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TOKYO, JAPAN

October 27, 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-11364

Subject: MHI's Responses to US-APWR DCD RAI No.832-6034 Revision 3 (SRP 19)

References: 1) "Request for Additional Information No. 832-6034 Revision 3, SRP Section: 19 – Probabilistic Risk Assessment and Severe Accident Evaluation," dated September 27, 2011.

With this letter, Mitsubishi Heavy Industries, Ltd. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") a document entitled "Responses to Request for Additional Information No. 832-6034 Revision 3".

As indicated in the enclosed materials, this submittal 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 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 submittals. His contact information is below.

Sincerely,

Y. Ogata

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

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Enclosure:

1. Affidavit of Yoshiki Ogata
2. Responses to Request for Additional Information No. 832-6034 Revision 3 (Proprietary Version)
3. Responses to Request for Additional Information No. 832-6034 Revision 3 (Non-Proprietary Version)

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

<u>Contact Information</u>

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Enclosure 1

Docket No. 52-021
MHI Ref: UAP-HF-11364

MITSUBISHI HEAVY INDUSTRIES, LTD.

AFFIDAVIT

I, Yoshiki Ogata, state as follows:

1. I am Group Manager, Licensing Promoting Group in 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 "Responses to Request for Additional Information No. 832-6034, Revision 3" dated October 2011, and have determined that portions of the document contain proprietary information that should be withheld from public disclosure. Those pages contain 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 design and methodology developed by MHI for performing the design of the US-APWR reactor

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 methodology related to the analysis.

B. Loss of competitive advantage of the US-APWR created by the benefits of the modeling information.

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 27th day of October 2011.

Y. Ogata

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

Docket No. 52-021
MHI Ref: UAP-HF-11364

Enclosure 3

UAP-HF-11364
Docket Number 52-021

Responses to Request for Additional Information No.832-6034 Revision 3

October, 2011
(Non-Proprietary)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

10/27/2011

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: NO. 832-6034 REVISION 3

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

APPLICATION SECTION: 19

DATE OF RAI ISSUE: 09/27/2011

QUESTION NO.: 19-550

The staff reviewed the uncertainty results for POSs 4-3 and 8-1 in Chapter 19 of the DCD. It does not appear that uncertainty from human errors, which often drives shutdown risk, was quantified in the results. The error factors (the square root of the ratio of the 95th percentile to the 5th percentile) are estimated as 4.3 and 4.2 respectively. Based on Chapter 9 of the PRA which describes the human reliability assessment, the error factors for all human error probabilities are assumed to be five. In some shutdown cutsets, there are 3 human errors. Also, the staff noted that the applicant did not develop a table for documenting how key sources of uncertainty were addressed in the low power and shutdown PRA. Therefore, the staff requests the applicant to:

1. Requantify the low power and shutdown results (including internal and external) including the uncertainty from human errors and document the results in Chapter 19 of the DCD and the low power and shutdown PRA..
2. Add a table in Chapter 19 of the DCD listing the key sources of uncertainty in the low power and shutdown PRA (including internal and external events) and how these sources of uncertainty were addressed in the PRA. Potential sources of uncertainty include: human error probabilities, the duration in hours of POS 4-3 and POS 8-1, the frequency of low power initiating events, and equipment outages (such as safety injection, charging, etc.)

ANSWER:

1.

As stated above, the error factor for each human error probability (HEP) is assumed to be five. The error factor has no dependency among other HEPs or failure probability of their relevant components. MHI has determined that the error factor for the total core damage frequency (CDF) is less than that of each HEP or component failure probability. Therefore, MHI re-quantified the uncertainty from human errors, as shown below.

With regard to the impact to the risk during low-power and shutdown (LPSD) operation from human error described in the DCD, sensitivity analyses for internal events PRA based on the lower (5%) and upper (95%) value were performed in order to study the uncertainty in each estimated HEP. Table 19.550-1 lists upper and lower HEPs, the re-quantified CDFs, and the ratio of the re-quantified CDFs to the base case CDF in DCD Rev.3 (1.8E-07/RY). In all of the sensitivity cases, the ratio was estimated to be less than five.

The following operator errors have impact on the LPSD risk.

1. **HPIOO02S** (Failure to start standby safety injection pumps)

2. **ACWOO02SC** (Failure to establish alternate charging pump cooling line by fire suppression water supply system and start the charging pump)

3. **EPSOO02RDG** (Failure to connect alternate ac power sources to Class 1E ac buses during a station blackout event)

4. **CHIOO02P+RWS** (Failure to recover reactor coolant system [RCS] inventory and to change the charging pump intake from the refueling water storage auxiliary tank [RWSAT] to refueling water storage pit [RWSP])

For the internal flood and internal fire PRA, sensitivity analyses were also performed using the four human errors impacting LPSD risk above. For all of the above human errors, the base case CDF was estimated using the lower HEP, because frequent operator training is required for these operations as discussed in DCD Table 19.1-119. Table 19.550-2 summarizes the sensitivity analysis results using upper HEPs for the above human errors. For all cases assuming an upper (95%) HEPs, the CDFs were estimated to be less than 1.0E-06/RY.

For the internal events PRA and internal fire LPSD PRA, human errors are the most dominant contributor to CDF and the uncertainty from the human errors should be captured to apply PRA for risk-informed decision. On the other hand, for the internal flood LPSD PRA, since equipment configuration including the outage equipment and its combination are more significant risk contributors in comparison to human errors, the uncertainty contribution from human errors has a relatively small impact on the risk

Considering this, uncertainty from not only human errors, but also equipment configuration including outage equipment and its combination need to be considered as key sources in the LPSD PRA. Also, the assumed frequent training of the aforementioned operator actions (and the correlating assumption for lower HEP) contributes some uncertainty to the LPSD PRA.

The sensitivity analyses results and table summarizing key sources of uncertainty will be incorporated, as shown in the attached markups.

2.

MHI agrees with the premise of the RAI question and will insert a description of how key sources of uncertainty are addressed and include a table listing the key sources of uncertainty during LPSD operation which will include discussion regarding Question 1 in this RAI response.

US-APWR PRA uses various assumptions for the unreliability of unique design features such as the gas turbine generator, digital I&C system, equipment configurations, duration of plant shutdown, operator actions to prevent or mitigate initiating events (including the failure probabilities) ,etc. While the assumptions are decidedly conservative, they may have large uncertainty resulting in large contribution to LPSD risk.

Table 19.550-3 lists the key sources of uncertainty that may have impact on the PRA results and types of uncertainty. The assessed areas of uncertainty are categorized into three types: one is parametric uncertainty associated with parametric values, second is completeness uncertainty associated with the possibility of unaccounted for initiating events, and the other is modelling uncertainty made in developing the PRA model. For these assumptions, uncertainty analysis and sensitivity analyses assuming various reliabilities were performed to clarify the contribution to the LPSD risk.

