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July 12, 2011

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-11213

Subject: MHI's Second Responses to US-APWR DCD RAI No. 752-5614 Revision 0 (SRP 19)

References:
1) "Request for Additional Information No. 752-5614 Revision 0, SRP Section: 19 – Probabilistic Risk Assessment and Severe Accident Evaluation, Application Section: SRP Chapter 19," dated May 3, 2011.
2) Letter MHI Ref: UAP-HF-11168 from Y. Ogata (MHI) to the U.S. NRC, "MHI's Responses to US-APWR DCD RAI No. 752-5614 Revision 0 (SRP19)" dated June 3, 2011.

With this letter, Mitsubishi Heavy Industries, Ltd. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") a document as listed in Enclosures.

Enclosed are the second responses to the RAIs contained within Reference 1, in which two responses to questions #19-520 and #19-521 are included. In the initial responses submitted within Reference 2, MHI committed to provide responses to #19-520, #19-521 and #19-522 by 2nd July 2011. However because of the continuing discussions regarding hydrogen mixing and control, the topic of question #19-522, the answer to #19-522 will be separately provided from this letter.

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. MHI's Second Responses to Request for Additional Information No. 752-5614 Revision 0

CC: J. A. Ciocco
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Contact Information

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Enclosure 1

UAP-HF-11213
Docket Number 52-021

MHI's Second Responses to Request for Additional Information
No. 752-5614 Revision 0

July 2011

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

7/12/2011

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: **NO. 752-5614 REVISION 0**

SRP SECTION: **19 – Probabilistic Risk Assessment and Severe Accident Evaluation**

APPLICATION SECTION: **SRP Chapter 19**

DATE OF RAI ISSUE: **5/3/2011**

QUESTION NO.: 19-520

Follow-up to Question 19-442:
Although low power and shutdown (LPSD) events are considered in the Environmental Report, the contribution to the CDF and the fission product source terms used in the offsite consequence analysis may be too low. As discussed in the response to Question 19-442, the CDF could be as high as 2.1E-5 per reactor year if all requirements not required by technical specifications were removed. Moreover, the source terms calculated in the Level 3 PRA for accident scenarios that could occur during mid-loop operations (POS 4 and POS 8) may be underestimated. Specifically, none of the release categories represent severe accidents that may occur during shutdown, when the containment is likely to be open as would the reactor coolant system during mid-loop operations. An arbitrary assumption was made by the applicant to consider the containment to be isolated (but not completely) when the accident occurs, such that the leakage would be 100% of containment volume per day at the design pressure. No justification was given for this assumption, and no procedures to do this were discussed. Based on this assumption, the volatile fission product releases were limited to be a few percent of the initial core inventory.

Since LPSD scenarios are significant contributors to the CDF and LRF, and source terms are expected to be large when containment is open, please determine fission product release fractions and the offsite consequences of not isolating the containment during mid-loop accident scenarios. Specifically, the analyses should use more realistic pathways for releases representative of LPSD scenarios. In addition, please perform sensitivity studies to determine maximum averted costs and which SAMDAs, if any, would become cost-beneficial if the LPSD CDF is increased to 2.1E-5 per reactor year, using release fractions characteristic of partial containment isolation, as well as for no isolation.

ANSWER:

In the LPSD Level 2 PRA, it is assumed that the containment is not isolated due to various maintenance activities, and the LRF is considered equivalent to the CDF. This is also considered a conservative assumption because it may be possible to completely isolate the containment before core damage would occur. In DCD Section 19.2.5, accident management during LPSD operation requires taking actions to isolate the containment when identifying some symptom, such as loss of decay heat removal capability and onset of boiling in the core. DCD Chapter 16 Technical Specification Section 3.9.4 provides the restrictions for opening of containment penetrations during refueling operation. Hence, it has been assumed that the containment can be temporarily isolated during LPSD operation, assuming the leakage rate as 100% of containment volume per day. However, the containment is not completely isolated so that in as an assumption for the LPSD Level 2 PRA, the CDF is equivalent to the LRF, and that is how the Level 2 PRA evaluations including SAMDA have been performed.

The procedures for LPSD operation, including containment isolation activities, are currently under development, which can also support the assumption of temporary containment isolation.

