MITSUBISHI HEAVY INDUSTRIES, LTD.
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TOKYO, JAPAN

May 16, 2008

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-08087

Subject: MHI's Responses to US-APWR DCD RAI No.1

References: 1) "Request for Additional Information No. 1 Revision 0, SRP Section19 - Probabilistic Risk Assessment and Severe Accident Evaluation Application Section: 19.1", dated April 17, 2008.

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.1 Revision 0".

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

As indicated in the enclosed materials, this document contains information that MHI considers proprietary, and therefore should be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4) and 10 C.F.R § 9.17 (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) and 10 C.F.R.§ 9.17 (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.

D081
NRO

Enclosures:

1. Affidavit of Yoshiki Ogata

2. Responses to Request for Additional Information No.1 Revision 0 (proprietary)

3. Responses to Request for Additional Information No.1 Revision 0 (non-proprietary)

CC: L. J. Burkhart
J. W. Chung
S. R. Monarque
C. K. Paulson

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ENCLOSURE 1

Docket No. 52-021
MHI Ref: UAP-HF-08087

MITSUBISHI HEAVY INDUSTRIES, LTD.

AFFIDAVIT

I, Yoshiki Ogata, state as follows:

1. I am General Manager, APWR Promoting Department, of Mitsubishi Heavy Industries, LTD ("MHI"), and have been delegated the function of reviewing MHI's US-APWR documentation to determine whether it contains information that should be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4) and 10 C.F.R.§ 9.17(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.1 Revision 0, May, 2008", and have determined that portions of the document contain proprietary information that should be withheld from public disclosure. Those pages containing proprietary information are identified with the label "Proprietary" on the top of the page and the proprietary information has been bracketed with an open and closed bracket as shown here "[ ]". The first page of the document indicates that all information identified as "Proprietary" should be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4).

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

4. The basis for holding the referenced information confidential is that it describes the unique 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 development of methodology related to the analysis.

B. Loss of competitive advantage of the US-APWR created by benefits of 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 16th day of May 2008.

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

Enclosure 3

UAP-HF-08087
Docket Number 52-021

Responses to Request for Additional Information No.1 Revision 0

May, 2008
(Non-Proprietary)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

5/16/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: **NO.1 REVISION 0**

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

APPLICATION SECTION: **19.1**

DATE OF RAI ISSUE: **4/17/2008**

QUESTION NO. : 19-1

Please provide more information on how the event tree top events MC (reactor coolant system (RCS) makeup by charging pump), SG (decay heat removed from the RCS via steam generators), GI (gravitational injection), and AC (offsite power recovery) were modeled and quantified for each Plant Operating State (POS), including a summary and results of success criteria calculations for each, at a level of detail similar to that provided for other top events. These events do not appear in the summary of front - line system failures (Table 19.1 - 82) or in the detailed modeling discussion in the Probabilistic Risk Assessment (PRA) Technical Report.

ANSWER:

Additional information on event tree top events MC, SG, GI, and AC are described below.

1. MC: RCS makeup by charging pump

   MC is described as "CHI21" in table 19.1-82 in the DCD chapter 19. Summary and results of success criteria calculations are provided in the technical report[1)] (PRA report) chapter 20, subsection 20.a.3.2.2. Excerpts from the PRA report are provided in Attachment A for Question19-1.

2. SG: Decay heat removed from the RCS via SGs

   This function is not available during POS8-1, and therefore, faults tree for this top event was not developed. For top events used in POSs other than POS8-1, probabilities of top events are conservatively estimated by human error probabilities considering the dependencies between tasks. This is because human error probabilities are greater than failure probabilities of components, especially when dependencies between tasks are considered.

3. GI: Gravitational injection

   This function is not available during POS8-1 and therefore, faults tree for this top event was not developed. As the same reason with top event "SG", probability of this top event was conservatively estimated by human error probabilities considering the dependencies between tasks.

4. AC: Offsite power recovery

   As described in page 19.1-103 of the DCD, the allowable time for offsite power recovery is assumed to be 1 hour for POS8-1. Probability of failure to recover within allowable time is estimated by data provided in NUREG/CR-6890. For other POSs, allowable time for offsite power recovery is conservatively assumed to be 1 hour. Accordingly, probabilities applied for top heading "AC" in other POSs are same with that used in POS8-1, which is 0.53.

1) US-APWR Probabilistic Risk Assessment, MUAP-07030, Mitsubishi Heavy Industries, December. 2007.

Impact on DCD

There is no impact on DCD from this RAI as the response contains only additional information.

Impact on COLA

There is no impact on DCD from this RAI as the response contains only additional information.

Impact on PRA

There is no impact on PRA from this RAI and the response as this additional information does not cause any changes to the PRA.

Attachment A for Question 19-1

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

5/16/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: **NO.1 REVISION 0**

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

APPLICATION SECTION: **19.1**

DATE OF RAI ISSUE: **4/17/2008**

QUESTION NO. : 19-2

For the GI top event, please discuss the operator actions needed given that the spent fuel pool (SFP) valves are shown as locked closed on system diagrams. Also, identify the minimum vent size needed for success of gravitational injection and discuss how the necessary vent is ensured.

ANSWER:

Operator actions are essential to perform gravitational injection. Simplified system diagram of the gravitational injection line is shown in Figure 1.

In order to perform gravitational injection, operator must conduct the following tasks.

1. Establish gravitational injection pass

   Open the following valves to initiate gravitational injection from SFP

   - RHR-VLV-031A(D)
   - RHR-VLV-032A(D)

2. Establish pass to supply water from RWSP to SFP

   Confirm that the following valves are closed, and if not, close valves.

   - RWS-VLV-021
   - SFS-VLV-103A

- NCS-VLV-066A(B)
- SFS-VLV-103B

Confirm that the following valves are opened, and if not, open valves.

- RWS-VLV-001
- RWS-MOV-002
- RWS-MOV-004
- RWS-VLV-006A(B)
- RWS-VLV-005
- RWS-VLV-013A(B)
- RWS-VLV-014
- SFS-VLV-028
- SFS-VLV-029
- SFS-VLV-015
- SFS-VLV-017
- SFS-VLV-021A(D)
- SFS-VLV-022

3. Supply water from RWSP to SFP

   Start refueling water recirculation pump

   - RWS-RPP-001A

In the PRA, gravitational injection is taken credit only when large RCS opening such as SG manhole lids are opened. This is a conservative assumption, but it is obvious that gravitational injection is effective when the injection line has been successfully accomplished.

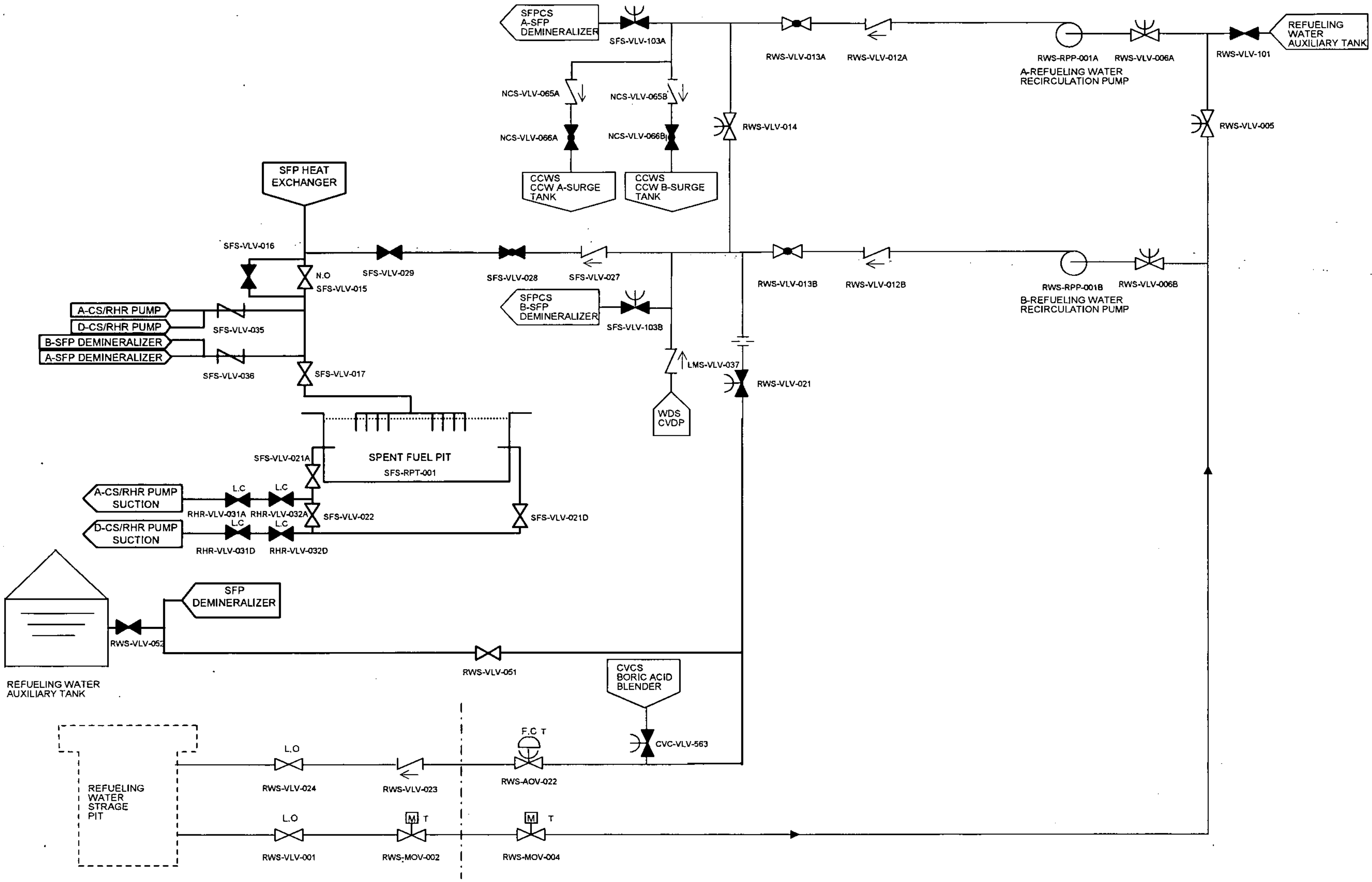

Figure 1 Gravitational Injection Simplified System Diagram

Impact on DCD

There is no impact on DCD from this RAI as the response contains only additional information.

Impact on COLA

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Impact on PRA

There is no impact on PRA from this RAI and the response as this additional information does not cause any changes to the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

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

APPLICATION SECTION: **19.1**

DATE OF RAI ISSUE: **4/17/2008**

QUESTION NO. : 19-3

At what point in a normal shutdown is the RCS expected to be opened (e.g., by opening a pressurizer manway)? Table 19.1 - 76 indicates that the RCS is closed until the SG manhole lids are removed and makes associated assumptions about the availability of the steam generators.

ANSWER:

The PRA assumes that the RCS is opened by opening of SG manhole lids. Other openings such as pressurizer manway or pressurizer safety valve vent are expected to be opened approximately in the same timing or after the opening of SG manhole lids.

The timing of the opening of SG manhole lids is described in Table.19.1-76 and Table19.1-79 of the DCD. The tables excerpted from the DCD are shown in the following pages.

Table 19.1-76 **Subdivided State of POS 4 (Mid-Loop Operation) for LPSD PRA**

	Open S/G manhole lid ↓	Install S/G nozzle lid ↓		Remarks
RCS water level	Mid-loop (nozzle center)			
POS	(POS4-1)	(POS4-2)	(POS4-3)	
RCS conditions	RCS close	RCS open	RCS close SG Isolated	
Mitigating systems				
SG and secondary systems	×	N/A	N/A	
Gravitational injection	N/A	×	N/A	
Initiating events				
Over-drain	×	N/A	N/A	
Fail to maintain water level	N/A	×	×	

Table 19.1-79 Duration Time of Each POS for LPSD PRA

Time		POS	Description	Duration time(hr)
1d	0:00			
		1	Low power operation	2.0
1d	2:00			
		2	Hot standby	7.7
1d	9:40			
		3	Hot and cold shutdown (RCS is filled with coolant)	2.3
1d	12:00			
		4-1	Cold shutdown (Mid-loop operation)	39.2
3d	3:10			
		4-2	Cold shutdown (Mid-loop operation)	12.0
3d	15:10			
		4-3	Cold shutdown (Mid-loop operation)	6.0
3d	21:10			
		5	Refueling cavity is filled with water	82.7
7d	7:50			
		6	No fuels in the core	108.0
11d	19:50			
		7	Refueling cavity is filled with water	75.8
14d	23:40			
		8-1	Cold shutdown (Mid-loop operation)	55.5
17d	7:10			
		8-2	Cold shutdown (Mid-loop operation)	12.0
17d	19:10			
		8-3	Cold shutdown (Mid-loop operation)	11.0
18d	6:10			
		9	Cold shutdown (RCS is filled with coolant)	10.0
18d	16:10			
		10	RCS leakage test (RHRS isolated from RCS)	20.5
19d	12:40			
		11	Cold and hot shutdown (RCS is filled with coolant)	43.5
21d	8:10			
		12	Hot standby	51.0
23d	11:10			
		13	Low power operation	4.0
23d	15:10			
			Total time	543
			Total days	22.6

Impact on DCD

There is no impact on DCD from this RAI as the response contains only additional information.

Impact on COLA

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Impact on PRA

There is no impact on PRA from this RAI and the response as this additional information does not cause any changes to the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

5/16/2008

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Docket No.52-021

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

APPLICATION SECTION: **19.1**

DATE OF RAI ISSUE: **4/17/2008**

QUESTION NO. : 19-4

Please clarify when (e.g., by identifying pressure, temperature, and vent status) the steam generators can be used for heat removal. Discuss how the depressurization (if any) required to use the steam generators for heat removal challenges temporary pressure boundaries such as nozzle caps or thimble seals.