Table 19.550-3 and all of sensitivity analyses results that provide a quantitative assessment for each assumption will be documented in the next DCD revision, as shown in the attached markups.

Table 19.550-1 Sensitivity Analysis Results for Internal Events PRA

#	Basic Event ID	HEP (DCD R3)	Lower Condition			Upper Condition		
			HEP (Lower)	CDF (/RY)	Ratio	HEP (Upper)	CDF (/RY)	Ratio
Case 1	ACWOO02SC	2.2E-02	2.2E-02	NA[NOTE1]	NA[NOTE1]	5.5E-01	7.3E-07	4.1
	ACWOO02SC-DP2	7.1E-02	7.1E-02			5.7E-01		
Case 2	CHIOO02RWS	1.7E-02	1.7E-02	NA[NOTE1]	NA[NOTE1]	4.4E-01	2.5E-07	1.4
	CHIOO02RWS-DP2	6.7E-02	6.7E-02			4.6E-01		
	CHIOO02RWS-DP3	1.6E-01	1.6E-01			5.2E-01		
Case 3	CHIOO02P	2.5E-03	3.2E-04	1.6E-07	0.9	7.9E-03	2.3E-07	1.3
Case 4	CHIOO02P+RWS	1.9E-02	1.9E-02	NA[NOTE1]	NA[NOTE1]	4.8E-01	3.3E-07	1.8
	CHIOO02P+RWS-DP2	6.8E-02	6.8E-02			5.0E-01		
	CHIOO02P+RWS-DP3	1.6E-01	1.6E-01			5.5E-01		
	CHIOO02P+RWS-DP4	5.1E-01	5.1E-01			7.4E-01		
Case 5	EPSOO02RDG	2.1E-02	2.1E-02	NA[NOTE1]	NA[NOTE1]	5.2E-01	5.1E-07	2.8
Case 6	HPIOO02S	4.9E-03	4.9E-03	NA[NOTE1]	NA[NOTE1]	1.2E-01	7.5E-07	4.1
	HPIOO02S-DP2	5.5E-02	5.5E-02			1.7E-01		
	HPIOO02S-DP3	1.5E-01	1.5E-01			2.5E-01		
Case 7	LOAOO02LC	2.5E-03	3.2E-04	1.7E-07	0.9	7.9E-03	2.1E-07	1.2
Case 8	LOAOO02OD	3.8E-03	4.7E-04	1.8E-07	1.0	1.2E-02	1.8E-07	1.0
Case 9	RSSOO02LINE+P	3.8E-03	4.7E-04	1.6E-07	0.9	1.2E-02	2.2E-07	1.2

Table 19.550-1 Sensitivity Analysis Results for Internal Events PRA

#	Basic Event ID	HEP (DCD R3)	Lower Condition			Upper Condition		
			HEP (Lower)	CDF (/RY)	Ratio	HEP (Upper)	CDF (/RY)	Ratio
Case 10	HPIOO01001A	3.9E-02	4.9E-03	1.8E-07	1.0	1.2E-01	1.8E-07	1.0
	HPIOO01001B	3.9E-02	4.9E-03			1.2E-01		
	HPIOO01001C	3.9E-02	4.9E-03			1.2E-01		
	HPIOO01001D	3.9E-02	4.9E-03			1.2E-01		
	RSSOO01CSS001A	3.9E-02	4.9E-03			1.2E-01		
	RSSOO01CSS001B	3.9E-02	4.9E-03			1.2E-01		
	RSSOO01CSS001C	3.9E-02	4.9E-03			1.2E-01		
	RSSOO01CSS001D	3.9E-02	4.9E-03			1.2E-01		
	RWSOO01RWAT	3.9E-02	4.9E-03			1.2E-01		
Case 11	SWSOO01ST001	8.7E-03	1.1E-03	1.8E-07	1.0	2.7E-02	1.8E-07	1.0
Case 12	RSSOO02P	2.5E-03	3.2E-04	1.4E-07	0.8	7.9E-03	2.7E-08	1.6
	RSSOO02P-DP2	5.2E-02	5.0E-02			5.8E-02		
Case 13	EPSOO01UATRAT	1.6E-02	2.0E-03	1.8E-07	1.0	5.0E-02	1.8E-07	1.0
Case 14	CHIOO01RECOV	5.8E-02	7.3E-03	1.8E-07	1.0	1.8E-01	1.8E-07	1.0

Note 1: Base case is assumed to be a lower HEP, and the CDF is equivalent to the base case.

Table 19.550-2 Sensitivity Analysis Results for Internal and External Events PRA

#	Basic Event ID	HEP		CDF [/RY]			
		DCD R3 (Lower)	Upper	Internal Events	Internal Flood	Internal Fire	Total
Case A	ACWOO02SC	2.2E-02	5.5E-01	7.3E-07	1.1E-07	2.5E-08	8.6E-07
	ACWOO02SC-DP2	7.1E-02	5.7E-01				
Case B	CHIOO02P+RWS	1.9E-02	4.8E-01	3.3E-07	9.5E-08	2.4E-08	4.5E-07
	CHIOO02P+RWS-DP2	6.8E-02	5.0E-01				
	CHIOO02P+RWS-DP3	1.6E-01	5.5E-01				
	CHIOO02P+RWS-DP4	5.1E-01	7.4E-01				
Case C	EPSOO02RDG	2.1E-02	5.2E-01	5.1E-07	9.5E-08	1.0E-07	7.0E-07
Case D	HPIOO02S	4.9E-03	1.2E-01	7.5E-07	1.0E-07	1.4E-07	9.9E-07
	HPIOO02S-DP2	5.5E-02	1.7E-01				
	HPIOO02S-DP3	1.5E-01	2.5E-01				
Base Case in DCD R3				1.8E-07	9.5E-08	1.8E-08	2.9E-07

In the DCD base case, HEPs listed in this table are assumed to be lower value due to their frequent training for the operators.