The LPSD PRA model for DCD Rev. 2 has been revised to address several changes, which caused changes in CDF calculation results in DCD Rev. 3. The major model changes include the following items:

- The frequency of the LOCA initiating event based on the MHI original calculation in Rev. 2 was revised to apply the EPRI data (Reference 1) in the Rev. 3 model.
- Detailed evaluation for POS 4-3 has been performed.
- The duration of all POSs have been converted from the duration expected based on the Japanese PWR plant operating experience to the US generic data (Reference 2)

This PRA model modification has resulted in improvement in the CDF evaluation, i.e. 2.2E-7/ry in the DCD Rev. 2 is reduced to 1.8E-7/ry.

The sensitivity analysis model to evaluate the conditional CDF based on Technical Specification has also been revised to address more realistic assumptions, especially for the relationship between the POSs and operating modes. Table 19-520-1 summarizes the CDFs and the conditional CDFs for each initiating event. Calculations for the maximum averted cost are also provided in the table.

Table 19-520-1 Summary of CDF and SAMDA evaluation results

Initiating event	CDF (/ry)			
	DCD R2	DCD R2 TS	DCD R3	DCD R3 TS
LOCA	1.3E-7	1.7E-5	5.8E-08	2.4E-06
RCS over drain	2.9E-9	4.8E-7	2.4E-09	3.5E-07
Fail to maintain RCS water level	4.1E-9	1.2E-6	4.1E-09	1.3E-06
Loss of RHR	5.8E-9	8.4E-7	7.2E-09	1.1E-06
Loss of CCW/ESW	1.5E-8	5.6E-7	2.2E-08	6.0E-07
Loss of offsite power	6.5E-8	1.4E-6	8.6E-08	2.4E-06
Total CDF	2.2E-7	2.1E-5	1.8E-07	8.1E-06
Maximum averted cost ($) (*1)	289K	1,598K	288K	786K

(*1) SAMDA evaluation is based on the DCD ER Rev.2 (MUAP-DC021, Rev. 2) and addresses only the LPSD evaluation results.

As shown in Table 19-520-1, the SAMDA evaluation results applying the conditional CDF based on the Technical Specification restrictions show that the maximum averted cost is less than the lowest price design alternative ($870K) and none of the SAMDA candidates are evaluated as cost-beneficial.

Reference:

(1) EPRI 1003113, An Analysis of Loss of Decay Heat Removal Trends and Initiating Event Frequencies (1989-2000), November 2001

(2) EPRI 1003465, Low Power and Shutdown Risk Assessment Benchmarking Study, December 2002

Impact on DCD

There is no impact on the DCD.

Impact on R-COLA

There is no impact on the R-COLA.

Impact on S-COLA

There is no impact on the S-COLA.

Impact on PRA

There is no impact on the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

7/12/2011

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: **NO. 752-5614 REVISION 0**

SRP SECTION: **19 – Probabilistic Risk Assessment and Severe Accident Evaluation**

APPLICATION SECTION: **SRP Chapter 19**

DATE OF RAI ISSUE: **5/3/2011**

QUESTION NO.: 19-521

Follow-up to Question 19-448:
In response to RAI 627-4926 Rev 2 (Question No: 19-448), the applicant performed sensitivity calculations using two different node sizes, 0.6 m (Model 1) and 0.2 m (Model 2), as compared to node size of 0.3 m used for the U.S. APWR design certification (DCD) calculations.

The applicant also performed several calculations to study the sensitivity to fragmentation model parameters. In the response to RAI, the applicant justified the ranges for the fragmentation model parameters based on the results of the benchmark studies for the original TEXAS-V code against the KROTOS experiment data. However, the jet break-up model used in the MHI-version of the TEXAS-V code is considerably different from the original TEXAS-V model. Therefore, the ranges of the fragmentation model parameters that are found to be appropriate for the original TEXAS-V model may not be applicable to the MHI-version of the code. Furthermore, the applicant's analysis showed that the peak cavity pressure is almost insensitive to one of the fragmentation model parameters. However, the NRC calculations have shown a much larger sensitivity in the estimated peak pressure and impulse loads over the same range of the selected fragmentation model parameters using the original (unmodified) TEXAS-V code.

The RAI response also stated that for the U.S. APWR design certification analysis, the evaluated pressure is increased by 10% in order to take into account the modeling uncertainties. Considering the above noted differences between the results of the original and the MHI-version of the TEXAS-V computer code calculations, the use of pressure/pressure impulse history predicted by the original TEXAS-V code in the U.S. APWR cavity structural analysis may lead to significantly lower margin between the calculated plastic strain and the maximum allowable strain. Therefore, please investigate the implications of using a larger range of uncertainties in the calculated peak pressure/pressure impulse (i.e., as much as 50% instead of the MHI-assumed 10%) associated with the steam explosions-induced dynamic loads on the cavity structural failure probability.