ANSWER:

POSs where decay heat can be removed by steam generators (SGs) are the followings:

- RCS fulfill (POS 3, POS 9, POS 11)

  During these POSs, heat removal via SGs can be accomplished when decay heat can be transferred to the SGs by natural circulation in RCS. Accordingly, the RCS boundary is closed (vent closed). In the PRA, postulated pressure and temperature during these POSs are 400 psig and 350 F, respectively.

- Mid-loop operation (POS 4-1, POS 8-3)

  During these POSs, heat removal via SGs can be accomplished when RCS is atmospheric pressure and no large openings in the RCS exist. Accordingly, vent is closed. SG manholes must be closed and SG nozzle lids open for heat removal via SGs. In the PRA, postulated temperature during this POS is to be 150 F.

It is assumed that there are no temporary pressure boundaries during these POSs, which take credit of heat removal via SGs.

Impact on DCD

There is no impact on DCD from this RAI as the response contains only additional information.

Impact on COLA

There is no impact on DCD from this RAI as the response contains only additional information.

Impact on PRA

There is no impact on PRA from this RAI and the response as this additional information does not cause any changes to the PRA.

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5/16/2008

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Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: **NO.1 REVISION 0**

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

APPLICATION SECTION: **19.1**

DATE OF RAI ISSUE: **4/17/2008**

QUESTION NO. : 19-5

The discussion on page 19.1 - 100 states that the SG function is "unavailable if there is a large breach in the RCS." Discuss how the assumed flow diversion from inadvertent transfer to the refueling water storage pit (RWSP) from residual heat removal (RHR) compares in size to the stated "large breach" Is the RWSP flow diversion the largest postulated breach during shutdown?

ANSWER:

The PRA assumes that when a continuous flow diversion from inadvertent transfer to the refueling water storage pit (RWSP) from residual heat removal (RHR) occurs, SG function is unavailable. This treatment is the same with an event of large breach in the RCS.

Generally, it is assumed in the PRA that rupture of pipes equivalent or smaller than 1/3 of the main piping, and pipe rupture in piping equivalent or smaller than 3/4 inches diameter do not degrade the function of a system. Rupture of piping with diameter larger than these is assumed to result in degrade in system function.

For breaches in the RCS that exceed the criteria discussed above, it is postulated that such events result in loss of RHR function and SG function if they cannot be isolated. Location or maximum size of breach is not discussed since it is conservatively assumed that any breach exceeding the criteria will potentially result in RHR function and SG function.

Impact on DCD

There is no impact on DCD from this RAI as the response contains only additional information.

Impact on COLA

There is no impact on DCD from this RAI as the response contains only additional information.

Impact on PRA

There is no impact on PRA from this RAI and the response as this additional information does not cause any changes to the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

5/16/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: **NO.1 REVISION 0**

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

APPLICATION SECTION: **19.1**

DATE OF RAI ISSUE: **4/17/2008**

QUESTION NO. : 19-6

Please provide a more detailed discussion, including specific references to previous PRA studies and design - specific calculations, related to the statement on page 19.1 - 104 that "it is assumed that reflux cooling with the SGs is effective" at mid - loop.

ANSWER:

Many experiments have been performed to confirm the effectiveness of reflux cooling. It is known that reflux cooling contributed in decay heat removal at the loss of RHR accident, which occurred during mid-loop operation at Diablo Canyon in April 1987 [Reference (1) and (2)]. Analysis based on information of the accident in Diablo Canyon is reported in NUREG-1269 [Reference (1)].

Analytical studies on the reflux cooling have been performed by and Idaho national laboratory [Reference (3) and (4)] and they have reported that reflux cooling can effectively remove decay heat.

Based on these previous studies and analysis reports, we judged that reflux cooling with SGs is effective,

(1) NUREG-1269, "Loss of Residual Heat Removal System," June 1987

(2) NUREG-1410, "Loss of Vital AC Power and the Residual Heat Removal System During Mid-Loop Operations at Vogtle Unit 1 on March, 1990," June 1990

(3) NUREG/CR-5820, "Consequences of the Loss of the Residual Heat Removal Systems in Pressurized Water Reactors," May 1992

(4) NUREG/CR-5855, "Thermal-Hydraulic Processes During Reduced Inventory Operation with Loss of Residual Heat Removal," April 1992

Impact on DCD

There is no impact on DCD from this RAI as the response contains only additional information.

Impact on COLA

There is no impact on DCD from this RAI as the response contains only additional information.

Impact on PRA

There is no impact on PRA from this RAI and the response as this additional information does not cause any changes to the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

5/16/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: **NO.1 REVISION 0**

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

APPLICATION SECTION: **19.1**

DATE OF RAI ISSUE: **4/17/2008**

QUESTION NO. : 19-7

Regulatory Guide (RG) 1.206 requests descriptions and mean values for significant initiating events, failures, and core - damage sequences for each POS. Chapter 19 of the Design Control Document (DCD) states only that POS 8 - 1 is a "bounding" POS and other POS were "evaluated conservatively using the values of human errors in consideration of the dependability between tasks." Please provide additional information in the DCD on the assessment of other shutdown POS, including the following topics: (a) Detailed description of the methodology used to evaluate other POS, similar to that provided in the PRA technical report (b) Identification of the significant initiating events, including both internal and external events, for each POS (c) Description of the significant core damage sequences and their estimated frequencies for each POS (d) Identification of significant functions; structures, systems, and components (SSCs); and operator actions for each POS. Tables 19.1 - 97 through 19.1 - 112 provide failure information for POS other than 8 - 1, but not to the level of detail of that provided for 8 - 1. It appears from the current tables that many of the significant failures are the same; therefore, it is appropriate to list only the differences for each POS. Tables 19.1 - 105 through 19.1 - 112 provide condensed information for 10 systems; a similar table for POS 8 - 1 would be helpful for comparison.

ANSWER:

(a)

As described in the question, first, the detailed analysis of POS 8-1 is carried out. Since almost all of mitigation systems of LPSD need operator action, the results of quantitative analysis are greatly dominated by human errors. For example, table 20.10-4 of the DCD shows that the dominant cutsets of

CDF are human errors, especially dependence between tasks. This result indicates that the effect of the dependency of human error is greater than that of hardware failure.

Although the plant states of other POSs differ from POS 8-1, the mitigation system of other POSs are equivalent to that of POS 8-1, or the decay heat removal via SGs or the gravitational injection can be additionally taken credit compared to POS 8-1. Therefore, the number of mitigation systems credited in other POSs does not decrease from that of POS 8-1. The conditional core damage probability (CCDP) of each sequence in other POS decreases as a result of increase in mitigation systems. Factors that contribute to the failures of the additional mitigation systems are the random failure of SSCs, the independent human error and the dependent of human error caused between tasks. Generally, failure probability of a system caused by random failure of SSCs is approximately 1E-03 to 1E-05, and the probability of independent human error is approximately 1E-02 to 1E-04. On the other hand, the value of the dependency of the human error is expected to be approximately 1E-02 to 1.0. Therefore, the failures of SSCs and independent human errors are generally smaller than the human errors caused probability by dependency, which is set as a conservative value. From the reason above, CCDP of other POS can be represented by human error probability caused by dependency between tasks. For the frequency evaluation of initial events (IEs), such as loss of CCW, contribution of human error is relatively small, so this approach is inapplicable. Therefore, frequency of IEs needs to be quantified by detailed analysis for each POS. Finally, the conservative CDF value of POSs other than POS 8-1 can be evaluated by the three values shown below;

- The frequency of IE which evaluated for each POS
- CCDP of POS 8-1
- The reduction factor of CCDP of POS 8-1 based on number of effective mitigation systems and human error dependency

The CDF of all POSs, which was evaluated by this approach, is given in Table 20.10-2 in the Technical report (PRA report) chapter 20. The total CDF for LPSD is 2.0E-07/RY.

Quantification of CDF of POSs other than POS 8-1, were performed applying the method described below.

CDF for each sequence is evaluated using the following equation.

$$CDF_{POSX,\ SequenceY} = IE_{POSX} \times CCDP_{POS8\text{-}1,\ SequenceY} \times factor_{POSX,\ SequenceY}$$

- $CDF_{POSX,\ SequenceY}$: CDF of the sequence Y in POS X
- IE_{POSX} : IE frequency of POS X
- $CCDP_{POS8\text{-}1,\ SequenceY}$: CCDP of the sequence Y in POS 8-1
- $factor_{POSX,\ SequenceY}$: Reduction factor of the sequence Y in POS X

The value of reduction factors are estimated as described below:

1. Sequences where there are more mitigation systems than POS 8-1 and the number of the

operation tasks which is not successful in the same sequence is two or less.

0.1 is applied as the reduction factor. The value of reduction factor 0.1 is computed from the value of low dependence (LD) of human error, using the equation below according to the THERP approach.

$$LD = (1+19N) / 20 = 0.0785 \fallingdotseq 0.1$$

N: 3.0E-02 (This is conservative value of independent human error)

Here, N is the unconditional human error probability.

However there are exceptions. For cases where the following combination is included in the same sequence, reduction factor of 0.2 is applied.

- Combination of 'CV and GI'
- Combination of 'GI, and SC'

These functions above require line-up from the same water source (RWSP), and therefore, dependencies of these functions are considered to be high. Reduction factor 0.2, which is that of moderate dependence as discussed below, is applied to reflect this dependency.

2. Sequences where there are more mitigation systems than POS 8-1 and the number of the operation tasks which is not successful in the same sequence is three.

0.2 is applied as the reduction factor. Reduction factor 0.2 is computed from the value of moderate dependence (MD) of human error, using the equation below.

$$MD = (1+6N) / 7 = 0.168 \fallingdotseq 0.2$$

N: 3.0E-02

However there are exceptions. For cases where the following combination is included in the same sequence, reduction factor of 0.5 is applied.

- Combination of 'CV and GI'
- Combination of 'GI, and SC'

These functions require line-up from the same water source (RWSP), and therefore, dependencies of these functions are considered to be high. Reduction factor 0.5, which is that of high dependence as discussed below, is applied to reflect this dependency.

3. Sequences where there are more mitigation systems than POS 8-1 and the number of the operation tasks which is not successful in the same sequence is four.

0.5 is applied as the reduction factor. The value of reduction factor 0.5 is computed from the value of high dependence of human error, using the equation below.

HD = (1+N) / 2= 0.515 ≒ 0.5

N: 3.0E-02

4. Sequences where there are same mitigation systems as POS 8-1 and the number of the operation tasks which is not successful in the same sequence is five or more.

   1.0 is applied as the reduction factor. A factor 1.0 is determined from the value of complete dependence of human error.

(b)

Significant initiating events for internal events

LOCA initiating event is significant for all POSs during low power and shutdown. For all POSs, LOCA is conservatively assumed to occur by opening of a single valve. Its frequency is higher than other initiating events that are caused by mechanical failures, hence largely contributes to the CDF. The LOCA frequencies do not vary with duration of each POSs because it is determined by human error probability. Since other initiating event frequencies vary with duration of its POS, LOCA frequencies tend to become relatively higher than other initiating events in POSs with short duration.

Significant initiating events for internal flooding events

Loss of CCW/ESW initiating event is significant for all POSs during low power and shutdown. As can be seen by at-power operation internal flooding PRA, the probability of consequential loss of CCW/ESW event caused by flooding is much higher than loss of other functions. In POSs where redundancy of CCW/ESW is degraded, the conditional core damage probability will increase. These features are common to all POSs and accordingly, loss of CCW/ESW is considered to be a significant initiating event.

Significant initiating events for internal fire events

LOCA and LOOP initiating events are potentially significant for all POSs. On the other hand, OVDR and FLWL are initiating events only considered in POSs representing mid-loop operation. Accordingly, LOCA and LOOP are significant in POSs where the RCS is full, while for POS of mid-loop operation, OVDR and/or FLWL are significant event other than LOCA and LOOP. In internal fire PRA for at-power operation, fire compartments (e.g. switchyard) that cause LOOP are significant fire scenarios. Similar events are considerably significant during low power and shutdown (Internal events).

(c)

Significant core damage sequences are shown in table 19.7 of the technical report.

Description of significant core damage sequences for each POS is shown below.

(POS 3)

The top three accident sequences contribute 94 percent of the Level 1 shutdown core damage frequency of POS 3. These dominant sequences are as follows:

- LOCA initiating event, with success of isolation and failure of RCS make-up, which contributes 55 percent of the CDF
- LOCA initiating event, with success of isolation and RCS makeup, which contributes 22 percent of the CDF
- LOCA initiating event, with failure of isolation, which contributes 18 percent of the CDF

The descriptions of the dominant sequences are provided in the following:

- LOCA initiating event, with success of isolation and failure of RCS make-up

  This is sequence # 11 of the LOCA ET of POS 3. The isolation of the source of the LOCA is successful. Since the RCS makeup fails, the RHRS as the mitigation system is unavailable. Since the RCS is not under atmospheric pressure after loss of decay heat removal function, the gravitational injection is unavailable either. Consequently, failures of heat removal by SGs and the injection to the RCS by charging pump and SI pump result in reactor core damage.

- LOCA with success of isolation and RCS makeup

  This is sequence # 6 of the LOCA ET of POS 3. The isolation of the source of the LOCA and the RCS makeup are successful. The gravitational injection is unavailable for the same reason described above. Consequently, failure of the decay heat removal by the RHRS and SGs, and failures of injection to the RCS by charging pump and SI pump result in reactor core damage.

- LOCA with failure of isolation

  This is sequence # 15 of the LOCA ET of POS 3. This sequence is that isolation of source of LOCA fails following a LOCA initiated by inadvertent opening of motor-driven valve. If the isolation fails after the LOCA occurs, decay heat removal by the SG and the RHRS are unavailable because coolant continues to out of flow the RCS. The gravitational injection is unavailable for the same reason described above. Consequently, failures of injection to the RCS by charging pump and SI pump result in reactor core damage.

