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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-09082

Subject: MHI's Responses to US-APWR DCD RAI No. 138-1704 Revision 1

Reference:
1) "Request for Additional Information No. 138-1704 Revision 1, SRP Section: 19-Probabilistic Risk Assessment and Severe Accident Evaluation, Application Section: 19.1.6.1," dated January 9, 2009
2) Letter MHI Ref: UAP-HF-09042 from Y. Ogata (MHI) to U.S. NRC, "MHI's Responses to US-APWR DCD RAI No.138-1704," dated February 6, 2009

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

Enclosed are the second responses to the RAIs contained within Reference 1. n the initial responses submitted with Reference 2, MHI committed to submit responses to 19-206, 19-207, 19-208, 19-209, 19-214, 19-218, 19-222, 19-230, 19-234, 19-236, 19-247 and 19-249 within 60 days after RAI issue date.

Please contact Dr. C. Keith Paulson, Senior Technical Manager, Mitsubishi Nuclear Energy Systems, Inc. if the NRC has questions concerning any aspect of the submittal. His contact information is below.

Sincerely,

Y. Ogata

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

Enclosure:

1. "Responses to Request for Additional Information No.138-1704 Revision 1"

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Enclosure 1

UAP-HF-09082
Docket No. 52-021

Response to Request for Additional Information
No. 138-1704 Revision 1

March 2009

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

3/10/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: **NO.138-1704 REVISION 1**

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

APPLICATION SECTION: **19.1.6.1**

DATE OF RAI ISSUE: **1/9/2009**

QUESTION NO. : 19-206

Justify the simplified treatment of plant operating states (POS) other than 8-1 in the USAPWR shutdown probabilistic risk assessment (PRA). In addition to generally discussing the rationale, address how each specific issue below affects the ability to obtain results and insights from the US-APWR shutdown PRA.

a. Uncertainty results are not available for total shutdown core damage frequency (CDF) or large release frequency (LRF).
b. Quantitative importance measures are not available for POS other than 8-1, so the qualitative assessment of importance used as input to the reliability assurance program (RAP) and other programs may over- or under-estimate the importance of specific equipment.
c. LRF may be overestimated, given that containment will not realistically be open in all POS.
d. The assessment of additional mitigating systems (secondary cooling by the steam generators (SG) and gravity injection (GI) from the spent fuel pool (SFP)) considers dependence between human actions, but does not carry forward equipment failures from other top events in the accident sequence that could disable the SG or GI mitigating systems.
e. Success criteria may be different for POS before and after refueling (see RAI 88, Question 19-139).

Answer:

Human errors dominate CDF of LPSD PRA, because operator actions are essential for operating the mitigation systems in LPSD state. The operator actions for the mitigation systems are not much different among POSs. Also risk during LPSD is depend on maintenance schedules and is not confirmed with the only one maintenance schedule on the DC phase.

MHI judged that the major object of the LPSD PRA at DC phase is to identify the key risk insights so that the evaluation of all POSs during LPSD would not essential for the DC phase LPSD PRA. Risk insights would be derived from POS 8-1 which is the typical risk significant mid-loop operation state. Also most available mitigation functions and initiating events are common between POS 8-1 and other POSs while there are few exceptions. In addition, it would be possible to estimate the risk of other POSs simply using the CCDP of POS 8-1.

Answer to the question a.

LPSD PRA on the DC phase includes many uncertainties because maintenance schedule is not specified. Thus, the manner that uses the uncertainty of POS 8-1 as a substitute for that of POS other than 8-1 is considered to be applicable. Additionally, most available mitigation functions and initiating events are common between POS 8-1 and other POSs with few exceptions, so uncertainties of other POSs are assumed to be similar with POS 8-1.

Answer to the question b.

Quantitative importance measures are not available for POSs other than 8-1. However, regarding to the identification of risk significant SSCs using the conservative method, MHI judged that it covers the extensive SSCs for programs in the DC stage.

Answer to the question c.