ANSWER:

Response of the main coolant pipes (MCP) and the wall structure in the reactor cavity are evaluated by employing the FEM techniques. The applied steam explosion load is increased by 50% from the original TEXAS-V calculation result addressing the various uncertainties involved in the steam explosion phenomena, as suggested by the NRC staff.

The evaluation results are presented in Appendix A to this RAI response.

As shown in Appendix A evaluation result, it is identified that steam explosion and the related structural capability issue involves a lot of uncertainties. Therefore, in order to address the various uncertainties involved in the steam explosion phenomena, a probabilistic evaluation is also performed. In the current Level 2 PRA model, the probability of containment failure due to steam explosion is assumed 0.01. The sensitivity of this failure probability is evaluated for the following two cases, and the evaluation results are shown in Table 19-521-1 and 2 for case (1) and (2), respectively.

Case (1): Probability of containment failure due to steam explosion: 0.1

Case (2): In addition to Case (1), containment failure due to RWSP hydrogen burn is assumed 1.0

As shown in the summary table, the NRC's probabilistic target for LRF to be less than 10^{-6}/ry is satisfied for both sensitivity cases.

Table 19-521-1 Sensitivity case (1) evaluation result

DCD Rev. 3	Base case			Sensitivity case (1)			Delta of LRF
	CDF	LRF	CCFP	CDF	LRF	CCFP	
Internal	1.03E-06	1.07E-07	0.10	1.03E-06	1.58E-07	0.15	47.8%
Fire	8.60E-07	1.87E-07	0.22	8.60E-07	2.16E-07	0.25	15.7%
Flood	8.91E-07	1.56E-07	0.18	8.91E-07	2.22E-07	0.25	42.3%
LPSD	1.80E-07	1.80E-07	1.00	1.80E-07	1.80E-07	1.00	0.0%
Total	2.96E-06	6.29E-07	0.21	2.96E-06	7.76E-07	0.26	23.2%
Total(Except LPSD)	2.78E-06	4.49E-07	0.16	2.78E-06	5.96E-07	0.21	32.6%

Table 19-521-2 Sensitivity case (2) evaluation result

DCD Rev. 3	Base case			Sensitivity case (1)			Delta of LRF
	CDF	LRF	CCFP	CDF	LRF	CCFP	
Internal	1.03E-06	1.07E-07	0.10	1.03E-06	2.22E-07	0.22	108.3%
Fire	8.60E-07	1.87E-07	0.22	8.60E-07	2.75E-07	0.32	47.0%
Flood	8.91E-07	1.56E-07	0.18	8.91E-07	2.22E-07	0.25	42.3%
LPSD	1.80E-07	1.80E-07	1.00	1.80E-07	1.80E-07	1.00	0.0%
Total	2.96E-06	6.29E-07	0.21	2.96E-06	8.99E-07	0.30	42.8%
Total(Except LPSD)	2.78E-06	4.49E-07	0.16	2.78E-06	7.19E-07	0.26	59.9%

Impact on DCD

There is no impact on the DCD.

Impact on R-COLA

There is no impact on the R-COLA.

Impact on S-COLA

There is no impact on the S-COLA.

Impact on PRA

There is no impact on the PRA.

Appendix-A Sensitivity analysis against ex-vessel steam explosion

To investigate the sensitivity of structural response against ex-vessel steam explosion, structural analysis against pressure multiplied by 110%, 150% (corresponding to range of uncertainties: 10%, 50%) was conducted.
The relationship between magnificent ratio of pressure and maximum plastic strain are shown in Figure A-1 (reactor coolant system pipe), Figure A-2 (reactor cavity wall model (a)) and Figure A-3 (reactor cavity wall model (b)). By increasing magnificent ratio of pressure from 110% to 150%, about twice maximum plastic strain occurred. The RCS pipe structures have sufficient capability to withstand the challenge by postulated ex-vessel steam explosion pressure load multiplied by 150%. However for the reactor cavity wall, the analysis showed that there is a possibility that the structural integrity cannot be maintained depending on the analysis model and partial collapse may be anticipated; although, the area of collapse is very limited. The reactor cavity wall model (a), which is originally reported in the PRA Report Ch. 15, shows that the plastic stain from 150% load exceeds the acceptance criterion. On the other hand, the reactor cavity wall model (b), which is another model developed during the study but not reported in the PRA Report Ch. 15, shows that the plastic strain is evaluated below the acceptance criterion. In the analysis using the model (a), it can be considered that some triangular elements smaller than other nearby elements may reduce the accuracy of continuity of steel plate deformation, accordingly unrealistically large plastic strain seems to occur very locally. On the other hand, regarding the model (b), because the mesh partitions are uniformly arranged as quadrangular elements in consideration of the direction of stress, the analysis using the model (b) seems to evaluate actual structural response with higher accurately.