Table 19.550-3 Key Sources of Uncertainty and Key Assumptions (LPSD Operation) (Sheet 1 of 5)

Key Sources of Uncertainty and Key Assumptions		Type (Note)	Summary Results of Qualitative Assessments	Quantitative Approach
Unique Equipments and their Duty to the US-APWR Design	Gas Turbine Generators	M	Sensitivity analyse of failure probability, failure rates and CCF parameters were performed.	Sensitivity Analysis (Case 1-1, 1-2)
	Digital I&C	M	Actuation of automatic signals and operator actions use the digital I&C. Uncertainty from CCF of basic software and application software impact reliability of these signals and operator actions. Sensitivity analyses of various failure probabilities of application and basic software CCF for digital I&C were performed.	Sensitivity Analysis (Case 6-1, 6-2)
Initiating Event Analysis	Initiating event frequency of loss of RHR caused by other failures (LORH) and loss of CCW/essential service water (LOCS)	M	Initiating event frequency of loss of RHR caused by other failures and loss of CCW/essential service water depends on equipment outage. Sensitivity analysis assuming no planned maintenance was performed.	Sensitivity Analysis (Case 3-1)
	Statistic uncertainty of initiating event frequency	P	(Statistical uncertainty is considered.)	Uncertainty Analysis
	Completeness of initiating events to the US-APWR design	C	Rare initiating events to the US-APWR design are assessed.	NA
	Outage types and their frequencies	M	Since human errors are the most dominant contributor to internal event risk during LPSD operation, the sources have less impact on the risk in comparison with the human errors. Sensitivity analysis assuming different outage types and their frequencies from the base case was performed.	Sensitivity Analysis (Case 3-2)
	Duration of plant shutdown	M	Sensitivity analysis using different duration from the base case was performed.	Sensitivity Analysis (Case 5-1)

Table 19.550-3 **Key Sources of Uncertainty and Key Assumptions (LPSD Operation) (Sheet 2 of 5)**

Key Sources of Uncertainty and Key Assumptions		Type (Note)	Summary Results of Qualitative Assessments	Quantitative Approach
Event Tree Analysis	Identification of accident sequences	M	Realistic accident sequences are considered.	NA
System Analysis	Plugging before events occurred is not modelled.	M	It would be hard to plug during LPSD operation in RCS and safety-related systems.	NA
	Class 1E electrical room HVAC are reliable and do not impact risk.	M	Even if losses of HVAC occurs, automatic signals to start Class 1E GTGs or AAC, and to actuate low pressure letdown line isolatino will actually complete prior to occurrence of RCS boiling of an initiating event. To relax room heat up after losses of Class 1E electrical room HVAC, the operator will be open the room door and utilize available temporary fans.	NA
	Equipment outage	M	Since human errors are the most dominant contributor to internal event risk during LPSD operation, the sources have less impact on the risk in comparison with the human errors. Sensitivity analysis assuming different outage types and their frequencies from the base case was performed.	Sensitivity Analysis (Case 3-1)

Table 19.550-3 Key Sources of Uncertainty and Key Assumptions (LPSD Operation) (Sheet 3 of 5)

Key Sources of Uncertainty and Key Assumptions		Type (Note)	Summary Results of Qualitative Assessments	Quantitative Approach
Data Analysis	Applicability of failure modes to the US-APWR equipment design	M	Potentially valuable generic data sources were collected. All the failure modes of the US-APWR component types were considered.	NA
	Failure probability and failure rates for diesel generators are applied to gas turbine generators.	M	Sensitivity analysis of failure probability and failure rates was performed.	Sensitivity Analysis (Case 1-1)
	Statistical uncertainty of failure rate	P	(Statistical uncertainty is considerable.)	Uncertainty Analysis
	Failure probability of digital I&C system	M	Actuation of automatic signals and operator actions use the digital I&C. Uncertainty from CCF of basic software and application software impact reliability of these signals and operator actions. Sensitivity analyses of various failure probabilities of application and basic software CCF for digital I&C were performed.	Sensitivity Analysis (Case 6-1, 6-2)
	Reliability of components	M	There is no plant-specific reliability data for the US-APWR. In the design stage, it is probable that the reliability of components of a newly designed plant is comparable to the component reliability of operating US plants. Therefore, US generic data is applicable.	NA

Table 19.550-3 Key Sources of Uncertainty and Key Assumptions (LPSD Operation) (Sheet 4 of 5)

Key Sources of Uncertainty and Key Assumptions		Type (Note)	Summary Results of Qualitative Assessments	Quantitative Approach
Common Cause Failure Analysis	CCF parameters of emergency diesel generators are applied to gas turbine generators.	M	Sensitivity analysis for the gas turbine generator CCF parameters was performed.	Sensitivity Analysis (Case 1-2)
	Statistical uncertainty of CCF probabilities.	P	(Statistical uncertainty is considerable.)	Uncertainty Analysis
	CCF for continually operating pumps	M	There is data published for CCF of continually operating pumps. Based on engineering judgment, the PRA applies a CCF parameter lower than those reported in the NUREGs for the CCW and ESW pumps. Uncertainty associated with the CCF parameters for continually running pumps impact the initiating event frequency for loss of CCW, which has large contribution to the CDF. The PRA treats CCF for continually running pumps and standby pumps alike and applies a value of 0.1. This value is deemed a conservative estimation since the running pumps and the standby pumps are initially in an asymmetric configuration.	NA

Table 19.550-3 Key Sources of Uncertainty and Key Assumptions (LPSD Operation) (Sheet 5 of 5)

Key Sources of Uncertainty and Key Assumptions		Type (Note)	Summary Results of Qualitative Assessments	Quantitative Approach
Human Reliability Analysis	Human error probability	M	Sensitivity analyses for post initiating event operator action failure probabilities were performed to study the impact of human errors to CDF, assuming a HEP of 0.0, lower value, mean value and upper value.	Sensitivity Analysis (Case 4-1, 4-2, 4-3)
	Statistical uncertainty of human error probability	P	(Statistical uncertainty is considered).	NA
	Visual display unit (VDU) interaction	M	A sensitivity analysis was performed assuming changing windows on the display is ineffective for reducing dependencies between actions and not be considered as actions performed in different locations.	Sensitivity Analysis (Case 4-6)
	Frequent training of operator actions	M	Sensitivity analysis assuming operators perform less frequent training was carried out.	Sensitivity Analysis (Case 4-2)
	Dependency among operator actions	M	Sensitivity analyses assuming varying dependency levels among operator actions were performed.	Sensitivity Analysis (Case 4-4, 4-5, 4-6)
Shutdown Condition	Outage types and their frequencies	M	Sensitivity analysis assuming different outage types and frequencies were performed.	Sensitivity Analysis (Case 3-2)
	Duration of plant shutdown	M	Sensitivity analysis using durations based on Japanese operating experience was performed.	Sensitivity Analysis (Case 5-1)
Note – Uncertainty sources are categorized into three types, Parametric (P), Modeling (M) or Completeness (C).				

Impact on DCD

Description of how the key sources of uncertainty for LPSD PRA were addressed and table summarizing the key sources will be inserted in the DCD next revision with the relevant sensitivity analysis results, as shown in the attached markups.
(See Attachment-1)

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

10/27/2011

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: **NO. 832-6034 REVISION 3**

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

APPLICATION SECTION: **19**

DATE OF RAI ISSUE: **09/27/2011**

QUESTION NO.: 19-551

The staffs requests information on how the automatic isolation of low pressure letdown (a key risk feature) was incorporated into the risk importance analyses. The initiating frequency of loss of RHR due to OVDR was evaluated by considering the automatic isolation failure of the low-pressure letdown line. It is estimated by quantifying the failure of the the loop level level signal and failure of the air operated valve to close.

