI 33a, to provide power history and measured temperature data that can be evaluated with the FINE code.

Figure 33b-1 shows the comparison between the measured and calculated temperatures for IFA-515 Rod A1.

Figure 33b-2 compares the measured and calculated temperatures for IFA-597.3 Rod 8. This rod had been irradiated in a BWR to a burnup of about 67 GWd/MTU at the temperature measurement elevation, and then irradiated in the Halden reactor for an additional burnup of approximately 3 GWd/MTU. Thermocouple measurements of the fuel centerline temperature have been obtained during the Halden irradiation.

Figure 33b-1 Temperature history of ETC1, Rod A1, IFA-515.15

Figure 33b-2 Temperature history of TFC2, Rod 8, IFA-597.3

- RAI 33c

Fuel temperature data have been obtained for IFA-681 Rod 2, Rod 4 and Rod 6 and for IFA-636 Rod 2. IFA-681 Rod 2 and Rod 4 have 2wt% Gd_2O_3 doped fuel pellets. Rod 6 of IFA-681 and Rod 2 of IFA-636 contain 8wt% Gd_2O_3 doped fuel pellets.

The TFDB data for these rods have been processed in the same way as for the other Halden temperature data to provide power history and measured temperature data that can be evaluated with the FINE code. The comparisons between the FINE code results and the measured temperatures are shown in Figure 33c-1 for IFA-681 Rod 2, Figure 33c-2 for IFA-681 Rod 4, Figure 33c-3 for IFA-681 Rod 6 and Figure 33c-4 for IFA-636 Rod 2. In addition to these, Figure 33c-5 shows only the FINE temperature prediction versus rod burnup because IFA-636 Rod4 has no temperature measurement data.

Figure 33c-1 Temperature history of TF2, Rod 2, IFA-681.1

Figure 33c-2 Temperature history of ET13, Rod 4, IFA-681.1

Figure 33c-3 Temperature history of ET15, Rod 6, IFA-681.1

Figure 33c-4 Temperature history of ET2, Rod 2, IFA-636.1

Figure 33c-5 Temperature history of Rod 4, IFA-636.1

- RAI 33d

Figures 33d-1 and -2 show the fuel temperature M – P data for the high power rods evaluated in RAI 33a, IFA-677 and IFA-681, as a function of burnup and power, respectively. The thermocouple data (TF data) are plotted as a function of the local burnup and power at the thermocouple position, while the extension thermometer (ET data) are plotted as a function of the rod average burnup and rod average power, since the ET temperatures are obtained from the thermal expansion of a wire running through the total fuel stack length.

Figures 33d-3 and -4 show the M – P data for the high burnup rods evaluated in RAI 33b, IFA -515 Rod A1 and IFA-597.3 Rod 8, as a function of burnup and fuel power, respectively.

Figures 33d-5 and -6 show the M – P data for the IFA-681 and IFA-636 gadolinia doped fuel rods evaluated in RAI 33c, as a function of burnup and power, respectively.

The upper-bound fuel temperature uncertainty has been recalculated using the data obtained in the RAI 33a, b and c evaluations. The results for this calculation of the FINE code temperature uncertainty based on the recent Halden data are:

Consistent with the data shown in Figures 33d-1 through 33d-6, the recalculated upper-bound is [ ]

Figure 33d-1 M-P vs Burnup for rods in RAI 33a

Figure 33d-2 M-P vs power for rods in RAI 33a

Figure 33d-3 M-P vs Burnup for rods in RAI 33b

Figure 33d-4 M-P vs Power for rods in RAI 33b

Figure 33d-5 M-P vs Burnup for Gd_2O_3 doped fuel rods in RAI 33c

Figure 33d-6 M-P vs Power for Gd_2O_3 doped fuel rods in RAI 33c

- RAI 33e

The M – P mean value and standard deviation has been calculated for the data with measured temperatures above 1300 deg.C, including all the data shown in MUAP-07008-P and the data discussed in the RAI 33a, b, and c. The results for this calculation are:.

This upper-bound uncertainty based on just these high temperature data is

Figure 33e-1 shows the high temperature M – P data as a function of burnup.

Figure 33e-1 M-P vs Burnup for the measured temperatures above 1300 deg.C

RAI 34

The code verification data for FGR contains a large quantity of data with a mixture of high and older moderate quality data. The use of this data set leads to several observations, comments, and requests. Some of the moderate quality data with high release (>5 percent) is underpredicted by an amount greater than the upperbound uncertainty used for FINE predictions. In addition, the FINE code predictions versus burn-up appear to follow the Vitanza threshold versus burn-up (Figure 4.2.3.3-3); however, Halden in recent years has discovered that the Vitanza model underpredicts the threshold in rods with local burn-ups greater than 50 GWd/MTU. The IFA-429 Rod DH predicted and measured gas release values (based on in-reactor pressure) are based on old data while more recent experimental data (end-of-life puncture FGR) is available for this rod than that presented in the submittal. Finally, there are three sets of high quality data that are absent from the FINE verification database. Therefore, to supplement the provided data and analyses, provide predictions of FGR for the following rods

a. Two Babcock & Wilcox Co. segmented fuel rods that were irradiated in Arkansas Nuclear Unit 1 to 62 GWd/MTU burn-up and then power ramp tested in Studsvik R2 resulting in measured cladding deformation and FGR following the power ramp (data presented in Proceedings of 1994 ANS International Topical Meeting on Light Water Reactor Fuel Performance, West Palm Beach, Florida, p. 343).