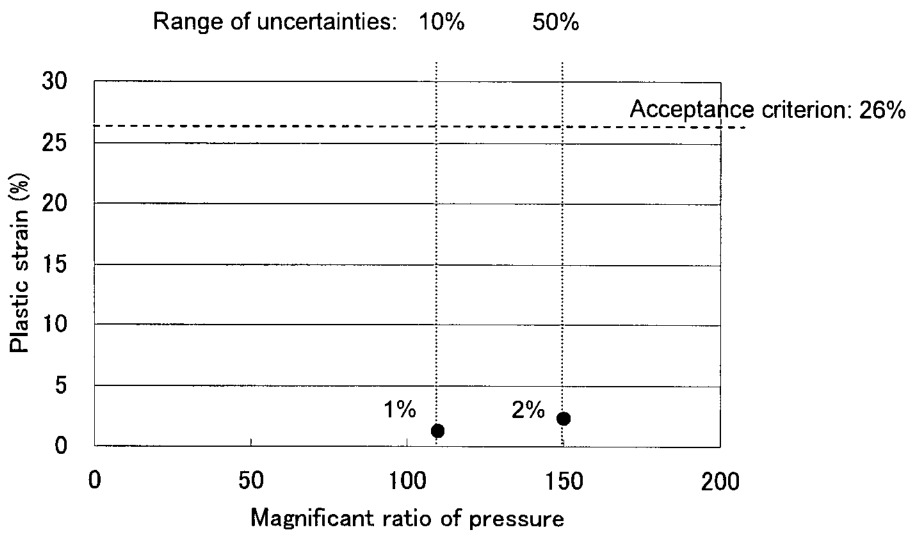


Figure A-1 Maximum plastic strain of RCS pipes

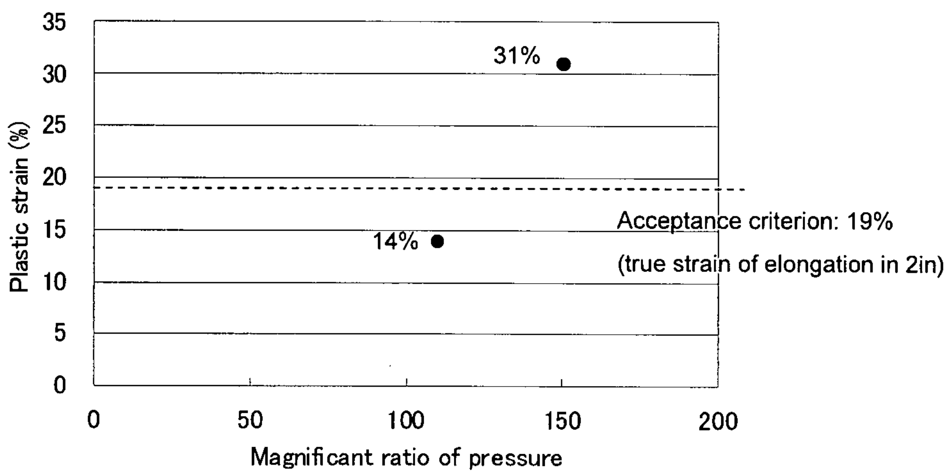


Figure A-2 Maximum plastic strain of reactor cavity wall model (a)

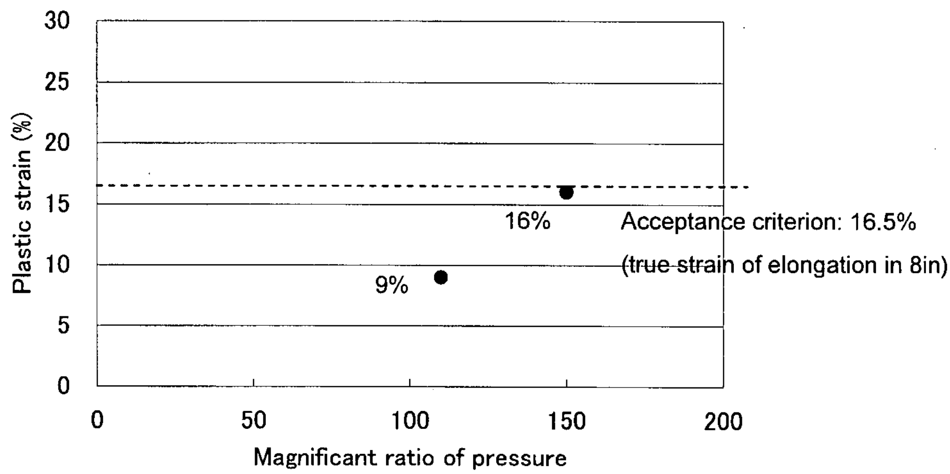


Figure A-3 Maximum plastic strain of reactor cavity wall model (b)