Since it appears that low power and shutdown initiating events were not included in the risk achievement worth analyses (RAW), the staff is concerned that the RAW value of the automatic isolation feature is greater than reported. Thus, the staff requests MHI to add low power and shutdown initiating events to the RAW analyses reported in Chapter 19 of the DCD and PRA. Also, the staff requests MHI to add components included in the initiating event frequency calculations to the RAW analyses (e.g. RCS loop low-level signal and failure of an air-operated valve to close).

ANSWER:

Importance measures listed in the DCD Tables 19.1-93, 19.1-94, (for POS 8-1) 19.1-164 and 19.1-165 (for POS 4-3) include components related to initiating events such as the RCS loop low-level signal or the low pressure letdown line isolation valve. The basis of how the importance measures for CDF were estimated is as follows:

Generally, the event heading of initiating event consists of only one basic event with a given initiating event frequency, which is applied to each initiating event; LOCA (loss-of-coolant accident), LORS (loss of RHR caused by other failure) , LOCS (loss of CCW/essential service water) and LOOP (loss of offsite power). On the other hand, the initiating events FLML (loss of RHR caused by failing to maintain water level) and OVDR (loss of RHR due to over-drain) are treated differently (with respect to their frequency) from the other aforementioned initiating events. As shown in Figures 19.551-1 and 19.551-2 (which are DCD Figures 19.1-17 and 19.1-24, respectively), the event heading for these initiating events consists of two basic events: One is system unavailability of the automatic isolation of low pressure letdown, and the other is system unavailability of chemical and volume control system (CVCS) for a FLML event, and human error probability for a OVDR event, respectively. The former event is estimated by a fault tree including failure of the RCS loop low-level signal from the digital I&C system and the air-operated isolation valves (RHS-AOV-024B and C) to close. The estimation for the latter is referred to MUAP-07030 Rev.3 "US-APWR Probabilistic Risk Assessment", Attachment 20B.1 (FLML) and Chapter 9 Table 9.3.2-1, respectively. Modelling the component failure in the event heading of initiating event enables initiating event frequencies and the relevant importance measure to estimate in quantification for core damage frequency (CDF).

This RAI question states that the RAW for the automatic isolation feature is greater than reported. RAW depends on POS because the initiating event likely to occur and equipment configuration is not the same throughout plant shutdown. The RAW for each POS or all POSs may be greater or less than reported. For example, the automatic isolation system is effective to prevent initiating event FLML or OVDR which are likely to occur during mid-loop operation of POSs 4 and 8. This means that the RAW is equivalent to 1.0 during POSs other than mid-loop operation.

In order to study the RAW for the automatic isolation of low pressure letdown, MHI estimated the RAW for each POS and all POSs. Figure 19.551-3 is a fault tree for the automatic isolation of low pressure letdown line. Failure of the RCS loop low-level signal or one of the two air-operated isolation valve to close will result in failure of the automatic isolation failure. With respect to a sensitivity analysis, this implies that the RAW for failure of RCS loop low-level signal or one air-operated isolation valve to close is equal to that of automatic isolation function and could be calculated as ratio of the quantified CDF, assuming no automatic isolation function, to the base case CDF. Table 19.551-1 lists the CDF for each POS and total CDF of the base case and sensitivity case assuming no automatic isolation function. In DCD Tables 19.1-94 and 19.1-165, the RAW for CVCAVCD024B/C in POSs 8-1 and 4-3 is estimated to be 5.1 (Table 19.1-94, Rank #621) and 42 (Table 19.1-165, Rank #159), respectively, which is congruent to the results in Table 19.551-1. Also, RAW for total CDF is estimated to be 12.6 and is between those of POSs 8-1 and 4-3. The RAW for the RCS loop low level signals is also equal to the ratio. The results will be documented in the next DCD revision, as shown in the attached markup.

Note:
RAW for the digital I&C system in DCD Tables 19.1-93 and 19.1-165 is not equal to the ratio of the estimated CDF and base case CDF. This is because the digital I&C system failure model includes the automatic isolation signal of low pressure letdown and human operations. The estimated RAW includes the risk importance measures regarding these failures.

Table 19.551-1 CDF for Each POS and Total CDF

POS	CDF [/RY]		Ratio (=RAW) Note 2
	Base	Sensitivity	
3 Note 1	1.3E-08	1.3E-08	1.0
4-1	1.3E-08	1.4E-07	10.3
4-2	5.3E-09	1.3E-07	24.8
4-3	3.0E-08	1.2E-06	41.6
8-1	8.0E-08	4.1E-07	5.1
8-2	5.6E-09	1.3E-07	23.4
8-3	1.1E-08	1.7E-07	16.5
9 Note 1	3.4E-09	3.4E-09	1.0
11 Note 1	1.8E-08	1.8E-08	1.0
TOTAL	1.8E-07	2.3E-06	12.6

Note 1: Automatic isolation of low pressure letdown is an effective function to reduce risk caused by FLML or OVDR. These initiating events occur during mid-loop operation of POSs 4 and 8.

Note 2: RAW of failure of RCS loop low-level signal and of air-operated isolation valve to close is equal to the ratio of the sensitivity CDF to the base case (in right column).

Figure 19.551-1 Loss of RHR caused by Over-drain Event Tree (Same as DCD Figure 19.1-17)

Figure 19.551-2 Loss of RHR caused by Failing to Maintain Water Level Event Tree (Same as DCD Figure 19.1-24)

Figure 19.551-3 Fault Tree of Automatic Isolation of Low Pressure Letdown

Impact on DCD

Risk importance measures for the digital I&C system will be documented in the next DCD revision shown in the attached markup. (See Attachment-1)

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.

Gas turbine generator reliability DCD_19-550

- Case ~~01~~1-1: Sensitivity to gas turbine generator failure rate

  This sensitivity study evaluates the impact of failure rate of the gas turbine generator on the CDF. For the base case study, the failure rate of the gas turbine generator is set to the failure rate of diesel generators described in NUREG/CR-6928 (Reference 19.1-16). In this sensitivity study, that failure rate is set to data of gas turbine generator described in NUREG/CR-6928.

  The sensitivity case produces a CDF of 2.0E-07/RY, which is an increase of 13 percent in the base case CDF of 1.8E-07/RY. Although a failure rate of gas turbine generator is ten times as high as one of diesel generator, it is indicated that the impact of failure rate of the gas turbine generator is small during plant shutdown conditions.