As NRC staff pointed out, the assumption that the LRF equals to the CDF in shutdown operation is conservative because the containment is not always open in the shutdown states. In the DC stage, however, MHI views this assumption as appropriate. Even in realistic conditions, the containment is expected to be open in many POSs. Because of this, the conservative assumption does not significantly affect the insights from PRA. Furthermore, mitigation procedures in LPSD states will be implemented assuming the conservative PRA conditions. In the future, separately from DC application, the shutdown PRA considering containment conditions will be done for configuration risk management if necessary.

Answer to the question d.

The secondary cooling by SGs depends on other mitigation systems supported by Class 1E power, while gravity injection is depending on Refueling Water Recirculation pump, RWSP and RHRS. Dependency between operator actions is dominant contributor than the dependency between component failures for mitigation systems. For example, Table 20.10-4 of the PRA report shows that the dominant cutsets of CDF are human errors, especially dependency between tasks. This result indicates that the effect of the dependency of human errors is greater than that of component failures considerably.

Answer to the question e.

As the response for the question 19-139 (RAI#88), only the RHR system during POS 4-1 has different success criteria. The success criterion of RHR system is one train operable for POSs other than POS 4-1. On the other hand, if the pressurizer spray vent line is opened at the initial stage of POS 4-1, the success criterion for RHR system is two trains operable. Now MHI address impact to PRA for changing success criteria of RHRS during POS 4-1 from one train required to two trains required. Also MHI would report this impact for the follow-up to the question 19-139.

Impact on DCD
There is no impact on DCD.

Impact on COLA
There is no impact on COLA.

Impact on PRA
There is no impact on PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

3/10/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: **NO.138-1704 REVISION 1**

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

APPLICATION SECTION: **19.1.6.1**

DATE OF RAI ISSUE: **1/9/2009**

QUESTION NO. : 19-207

(Follow-up to Question 19-59) The response to Question 19-59 indicates that Design Control Document (DCD) Table 19.1-113 "will be revised adding the requested dispositions for all entries, and include the assumptions and insights provided in response to Question 19-27."

a. Provide marked-up pages of an interim revision so the staff can review the proposed changes.
b. Discuss the process (e.g., internal procedure) for communicating these assumptions and insights to the developers of procedures and training.
c. State the mechanism (e.g., combined license (COL) item) for ensuring that these assumptions and insights remain valid in the PRA for the as-built, as-operated plant. (Note that the COL items provided in DCD Section 19.3.3 direct the COL applicant only to update external events and fragilities.)
d. Ensure that the revised table includes (or justifies exclusion of) all key assumptions and PRA-based insights identified in the PRA, DCD Chapter 19, and RAI responses. Several examples related to shutdown risk are (note that this list is not inclusive):

* DCD page 19.1-101: assumption related to the use of freeze seals
* DCD pages 19.1-108 to 19.1-109: key assumptions related to shutdown initiating events and system models
* DCD page 19.1-131 and PRA page 22-18: assumptions related to internal flooding during shutdown (note that PRA description is more detailed)
* PRA page 9-1: assumption that revised evaluation of operator actions can be performed as more specific design information becomes available later
* PRA page 22-20: insights from shutdown internal flooding assessment
* Response to Question 19-11: assumption that strainer plugging failure is no more likely during shutdown than at power
* Response to Question 19-26: assumption that NUMARC 91-06 will be satisfied in the shutdown response guideline
* Response to Question 19-27: insights about shutdown-related design features and key

operator actions, instruments, and equipment
* Response to Question 19-45: assumptions about operator actions required to initiate reflux cooling and use of nitrogen to drain SG tubes
* Response to Question 19-45: insight that vent must be closed in certain conditions to prevent core damage
* Response to Question 19-50: assumptions about control of doors during shutdown
* Response to Question 19-63: assumptions used to justify exclusion of boron dilution as an initiating event in the shutdown PRA
* Response to Question 19-66: insight about reactor vessel penetrations
* Response to Question 19-73: assumption that indication will be available during shutdown

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ANSWER:

Attached table is the list of the insights and assumptions that are identified in the Chapter 19 of DCD Revision 1, the PRA report (MUAP-07030(R1), and the responses to RAIs related to DCD Chapter 19. This table is the interim revision of Table 19.1-115 of DCD Revision 1 (Table 19.1-113 of DCD Revision 0). The revised portions will be marked-up with underlines.