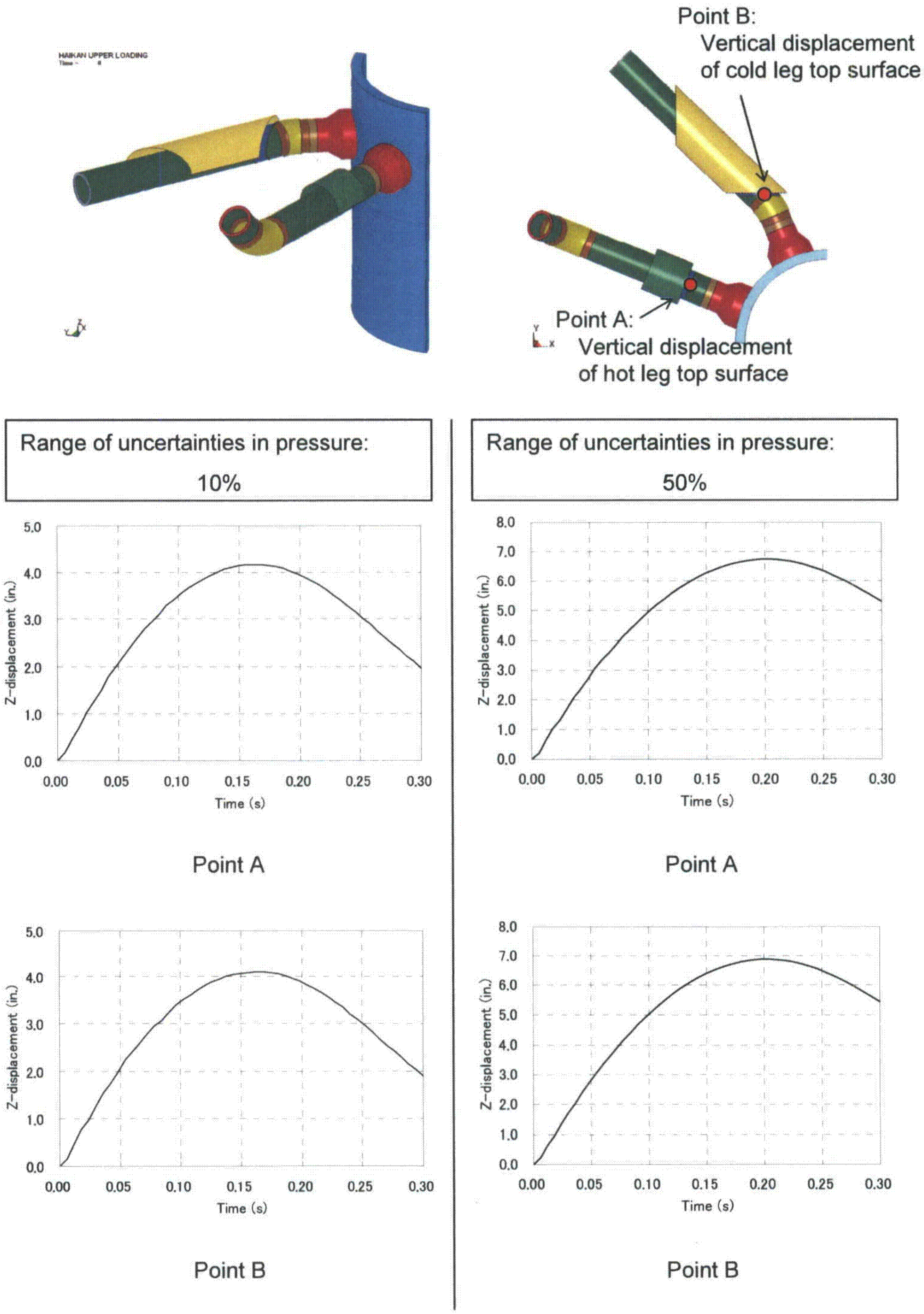
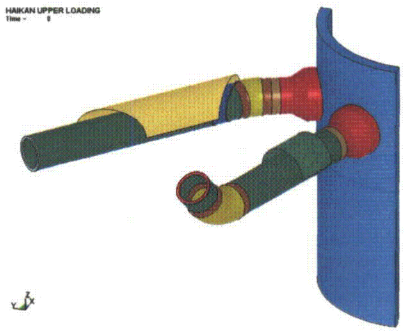
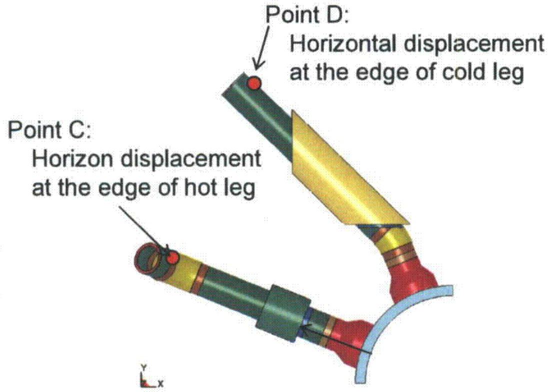
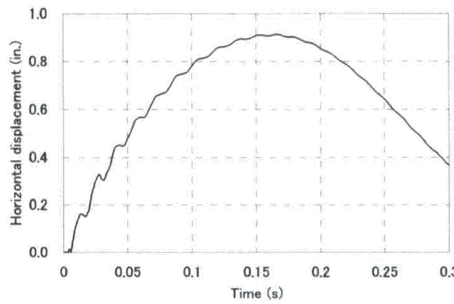