(POS 4-1)

The top six accident sequences contribute 95 percent of the Level 1 shutdown core damage frequency of POS 4-1. These dominant sequences are as follows:

- LOCA initiating event, with success of isolation and failure of RCS make-up, which contributes 28 percent of the CDF
- LOCS initiating event, with failure of the injection to the RCS using alternate component cooling, which contributes 22 percent of the CDF
- LOOP initiating event, with failure of the power supplying by all of ac power, which contributes 14 percent of the CDF
- LOOP initiating event, with success of the power supplying by class 1E gas turbine generators and failure of mitigation systems such as RHRS, which contributes 11 percent of the CDF
- LOCA initiating event, with success of isolation and RCS makeup, which contributes 11 percent of the CDF
- LOCA initiating event, with failure of isolation, which contributes 9 percent of the CDF

The descriptions of the dominant sequences are provided in the following:

- LOCA initiating event, with success of isolation and failure of RCS make-up

  This is sequence # 11 of the LOCA ET of POS 4-1. The isolation of the source of the LOCA is successful. Since the RCS makeup fails, the RHRS as the mitigation system is unavailable. Since the RCS is not under atmospheric pressure after loss of decay heat removal function, the gravitational injection is unavailable either. Consequently, failures of heat removal by SGs and the injection to the RCS by charging pump and SI pump result in reactor core damage.

- LOCS initiating event, with failure of the injection to the RCS using alternate component cooling

  This is sequence # 3 of the LOCS ET of POS 4-1. This sequence has a loss of CCW/essential service water initiator. The mitigation systems such as RHRS, SG, CVCS, and high head injection that are supported by CCW/essential service water are unavailable for this initiating event. (The SG is required for HVAC that is supported by essential service water). Moreover, the gravitational injection is unavailable for the same reason described above. Consequently, failure of injection by charging pump using the alternate component cooling system results in reactor core damage.

- LOOP with failure of the power supplying by all of ac power

  This is sequence # 28 of the LOOP ET of POS 4-1. This is station blackout sequence. Class 1E gas turbine generators and alternative gas turbine generators fail following the Initiating event. The recovery of offsite power is not successful either. It is assumed that any mitigation systems which are supported by ac power are unavailable. Therefore, this sequence results in reactor core damage.

- LOOP with success of the power supplying by class 1E gas turbine generators and failure of mitigation systems

This is sequence # 6 of the LOOP ET of POS 4-1. This sequence is that the power supply by class 1E gas turbine generators succeeds to start and run automatically following the initiating event. The gravitational injection is unavailable for the same reason described above. Consequently, failures of decay heat removal by RHRS and SGs, and injection to the RCS by charging pump and SI pump result in reactor core damage.

- LOCA with success of isolation and RCS makeup

  This is sequence # 6 of the LOCA ET of POS 4-1. The isolation of the source of the LOCA and the RCS makeup are successful. The gravitational injection is unavailable for the same reason described above. Consequently, failure of the decay heat removal by the RHRS and SGs, and failures of injection to the RCS by charging pump and SI pump result in reactor core damage.

- LOCA with failure of isolation

  This is sequence # 15 of the LOCA ET of POS 4-1. This sequence is that isolation of source of LOCA fails following a LOCA initiated by inadvertent opening of motor-driven valve. If the isolation fails after the LOCA occurs, decay heat removal by the SG and the RHRS are unavailable because coolant continues to out of flow the RCS. The gravitational injection is unavailable for the same reason described above. Consequently, failure of injection to the RCS by charging pump and SI pump result in reactor core damage.

(POS 4-2)

The top six accident sequences contribute 95 percent of the Level 1 shutdown core damage frequency of POS 4-2. These dominant sequences are as follows:

- LOCA initiating event, with success of isolation and failure of RCS make-up, which contributes 50 percent of the CDF
- LOCA initiating event, with success of isolation and RCS makeup, which contributes 20 percent of the CDF
- LOCA initiating event, with failure of isolation, which contributes 8 percent of the CDF
- LOOP initiating event, with failure of the power supplying by all of ac power, which contributes 8 percent of the CDF
- LOOP initiating event, with success of the power supplying by class 1E gas turbine generators and failure of mitigation systems such as RHRS, which contributes 6 percent of the CDF
- FLML initiating event, with success of the isolation and failure of the injection to the RCS by the SI pump, which contributes 3 percent of the CDF

The descriptions of the dominant sequences are provided in the following:

- LOCA initiating event, with success of isolation and failure of RCS make-up

This is sequence # 11 of the LOCA ET of POS 4-2. The isolation of the source of the LOCA is successful. Since the RCS makeup fails, the RHRS as the mitigation system is unavailable. Since POS 4-2 is the mid-loop operation and the SG manhole lid is open, the heat removal by SGs is unavailable either. Consequently, failures of the injection to the RCS by charging pump, SI pump and the gravitational injection result in reactor core damage.

- LOCA with success of isolation and RCS makeup

  This is sequence # 6 of the LOCA ET of POS 4-2. The isolation of the source of the LOCA and the RCS makeup are successful. The heat removal by SGs is unavailable for the same reason described above. Consequently, failure of the decay heat removal by the RHRS, and failures of injection to the RCS by charging pump, SI pump and gravitational injection result in reactor core damage.

- LOCA with failure of isolation

  This is sequence # 15 of the LOCA ET of POS 4-2. This sequence is that isolation of source of LOCA fails following a LOCA initiated by inadvertent opening of motor-driven valve. If the isolation fails after the LOCA occurs, decay heat removal by the SG and the RHRS are unavailable because coolant continues to out of flow the RCS. Consequently, failure of injection to the RCS by charging pump, SI pump and gravitational injection result in reactor core damage.

- LOOP with failure of the power supplying by all of ac power

  This is sequence # 28 of the LOOP ET of POS 4-2. This is station blackout sequence. Class 1E gas turbine generators and alternative gas turbine generators fail following the Initiating event. The recovery of offsite power is not successful either. It is assumed that any mitigation systems which are supported by ac power are unavailable. Therefore, this sequence results in reactor core damage.

- LOOP with success of the power supplying by class 1E gas turbine generators and failure of mitigation systems

  This is sequence # 6 of the LOOP ET of POS 4-2. This sequence is that the power supply by class 1E gas turbine generators succeeds to start and run automatically following the initiating event. The heat removal by SGs is unavailable for the same reason described above. Consequently, failures of decay heat removal by RHRS, and failure s of injection to the RCS by charging pump, SI pump and the gravitational injection result in reactor core damage.

- FLML initiating event, with success of the isolation and failure of the injection to the RCS by the SI pump

  This is sequence # 11 of the FLML ET of POS 4-2. The isolation of the source of the FLML is successful. Since this initiating event is assumed to be caused by failure of the CVCS, the RCS makeup and the injection to the RCS by charging pump which use the CVCS system cannot be used as a mitigation system of this event. The heat removal by SGs is unavailable for the same reason described above. Consequently, failures of the injection to the RCS by the SI pump and gravitational injection result in reactor core damage.

(POS4-3)

The top five accident sequences contribute 95 percent of the Level 1 shutdown core damage frequency of POS 4-3. These dominant sequences are as follows:

- LOCA initiating event, with success of isolation and failure of RCS make-up, which contributes 47 percent of the CDF
- LOCA initiating event, with success of isolation and RCS makeup, which contributes 19 percent of the CDF
- FLML initiating event, with success of the isolation and failure of the injection to the RCS by the SI pump, which contributes 16 percent of the CDF
- LOCA initiating event, with failure of isolation, which contributes 8 percent of the CDF
- LOOP initiating event, with success of the power supplying by class 1E gas turbine generators and failure of mitigation systems such as RHRS, which contributes 3 percent of the CDF
- LOCS initiating event, with failure of the injection to the RCS using alternate component cooling, which contributes 3 percent of the CDF

The descriptions of the dominant sequences are provided in the following:

- LOCA initiating event, with success of isolation and failure of RCS make-up

  This is sequence # 11 of the LOCA ET of POS 4-3. The isolation of the source of the LOCA is successful. Since the RCS makeup fails and the SGs nozzle lids are closed in POS 4-3, the RHRS and the SGs as the mitigation system are unavailable. Since the RCS is not under atmospheric pressure after loss of decay heat removal function, the gravitational injection is unavailable either. Consequently, failures of the injection to the RCS by charging pump and SI pump result in reactor core damage.

- LOCA with success of isolation and RCS makeup

  This is sequence # 6 of the LOCA ET of POS 4-3. The isolation of the source of the LOCA and the RCS makeup are successful. The heat removal by SGs and the gravitational injection are unavailable for the same reason described above. Consequently, failure of the decay heat removal by the RHRS, and failures of injection to the RCS by charging pump and SI pump result in reactor core damage.

- FLML initiating event, with success of the isolation and failure of the injection to the RCS by the SI pump

  This is sequence # 11 of the FLML ET of POS 4-3. The isolation of the source of the FLML is successful. Since this initiating event is assumed to be caused by failure of the CVCS, the RCS makeup and the injection to the RCS by charging pump which use the CVCS system cannot be used as a mitigation system of this event. The heat removal by SGs and the gravitational injection are unavailable for the same reason described above. Consequently, the failure of the injection to the RCS by the SI pump results in reactor core damage.

- LOCA with failure of isolation

  This is sequence # 15 of the LOCA ET of POS 4-3. This sequence is that isolation of source of LOCA fails following a LOCA initiated by inadvertent opening of motor-driven valve. If the isolation fails after the LOCA occurs, decay heat removal by the RHRS is unavailable because coolant continues to out of flow the RCS. The heat removal by SGs and the gravitational injection are unavailable for the same reason described above. Consequently, failure of injection to the RCS by charging pump and SI pump result in reactor core damage.

- LOOP with success of the power supplying by class 1E gas turbine generators and failure of mitigation systems

  This is sequence # 6 of the LOOP ET of POS 4-3. This sequence is that the power supply by class 1E gas turbine generators succeeds to start and run automatically following the initiating event. The heat removal by SGs and the gravitational injection are unavailable for the same reason described above. Consequently, failures of decay heat removal by RHRS, and injection to the RCS by charging pump and SI pump result in reactor core damage.

- LOCS initiating event, with failure of the injection to the RCS using alternate component cooling

  This is sequence # 3 of the LOCS ET of POS 4-3. This sequence has a loss of CCW/essential service water initiator. The mitigation systems such as RHRS, SG, CVCS, and high head injection that are supported by CCW/essential service water are unavailable for this initiating event. (The SG is required for HVAC that is supported by essential service water). Moreover, the gravitational injection is unavailable for the same reason described above. Consequently, failure of injection by charging pump using the alternate component cooling system results in reactor core damage.

(POS 8-2)

The top five accident sequences contribute 96 percent of the Level 1 shutdown core damage frequency of POS 8-2. These dominant sequences are as follows:

- LOCA initiating event, with success of isolation and failure of RCS make-up, which contributes 52 percent of the CDF
- LOCA initiating event, with success of isolation and RCS makeup, which contributes 21 percent of the CDF
- LOCA initiating event, with failure of isolation, which contributes 8 percent of the CDF
- LOOP initiating event, with failure of the power supplying by all of ac power, which contributes 8 percent of the CDF
- LOOP initiating event, with success of the power supplying by class 1E gas turbine generators and failure of mitigation systems such as RHRS, which contributes 7 percent of the CDF

The descriptions of the dominant sequences are provided in the following:

- LOCA initiating event, with success of isolation and failure of RCS make-up

  This is sequence # 11 of the LOCA ET of POS 8-2. The isolation of the source of the LOCA is successful. Since the RCS makeup fails, the RHRS as the mitigation system is unavailable. Since POS 8-2 is the mid-loop operation and the SG manhole lid is open, the heat removal by SGs is unavailable either. Consequently, failures of the injection to the RCS by charging pump, SI pump and the gravitational injection result in reactor core damage.

- LOCA with success of isolation and RCS makeup

  This is sequence # 6 of the LOCA ET of POS 8-2. The isolation of the source of the LOCA and the RCS makeup are successful. The heat removal by SGs is unavailable for the same reason described above. Consequently, failure of the decay heat removal by the RHRS, and failures of injection to the RCS by charging pump, SI pump and gravitational injection result in reactor core damage.

- LOCA with failure of isolation

  This is sequence # 15 of the LOCA ET of POS 8-2. This sequence is that isolation of source of LOCA fails following a LOCA initiated by inadvertent opening of motor-driven valve. If the isolation fails after the LOCA occurs, decay heat removal by the SG and the RHRS are unavailable because coolant continues to out of flow the RCS. Consequently, failure of injection to the RCS by charging pump, SI pump and gravitational injection result in reactor core damage.

- LOOP with failure of the power supplying by all of ac power

  This is sequence # 28 of the LOOP ET of POS 8-2. This is station blackout sequence. Class 1E gas turbine generators and alternative gas turbine generators fail following the Initiating event. The recovery of offsite power is not successful either. It is assumed that any mitigation systems which are supported by ac power are unavailable. Therefore, this sequence results in reactor core damage.

- LOOP with success of the power supplying by class 1E gas turbine generators and failure of mitigation systems

  This is sequence # 6 of the LOOP ET of POS 8-2. This sequence is that the power supply by class 1E gas turbine generators succeeds to start and run automatically following the initiating event. The heat removal by SGs is unavailable for the same reason described above. Consequently, failures of decay heat removal by RHRS, and failure s of injection to the RCS by charging pump, SI pump and the gravitational injection result in reactor core damage.