b. Halden IFA-597.3 Rod 8 that was re-fabricated by Halden from a BWR fuel rod commercially irradiated in Ringhal's 1 to ~68 GWd/MTU at discharge and then re-irradiated in Halden to ~71 GWd/MTU at a moderately high linear heat generation rate between 21 to 31 Kilowatts per meter for this high burn-up level (see RAI number 9.b above). The latter rod was a boiling water reactor (BWR) rod but the difference between BWR and pressurized water reactor (PWR) rods is not significant at this burn-up level because the fuel-cladding gap is closed and this data is particularly valuable because fuel temperatures were measured as well as FGR.

c. Halden IFA-534.14 Rods 18 and 19 were irradiated in Gosgen PWR to ~ 59 GWd/MTU burn-up, re-fabricated with pressure instrumentation, and then re-irradiated in Halden to ~63 GWd/MTU. These two rods had different fuel grain sizes and FGR was measured by rod puncture (data in HWR-558 and in the OECD NEA database).

d. Provide a revised prediction of the IFA-429 Rod DH based on the latest power history and FGR puncture data (data in HWR-668).

e. Recalculate the standard deviation, mean and upper-bound for all of the FGR data including those in a., b., c, and d. above.

f. Calculate the standard deviation, mean and upper-bound for only those with measured release data ≥ 5 percent and plot predicted-minus-measured versus fuel burn-up for this high release data. The peak rod pressure rods in a core usually have predicted release values greater than five percent such that the accuracy of predicting these data is of primary concern.

RAI 34 RESPONSE

- RAI 34a

Table 34a-1 compares the FINE FGR results with the PIE FGR data given in Reference (34-1) for the B&W rod 1 and rod 3. FINE overpredicts the FGR for these rods, as it does for the power ramp FGR data described in Section 4.3.2 of Reference (34-2).

The FINE evaluation results for the cladding deformation of these rods are described in the response to RAI35.

Table 34a-1 FGR results for B&W rod 1 and 3

[ ]

- RAI 34b

Table 34b-1 compares the FINE FGR result for IFA-597.3 rod 8 with the PIE results in the OECD NEA database. Figure 34b-1 shows the comparison between the measured data and the FINE predictions for the internal pressure during the Halden irradiation. [ ]

[ ]

[ ]

Table 34b-1 FGR results for IFA-597.3 rod 8

[ ]

Figure 34b-1 Rod internal pressure results for IFA-597.3 rod 8

- RAI 34c

Table 34c-1 compares the FINE FGR result for IFA-534.14 rod 19 with the PIE results in the OECD NEA database. The US-APWR fuel design uses standard grain size pellets and IFA-534.14 rod 18, which used large grain size pellets, is not applicable to the US-APWR. Figure 34c-1 shows the comparison between the measured and FINE predictions for the internal pressure during the Halden irradiation.

Table 34c-1 FGR results for IFA-534.14 rod 19

Figure 34c-1 Rod internal pressure results for IFA-534.14 rod 19

- RAI 34d

Table 34d-1 compares the FINE FGR results for IFA-429 rod DH with the PIE data given in Reference (34-5).[ ]

[ ]

It should be noted that the previous FINE results for rod IFA-429 DH, given in Reference (34-2), were obtained with inputs derived from the data given in References (34-6) and (34-7). The IFA-429 Rod DH calculations for this RAI response are based on input derived from the Halden TFDB data. This difference in the inputs used here and for the Reference (34-2) analysis is the reason why these results differ from those previously given in Reference (34-2).

Table 34d-1 FGR results for IFA-429 rod DH

[ ]

[ ]

Figure 34d-1 Rod internal pressure results for IFA-429 rod DH

- RAI 34e

The recalculated results for the standard deviation, mean and upper-bound for all of the FGR data, including the data obtained in the responses to RAI 34a through 34d above, are shown in Table 34e-1.

Table 34e-1 Re-evaluation of the FGR uncertainty including the additional data

- RAI 34f

The results for the standard deviation, mean and upper-bound considering only the FGR data with a measured FGR ≥ 5 percent, including the data obtained in the responses to RAI 34a through 34d above, are shown in Table 34f-1.[ ]

Figure 34f-1 shows the Predicted - Measured FGR versus burn-up for the data with measured FGR greater than 5%.[ ]

Table 34f-1 Evaluation of the FGR uncertainty for only the data with measured FGR ≥ 5 percent

[ ]

Figure 34f-1 Predicted - Measured FGR versus burn-up for the data with measured FGR ≥ 5 percent

REFERENCES

(34-1) D.A. Wesley at el, "Mark-BEB Ramp Testing Program", Proceedings of 1994 ANS International Topical Meeting on Light Water Reactor Fuel Performance, West Palm Beach, Florida, p. 343.