Figure A-4 Time profile of vertical displacement of RCS pipes

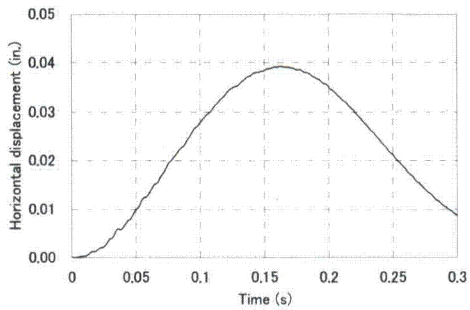




Range of uncertainties in pressure:
10%

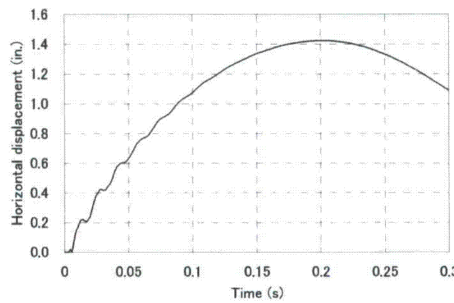


Point C

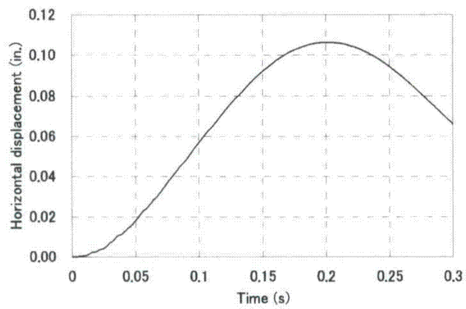


Point D