(POS 8-3)

The top six accident sequences contribute 95 percent of the Level 1 shutdown core damage frequency of POS 8-3. These dominant sequences are as follows:

- LOCA initiating event, with success of isolation and failure of RCS make-up, which contributes 47 percent of the CDF

- LOCA initiating event, with success of isolation and RCS makeup, which contributes 19 percent of the CDF
- LOCA initiating event, with failure of isolation, which contributes 15 percent of the CDF
- LOOP initiating event, with failure of the power supplying by all of ac power, which contributes 7 percent of the CDF
- LOOP initiating event, with success of the power supplying by class 1E gas turbine generators and failure of mitigation systems such as RHRS, which contributes 5 percent of the CDF
- LOCS initiating event, with failure of the injection to the RCS using alternate component cooling, which contributes 4 percent of the CDF

The descriptions of the dominant sequences are provided in the following:

- LOCA initiating event, with success of isolation and failure of RCS make-up

  This is sequence # 11 of the LOCA ET of POS 8-3. The isolation of the source of the LOCA is successful. Since the RCS makeup fails, the RHRS as the mitigation system is unavailable. Since the RCS is not under atmospheric pressure after loss of decay heat removal function, the gravitational injection is unavailable either. Consequently, failures of heat removal by SGs and the injection to the RCS by charging pump and SI pump result in reactor core damage.

- LOCA with success of isolation and RCS makeup

  This is sequence # 6 of the LOCA ET of POS 8-3. The isolation of the source of the LOCA and the RCS makeup are successful. The gravitational injection is unavailable for the same reason described above. Consequently, failure of the decay heat removal by the RHRS and SGs, and failures of injection to the RCS by charging pump and SI pump result in reactor core damage.

- LOCA with failure of isolation

  This is sequence # 15 of the LOCA ET of POS 8-3. This sequence is that isolation of source of LOCA fails following a LOCA initiated by inadvertent opening of motor-driven valve. If the isolation fails after the LOCA occurs, decay heat removal by the SG and the RHRS are unavailable because coolant continues to out of flow the RCS. The gravitational injection is unavailable for the same reason described above. Consequently, failure of injection to the RCS by charging pump and SI pump result in reactor core damage.

- LOOP with failure of the power supplying by all of ac power

  This is sequence # 28 of the LOOP ET of POS 8-3. This is station blackout sequence. Class 1E gas turbine generators and alternative gas turbine generators fail following the Initiating event. The recovery of offsite power is not successful either. It is assumed that any mitigation systems which are supported by ac power are unavailable. Therefore, this sequence results in reactor core damage.

- LOOP with success of the power supplying by class 1E gas turbine generators and failure of mitigation systems

This is sequence # 6 of the LOOP ET of POS 8-3. This sequence is that the power supply by class 1E gas turbine generators succeeds to start and run automatically following the initiating event. The gravitational injection is unavailable for the same reason described above. Consequently, failures of decay heat removal by RHRS and SGs, and injection to the RCS by charging pump and SI pump result in reactor core damage.

- LOCS initiating event, with failure of the injection to the RCS using alternate component cooling

This is sequence # 3 of the LOCS ET of POS 8-3. This sequence has a loss of CCW/essential service water initiator. The mitigation systems such as RHRS, SG, CVCS, and high head injection that are supported by CCW/essential service water are unavailable for this initiating event. (The SG is required for HVAC that is supported by essential service water). Moreover, the gravitational injection is unavailable for the same reason described above. Consequently, failure of injection by charging pump using the alternate component cooling system results in reactor core damage.

(POS 9)

The top five accident sequences contribute 95 percent of the Level 1 shutdown core damage frequency of POS 9. These dominant sequences are as follows:

- LOCA initiating event, with success of isolation and failure of RCS make-up, which contributes 48 percent of the CDF
- LOCA initiating event, with success of isolation and RCS makeup, which contributes 20 percent of the CDF
- LOCA initiating event, with failure of isolation, which contributes 16 percent of the CDF
- LOOP initiating event, with failure of the power supplying by all of ac power, which contributes 6 percent of the CDF
- LOOP initiating event, with success of the power supplying by class 1E gas turbine generators and failure of mitigation systems such as RHRS, which contributes 5 percent of the CDF
- LOCS initiating event, with failure of the injection to the RCS using alternate component cooling, which contributes 4 percent of the CDF

The descriptions of the dominant sequences are provided in the following:

- LOCA initiating event, with success of isolation and failure of RCS make-up

This is sequence # 11 of the LOCA ET of POS 9. The isolation of the source of the LOCA is successful. Since the RCS makeup fails, the RHRS as the mitigation system is unavailable. Since the RCS is not under atmospheric pressure after loss of decay heat removal function, the gravitational injection is unavailable either. Consequently, failures of heat removal by SGs and the injection to the RCS by charging pump and SI pump result in reactor core damage.

- LOCA with success of isolation and RCS makeup

  This is sequence # 6 of the LOCA ET of POS 9. The isolation of the source of the LOCA and the RCS makeup are successful. The gravitational injection.is unavailable for the same reason described above. Consequently, failure of the decay heat removal by the RHRS and SGs, and failures of injection to the RCS by charging pump and SI pump result in reactor core damage.

- LOCA with failure of isolation

  This is sequence # 15 of the LOCA ET of POS 9. This sequence is that isolation of source of LOCA fails following a LOCA initiated by inadvertent opening of motor-driven valve. If the isolation fails after the LOCA occurs, decay heat removal by the SG and the RHRS are unavailable because coolant continues to out of flow the RCS. The gravitational injection is unavailable for the same reason described above. Consequently, failure of injection to the RCS by charging pump and SI pump result in reactor core damage.

- LOOP with failure of the power supplying by all of ac power

  This is sequence # 28 of the LOOP ET of POS 9. This is station blackout sequence. Class 1E gas turbine generators and alternative gas turbine generators fail following the Initiating event. The recovery of offsite power is not successful either. It is assumed that any mitigation systems which are supported by ac power are unavailable. Therefore, this sequence results in reactor core damage.

- LOOP with success of the power supplying by class 1E gas turbine generators and failure of mitigation systems

  This is sequence # 6 of the LOOP ET of POS 9. This sequence is that the power supply by class 1E gas turbine generators succeeds to start and run automatically following the initiating event. The gravitational injection is unavailable for the same reason described above. Consequently, failures of decay heat removal by RHRS and SGs, and injection to the RCS by charging pump and SI pump result in reactor core damage.

(POS 11)

The top six accident sequences contribute 97 percent of the Level 1 shutdown core damage frequency of POS 11. These dominant sequences are as follows:

- LOCA initiating event, with success of isolation and failure of RCS make-up, which contributes 27 percent of the CDF

- LOCS initiating event, with failure of the injection to the RCS using alternate component cooling, which contributes 23 percent of the CDF

- LOOP initiating event, with failure of the power supplying by all of ac power, which contributes 15 percent of the CDF

- LOOP initiating event, with success of the power supplying by class 1E gas turbine generators and failure of mitigation systems such as RHRS, which contributes 12 percent of the CDF

- LOCA initiating event, with success of isolation and RCS makeup, which contributes 11 percent of the CDF

- LOCA initiating event, with failure of isolation, which contributes 9 percent of the CDF

The descriptions of the dominant sequences are provided in the following:

- LOCA initiating event, with success of isolation and failure of RCS make-up

  This is sequence # 11 of the LOCA ET of POS 11. The isolation of the source of the LOCA is successful. Since the RCS makeup fails, the RHRS as the mitigation system is unavailable. Since the RCS is not under atmospheric pressure after loss of decay heat removal function, the gravitational injection is unavailable either. Consequently, failures of heat removal by SGs and the injection to the RCS by charging pump and SI pump result in reactor core damage.

- LOCS initiating event, with failure of the injection to the RCS using alternate component cooling

  This is sequence # 3 of the LOCS ET of POS 11. This sequence has a loss of CCW/essential service water initiator. The mitigation systems such as RHRS, SG, CVCS, and high head injection that are supported by CCW/essential service water are unavailable for this initiating event. (The SG is required for HVAC that is supported by essential service water). Moreover, the gravitational injection is unavailable for the same reason described above. Consequently, failure of injection by charging pump using the alternate component cooling system results in reactor core damage.

- LOOP with failure of the power supplying by all of ac power

  This is sequence # 28 of the LOOP ET of POS 11. This is station blackout sequence. Class 1E gas turbine generators and alternative gas turbine generators fail following the Initiating event. The recovery of offsite power is not successful either. It is assumed that any mitigation systems which are supported by ac power are unavailable. Therefore, this sequence results in reactor core damage.

- LOOP with success of the power supplying by class 1E gas turbine generators and failure of mitigation systems

  This is sequence # 6 of the LOOP ET of POS 11. This sequence is that the power supply by class 1E gas turbine generators succeeds to start and run automatically following the initiating event. The gravitational injection is unavailable for the same reason described above. Consequently, failures of decay heat removal by RHRS and SGs, and injection to the RCS by charging pump and SI pump result in reactor core damage.

- LOCA with success of isolation and RCS makeup

  This is sequence # 6 of the LOCA ET of POS 11. The isolation of the source of the LOCA and the RCS makeup are successful. The gravitational injection is unavailable for the same reason described above. Consequently, failure of the decay heat removal by the RHRS and

SGs, and failures of injection to the RCS by charging pump and SI pump result in reactor core damage.

- LOCA with failure of isolation

  This is sequence # 15 of the LOCA ET of POS 11. This sequence is that isolation of source of LOCA fails following a LOCA initiated by inadvertent opening of motor-driven valve. If the isolation fails after the LOCA occurs, decay heat removal by the SG and the RHRS are unavailable because coolant continues to out of flow the RCS. The gravitational injection is unavailable for the same reason described above. Consequently, failure of injection to the RCS by charging pump and SI pump result in reactor core damage.

(d)

Tables revised to contain the requested information are shown below.

Table19.1-97 Differences of Important Operator Action between POS 3 and POS 8-1

No	System	Operator Action Description	Remarks
1	LOW PRESSURE LETDOWN LINE	N/A	This system would be modeled as a mitigation system which prevents reduction of a RCS water level only at the time of mid-loop operation. Therefore, this system is not modeled in POS 3.
2	EMERGENCY FEED WATER SYSTEM	OPERATOR FAILS TO START STANDBY EFW PUMP (HE)	This system is unavailable in POS 8-1 because SG is isolated from the RCS. But it is available in POS 3.

Operator actions other than systems mentioned above are the same as POS 8-1.

Table19.1-98 Differences of Important Operator Action between POS 4-1 and POS 8-1

No	System	Operator Action Description	Remarks
1	EMERGENCY FEED WATER SYSTEM	OPERATOR FAILS TO START STANDBY EFW PUMP (HE)	This system is unavailable in POS 8-1 because SG is isolated from the RCS. But it is available in POS 4-1.

Operator actions other than systems mentioned above are the same as POS 8-1.

Table19.1-99 Differences of Important Operator Action between POS 4-2 and POS 8-1

No	System	Operator Action Description	Remarks
1	GRAVITATIONAL INJECTION SYSTEM	OPERATOR FAILS TO ESTABLISH GRAVITATIONAL INJECTION (HE)	This system is unavailable in POS 8-1 because the RCS is not under atmospheric pressure. But it is available in POS 4-2.

Operator actions other than systems mentioned above are the same as POS 8-1.

Table19.1-100 Differences of Important Operator Action between POS 4-3 and POS 8-1

No	System	Operator Action Description	Remarks
1	None	None	None

All operator actions in POS 4-3 are the same as POS 8-1.

Table19.1-101 Differences of Important Operator Action between POS 8-2 and POS 8-1

No	System	Operator Action Description	Remarks
1	GRAVITATIONAL INJECTION SYSTEM	OPERATOR FAILS TO ESTABLISH GRAVITATIONAL INJECTION (HE)	This system is unavailable in POS 8-1 because the RCS is not under atmospheric pressure. But it is available in POS 8-2.

Operator actions other than systems mentioned above are the same as POS 8-1.

Table19.1-102 Differences of Important Operator Action between POS 8-3 and POS 8-1

No	System	Operator Action Description	Remarks
1	EMERGENCY FEED WATER SYSTEM	OPERATOR FAILS TO START STANDBY EFW PUMP (HE)	This system is unavailable in POS 8-1 because SG is isolated from the RCS. But it is available in POS 8-3.

Operator actions other than systems mentioned above are the same as POS 8-1.

Table19.1-103 Differences of Important Operator Action between POS 9 and POS 8-1

No	System	Operator Action Description	Remarks
1	LOW PRESSURE LETDOWN LINE	N/A	This system would be modeled as a mitigation system which prevents reduction of a RCS water level only at the time of mid-loop operation. Therefore, this system is not modeled in POS 9.
2	EMERGENCY FEED WATER SYSTEM	OPERATOR FAILS TO START STANDBY EFW PUMP (HE)	This system is unavailable in POS 8-1 because SG is isolated from the RCS. But it is available in POS 9.

Operator actions other than systems mentioned above are the same as POS 8-1.

Table19.1-104 Differences of Important Operator Action between POS 11 and POS 8-1

No	System	Operator Action Description	Remarks
1	LOW PRESSURE LETDOWN LINE	N/A	This system would be modeled as a mitigation system which prevents reduction of a RCS water level only at the time of mid-loop operation. Therefore, this system is not modeled in POS 11.
2	EMERGENCY FEED WATER SYSTEM	OPERATOR FAILS TO START STANDBY EFW PUMP (HE)	This system is unavailable in POS 8-1 because SG is isolated from the RCS. But it is available in POS 11.

Operator actions other than systems mentioned above are the same as POS 8-1.