(34-2) "Mitsubishi Fuel Design Criteria and Methodology", MUAP-07008-P (Proprietary) and MUAP-07008-NP (Non-Proprietary), May 2007.

(34-3) K. Malen et al, "PIE of High Burn-up BWR Fuel Rod IFA-597.3 (Rod 8)", HRP-356/18.

(34-4) I. Matsson and J.A. Turnbull, "The integral fuel rod behaviour test IFA-597.3: Analysis of the measurements," HWR-543, January 1998.

(34-5) J.A. Turnbull, "CONCLUDING REPORT ON THREE PWR RODS IRRADIATED TO 90 MWD/KG UO2 IN IFA-519.9: ANALYSIS OF MEASUREMENTS OBTAINED IN-PILE AND BY PIE", HWR-668, January 2001.

(34-6) D.D. Lanning et.al, "FRAPCON-3: Integral Assessment", NUREG/CR-6534 Vol. 3, PNNL-11513, December 1997.

(34-7) D.D. Lanning et.al, "FRAPCON-3 Updates, Including Mixed-Oxide Fuel Properties", NUREG/CR-6534, Vol.4, PNNL-11513, May 2005.

RAI 35

The code verification data for prediction of cladding strains during power ramps is very limited to only 2 test rods at ~ 30 GWd/MTU. There are several ramp test programs that have measured cladding plastic strains greater than 0.3 percent as a result of power ramping, such as RISØ FGP3 ANF rods, B&W Studsvik, Super-Ramp, and Over-Ramp rods. Provide comparisons of FINE predicted strains to this or other data from rods (such as those from Vandellos) subjected to power ramps with measured plastic strains (hoop and axial) greater than 0.3 percent to verify the code's ability to predict cladding deformation. Identify those rods where gas bubble swelling has contributed to predicted deformation.

RAI 35 RESPONSE

FINE code results for the cladding diameter change due to power ramps have been obtained for the RISO FGP3 ANF rods AN2 and AN8, the B&W Studsvik rods 1 and 3, the Super-Ramp rods PK2/3 and PW3/2, the Over-Ramp rod PW5/1 and the Vandellos 2 rods WI-I07, WZt-I03, MIt-I13, MIt-I14, WI-I23, MZt-I20 and WI-I24 that were ramp tested in the Studsvik R2 reactor after steady state irradiation in the Vandellos 2 reactor. The fuel specification parameters and irradiation data for the Vandellos 2 test rods are summarized in Table 35-1The ramp test conditions and peak measured cladding strains for the Vandellos 2 rods are summarized in Table 35-2.

Results for the pre- and post-ramp test cladding diameters, and the diameter change due to the overpower ramp are compared in Figure 35-1 for RISØ FGP3 ANF AN2 and AN8 rods, in Figure 35-2 for B&W Studsvik rods 1 and 3, in Figure 35-3 for Super-Ramp PK2/3 and PW3/2 rods, in Figure 35-4 for Over-Ramp PW5/1 rod and in Figures 35-5 through 35-8 for the typical Vandellos 2 – Studsvik R2 ramp test rods. Table 35-3 summarizes the results for the measured and calculated peak diameter changes due to the overpower ramp.

Table 35-4 summarizes the FINE results for the fission gas release due to the ramp tests. A large fission gas release during the ramp test is calculated for the RISO FGP3 ANF rods, the B&W Studsvik rods, the Super-Ramp PK2/3 rod, and the Vandellos 2 – Studsvik R2 ramp test rods.

<u>Reference</u>

35-1 "Mitsubishi Fuel Design Criteria and Methodology", MUAP-07008-P (Proprietary) and MUAP-07008-NP (Non-Proprietary), May 2007.

Table 35-1 Fuel specification parameters and irradiation data for the Vandellos 2 – Studsvik R2 ramp test rods

Table 35-2 Ramp conditions and measured diameter change for the Vandellos 2 – Studsvik R2 ramp tests

[ ]

Table 35-3 Comparison of measured and calculated ramp test peak diameter change results

Table 35-5 FINE calculated results for the fission gas release during the ramp tests

Figure 35-1 (1) RISØ FGP3 ANF rod AN2

Figure 35-1 (2) RISØ FGP3 ANF rod AN8

Figure 35-2 (1) B&W Studsvik rod 1

Figure 35-2 (2) B&W Studsvik rod 3

Figure 35-3 (1) Super Ramp rod PK2/3

Figure 35-3 (2) Super Ramp rod PW3/2

Figure 35-4 Over Ramp rod PW5/1

[ ]

Figure 35-5 Vandellos 2 – R2 ramp rod WI-I23 (Low Tin)

Figure 35-6 Vandellos 2 – R2 ramp rod MZt-I20 (ZIRLO)

Figure 35-7 Vandellos 2 – R2 ramp rod MZt-I03 (ZIRLO)

Figure 35-8 Vandellos 2 – R2 ramp rod WI-I07 (Low Tin)