- Case 1-2: Sensitivity to gas turbine generator common cause failure DCD_19-550

  This sensitivity analysis estimates the impact of CCF parameter for the gas turbine generator on the CDF. For the base case, the CCF parameter of the gas turbine generator is set to that of diesel generators. In the sensitivity case, CCF parameters based on the generic parameter reported in NUREG/CR-5485 (Reference 19.1-24) is applied to the US-APWR gas turbine generators.

  The estimated CDF is 1.7E-07/RY, which is a decrease of 7.2 percent from the base case. The results indicate that the impact of CCF parameter for the gas turbine generator is small during plant shutdown conditions.

Initiating Event Frequency

- Case ~~02~~2-1: Sensitivity to the frequency of LOOP

  For this sensitivity case, in order to confirm how the CDF of LOOP is sensitive to total CDF, the frequency of the LOOP is set to be three times higher than the base case.

  The sensitivity case produces a CDF of 3.4E-07/RY, which is an increase of 91 percent in the base case CDF. For this reason, it is indicated that the LOOP in LPSD PRA has a small impact on total CDF.

Outage schedule DCD_19-550

- Case ~~03~~3-1: Sensitivity to the planned maintenance during the LPSD

  In the base case, some components or systems are unavailable due to the planned maintenance during the LPSD. The assumption of their planned maintenance used in the base case is documented in Table 19.1-83.

  This sensitivity study evaluates the impact not allowing the planned maintenance during the LPSD. In this sensitivity, unavailability due to the planned maintenance is not modeled for any component and system in the event trees. The schedule not allowing the planned maintenance for this sensitivity study is described in Table 19.1-92. This sensitivity is designed to assess the impact on the base case CDF, if some components and systems are not unavailable due to the planned maintenance.

This sensitivity case produces a CDF of 1.6E-07/RY, which is a decrease of 12 percent in the base case CDF. This result indicates that the assumption of the planned maintenance is not risk-important.

- Case ~~04~~3-2: Frequency of outages DCD_19-550

The PRA evaluates the LPSD risk from refueling outages scheduled every 24 months as a typical analysis case. Sensitivity studies were performed to evaluate the LPSD risk assuming different outage types and frequencies. The results and assumed conditions of the sensitivity cases are shown below.

Case	Shutdown frequency			LPSD CDF
	Refueling outages (Type C outage)	Forced outages with drain (Type B outage)	Forced outages without drain (Type C outage)	
Base case	0.5 /Y	-	-	1.8E-07 /RY
Case ~~04-1~~A	0.67 /Y	-	-	2.4E-07 /RY
Case ~~04-2~~B	0.5 /Y	0.5 /Y	0.29 /Y	3.3E-07 /RY
Case ~~04-3~~C	0.5 /Y	0.05 /Y	1.5 /Y	3.6E-07 /RY

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The first case, case ~~04-1~~A, evaluates the LPSD risk assuming a shorter refueling outage cycle. If refueling outages are scheduled every 18 months, the shutdown frequency will be 0.67 per year and the CDF increases to 2.4E-07 /RY. DCD_19-550

The second and third case, cases ~~04-02~~B and ~~04-3~~C, evaluates the impact of forced outages to the LPSD risk. In the sensitivity analysis, forced outages with drain are assumed to involve POS 3, POS 4-1, POS 4-2, POS 9 and POS 11. Forced outages without drain are assumed to involve only POS 3 and POS 11. DCD_19-550

Case ~~04-2~~B assumes force outages with drain to occur with a frequency of 0.5 per year. In this case, drained maintenance is performed once per year, either by refueling outage or forced outage. This gives a conservative condition for drained maintenance since US-APWR does not plan to perform steam generator inspection every year. The resulting CDF is 3.3E-07 /RY. DCD_19-550

Case ~~04-3~~C assumes forced outages without drain to occur with a frequency of 1.5 per year. The total frequency of shutdown per year is approximately two in this sensitivity case. The resulting CDF is 3.6E-07 /RY. DCD_19-550

Human error rate sensitivity DCD_19-550

- Case ~~05~~4-1: Sensitivity to human error probabilities set to 0.0

  This sensitivity study evaluates the impact of having perfect operators (i.e., setting all human error probabilities to 0.0 in the baseline shutdown core damage quantification).

  This sensitivity produces a CDF of 2.7E-08/RY, which is decrease of 85 percent in the base CDF. This indicates that the operator actions are risk important at the level of plant risk obtained from the base case study.

- Case ~~06~~4-2: All HEPs set to mean value DCD_19-550

  In this sensitivity analysis, mean HEPs, rather than lower bound value, are applied for human actions that will have frequent training. The resulting CDF is 7.9E-07/RY, which is 4.4 times of base case CDF.

- Case 4-3: HEP set to lower or higher value DCD_19-550

  This sensitivity analyses are performed to study uncertainty from each human error. The base case assumes that some HEPs is mean value and HEPs associated with frequent training in Table 19.1-119 have lower values. The results of sensitivity analyses assuming HEP of lower or upper value are summarized as follows:

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Basic Event ID	Lower Condition		Upper Condition	
	CDF [/RY]	Ratio NOTE 2	CDF [/RY]	Ratio NOTE 2
ACWOO02SC ACWOO02SC-DP2	NA NOTE 1	NA NOTE 1	7.3E-07	4.1
CHIOO02RWS CHIOO02RWS-DP2 CHIOO02RWS-DP3	NA NOTE 1	NA NOTE 1	2.5E-07	1.4
CHIOO02P	1.6E-07	0.9	2.3E-07	1.3

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Basic Event ID	Lower Condition		Upper Condition	
	CDF [/RY]	Ratio NOTE 2	CDF [/RY]	Ratio NOTE 2
CHIOO02P+RWS	NA NOTE 1	NA NOTE 1	3.3E-07	1.8
CHIOO02P+RWS-DP2				
CHIOO02P+RWS-DP3				
CHIOO02P+RWS-DP4				
EPSOO02RDG	NA NOTE 1	NA NOTE 1	5.1E-07	2.8
HPIOO02S	NA NOTE 1	NA NOTE 1	7.5E-07	4.1
HPIOO02S-DP2				
HPIOO02S-DP3				
LLOAOO02LC	1.7E-07	0.9	2.1E-07	1.2
LOAOO02OD	1.8E-07	1.0	1.8E-07	1.0
RSSOO02LINE+P	1.6E-07	0.9	2.2E-07	1.2
HPIOO01001A	1.8E-07	1.0	1.8E-07	1.0
HPIOO01001B				
HPIOO01001C				
HPIOO01001D				
RSSOO01CSS001A				
RSSOO01CSS001B				
RSSOO01CSS001C				
RSSOO01CSS001D				
RSWOO01RWAT				
SWSOO01RWAT	1.8E-07	1.0	1.8E-07	1.0
RSSOO02P	1.4E-07	0.8	2.7E-08	1.6
RSSOO02P-DP2				
EPSOO01UATRAT	1.8E-07	1.0	1.8E-07	1.0
CHIOO01RECOV	1.8E-07	1.0	1.8E-07	1.0

Note 1: Base case assumes lower HEP.
Note 2: Ratio of the sensitivity and base cases. DCD_19-550

For external event such as fire and flood, sensitivty analyses applying the HEPs that have impact on internal PRA to external PRA were also performed to study uncertainty from the human errors and the result are shown below.