Table19.1-113 Important SSCs of each System in POS 8-1

No	System	Description	Remarks
1	LOW PRESSURE LETDOWN LINE	LOW PRESSURE LETDOWN LINE ISOLATION VALVES (A,D) LOW PRESSURE LETDOWN LINE AIR OPERATED VALVE	
2	RESIDUAL HEATA REMOVAL SYSTEM	RHR PUMP SUCTION MOTOR OPERATED ISOLATION VALVES (9000A,B,C, 9001A,B,C) RHR PUMP (A,B,C) RHR LINE CONTAINMNET ISOLATION MOTOR OPERATED VALVES (9015A,B,C) RCS COLD LEG INJECTION LINE MOTOR OPERATED VALVES (9014A,B,C)	RHR D-train is outage.
3	EMERGENCY FEED WATER SYSTEM	N/A	This system is unavailable in POS 8-1.
4	HIGH HEAD INJECTION SYSTEM	SI PUMP (A,B)	SI pump C,D are outage.
5	CHEMICAL VOLUME CONTROL SYSTEM	CHARGING PUMP A,B VOLUME CONTROL TANK DISCHARGE LINE MOTOR OPERATED VALVES (121B,C) CHARGING PUMP RWAT SUCTION ISOLATION VALVES MOTOR OPERATED (121D,E) REFUELING WATER AUXILIARY TANK SUCTION LINE MANUAL VALVE FAIL TO OPEN (026) REFUELING WATER AUXILIARY TA	
6	GRAVITATIONAL INJECTION SYSTEM	N/A	This system is unavailable in POS 8-1.
7	EMERGENCY ELECTRIC POWER SUPPLY SYSTEM	EMERGENCY GAS TURBINE GENERATOR (GTG A,B,C) 6.9KV AC BUS INCOMER CIRCUIT BREAKER (6H A,B,C) AAC GAS TURBINE GENERATOR (GTG P1,2)	GTG D-train is outage.
8	COMPONENT COOLING WATER SYSTEM	CCW PUMP (A,B,C) CCW HEAT EXCHANGER (A,B,C)	CCW D-train is outage.
9	ESSENTIAL SERVICE WATER SYSTEM	ESW PUMP (A,B,C,D)	
10	ALTERNATE COMPONENT COOLING WATER SYSTEM	MOTOR DRIVEN / DEISEL DRIVEN FIRE SUPPRESSION PUMP ALTERNATE COMPONENT COOLING WATER LINE MOTOR OPERATED VALVES (ACWCH1B,ACWCH2B,ACWCH3B,ACWCH4B) CHARGING PUMP COOLING LINE ISOLATION MOTOR OPERATED VALVES (ACWCH6B)	

Table19.1-114 Differences of Important SSCs of each System in POS 3

No	System	Description	Remarks
1	LOW PRESSURE LETDOWN LINE	N/A	This system would be modeled as a mitigation system which prevents reduction of a RCS water level only at the time of mid-loop operation. Therefore, this system is not modeled in POS 3.
2	RESIDUAL HEATA REMOVAL SYSTEM	Main active components of RHRS are the same as POS 8-1. However, trains which are available differ from POS 8-1 as follows;. POS 8-1: A,B,C trains, POS 3: B,C,D trains.	RHR A-train is outage.
3	EMERGENCY FEED WATER SYSTEM	MOTOR DRIVEN EFW PUMP C	This system is unavailable in POS 8-1 because SG is isolated from the RCS. But it is available in POS 3. Motor driven EFW pump B is outage.
4	HIGH HEAD INJECTION SYSTEM	Main active components of HHIS are the same as POS 8-1. However, trains which are available differ from POS 8-1 as follows;. POS 8-1: A,B trains, POS 3: C,D trains.	SI pump A,B are outage.
5	CHEMICAL VOLUME CONTROL SYSTEM	Main active components of CVCS are the same as POS 8-1. However, trains which are available differ from POS 8-1 as follows;. POS 8-1: A,B charging pumps, POS 3: B charging pump.	Charging pump A is outage.
6	EMERGENCY ELECTRIC POWER SUPPLY SYSTEM	Main active components of EPS are the same as POS 8-1. However, trains which are available differ from POS 8-1 as follows;. POS 8-1: A,B,C trains, POS 3: A,B,C,D trains.	
7	COMPONENT COOLING WATER SYSTEM	Main active components of CCWS are the same as POS 8-1. However, trains which are available differ from POS 8-1 as follows;. POS 8-1: A,B,C trains, POS 3: B,C,D trains.	CCW A-train is outage.

Table19.1-115 Differences of Important SSCs of each System in POS 4-1

No	System	Description	Remarks
1	RESIDUAL HEATA REMOVAL SYSTEM	Main active components of RHRS are the same as POS 8-1. However, trains which are available differ from POS 8-1 as follows;. POS 8-1: A,B,C trains, POS 4-1: B,C,D trains.	RHR A-train is outage.
2	EMERGENCY FEED WATER SYSTEM	MOTOR DRIVEN EFW PUMP C	This system is unavailable in POS 8-1 because SG is isolated from the RCS.But it is available in POS 3. Motor driven EFW pump B is outage.
3	HIGH HEAD INJECTION SYSTEM	Main active components of HHIS are the same as POS 8-1. However, trains which are available differ from POS 8-1 as follows;. POS 8-1: A,B trains, POS 4-1: C,D trains.	SI pump A,B are outage.
4	CHEMICAL VOLUME CONTROL SYSTEM	Main active components of CVCS are the same as POS 8-1. However, trains which are available differ from POS 8-1 as follows;. POS 8-1: A,B charging pumps, POS 4-1: B charging pump.	Charging pump A is outage.
5	EMERGENCY ELECTRIC POWER SUPPLY SYSTEM	Main active components of EPS are the same as POS 8-1. However, trains which are available differ from POS 8-1 as follows;. POS 8-1: A,B,C trains, POS 4-1: B,C,D trains.	GTG A-train is outage.
6	COMPONENT COOLING WATER SYSTEM	Main active components of CCWS are the same as POS 8-1. However, trains which are available differ from POS 8-1 as follows;. POS 8-1: A,B,C trains, POS 4-1: B,C,D trains.	CCW A-train is outage.

Table19.1-116 Differences of Important SSCs of each System in POS 4-2

No	System	Description	Remarks
1	RESIDUAL HEATA REMOVAL SYSTEM	Main active components of RHRS are the same as POS 8-1. However, trains which are available differ from POS 8-1 as follows;. POS 8-1: A,B,C trains, POS 4-2: B,C,D trains.	RHR A-train is outage.
2	HIGH HEAD INJECTION SYSTEM	Main active components are the same as POS 8-1. However, trains which are available differ from POS 8-1 as follows;. POS 8-1: A,B trains, POS 4-2: C,D trains.	SI pump A,B are outage.
3	CHEMICAL VOLUME CONTROL SYSTEM	Main active components of CVCS are the same as POS 8-1. However, trains which are available differ from POS 8-1 as follows;. POS 8-1: A,B charging pumps, POS 4-2: B charging pump.	Charging pump A is outage.
4	GRAVITATIONAL INJECTION SYSTEM	SPENT FUEL PIT CS/RHR-SPENT FUEL PIT BOUNDARY MANUAL VALVES (SUCTION LINE) (SFP01A,D, 020A,D) REFUELING WATER RECIRCULATION PUNP (A,B) SPENT FUEL PIT SUCTION LINE FROM REFUELING WATER STORAGE PIT	This system is unavailable in POS 8-1 because the RCS is not under atmospheric pressure. But it is available in POS 4-2.
5	EMERGENCY ELECTRIC POWER SUPPLY SYSTEM	Main active components of EPS are the same as POS 8-1. However, trains which are available differ from POS 8-1 as follows;. POS 8-1: A,B,C trains, POS 4-2: B,C,D trains.	GTG A-train is outage.
6	COMPONENT COOLING WATER SYSTEM	Main active components of CCWS are the same as POS 8-1. However, trains which are available differ from POS 8-1 as follows;. POS 8-1: A,B,C trains, POS 4-2: B,C,D trains.	CCW A-train is outage.

Table19.1-117 Differences of Important SSCs of each System in POS 4-3

No	System	Description	Remarks
1	RESIDUAL HEATA REMOVAL SYSTEM	Main active components of RHRS are the same as POS 8-1. However, trains which are available differ from POS 8-1 as follows;. POS 8-1: A,B,C trains, POS 4-3: B,C,D trains.	RHR A-train is outage.
2	HIGH HEAD INJECTION SYSTEM	Main active components are the same as POS 8-1. However, trains which are available differ from POS 8-1 as follows;. POS 8-1: A,B trains, POS 4-3: C,D trains.	SI pump A,B are outage.
3	CHEMICAL VOLUME CONTROL SYSTEM	Main active components of CVCS are the same as POS 8-1. However, trains which are available differ from POS 8-1 as follows;. POS 8-1: A,B charging pumps, POS 4-3: B charging pump.	Charging pump A is outage.
4	EMERGENCY ELECTRIC POWER SUPPLY SYSTEM	Main active components of EPS are the same as POS 8-1. However, trains which are available differ from POS 8-1 as follows;. POS 8-1: A,B,C trains, POS 4-3: B,C,D trains.	GTG A-train is outage.
5	COMPONENT COOLING WATER SYSTEM	Main active components of CCWS are the same as POS 8-1. However, trains which are available differ from POS 8-1 as follows;. POS 8-1: A,B,C trains, POS 4-3: B,C,D trains.	CCW A-train is outage.

Table19.1-118 Differences of Important SSCs of each System in POS 8-2

No	System	Description	Remarks
1	RESIDUAL HEATA REMOVAL SYSTEM	Main active components of RHRS are the same as POS 8-1. However, trains which are available differ from POS 8-1 as follows;. POS 8-1: A,B,C trains, POS 8-2: A,B,C,D trains.	
2	GRAVITATIONAL INJECTION SYSTEM	SPENT FUEL PIT CS/RHR-SPENT FUEL PIT BOUNDARY MANUAL VALVES (SUCTION LINE) REFUELING WATER RECIRCULATION PUNP (A,B) SPENT FUEL PIT SUCTION LINE FROM REFUELING WATER STORAGE PIT	
3	COMPONENT COOLING WATER SYSTEM	Main active components of CCWS are the same as POS 8-1. However, trains which are available differ from POS 8-1 as follows;. POS 8-1: A,B,C trains, POS 8-2:A,B,C,D trains.	

Table19.1-119 Differences of Important SSCs of each System in POS 8-3

No	System	Description	Remarks
1	RESIDUAL HEATA REMOVAL SYSTEM	Main active components of RHRS are the same as POS 8-1. However, trains which are available differ from POS 8-1 as follows;. POS 8-1: A,B,C trains, POS 8-3: A,B,C,D trains.	
2	EMERGENCY FEED WATER SYSTEM	MOTOR DRIVEN EFW PUMP B	This system is unavailable in POS 8-1 because SG is isolated from the RCS.But it is available in POS 8-3. Motor driven EFW pump C is outage.
3	COMPONENT COOLING WATER SYSTEM	Main active components of CCWS are the same as POS 8-1. However, trains which are available differ from POS 8-1 as follows;. POS 8-1: A,B,C trains, POS 8-3:A,B,C,D trains.	

Table19.1-120 Differences of Important SSCs of each System in POS 9

No	System	Description	Remarks
1	LOW PRESSURE LETDOWN LINE	N/A	This system would be modeled as a mitigation system which prevents reduction of a RCS water level only at the time of mid-loop operation. Therefore, this system is not modeled in POS 9.
2	RESIDUAL HEATA REMOVAL SYSTEM	Main active components of RHRS are the same as POS 8-1. However, trains which are available differ from POS 8-1 as follows;. POS 8-1: A,B,C trains, POS 9: A,B,C,D trains.	
3	EMERGENCY FEED WATER SYSTEM	MOTOR DRIVEN EFW PUMP B	This system is unavailable in POS 8-1 because SG is isolated from the RCS.But it is available in POS 9. Motor driven EFW pump C is outage.
4	HIGH HEAD INJECTION SYSTEM	Main active components of HHIS are the same as POS 8-1. However, trains which are available differ from POS 8-1 as follows;. POS 8-1: A,B trains, POS 9: A,B,C,D trains.	
5	COMPONENT COOLING WATER SYSTEM	Main active components of CCWS are the same as POS 8-1. However, trains which are available differ from POS 8-1 as follows;. POS 8-1: A,B,C trains, POS 9:A,B,C,D trains.	

Table19.1-121 Differences of Important SSCs of each System in POS 11

No	System	Description	Remarks
1	LOW PRESSURE LETDOWN LINE	N/A	This system would be modeled as a mitigation system which prevents reduction of a RCS water level only at the time of mid-loop operation. Therefore, this system is not modeled in POS 11.
2	RESIDUAL HEATA REMOVAL SYSTEM	Main active components of RHRS are the same as POS 8-1. However, trains which are available differ from POS 8-1 as follows;. POS 8-1: A,B,C trains, POS 11: A,B,C,D trains.	
3	EMERGENCY FEED WATER SYSTEM	MOTOR DRIVEN EFW PUMP B	This system is unavailable in POS 8-1 because SG is isolated from the RCS.But it is available in POS 11. Motor driven EFW pump C is outage.
4	HIGH HEAD INJECTION SYSTEM	Main active components of HHIS are the same as POS 8-1. However, trains which are available differ from POS 8-1 as follows;. POS 8-1: A,B trains, POS 11: A,B,C,D trains.	
5	EMERGENCY ELECTRIC POWER SUPPLY SYSTEM	Main active components of EPS are the same as POS 8-1. However, trains which are available differ from POS 8-1 as follows;. POS 8-1: A,B,C trains, POS 11: A,B,C,D trains.	
6	COMPONENT COOLING WATER SYSTEM	Main active components of CCWS are the same as POS 8-1. However, trains which are available differ from POS 8-1 as follows;. POS 8-1: A,B,C trains, POS 11:A,B,C,D trains.	

Impact on DCD

The DCD will be revised reflecting this response to this RAI as this additional information is requested in Regulatory Guide (RG) 1.206.

Impact on COLA

This RAI and its response will impact the COLA, which refers the DCD, as a result as its impact on DCD.

Impact on PRA

There is no impact on PRA from this RAI and the response as this additional information does not cause any changes to the PRA.