Range of uncertainties in pressure:
50%

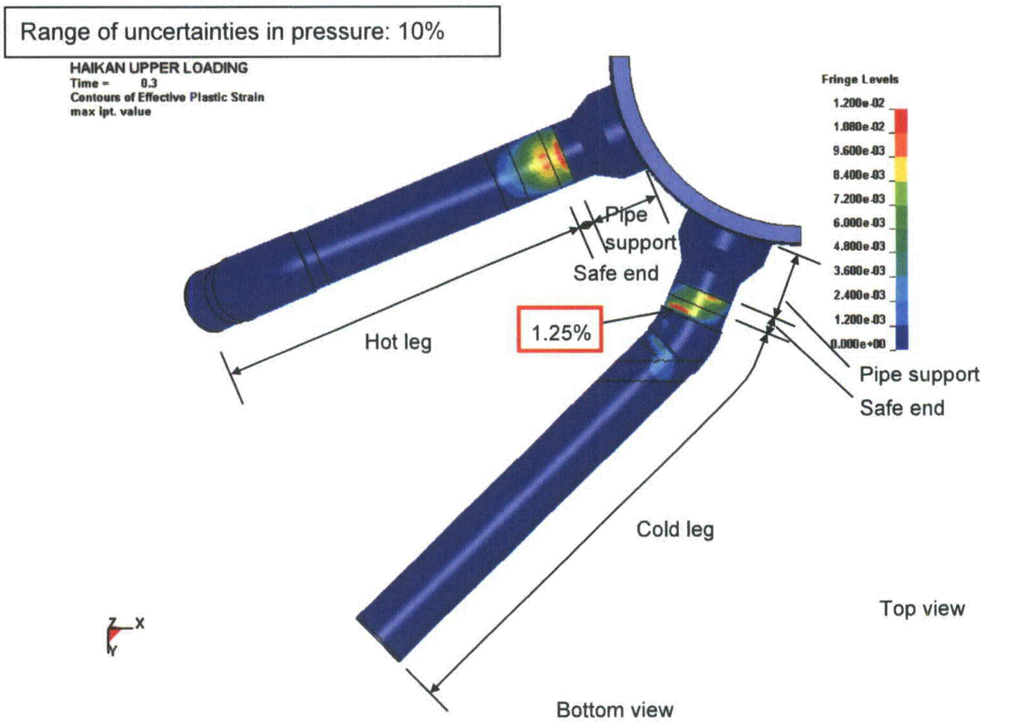


Point C



Point D

Figure A-5 Time profile of horizontal displacement of RCS pipes



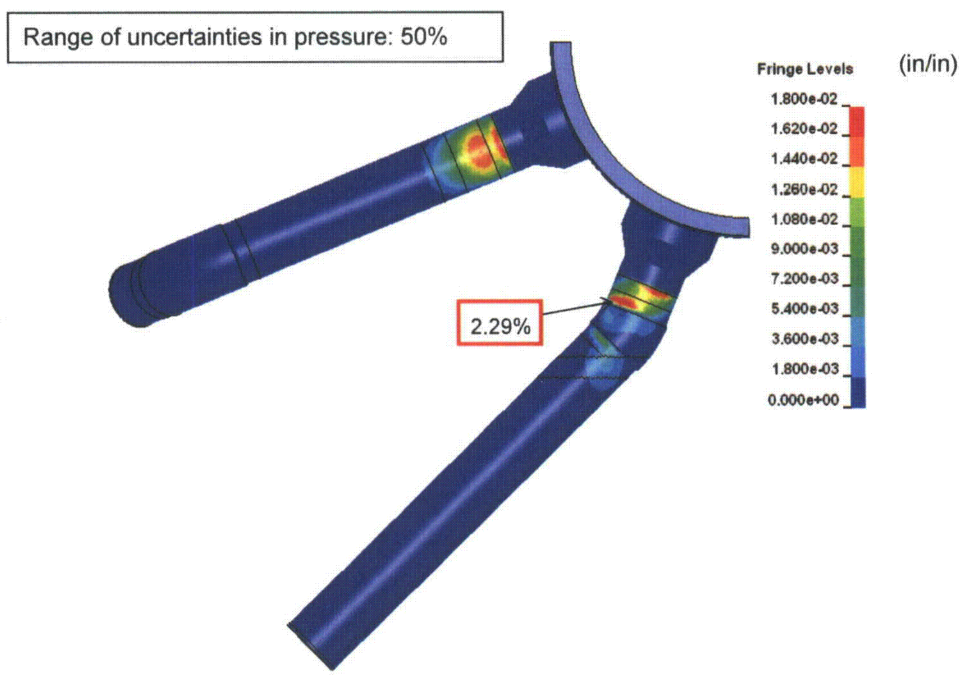
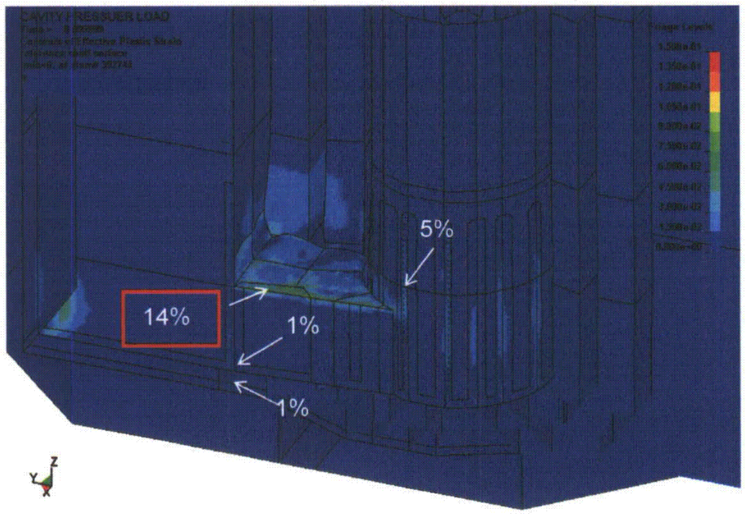