Basic Event ID	CDF [/RY]			
	Internal	Flood	Fire	Total
ACWOO02SC	7.3E-07	1.1E-07	2.5E-08	8.6E-07
ACWOO02SC-DP2				
CHIOO02P+RWS	3.3E-07	9.5E-08	2.4E-08	4.5E-07
CHIOO02P+RWS-DP2				
CHIOO02P+RWS-DP3				
CHIOO02P+RWS-DP4				
EPSOO02RDG	5.1E-07	9.5E-08	1.0E-07	7.0E-07
HPIOO02S	7.5E-07	1.0E-07	1.4E-07	9.9E-07
HPIOO02S-DP2				
HPIOO02S-DP3				

- Case ~~07~~4-4: Sensitivity to dependency of human error to CD(complete dependency)

  This sensitivity study evaluates the impact of setting dependency level of human error to CD. That is, the sensitivity case most conservatively assumes that operator actions have a complete dependency on a previously failed action.

  This sensitivity produces a CDF of 8.6E-06/RY, which is approximately 48 times of the base CDF. This indicates that assumption of dependency of human error provide significant impact to result of PRA during shutdown, and the operators play a significant role in maintaining a very low CDF during shutdown conditions.

- Case ~~08~~4-5: Sensitivity to dependency of human error to ZD (zero dependency) DCD_19-550

  This sensitivity study evaluates the impact of setting dependency level of human error to ZD. That is, the sensitivity case most non-conservatively assumes that operator actions are independent absolutely between prior mitigation system and post mitigation system.

This sensitivity produces a CDF of 7.9E-08/RY, which is decrease of 56 percent in the base CDF. This indicates that assumption on dependency of human error provide meaningful sensitivity to result of PRA during shutdown.

- Case ~~09~~4-6: Sensitivity to higher dependency of human error DCD_19-550

This sensitivity study evaluates impact of setting higher dependency level between operator actions, which assumes that changing window on display is not effective. That is, dependency level is considered to be performed in the same location.

This sensitivity produces a CDF of 3.5E-07/RY, which is approximately 1.9 times of the base case CDF.

Duration during LPSD operation DCD_19-550

- Case ~~10~~5-1: Sensitivity to POS duration based on operational Japanese PWR plant Experience

This sensitivity study evaluates impact of POS duration based on operational Japanese PWR plant data. This postulated POS duration is shown in Table 19.1.82.

This sensitivity produces a CDF of 1.8E-07/RY, which is decrease of 2 percent in the base case. This indicates that the POS duration based on the Japanese data has a small impact on the shutdown risk.

Digital I&C reliability DCD_19-550

- Case 6-1: Common cause failure of application software

The base case assumes that application software CCF of safety system (i.e., PSMS) is 1.0E-05/demand. Since this probability has high uncertainty, sensitivity analyses concerning software CCF have been performed.

In this sensitivity analysis, CCF probability of application software used for operator actions and all signals such as automatic isolation of low power and shutdown, excluding that of the AAC system, is common and has no diversity. Application software CCF will therefore result in failure of operator actions and all signals modeled in the PRA besides that of the AAC. Three cases listed below were considered as part of the sensitivity analysis.

Case1: Application software CCF = 2.0E-05/demand

If application software CCFs are assumed to occur 2.0E-05 /demand, which is twice the value considered in the base case, the resulting CDF is 1.8E-07/RY. This value is 2.2% higher than the base case CDF.

Case 2: Application software CCF = 5.0E-05 /demand DCD_19-550

If application software CCFs are assumed to occur 5.0E-05 /demand, the CDF is 2.0E-07/RY, which is 9.4% higher than the base case CDF.

Case 3: Application software CCF = 1.0E-04 /demand

If application software CCFs are assumed to occur 1.0E-04 /demand, which is ten times of the base case, the CDF is 2.2E-07RY, which is 21% higher than the base case CDF.

Results of sensitivity analyses show that if the probability of software CCF that results in failure of all safety related signals operator actions and modeled in the PRA occur with a probability of 1.0E-04 /demand, which is ten times higher than the application software CCF probability assumed in the base case, the CDF is 2.2E-07/RY. This value is approximately 1.2 times the base case CDF.

- Case 6-2: Common cause failure of basic software

The base case assumes that basic software CCF probability is 1.0E-07/demand. Since this probability has high uncertainty, sensitivity analyses concerning basic software CCF have been performed to study the uncertainty.

Case1: Basic software CCF = 2.0E-07 /demand

If basic software CCFs are assumed to occur 2.0E-07/demand, which is twice the value considered in the base case, the resulting CDF is 1.8E-07/RY. This value is 0.12% higher than the base case CDF.

Case 2: Basic software CCF = 5.0E-07 /demand

If basic software CCFs are assumed to occur 5.0E-07/demand, the CDF is estimated to be 1.8E-07/RY, which is 1.1% higher than the base case CDF.

Case 3: Basic software CCF = 1.0E-06 /demand

If basic software CCFs are assumed to occur 1.0E-06/demand, the CDF is 1.9E-07/RY, which is 2.8% higher than the base case CDF.

The above results show that if the probability of basic software CCF, which causes failure of all automatic signals and operator actions using PSMS and PCMS, occurs with ten time times probability of the base case, the resulting CDF is 1.9E-07/RY. The results is approximately 3% higher than the base case.

Importance assessment has been respectively performed in POS 4-3 and POS 8-1 because detailed analyses of CDF are limited to POS 4-3 and POS 8-1 for the LPSD PRA. These analyses have been performed to determine the following:

- Basic event importance
- Common cause failure importance

Component importance

In this subsection, component (single failure of hardware) importance is documented.

The top ten FV importance of component basic events for POSs 4-3 and 8-1 are shown in Table 19.1-170 and Table 19.1-99, respectively, and the top ten RAW basic events for POSs 4-3 and 8-1 are shown in Table 19.1-171 and Table 19.1-100, respectively.

For POS 4-3, there are only two single failure basic events that have a FV importance greater than 1.0E-02. The most significant single failure basic event based on FV importance is **CVCAVCD024B and CVCAVCD024C**, which represent the failure of air-operated valve on low-pressure letdown line to close, with a FV importance of 4.9E-02. For POS 8-1, there are only four single failure basic events that have a FV importance greater than 1.0E-02. The most significant single failure basic event based on FV importance is **EPSDLLRAACA**, which represent the failure of AAC in A train to run, with a FV importance of 3.9E-02.