This table will include the all key assumptions and PRA-based insights identified in the PRA, DCD Chapter 19, and RAI responses. Also the column "Dispositions" will involve the programs or documents to develop the procedures/trainings or to validate that the assumptions and insights are valid in the PRA for the as-built, as-operated plant.

Please note that the completed table will be prepared by the next revision of DCD.

Impact on DCD
Revise the Table 19.1-115 of the DCD revision 1 by the next revision.

Impact on COLA
There is no impact on COLA.

Impact on PRA
Incorporate the revision of the key insights and assumptions into the PRA report.

Key Insights and Assumptions	Dispositions
Design features and insights	
1. The high head safety injection system consists of four independent and dedicated SI pump trains. The SI pump trains are automatically initiated by a SI signal, and supply borated water from the RWSP to the reactor vessel via direct vessel injection line. This system provides safety injection function during LOCA events and feed and bleed operation. During plant shutdown, high head safety injection system provides RCS makeup function in case RHR function is lost.	6.3
2. There are four accumulators, one supplying each reactor coolant cold leg. The accumulators incorporate internal passive flow dampers, which function to inject a large flow to refill the reactor vessel in the first stage of injection, and then reduce the flow as the accumulator water level drops. Thus the accumulators provide integrated function of low head injection system in the event of LOCA.	6.3
3. Charging injection is provided by the chemical volume control system. The charging and letdown system provides a function to maintain programmed water level in the pressurizer and maintain appropriate reactor coolant inventories in reactor coolant system (RCS) during all phases of plant operation. In case small leak of the reactor coolant occur without generating safety injection signal, the volume of the reactor coolant can be recovered with the charging pump, provided that the water of the refueling water storage pit (RWSP) is supplied to the tank. During plant shutdown, charging injection provides RCS makeup function in case RHR function is lost.	9.3.4
4. The CS/RHRS consists of four independent subsystems, each of which receives electrical power from one of four safety buses. Each subsystem includes one CS/RHR pump and one CS/RHR heat exchanger, which have functions in both the CS system and the RHRS. CS/RHRS provides multiple functions such as, (1) containment spray to decrease pressure and temperature in the CV, (2) alternate core cooling in case all safety injection systems fails at the LOCA, (3) RHR operation for long term core cooling, and (4) heat removal function for long term C/V cooling. During plant shutdown, RHRS provides function to remove decay heat from the RCS.	5.4.7 6.2.2
5. Reactor trip signal is provided by the RPS, which consists of four redundant and independent trains. Four redundant measurements using sensors from the four separate trains are made for each variable used for reactor trip. In addition, diverse actuation system is provided as a countermeasure against software failure of the digital I&C system.	7.2

Key Insights and Assumptions	Dispositions
6. EFWS consists of two motor-driven pumps and two steam turbine-driven pumps with two emergency feedwater pits. The EFWS supplies feedwater to the steam generators at a sufficient flow rate required for the transient conditions or postulated accidents and hot standby.	10.4.9
7. The pressurizer is sized to have sufficient volume to accomplish the preceding requirements without the need of power-operated relief valves. Safety depressurization valves (SDVs) are provided at top head of the pressurizer in order to cool the reactor core by feed and bleed operation when loss of heat removal from steam generator (SG) occurs.	5.4.12
8. MSIVs are installed in each of the main steam lines to (1) limit uncontrolled steam release from one steam generator in the event of a steam line break, and to (2) isolate the faulted SG in the event of SGTR.	10.3
9. The CCW system provides cooling water required for various components during all plant operating conditions, including normal plant operating, abnormal and accident conditions. During plant operation, CCW provides cooling water for the thermal barrier of the RCP to maintain RCP seal integrity. The CCW also functions as the heat sink for the CS/RHR system as well as the alternative containment cooling.	9.2.2
10. Four class 1E gas turbine generators (GTGs) are provided to supply power to their dedicated safety bus as a counter measure against loss of offsite power. When loss of offsite power occurs, GTGs automatically start and would accept load in less than or equal to 100 seconds after receiving the start signal.	8.3
11. Common cause failure between class 1E GTG and non-class 1E GTG supply is minimized by design characteristics. Different rating GTGs with diverse starting system, independent and separate auxiliary and support systems are provided to minimize common cause failure.	8.4.1.3
12. In addition to the class 1E GTGs, two non-classes 1E GTGs are provided to supply power to permanent buses. These two GTGs also functions as an alternative ac power source (AAC), which can supply power to any two of the four safety buses in case class 1E GTGs fail during loss of offsite power. To minimize the potential for common cause failures with the class 1E GTGs, different rating GTGs with diverse starting system are provided. Furthermore, the auxiliary and support systems for the AAC GTGs are independent and separate from the class 1E GTGs to minimize the potential for common cause failures.	8.4.1.3
13. Instrumentations for detecting core damage with high reliability are provided.	7.3 & 7.5