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

APPLICATION SECTION: **19.1**

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QUESTION NO. : 19-8

The success criteria for the chemical and volume control system (CVCS) provided in Table 19.1 - 81, sheet 1, state that one of two charging pumps is required in POS 8 - 1 and that both pumps are in a standby state. Table 19.1 - 80 shows one pump in outage and one in standby in POS 3 and 4; both pumps in standby in POS 8 - 1; and one pump running and one in standby in POS 8 - 2, 8 - 3, 9, and 11. The technical specification (TS) for the low temperature overpressure protection (LTOP) system (TS 3.4.12), however, states that "a maximum of ... one charging pump capable of injecting into the RCS" shall be operable in MODE 4 when any RCS cold leg temperature is lower than the LTOP arming temperature, MODE 5, and MODE 6 when the reactor vessel head is on. Please provide further discussion of the system availability during shutdown: (a) How is the LTOP requirement of a maximum of one charging pump "capable of injecting" satisfied during shutdown (e.g., by locked closed valves or similar)? (b) If only one charging pump is capable of injecting, please justify the classification of the pumps as running or standby (rather than one in outage) in POS 8, 9, and 11, with sensitivity studies for success criteria as appropriate. (c) Similarly, justify the indication in Table 19.1 - 80 that four safety injection (SI) pumps are in a standby state in POS 9 and 11, given that TS 3.4.12 requires that a maximum of two SI pumps be capable of injecting into the RCS. (Note that the success criteria in Table 19.1 - 81, sheet 1, indicate 1 of 2 pumps required in accordance with the LTOP requirement.).

ANSWER:

This question will be answered later, within 60 days after RAI issue date.

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APPLICATION SECTION: 19.1

DATE OF RAI ISSUE: 4/17/2008

QUESTION NO. : 19-9

Please specify which motor - driven valve is referred to in this statement on page 19.1 - 99: "In this evaluation, inadvertent transfer to the RWSP from the RHR [residual heat removal system] is assumed. This diversion can happen if a motor - driven valve is opened." .

ANSWER:

"Inadvertent transfer to the RWSP from the RHR is assumed. This diversion can happen if a motor-driven valve (9815A~D) is opened.

Impact on DCD

There is no impact on DCD from this RAI as the response contains only additional information.

Impact on COLA

There is no impact on DCD from this RAI as the response contains only additional information.

Impact on PRA

There is no impact on PRA from this RAI and the response as this additional information does not cause any changes to the PRA.

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APPLICATION SECTION: **19.1**

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QUESTION NO. : 19-10

Section 19.1.6.3.3 includes a statement that the at - power flooding frequency is applied to the shutdown state. Please discuss how potential removal of flood barriers during shutdown is expected to affect flood propagation.

ANSWER:

Flooding propagation also treated the same at shutdown as at power. The effects of the flood barriers are considered only separation barriers between the east side and the west side in the reactor building.

Impact on DCD

There is no impact on DCD from this RAI as the response contains only additional information.

Impact on COLA

There is no impact on DCD from this RAI as the response contains only additional information.

Impact on PRA

There is no impact on PRA from this RAI and the response as this additional information does not cause any changes to the PRA.

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DATE OF RAI ISSUE: **4/17/2008**

QUESTION NO. : 19-11

Key assumption (h) on page 19.1 - 106 states that "potential plugging of the suction strainers due to debris is excluded from the PRA modeling." Please clarify whether the plugging failure mode is removed completely from the model, or whether the failure rate is simply unchanged from the at-power model as a result of foreign materials exclusion program assumptions during shutdown.

ANSWER:

The failure rate of suction strainer is simply unchanged from that of the at-power model.

Impact on DCD

There is no impact on DCD from this RAI as the response contains only additional information.

Impact on COLA

There is no impact on DCD from this RAI as the response contains only additional information.

Impact on PRA

There is no impact on PRA from this RAI and the response as this additional information does not cause any changes to the PRA.

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APPLICATION SECTION: **19.1**

DATE OF RAI ISSUE: **4/17/2008**

QUESTION NO. : 19-12

The discussion of low power states in section 19.1.6.1 states that low power states are "usually bounded by the full power case" and are "not explicitly analyzed herein at this stage." Under what conditions might a low power state not be bounded by the full power case? Is it expected that combined license (COL) applicants will analyze other low power states explicitly at another stage of the design?

ANSWER:

Other low power states will be analyzed during plant operation stage, when risk insights on low power mode will be used for making decisions. Under such situations, it may not be appropriate to bound low power state by full power case.

Impact on DCD

There is no impact on DCD from this RAI as the response contains only additional information.

Impact on COLA

There is no impact on DCD from this RAI as the response contains only additional information.

Impact on PRA

There is no impact on PRA from this RAI and the response as this additional information does not cause any changes to the PRA.

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APPLICATION SECTION: **19.1**

DATE OF RAI ISSUE: **4/17/2008**

QUESTION NO. : 19-13

Please provide a more detailed description of POS 10, the RCS leakage test state, in addition to that provided in Table 19.1 - 78. Table 19.1 - 79 indicates that this POS is expected to last 20.5 hours. How is decay heat removed during this POS? How do the time to boil and time to core uncovery compare to the expected duration of this POS?

ANSWER:

Decay heat is removed via SGs during POS 10. RCS is filled with water and RHR is isolated during this POS. Heat removal via SGs is the only considerable pass for heat removal from the RCS.

Moreover, decay heat intensity is relaxed at the time of POS 10, and therefore, it is expected that there is sufficient time to repair the faulted function to avoid bulk boiling of RCS. Accordingly, it is judged that core uncover will not occur even if heat removal function is lost.

Impact on DCD

There is no impact on DCD from this RAI as the response contains only additional information.

Impact on COLA

There is no impact on DCD from this RAI as the response contains only additional information.

Impact on PRA

There is no impact on PRA from this RAI and the response as this additional information does not cause any changes to the PRA.

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

APPLICATION SECTION: **19.1**

DATE OF RAI ISSUE: **4/17/2008**

QUESTION NO. : 19-14

Page 19.1 - 110 provides the results of a sensitivity study in which all human error probabilities (HEPs) are set to 0. Please provide a discussion of how important operator failures are to overall shutdown risk (i.e., what is the effect of the reverse situation, in which HEPs are set to a high value such as the 95th percentile values or 1.0?).

ANSWER:

During shutdown, operator actions are essential for mitigation of initiating events. If HEPs are set to high values, the conditional core damage probability (CCDP) will increase and will depend on the error assumed.

If HEPs were set to 1.0, CCDP given an initiating event will be 1.0.

Impact on DCD

There is no impact on DCD from this RAI as the response contains only additional information.

Impact on COLA

There is no impact on DCD from this RAI as the response contains only additional information.

Impact on PRA

There is no impact on PRA from this RAI and the response as this additional information does not cause any changes to the PRA.

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APPLICATION SECTION: **19.1**

DATE OF RAI ISSUE: **4/17/2008**

QUESTION NO. : 19-15

The PRA uses emergency diesel generator (EDG) failure data rather than gas turbine generator GTG) failure data. Please discuss whether EDG common - cause failure probabilities were used as well, and justify use of EDG failure data for common - cause failure of the GTGs.

ANSWER:

Common cause failure probabilities of EDGs were applied to GTGs in the US-APWR PRA.

At this time, there has been common cause failure data analysis for GTGs due to lack of operating experience.

For components with no common cause failure data, two approaches can be considered to estimate the common cause failure parameter values: (1) applying generic common cause failure parameters, and (2) applying common cause failure parameters of similar components.

(1) Generic common cause failure parameters

Common cause parameter applicable for components with no operating data is discussed in NUREG/CR-5485. Generic common cause failure parameters for such components were developed and provided in the report. The generic values for MGL parameters are shown in table 1 and 2.

(2) Common cause failure parameters of similar components

The component that is most similar to GTG, and has available common cause data, is the EDG. The GTG and EDG are composed by similar parts although they have difference in the engine itself. Table 7.4-1 shows the comparison of the composite parts between GTG and DG. As it can be seen in table, GTG and EDG have similar auxiliary systems composed by basically same quantity of parts. This implies that similar types of common cause failure, induced by failure of its auxiliary systems, can potentially occur in GTGs and EDGs. MGL parameters of EDGs are shown in table 1 and 2.

Common cause failure parameters applicable to GTGs:

Comparisons of MGL parameters for EDGs and generic components are shown in Tables 1 and 2. The MGL parameters of EDGs show higher values than that of generic values reported in NUREG/CR-5485. Taking into account that the EDG and GTG are composed by similar parts, there is a possibility that MGL parameters of GTG are close to EDGs and the generic MGL parameter underestimate the parameters of GTGs.

MGL parameter of safety injection (SI) pump, which is an active standby component, is also provided in table 2 and 3. The parameters of EDG and SI pump are quite similar compared to the difference from the generic parameter. This also implies that the parameters of EDG may better represent that of the GTG, which is also an active standby component.

For the reasons above, MGL parameters of EDGs are judged to be applicable to GTGs in the US-APWR PRA.

Table 1 Common cause failure parameter for fail to start

	MGL parameters			Reference
	β	γ	δ	
Generic value	0.05	0.5	0.5	NUREG/CR-5485
EDG	0.099	0.74	0.57	NUREG/CR-5497
SI pump	0.14	0.68	0.79	NUREG/CR-5497

Table 2 Common cause failure parameter for fail to run

	MGL parameters			Reference
	β	γ	δ	
Generic value	0.05	0.5	0.5	NUREG/CR-5485
EDG	0.13	0.7	0.57	NUREG/CR-5497
SI pump	0.072	0.62	0.46	NUREG/CR-5497

Table 3 Comparison of quantity of parts between GTG and DG

Parts	GTG	DG
1. Engine Generator	273 sorts of parts (for the type applied to US-APWR)	More than 3 times of GTG
2. Starting System	Almost same as DG	base
3. Fuel System	Almost same as DG	base
4. Lubricant Oil System	Slightly less than DG	base
5. Air Intake and Exhaust System	Almost same as DG	Base (Room Ventilation system included)
6. Cooling Water System	None	Required

(*) This table refers technical report MUAP-07024-P(R0) "Qualification and Test Plan of Class 1E Gas Turbine Generator System", submitted in December 2007.

Impact on DCD

There is no impact on DCD from this RAI as the response contains only additional information.

Impact on COLA

There is no impact on DCD from this RAI as the response contains only additional information.

Impact on PRA

There is no impact on PRA from this RAI and the response as this additional information does not cause any changes to the PRA.

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

APPLICATION SECTION: 19.1

DATE OF RAI ISSUE: 4/17/2008

QUESTION NO. : 19-16

Please clarify this statement on page 19.1 - 99: "Two normally closed motor - operated valves are aligned in series in each of four RHR train suction lines between the RCS and the [containment spray] CS/RHR pump." Does the statement refer to valves 9000 and 9001 (shown in Figure 19.1 - 2, sheets 4 and 5)? Please clarify that these valves are normally closed (9001 appears to be locked closed as well) during operation, but normally open during shutdown RHR operation.

ANSWER:

Yes, this statement refers to valves 9000 and 9001.

These valves are normally open during shutdown RHR operation.

Impact on DCD

There is no impact on DCD from this RAI as the response contains only additional information.

Impact on COLA

There is no impact on DCD from this RAI as the response contains only additional information.

Impact on PRA

There is no impact on PRA from this RAI and the response as this additional information does not cause any changes to the PRA.

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

APPLICATION SECTION: **19.1**

DATE OF RAI ISSUE: **4/17/2008**

QUESTION NO. : 19-17

The discussion of failure to maintain water level on page 19.1 - 101 states that this sequence does not apply to POS 8 - 1. Please describe this event tree as it applies to other POS and include with the other event tree figures.

ANSWER:

The sequence of loss of RHR caused by failing to maintain water level (FLML) is provided in the technical report[1)] (PRA report) chapter 20, subsection 20.10.3.2. Excerpts from the PRA report are provided in attachment A. The figure of FLML event tree is also provided Attachment A for Question 19-17. The event tree of FLML is the same as OVDR except that the CVCS is unavailable.

1) US-APWR Probabilistic Risk Assessment, MUAP-07030, Mitsubishi Heavy Industries, December. 2007.

Impact on DCD

There is no impact on DCD from this RAI as the response contains only additional information.

Impact on COLA

There is no impact on DCD from this RAI as the response contains only additional information.

Impact on PRA

There is no impact on PRA from this RAI and the response as this additional information does not cause any changes to the PRA.

Attachment A for Question 19-17

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

APPLICATION SECTION: 19.1

DATE OF RAI ISSUE: 4/17/2008

QUESTION NO. : 19-18

Section 19.1.6.3.2 provides a set of internal fire initiating events that excludes three initiators evaluated in the internal events shutdown analysis: loss of RHR caused by failing to maintain water level, loss of RHR caused by all other failures, and loss of component cooling water (CCW) or essential service water (ESW). Additionally, loss of CCW was evaluated in the at - power fire analysis. Please justify the exclusion of these three initiating events.

ANSWER:

1. OVDR (Loss of RHR caused by failing to maintain water level)

This initiating event is not excluded from fire PRA. This initiating event only occurs during mid-loop operations, which are POS 4-2, POS 4-3, POS 8-2 and POS 8-3. It does not occur during POS 8-1.

2. LORH (Loss of RHR caused by all other failures)

LORH was excluded from fire initiating events according to the following considerations:

- Safety related systems of the US-APWR consist of 4 trains, each physically separated by fire resistant walls or doors. Fire in one fire compartment does not result in loss of two or more trains of the RHR system.

- Nether of the RHR pump room are not adjacent to each other. Accordingly, even by fire affecting multiple fire compartments do not result in loss of in loss of two or more trains of the RHR system.

- At least three RHR pumps are normally available. LORH can be initiated when one train of the RHR system is lost by fire and the other two trains are lost by random failures. The frequency of such an event is considerably low in all POSs.

3. LOCS (Loss of CCWS/ESWS)

LOCS was excluded from fire initiating events according to the following considerations:

- Safety related systems of the US-APWR consist of 4 trains, each physically separated by fire resistant walls or doors. Fire in one fire compartment does not result in loss of two or more trains of the CCW system or ESW system.

- There are no penetrations between the CCW pump rooms. Accordingly, even by fire affecting multiple fire compartments do not result in loss of in loss of two or more trains of the CCW system or ESW system.

- At least three CCW trains are normally available. LOCS can be initiated when one train of the CCW system is lost by fire and the other two trains are lost by random failures. The frequency of such an event is considerably low in all POSs. This is the same for ESW system.