For POS 4-3, there are more than 22 basic events that have a RAW which value is approximately 2.3E+03. These are basic events that represent large external leak from components and piping. For POS 8-1, there are more than 45 basic events that have a RAW which value is approximately 3.0E+02. These are basic events that represent large external leak from components and piping.

US-APWR Unique Design Importance DCD_19-551

In this subsection, component importance for US-APWR unique design, i.e., the automatic isolation of low pressure letdown, is documented.

The Automatic isolation of low pressure letdown is an effective function to reduce risk caused by initiating event FLML or OVDR likely to occur during mid-loop operation of POSs 4 and 8. The isolation system consist of RCS low-level signal and air-operated isolation valves (RHS-AOV-024B and C). The following is estimated RAW for each component associated with the automatic isolation. The RAW is applicable to that for failure of RCS low-level signal or air-operated isolation valve to close.

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POS	CDF [/RY]		Ratio (=RAW)
	Base	Sensitivity	Note 2
3 Note 1	1.3E-08	1.3E-08	1.0
4-1	1.3E-08	1.4E-07	10.3
4-2	5.3E-09	1.3E-07	24.8
4-3	3.0E-08	1.2E-06	41.6
8-1	8.0E-08	4.1E-07	5.1
8-2	5.6E-09	1.3E-07	23.4
8-3	1.1E-08	1.7E-07	16.5
9 Note 1	3.4E-09	3.4E-09	1.0
11 Note 1	1.8E-08	1.8E-08	1.0
TOTAL	1.8E-07	2.3E-06	12.6

Note 1: Automatic isolation of low pressure letdown is an effective function to reduce risk caused by FLML or OVDR likely to occur during mid-loop operation of POSs 4 and 8. DCD_19-551

Note 2: RAW for failure of RCS loop low-level signal and of air-operated isolation valve to close is equal to the ratio.

The important SSCs and operator actions of other POSs are qualitatively extracted based on the mitigation system that is available for each POS and the importance results of POSs 4-3 and 8-1. SSCs and operator actions that have been identified to be risk important in POSs 4-3 and 8-1 were considered to be risk important in other POSs. SSCs and operator actions that have been credited in other POSs but not in POSs 4-3 and 8-1 were also considered to be risk important. Important operator actions of POSs 4-3 and 8-1 and other POSs are shown in Table 19.1-101 through 19.1-109. Important SSCs of POSs 4-3 and 8-1 and other POSs are shown in Table 19.1-110 to Table 19.1-118. These results are used as the input to the reliability assurance program and human factor engineering. Quantification results of POSs 4-3 and 8-1 have been considered applicable to identify SSCs (and operator actions) that are important to the overall LPSD risk for the reasons described below.

- The contributions of POSs 4-3 and 8-1 to the total CDF are larger than those of other POSs. SSCs that are important for these POSs are also important for the total LPSD risk.
- Loss of offsite power (LOOP) and LOCA initiating events have large contribution to the total CDF. Loss of RHR due to over-drain (OVDR) and caused by failing to maintain water level (FLML) has small impact on total CDF. This tendency is the same in each POS. This implies that the risk profile is similar for all POSs.
- POSs 4-3 and 8-1 have the least number of mitigation functions in all POSs. POSs other than POSs 4-3 and 8-1 have additional mitigation functions that are not available during POSs 4-3 and 8-1 (e.g. RCS cooling by SGs and gravity injection). Since number of mitigation functions credited inPOSs other than POSs 4-3 and 8-1 is equal or more than that of POSs 4-3 and 8-1, the risk importance of SSCs quantified for POSs 4-3 and 8-1 will have lower or similar values in other POSs. It is unlikely that SSCs that are below the quantitative thresholds in POSs 4-3 and 8-1 to become risk important in other POSs.
- SSCs that are used for mitigation systems not credited in POSs 4-3 and 8-1 (i.e., decay heat removal via SGs and gravity injection) may be risk important if all POSs were quantified together. SSCs of mitigation functions unique to other POS are all included in the list of risk important SSCs to assure that the list includes all risk important SSCs.

The uncertainties of the CDF for POSs 4-3 and 8-1 hhave been calculated and are summarized in Figure 19.1-21. The mean value, median, 5th percentile and 95th percentile of the distribution are calculated. The error factor (EF) is estimated by the square root of the ratio of the 95th percentile to the 5th percentile.

	POS 4-3	POS 8-1
Upper	8.3E-08/RY	2.3E-07/RY
Mean	2.9E-08/RY	8.1E-08/RY
Medium	1.8E-08/RY	4.8E-08/RY
Lower	4.4E-09/RY	1.3E-08/RY
EF	4.3	4.2
(Point Estimate)	3.0E-08/RY	8.0E-08/RY

The uncertainty range for POS 4-3 CDF is found to be 4.4E-09/RY – 8.3E-08/RY for the 5% to 95% interval. This indicates that there is 95% confidence that the POS 4-3 CDF is no greater than 8.3E-08/RY. The EF for the POS 4-3 CDF is 4.3. The point estimate CDF for POS 4-3 is 3.0E-08/RY.

The uncertainty range for the POS 8-1 CDF is found to be 1.3E-08/RY - 2.3E-07/RY for the 5% to 95% interval. This indicates that there is 95% confidence that the POS 8-1 CDF is no greater than 2.3E-07/RY. The EF for the POS 8-1 CDF is 4.2. The point estimate CDF for POS 8-1 is 8.0E-08/RY.

The estimation of uncertainty in the analysis from human error is discussed in sensitivity analysis Case 4-3. The ratio of the upper HEP values to lower HEP values are less than 5.0. DCD_19-550

US-APWR PRA uses various assumptions for the unreliability of unique designs such as the GTG, digital I&C system, equipment configurations, duration of plant shutdown, operator action to prevent or mitigate initiating events including their probabilities. While the assumptions are decidedly conservative, they man have large uncertainty resulting in large impact on LPSD risk.

Table 19.1-181 lists the key sources of uncertainty that may have impact on the PRA results and types or uncertainty. The assessed areas of uncertainty are categorized into three types: one is parametric uncertainty associated with parametric values, second is completeness uncertainty regarding the possibility of unaccounted for initiating events, and the other is modelling uncertainty made in developing the PRA model. Uncertainty analysis and sensitivity analyses are summarized in Table 19.1-181 to discuss the contribution to the LPSD risk.

In the LPSD Level 2 PRA, the release probability under the condition that core damage occurs is assumed to be 1.0. Therefore, the LRF, which equals the CDF, is 1.8E-07/RY.