Key Insights and Assumptions	Dispositions
14. The containment spray system is designed to perform two major functions, i.e. (1) containment heat removal and (2) fission product removal. As for the features for mitigation of the consequences of core damage and prevention of release from containment, the above function (1) is expected. The containment spray system also takes a fundamental role for the reactor cavity flooding.	6.2.2 & SAMG
15. Hydrogen control system that consists of igniters is provided to limit the combustible gas concentration. The igniters start with the safety injection signal and are powered by two non-class 1E buses with non-class 1E GTGs.	ITAAC & SAMG
16. RCS depressurization system dedicated for severe accident is provided to prevent high pressure melt ejection. The location of release point from the valve is in containment dome area. This operation is implemented after onset of core damage and before reactor vessel breach.	ITAAC & SAMG
17. Reactor cavity flooding system by firewater injection is provided to enhance heat removal from molten core ejected into the reactor cavity. This system is available as a countermeasure against severe accidents even in case of fire. This operation is implemented after onset of core damage and before reactor vessel breach.	ITAAC & SAMG
18. Alternate containment cooling system using the containment fan cooler units is provided to prevent containment over pressure even in case of containment spray system failure. This operation is implemented after containment pressure reaches the design pressure.	ITAAC & SAMG
19. The FSS is also utilized to promote condensation of steam. The FSS is lined up to the containment spray header when the CSS is not functional, and provides water droplet from top of containment. This will temporarily depressurize containment.	ITAAC & SAMG
20. Reactor cavity has a core debris trap area to prevent entrainment of the molten core to the upper part of the containment.	ITAAC
21. The other cavity flooding system is a set of drain lines from SG compartment to the reactor cavity. Spray water which flows into the SG compartment drains to the cavity and cools down the molten core after reactor vessel breach.	ITAAC
22. Reactor cavity is designed to ensure thinly spreading debris by providing sufficient floor area and appropriate depth.	ITAAC
23. Reactor cavity floor concrete is provided to protect against challenge to liner plate melt through.	ITAAC
24. The containment prevents or limits the release of fission products to the environment.	19.2.4
25. Main penetrations through containment vessel are isolated automatically with the containment penetration signal even in case of SBO.	6.2.4

Key Insights and Assumptions	Dispositions
26. Main equipments and instrumentations used for severe accident mitigation are designed to perform their function in the environmental conditions such as containment overpressure and temperature rise following hydrogen combustion.	19.2.3.3.7
27. Risk significant SSCs are identified for the RAP	17.4
28. In US-APWR, the instrumentation piping are installed at upside of the RV. Therefore, the reactor vessel bottom head has no penetrations that could lead to inadvertent draining of the RCS during shutdown.	(RAI19-66)
Seismic risk insights	19.1.5.1.2
1. There are some important safety-related SSCs for which seismically induced failure would lead directly to core damage. In this SMA study, these SSCs have HCLPF values in excess of 0.50 g. If any of these SSCs were built with a HCLPF lower than 0.50g, the plant HCLPF would also be lower than 0.50 g.	19.1.5.1.2
2. The plant HCLPF is dominated by HVAC chillers (0.50g), safety power source buildings (0.50g), essential service water Intake structure (0.50 g), essential service water pipe tunnel (0.50g), fuel assembly (0.50g) and class 1E gas turbine generators (0.50g). If those SSCs HCLPF value were to be increased to any value above 0.53 g, the plant HCLPF would increase to 0.53 g and would be dominated by the cable tray (0.53 g).	19.1.5.1.2
3. The analysis did not identify any important sequence containing mixed cutsets. The only sequences containing mixed cutsets are LOOP sequences induced by the failure of ceramic insulators (0.08 g) with the random failures. However, the probability of such random failures occurring is low (i.e., less than 1.0E-03). This means that random failures are unlikely to occur in a seismically-initiated accident sequence.	19.1.5.1.2
4. No credit is taken for operator actions in this study. The plant HCLPF is dominated by failures of SSCs result in core damage directly, such as the failure of structures.	19.1.5.1.2