Loss of CCW (LOCS) is not evaluated in the at-power fire analysis according to the same reason described above.

Although Table 19.1-54 of the DCD states "yes" to total loss of component cooling water event, this is an editorial error. We will fix this error in the next DCD revision. "No" will be stated for Total loss of component cooling water event in this table.

Impact on DCD

There is no impact on DCD from this RAI as the response contains only additional information. However, editorial error pointed out in this additional information will be fixed.

Impact on COLA

There is no impact on DCD from this RAI as the response contains only additional information.

Impact on PRA

There is no impact on PRA from this RAI and the response as this additional information does not cause any changes to the PRA.

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Docket No.52-021

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

APPLICATION SECTION: **19.1**

DATE OF RAI ISSUE: **4/17/2008**

QUESTION NO. : 19-19

Please clarify this statement on page 19.1 - 114: "For internal fires, risk significant POS 8 - 1 of LPSD [low power and shutdown] has been estimated using the same methodology at power though the transient fire due to welding and cutting works and access for maintenance works have been specially reflected." How were fire initiation and propagation values adjusted to account for hot work and maintenance, which could both increase the likelihood of a fire and remove barriers to propagation of fires?

ANSWER:

1. Fire initiating frequencies to account for hot works and maintenance works

Maintenance activities were assumed in the compartments containing equipments that are out of service during the POS.

For any compartment that has equipment out of service, it is assumed that the fire ignition frequency caused by temporary combustibles increases. This is modeled by considering a weighting coefficient taking into account the impact of temporarily located combustibles. Thus hot work and maintenance work were considered in the fire initiation frequency.

Referring to NUREG/CR-6850 section 6.5.7.2, weighting coefficient values for cable fire (welding, cutting), transient (welding, cutting) and transient were set to "high" for compartments that contains equipments that is out of service.

2. Consideration of fire barriers during maintenance

Fire doors were assumed to be opened for adjacent compartments containing equipments that are out of service during the POS.

Impact on DCD

There is no impact on DCD from this RAI as the response contains only additional information.

Impact on COLA

There is no impact on DCD from this RAI as the response contains only additional information.

Impact on PRA

There is no impact on PRA from this RAI and the response as this additional information does not cause any changes to the PRA.

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QUESTION NO. : 19-20

Table 19.1 - 1 describes design features to reduce losses of RHR during plant shutdown. However, most of these features appear to have limited or no coverage in TS. Considering this table, please: (a) Discuss how the availability of these features designed to reduce shutdown risk will be ensured in TS or by other administrative controls. (b) Discuss how each feature is credited in the shutdown PRA. (c) Provide a sensitivity study for the shutdown PRA that credits only the systems required to be operable according to TS, since voluntary measures that are not required by current regulations could be withdrawn by licensees without NRC approval.

ANSWER:

This question will be answered later, within 60 days after RAI issue date.

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QUESTION NO. : 19-21

Page 19.1 - 102 states that "the configuration of offsite power in the shutdown PRA is considered the same as for the full power operations PRA." Table 19.1 - 80 indicates that the offsite power transformers are in standby status during shutdown. Please clarify the assumptions made about switchyard maintenance and other activities that could affect offsite power availability during shutdown.

ANSWER:

The base model for shutdown PRA assumes that switchyard maintenance is not performed during shutdown. Accordingly, availability of offsite power is assumed to be same with at power.

Sensitivity analysis was performed for cases where switchyard maintenance is performed during all POSs. In the sensitivity analysis, loss of offsite power frequency bases on low power shutdown data (NUREG/CR-6890) is applied to reflect the effect of maintenance activities on offsite power availabilities. This sensitivity analysis is named "Case 02" and documented in both the DCD and technical report. The sensitivity analysis showed that if the frequency of loss of offsite power events increases 3 times, LPSD CDF increases approximately 40%.

Impact on DCD

There is no impact on DCD from this RAI as the response contains only additional information.

Impact on COLA

There is no impact on DCD from this RAI as the response contains only additional information.

Impact on PRA

There is no impact on PRA from this RAI and the response as this additional information does not cause any changes to the PRA.

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QUESTION NO. : 19-22

POS 3 and 11 both span MODES 4 and 5, which have different TS requirements and potentially different systems available to mitigate accidents. Therefore, please justify the modeling of these two POS as single states and describe how available mitigation strategies in each MODE are addressed in the detailed system modeling.

ANSWER:

System availabilities are base on practical plant configurations, and not necessarily represented by the most severe condition given by TS requirement. Plant configurations in POS 3 and 4 are not represented by limiting conditions of MODES 4 and 5, and therefore, these two POS is modeled as single states.

POS categorization is described in DCD and in subsection 20.1.2 the technical report[1)].

1) US-APWR Probabilistic Risk Assessment, MUAP-07030, Mitsubishi Heavy Industries, December. 2007.

Impact on DCD

There is no impact on DCD from this RAI as the response contains only additional information.

Impact on COLA

There is no impact on DCD from this RAI as the response contains only additional information.

Impact on PRA

There is no impact on PRA from this RAI and the response as this additional information does not cause any changes to the PRA.

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QUESTION NO. : 19-23

Please clarify the isolation of letdown line (LOB) top event used in the "loss of RHR due to overdrain" (OVDR) event tree. Following an over - drain event, is it expected that the operator must always close the CLVD01 air - operated valve? (Note that this valve is downstream of the letdown isolation valves that are automatically closed on low loop level.) How is the failure probability of the automatically operated letdown isolation valves treated in the OVDR model?

ANSWER:

Air-operated letdown isolation valves are design to close automatically upon detection of low loop level. Over-drain initiating event is assumed to occur when the letdown isolation valve fails to close automatically following an over-drain event. Fault tree used to quantify the failure probability of automatically operated letdown isolation valves is shown in Figure 1

Operator action to manually close the air operated valve located downstream the letdown isolation valves is taken credit as a mitigation function following over-drain initiating event.

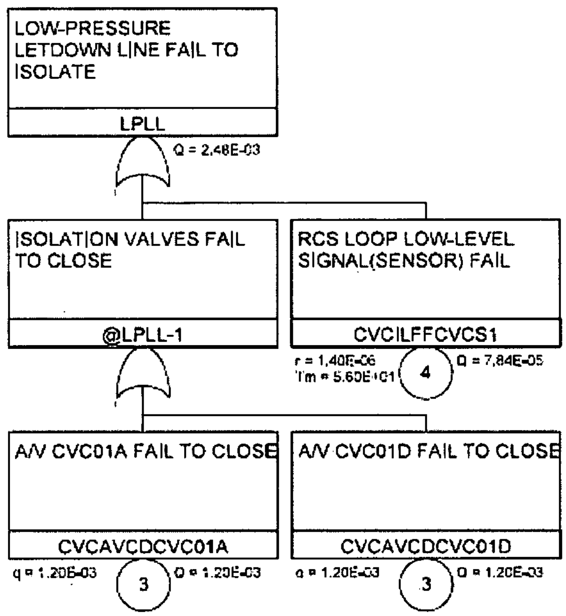

Figure 1 Fault tree of automatic low pressure letdown line isolation

Impact on DCD

There is no impact on DCD from this RAI as the response contains only additional information.

Impact on COLA

There is no impact on DCD from this RAI as the response contains only additional information.

Impact on PRA

There is no impact on PRA from this RAI and the response as this additional information does not cause any changes to the PRA.

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APPLICATION SECTION: **19.1**

DATE OF RAI ISSUE: **4/17/2008**

QUESTION NO. : 19-24

Please provide additional information on the operator actions required to successfully provide alternate component cooling water via the fire suppression system. The fault trees provided in the PRA Technical Report do not appear to include operator actions; however, Table 19.1 - 113 states that the operator will establish "the injection flow path from fire suppression tank to charging pump and from charging pump to fire suppression tank, and starting the fire suppression pump." What assumptions are made about procedures, cues, and other performance shaping factors related to these actions? For comparison, the human reliability analysis (HRA) portion of the PRA Technical Report includes an analysis of a similar failure (ACWOO02FS) with a failure probability that is higher than the front - line system failure probability for the shutdown PRA cited in Table 19.1 - 82.

ANSWER:

The operator action of alternate component cooling water via the fire suppression system is provided in the technical report[1)] (PRA report) chapter 9, (39) ACWOO02SC. Excerpts from the PRA report are provided in Attachment A for Question 19-24.

1) US-APWR Probabilistic Risk Assessment, MUAP-07030, Mitsubishi Heavy Industries, December. 2007.

Impact on DCD

There is no impact on DCD from this RAI as the response contains only additional information.

Impact on COLA

There is no impact on DCD from this RAI as the response contains only additional information.

Impact on PRA

There is no impact on PRA from this RAI and the response as this additional information does not cause any changes to the PRA.

Attachment A for Question 19-24

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APPLICATION SECTION: **19.1**

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QUESTION NO. : 19-25

The discussion of outage types in section 19.1.6.1 acknowledges that reduced inventory states can occur in outage type B (maintenance shutdown), but only the expected frequency of type C outages (refueling outages) is considered when calculating initiating event frequencies, as for LOCA on page 19.1 - 99. Please justify the 0.5 events per year shutdown frequency based on the expected frequency of all types of outages that result in reduced inventory.

ANSWER:

A sensitivity analysis which considers the shutdown frequency of all Outage types is carried out. The definition of Outage Type assumed in LPSD PRA for US-APWR is shown below.

- Type A: Shutdown for maintenance, and restart without reducing RCS inventory and refueling. RCS is closed and coolant inventory in the pressurizer is retained. Although a single SG may be unavailable either for the forced outage or for the planned maintenance outage, the other SGs are available for heat removal.

- Type B: Shutdown for maintenance with below normal RCS inventory and restart without refueling. In contrast to type A, the RCS inventory is reduced and/or the RCS boundary is opened. During the period when the RCS is open, SGs are not used for heat removal. Alternate heat removal function would be provided and planned.

- Type C: Refueling shutdown, which includes both type A and B conditions. In contrast to type A and B, there may be at times a large amount of additional water over the fuel during refueling, and the fuel may be unloaded from the RV to the SFP during the major maintenance activities. Reduced inventory condition states (mid-loop) may exist for periods before or after refueling.

As described above, type A outage involves plant shutdown for maintenance and then startup. If the RCS requires draining or opening, then the outage is type B and it invokes additional actions. Type C refueling outages invoke the same set of actions as a type B outage. In addition, type C outages invoke actions associated with removing the reactor vessel head, filling the refueling cavity, refueling draining the refueling cavity, installing the vessel head, and reducing inventory after refueling. In the LPSD PRA, outage type C "Refueling shutdown" is treated as a representative outage type. Therefore, shutdown frequency used in the base case only considers contribution of type C outages.

In this sensitivity analysis, shutdown frequencies of all three types of shutdowns are considered for each POSs as adequate to assess the impact of type A and B shutdown events. Outage types that need to be considered for evaluating the annual frequency of each POS are organized in table 1.

Reactor trip frequencies assessed for the LPSD PRA are provided in Table 2. General transient is defined as a transient that results in automatic reactor trip or manual reactor trip but does not directly degrade plant response for shutdown. Manual reactor trips occurred in 20% of all reactors trip events (Refer to NUREG/CR-5750). Frequencies of reactor trip frequencies are generally based on initiating event frequencies of at power PRA.

The breakdown of accident types contributing to reactor trip frequency is shown in Table 2. The total frequency of reactor trip due to transients, loss of support systems, and loss of offsite power, is estimated to be 0.88 events per year. The estimated frequency of controlled shutdown other than refueling outages is also shown in Table 2. The frequency of manual trip, which is a controlled shutdown, is estimated to be 0.16 events per year.

It is assumed that a refueling shutdown is scheduled every 24 months. Therefore, the frequency of refueling outages is 0.5 events per year.

The plant may be brought to cold shutdown condition following plant trip due to transients or other initiators caused during at power condition. In the PRA, it is assumed that, for events no further component failures occur (i.e. events with successful reactor trip and core cooling systems operation), 20 percent of the events require transition to cold shutdown state. Furthermore, it is assumed that only 10 percent of the cold shutdown events require drained maintenance. In other words, 90 percent of the cold shutdown events do not require drained maintenance. This percentage is based on engineering judgment.

Based on the shutdown frequencies outlines above and the percentages of forced outages that could require plant to transit to cold shutdown and drained conditions, the frequencies of the plant to transit to cold shutdown conditions (i.e., reactor coolant system non-drained and drained conditions) are calculated as follows:

- Type A: Non-drained maintenance with forced outage

  This frequency is calculated as the product of the trip frequency (0.88 events per year) and the percentage of shutdowns that takes the plant to cold shutdown with the reactor coolant system remaining filled (20 percent × 90 percent). This frequency is as below.

  $$(0.88 \times 0.20 \times 0.90) = 0.16 \text{ events/year} \qquad \text{(A)}$$

- Type B: Drained maintenance with forced outage

This frequency is calculated as the product of the trip frequency (0.88 events per year) and the percentage of shutdowns that takes the plant to cold shutdown with the reactor coolant system drained (20 percent × 10 percent). This frequency is as below.

$$(0.88 \times 0.20 \times 0.10) = 0.02 \text{ events/year} \qquad \text{(B)}$$

The frequency of manual trip which is a controlled shutdown, excluding refueling shutdown, is assumed to be 0.16 events per year. It is assumed that 80 percent of those cases require non-drained maintenance, 19 percent require drained maintenance without fuel removal, and 1 percent requires drained maintenance with fuel removal. This percent is based on engineering judgment.