The release categories for the LPSD operation are evaluated as follows:

- Filled RCS state: 3.5E-08/RY

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Table 19.1-181 Key Sources of Uncertainty and Key Assumptions (LPSD Operation) (Sheet 1 of 5)

Key Sources of Uncertainty and Key Assumptions		Type (Note)	Summary Results of Qualitative Assessment	Quantitative Approach
Unique Equipments and their Duty to the US-APWR Design	Gas Turbine Generators	M	Sensitivity analyse of failure probability, failure rates and CCF parameters were performed.	Sensitivity Analysis (Case 1-1, 1-2)
	Digital I&C	M	Sensitivity analyses of various failure probabilities of application and basic software CCF for digital I&C were performed.	Sensitivity Analysis (Case 6-1, 6-2)
Initiating Event Analysis	Initiating event frequency of loss of RHR caused by other failures (LORH) and loss of CCW/essential service water (LOCS)	M	Initiating event frequency of loss of RHR caused by other failures and loss of CCW/essential service water depends on equipment outage. Sensitivity analysis assuming no planned maintenance was performed.	Sensitivity Analysis (Case 3-1)
	Statistic uncertainty of initiating event frequency	P	(Statistical uncertainty is considered.)	Uncertainty Analysis
	Completeness of initiating events to the US-APWR design	C	Rare initiating events to the US-APWR design are assessed.	NA
	Outage types and their frequencies	M	Since human errors are the most dominant contributor to internal event risk during LPSD operation, the sources have less impact on the risk in comparison with the human errors. Sensitivity analysis assuming different outage types and their frequencies from the base case was performed.	Sensitivity Analysis (Case 3-2)
	Duration of plant shutdown	M	Sensitivity analysis using different duration from the base case was performed.	Sensitivity Analysis (Case 5-1)

Table 19.1-181 Key Sources of Uncertainty and Key Assumptions (LPSD Operation) (Sheet 2 of 5) DCD_19-550

Key Sources of Uncertainty and Key Assumptions		Type (Note)	Summary Results of Qualitative Assessment	Quantitative Approach
Event Tree Analysis	Indication of accident sequences	M	Realistic accident sequences are considered.	NA
System Analysis	Plugging before events occurred is not modelled	M	It would be hard to plug during LPSD operation in RCS and safety-related systems.	NA
	Class 1E electrical room HVAC are reliable and do not impact risk	M	Even if losses of HVAC occurs, automatic signals to start Class 1E GTGs or AAC, and to actuate low pressure letdown line isolatino will actually complete prior to occurrence of RCS boiling of an initiating event. To relax room heat up after losses of Class 1E electrical room HVAC, the operator will be open the room door and utilize available temporary fans.	NA
	Equipment outage	M	Since human errors are the most dominant contributor to internal event risk during LPSD operation, the sources have less impact on the risk in comparison with the human errors. Sensitivity analysis assuming different outage types and their frequencies from the base case was performed.	Sensitivity Analysis (Case 3-1)

Table 19.1-181 Key Sources of Uncertainty and Key Assumptions (LPSD Operation) (Sheet 3 of 5)

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Key Sources of Uncertainty and Key Assumptions		Type (Note)	Summary Results of Qualitative Assessment	Quantitative Approach
Data Analysis	Applicability of failure modes to the US-APWR equipment design	M	Potentially valuable generic data sources were collected. All the failure modes of the US-APWR component types were considered.	NA
	Failure probability and failure rates for diesel generators are applied to gas turbine generators	M	Sensitivity analysis failure probability and failure rates was performed.	Sensitivity Analysis (Case 1-1)
	Statistical uncertainty of failure rate	P	(Statistical uncertainty is considerable.)	Uncertainty Analysis
	Failure probability of digital I&C system	M	Sensitivity analyses of various failure probabilities of application software CCF for I&C were performed.	Sensitivity Analysis (Case 6-1, 6-2)
	Reliability of components	M	There is no plant-specific reliability data for the US-APWR. In the design stage, it is probable that the reliability of components of a newly designed plant is comparable to the component reliability of operating US plants. Therefore, US generic data is applicable.	NA

Table 19.1-181 Key Sources of Uncertainty and Key Assumptions (LPSD Operation) (Sheet 4 of 5)

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Key Sources of Uncertainty and Key Assumptions		Type (Note)	Summary Results of Qualitative Assessment	Quantitative Approach
Common Cause Failure Analysis	CCF parameters of emergency diesel generators are applied to gas turbine generators	M	Sensitivity analysis for the gas turbine generator CCF parameters was performed.	Sensitivity Analysis (Case 1-2)
	Statistical uncertainty of CCF probabilities	P	(Statistical uncertainty is considerable.)	Uncertainty Analysis
	CCF for continually operating pumps	M	There is data published for CCF of continually operating pumps. Based on engineering judgment, the PRA applies a CCF parameter lower than those reported in the NUREGs for the CCW and ESW pumps. Uncertainty associated with the CCF parameters for continually running pumps impact the initiating event frequency for loss of CCW, which has large contribution to the CDF. The PRA treats CCF for continually running pumps and standby pumps alike and applies a value of 0.1. This value is deemed a conservative estimation since the running pumps and the standby pumps are initially in an asymmetric configuration.	NA

Table 19.1-181 Key Sources of Uncertainty and Key Assumptions (LPSD Operation) (Sheet 5 of 5)

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Key Sources of Uncertainty and Key Assumptions		Type (Note)	Summary Results of Qualitative Assessment	Quantitative Approach
Human Reliability Analysis	Human error probability	M	Sensitivity analyses of post initiating event operator action failure probabilities were performed to study the impact of human errors to CDF, assuming the HEP to 0.0, lower value, mean value or upper value.	Sensitivity Analysis (Case 4-1, 4-2, 4-3)
	Statistical uncertainty of human error probability	P	(Statistical uncertainty is considered).	NA
	Visual display unit (VDU) interaction	M	Sensitivity analysis was performed assuming that changing windows on the display is not effective to reduce dependencies between actions and cannot be perceived as action performed in different locations.	Sensitivity Analysis (Case 4-6)
	Frequent training of operator actions	M	Sensitivity analysis assuming that operators perform less frequent training was carried out.	Sensitivity Analysis (Case 4-2)
	Dependency among operator actions	M	Sensitivity analyses assuming various dependency level among operator actions were performed.	Sensitivity Analysis (Case 4-4, 4-5, 4-6)
Shutdown Condition	Outage types and their frequencies	M	Sensitivity analysis assuming different outage types and frequencies were performed.	Sensitivity Analysis (Case 3-2)
	Duration of plant shutdown	M	Sensitivity analysis using duration based on Japanese operating experience was performed.	Sensitivity Analysis (Case 5-1)
Note - Uncertainty sources are categorized into three types, Parametric (P), Modeling (M) or Completeness (C).				