Figure A-6 Plastic strain contour of RCS pipes (T=0.3(s))

Range of uncertainties in pressure: 10%



Range of uncertainties in pressure: 50%

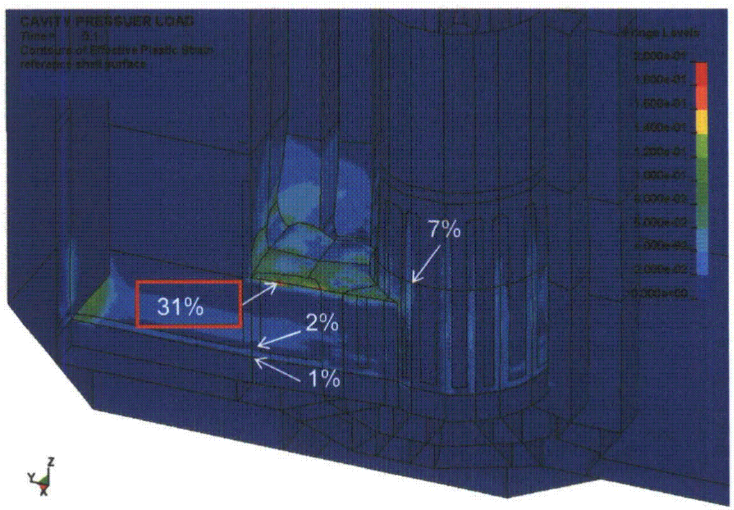
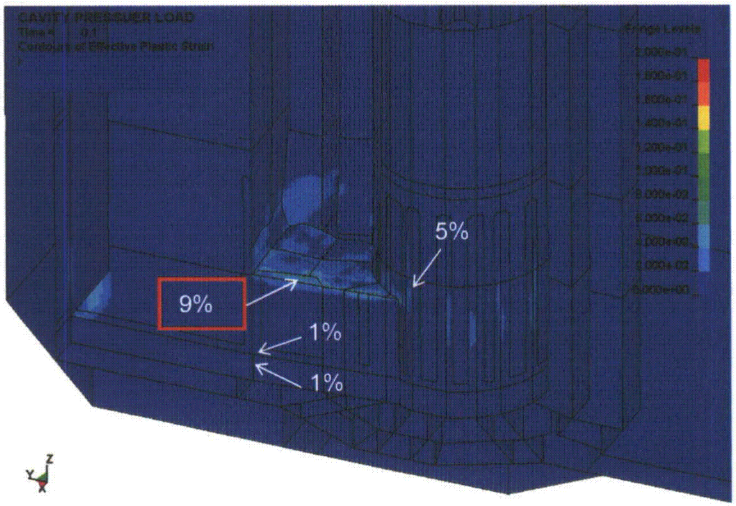


Figure A-7 Plastic strain contour of steel plate (t = 0.1sec, membrane strain) for reactor cavity wall model (a)

Range of uncertainties in pressure: 10%



Range of uncertainties in pressure: 50%

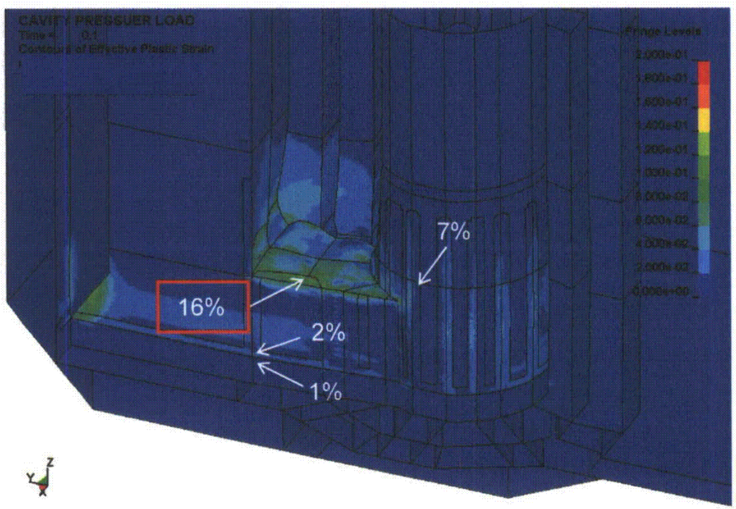


Figure A-8 Plastic strain contour of steel plate (t = 0.1sec, membrane strain) for reactor cavity wall model (b)