Based on the controlled shutdown frequencies outlined above and the percentages of these outages that could take the plant to cold shutdown and drained conditions, the frequencies of the plant in cold shutdown conditions (i.e., reactor coolant system non-drained and drained conditions) are calculated as follows:

- Type A: Non-drained maintenance with controlled shutdown

  This frequency is calculated as the product of the controlled shutdown frequency (0.16 events per year) and the percentage of shutdowns that takes the plant to cold shutdown with the reactor coolant system kept filled (80 percent). This frequency is:

$$(0.16 \times 0.80) = 0.13 \text{ events/year} \qquad \text{(C)}$$

- Type B: Drained maintenance with controlled shutdown

  This frequency is calculated as the product of the controlled shutdown frequency (0.16 events per year) and the percentage of shutdowns that takes the plant to cold shutdown with the reactor coolant system drained (19 percent). This frequency is:

$$(0.16 \times 0.19) = 0.03 \text{ events/year} \qquad \text{(D)}$$

- Type C: Shutdown frequency for fuel removal maintenance

  This frequency is calculated as the product of the controlled shutdown frequency (0.16 events per year) and the percentage of shutdowns that takes the plant to refueling mode (1 percent), plus the refueling outage frequency (0.5 events per year). This frequency is:

$$(0.5 + (0.16 \times 0.01)) = 0.50 \text{ events/year} \qquad \text{(E)}$$

Based on the calculations above, the yearly frequencies of the three types of plant shutdown events are calculated as follows:

Type A: Non-drained maintenance frequency is

Sum of the forced and controlled shutdown frequencies

$(A + C) = 0.16 + 0.13 = 0.29$ events/year

Type B: Drained maintenance frequency is

Sum of the forced and controlled shutdown frequencies

(B + D) = 0.02 + 0.03 = 0.05 events/year

Type C: Refueling outages frequency is

Controlled shutdown frequencies

(E) = 0.50 events/year

Based on shutdown frequencies calculated above of the three maintenance activities, the frequencies of the plant in the different shutdown phases are formulated as follows:

- The yearly frequency of POS which exists in Type A, B, and C is as follows.

  0.29 + 0.05 + 0.50 = 0.84 events/year

- The yearly frequency of POS which exists in Type B, and C is as follows.

  0.05 + 0.50 = 0.55 events/year

- The yearly frequency of POS which exists in Type C is as follows.

  0.50 events/year

Impact on results

It is assumed that during type C outages, LOCA event, which is postulated as a loss of loop coolant via RHR to the RWSP caused by operator action, may occur during any POSs.

In some POSs loss of coolant may occur from other reasons, such as leak from CVCS. However, since it cannot be determined at this stage what operator actions will be performed in each POS, and owing to the fact that loss of coolant via the RHR can occur by single operator error, which results in relatively high event probability compared to other leaks, the PRA assumes that this probability can be conservatively applied to other POSs.

For outage type A and B, this operator action assumed in type C outage may not be performed throughout the outage. Accordingly, it may not be necessary to assume this kind of operator error. However, in this sensitivity analysis, this type of operator error is assumed.

Annual occurrence of LOCA events is shown in table 1. CDF from LOCA events based on the values in table 1 is shown in table 3.

The sensitivity case produces a LOCA CDF of 1.5E-07/RY, which is an increase of 15 percent in the base case CDF of 1.3E-07/RY. It is indicated that the impact of the frequency events per year which considered various outage types is small during plant shutdown conditions.

In level 2 PRA, conditional containment failure probability (CCFP) is conservatively assumed to be 1, since there are probabilities that the CV hatch is opened. For outage type A and B, CV hatch may not be opened throughout the outage period. However, this may vary with each situation, so we conservatively assume CCFP =1 fore these outage types. Since increase in CDF is small as discussed above, LERF will meet its goal even if CCFP=1 is assumed.

Table 1 Occurrence per year of each POS in US-APWR Shutdown PRA

POS	Description	Type			Occurrence Per Year	Initiating event frequency (LOCA)	
		A	B	C		Base case	Sensitivity case
POS 3	Hot and cold shutdown (RCS is filled with coolant)	×	×	×	0.84	1.0E-04	1.7E-04
POS 4-1	Cold shutdown (mid-loop operation, RCS closed)	N/A	×	×	0.55	1.0E-04	1.1E-04
POS 4-2	Cold shutdown (mid-loop operation, RCS opened)	N/A	×	×	0.55	1.0E-04	1.1E-04
POS 4-3	Cold shutdown operation (mid-loop operation, SG isolated)	N/A	N/A	×	0.50	1.0E-04	1.0E-04
POS 8-1	Cold shutdown (mid-loop operation, SG isolated)	N/A	N/A	×	0.50	1.0E-04	1.0E-04
POS 8-2	Cold shutdown (mid-loop operation, RCS opened)	N/A	N/A	×	0.50	1.0E-04	1.0E-04
POS 8-3	Cold shutdown (mid-loop operation, RCS closed)	N/A	N/A	×	0.50	1.0E-04	1.0E-04
POS 9	Cold shutdown (RCS is filled with coolant)	N/A	×	×	0.55	1.0E-04	1.1E-04
POS 11	Cold shutdown (RCS is filled with coolant)	×	×	×	0.84	1.0E-04	1.7E-04

Table 2 Initiating Event frequencies used in Event tree quantification

	Event	Frequency (Event/year)
-	General transient	8.0E-01
a)	Manual Trip (20% of General transient)	1.6E-01
b)	Transients (Automatic reactor trip, 80% of General transient) Loss of Feedwater Flow Main Steam Line Break Upstream of Main Steam Isolation Valves Main Steam Line Break Downstream of Main Steam Isolation Valves	6.4E-01 1.9E-01 1.0E-03 1.0E-02
C)	Loss of support System Loss of Component Cooling System or Service Water System Partial Loss of Component Cooling Water or Service Water System	 2.3E-05 3.2E-03
D)	Loss of Offsite Power	4.0E-02
	Total (b + c+ d)	8.8E-01

Table 3 Core Damage Frequency of LOCA

POS	Description	CDF	
		Base case	Sensitivity case
POS 3	Hot and cold shutdown (RCS is filled with coolant)	1.3E-08	2.1E-08
POS 4-1	Cold shutdown (mid-loop operation, RCS closed)	1.3E-08	1.4E-08
POS 4-2	Cold shutdown (mid-loop operation, RCS opened)	1.1E-08	1.2E-08
POS 4-3	Cold shutdown operation (mid-loop operation, SG isolated)	2.3E-08	2.3E-08
POS 8-1	Cold shutdown (mid-loop operation, SG isolated)	2.3E-08	2.3E-08
POS 8-2	Cold shutdown (mid-loop operation, RCS opened)	1.1E-08	1.1E-08
POS 8-3	Cold shutdown (mid-loop operation, RCS closed)	1.3E-08	1.3E-08
POS 9	Cold shutdown (RCS is filled with coolant)	1.3E-08	1.4E-08
POS 11	Cold shutdown (RCS is filled with coolant)	1.3E-08	2.1E-08
	Total CDF of LOCA	1.3E-07	1:5E-07

Impact on DCD

There is no impact on DCD from this RAI as the response contains only additional information.

Impact on COLA

There is no impact on DCD from this RAI as the response contains only additional information.

Impact on PRA

There is no impact on PRA from this RAI and the response as this additional information does not cause any changes to the PRA.

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QUESTION NO. : 19-26

Please identify how the US - APWR design has considered the Shutdown Management Guidelines in NUMARC 91 - 06, including how containment closure can be achieved in sufficient time to prevent potential fission product release (NUMARC Guideline 4.5).

ANSWER:

The shutdown response guideline will be developed making sure NUMARC 91-06 is satisfied. Shutdown response guideline will be prepared by COL applicant before fuel load.

Impact on DCD

There is no impact on DCD from this RAI as the response contains only additional information.

Impact on COLA

There is no impact on DCD from this RAI as the response contains only additional information.

Impact on PRA

There is no impact on PRA from this RAI and the response as this additional information does not cause any changes to the PRA.

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QUESTION NO. : 19-27

Table 19.1 - 113 provides several assumptions related to operator actions at shutdown, but no disposition (e.g., Tier 2, Tier 1, TS, or emergency response guidelines). Additionally, the table is missing key US - APWR features that reduce shutdown risk and their disposition. Please augment this table in the following areas of shutdown risk (the examples are not inclusive): (a) Key design features or SSCs that reduce the potential of reactor coolant diversion from the vessel through the RHR/CVCS systems (e.g., automatic closure of CVCS low - pressure letdown isolation valves on low RCS loop level) (b) Key design features, if any, that automate the response to losses of RHR (c) Key design features, if any, that automate RCS injection following loss of RHR, reactor coolant diversions, and LOCAs (d) Key operator actions and key pieces of instrumentation that are needed to support the associated operator actions (e.g., operator opening a gravity injection flow path) (e) Key SSCs that need to be available at shutdown to provide an alternate decay heat removal path using low pressure makeup and primary pressure relief (f) Key SSCs that are needed to reduce fire risk at shutdown and validate fire risk estimates (e.g., capability of fire watches when fire barriers are not intact).

ANSWER:

(a)

For the US-APWR, low-pressure letdown line isolation valves are installed. One normally closed air-operated valve is installed in each of two low-pressure letdown lines that are connected to two of four RHR trains. During normal plant cooldown operation, these valves are opened to divert part of the normal RCS flow to the CVCS for purification and the RCS inventory control. These valves are

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QUESTION NO. : 19-27

Table 19.1 - 113 provides several assumptions related to operator actions at shutdown, but no disposition (e.g., Tier 2, Tier 1, TS, or emergency response guidelines). Additionally, the table is missing key US - APWR features that reduce shutdown risk and their disposition. Please augment this table in the following areas of shutdown risk (the examples are not inclusive): (a) Key design features or SSCs that reduce the potential of reactor coolant diversion from the vessel through the RHR/CVCS systems (e.g., automatic closure of CVCS low - pressure letdown isolation valves on low RCS loop level) (b) Key design features, if any, that automate the response to losses of RHR (c) Key design features, if any, that automate RCS injection following loss of RHR, reactor coolant diversions, and LOCAs (d) Key operator actions and key pieces of instrumentation that are needed to support the associated operator actions (e.g., operator opening a gravity injection flow path) (e) Key SSCs that need to be available at shutdown to provide an alternate decay heat removal path using low pressure makeup and primary pressure relief (f) Key SSCs that are needed to reduce fire risk at shutdown and validate fire risk estimates (e.g., capability of fire watches when fire barriers are not intact).

ANSWER:

(a)

For the US-APWR, low-pressure letdown line isolation valves are installed. One normally closed air-operated valve is installed in each of two low-pressure letdown lines that are connected to two of four RHR trains. During normal plant cooldown operation, these valves are opened to divert part of the normal RCS flow to the CVCS for purification and the RCS inventory control. These valves are automatically closed and the CVCS is isolated from the RHRS by the RCS loop low-level signal to prevent loss of RCS inventory at mid-loop operation during plant shutdown.

(b)

There are no features that automate the response to loss of RHR.

(c)

There are no features that automate RCS injection.

(d)

Key operator actions and key pieces of instrumentation that are needed to support the associated operator actions are listed below.

- Operator opening alternative CCW flow path

  Key instrumentations are CCW flow indicators and pressure indicators to identify "CCW flow Low".

- Operator opening a RWSAT supply flow path from RWSP to be able to inject to the RCS by charging pump

  Key instrumentations are RCS temperature indicators to identify "increasing RCS temperature" and RCS water level indicators to identify "decreasing RCS water level" during RHR/SG/CHI cooling.

- Operator connecting the alternative power to emergency power bus

  Key instrumentations are voltage indicators of the safety buses to identify "no voltage on the all bus line" or "no voltage on the either bus line.".

- Operator actuating the high head injection pump

  Key instrumentations are RCS temperature indicators to identify "increasing of RCS temperature" and RCS water level indicators to identify "decreasing RCS water level" during RHR/SG/CHI cooling.".

- Operator closing the valves on RHR pump suction line

  Key instrumentations are RCS water level indicators to identify "low RCS water level ".

- Operator closing the low-pressure letdown line isolation valve

Key instrumentations are the RCS water level indicators to "identify low RCS water level during drain operation.".

- Operator actuating a standby RHR train flow path

  Key instrumentations are the status indicators of RHR pump to identify failure of running RHR pump.

- Operator actuating a standby CCW train flow path

  Key instrumentations are CCW flow indicators and pressure indicators to identify failure of running CCW pump.

- Operator actuating a standby ESW train flow path

  Key instrumentations are ESW pressure indicators to identify failure of running ESW pump.

- Operator opening a standby strainer line

  Key instrumentations are ESW pressure indicator and flow indicator to identify "ESW strainer plugging" and the differential pressure indicator to identify the plugged strainer.

Task and human error probabilities of key operator actions are provided in the technical report [1)] (PRA report). Excerpts from the PRA report are provided in Attachment A for Question 19-27.

(e)

Key SSCs that need to be available at shutdown to provide an alternate decay heat removal path using low pressure makeup and primary pressure relief are as follows:

- RCS injection line from charging pumps
- Water supply line from RWSP to RWAST, which is water source of charging pumps
- RCS injection line from safety injection pump
- RCS injection line for gravitational injection
- Water supply line from RWSP to SFP, which is the water source of gravitational injection

(f)

Key SSCs that are needed to reduce fire risk at shutdown and validate fire risk estimates are as follows:

- Management procedures and controls necessary to ensure fire protection hazards introduced during maintenance and refueling is provided.

- Essential systems are separated by fire barriers.
- Cables of each train run in separate raceways (cable tray and conduits) and are physically separated from cables of other trains.
- Cable of each trains are laid in different and separated room/area, except in main control room (MCR) and containment vessel (CV).
- For MCR, train separation is maintained from remote shutdown room. Fire is accommodated by remote shutdown room. For CV, cables are not crossed over with other trains and train thus separation is maintained.

Fire watchers are not taken credit in the LPSD fire PRA.

1) US-APWR Probabilistic Risk Assessment, MUAP-07030, Mitsubishi Heavy Industries, December. 2007.

Impact on DCD

There is no impact on DCD from this RAI as the response contains only additional information.

Impact on COLA

There is no impact on DCD from this RAI as the response contains only additional information.

Impact on PRA

There is no impact on PRA from this RAI and the response as this additional information does not cause any changes to the PRA.

Attachment A for Question 19-27

19-27-A8