Key Insights and Assumptions	Dispositions
5. Depending on whether offsite power is available, different scenarios to trip the reactor are considered. In the case offsite power failed (i.e., a LOOP initiating event), the control rod motor generator sets would be de-energized following LOOP and succeed in the release of control rods into the core even if the reactor trip function failed. Only when the control rod system is failed would the reactor trip be failed. This scenario is considered in this study and the HCLPF value for this event is 0.67 g (dominated by the control rod HCLPF). In case offsite power is available, the failure of the reactor trip function should be considered. However, the HCLPF for the reactor trip system would be higher than 0.67 g determined when offsite power is lost. This is because HCLPFs for electrical equipment and sensors/transmitters to trip the reactor are above 0.67 g. Thus, whether offsite power is available or not, the HCLPF value (i.e., seismic capacity) to trip the reactor is higher than the plant HCLPF of 0.50 g.	19.1.5.1.2
6. There are no vulnerabilities for containment performance (i.e., containment integrity, containment isolation and prevention of bypass functions) due to a seismic event.	19.1.5.1.2
Internal fire risk insights	19.1.5.2
1. Fire protection seals are provided for walls, floors, and ceilings, which compose the fire area boundaries divided by four train areas.	19.1.5.2.2
2. Bus ducts and circuit breaker panels of safety ac system and alternative ac system in the T/B Electric Room are segregated into two groups by qualified fire barriers.	19.1.5.2.2
Internal Flood risk insights	
1. East side and west side of reactor building are physically separated by flood propagation preventive equipment and the connections are kept closed and locked.	19.1.5.3.2
2. Areas between the reactor building and the turbine building are physically separated by flood propagation prevention equipment.	19.1.5.3.2
3. The flood barriers that separate the reactor building between east side and west side are important to safety for the operation of the facility. These doors should be monitored and controlled during plant operation and maintenance.	(RAI 19-50) COL13.5(1) COL13.5(7)

Key Insights and Assumptions	Dispositions
Operator actions and assumptions Operator actions modeled in the PRA are based on symptom oriented procedures. Risk significant operator actions identified in the PRA will be address in the EOP.	EOP/ERG
Operator actions during at power events and assumptions	19.1.4
1. In the case of loss of CCW cooling function, with CCW flow rate – low signal or CCW pressure – low signal, operators connect the non-essential chilled water system and the fire suppression system to the CCWS in order to cool the charging pump and maintain RCP seal water injection.	EOP/ERG
2. In the case of loss of running train CCW cooling function, with running train CCW flow rate – low signal, CCW pressure – low signal, and running charging pump condition, operators start another stand-by charging pump in order to maintain RCP seal water injection.	EOP/ERG
3. When station blackout occurs, with emergency bus voltage – low signal after connecting class 1E GTGs, operators connect the alternative ac power with alternate Gas turbines to class 1E bus in order to recovery emergency ac power.	EOP/ERG
4. If emergency feed water pumps cannot feed water to two intact SGs, operators will attempt to open the cross tie-line of EFW pump discharge line in order to feed water to two more than SGs by one pump.	EOP/ERG
5. The CS/RHR System has the function to inject the water from RWSP into the cold leg piping by switching over the CS/RHR pump lines to the cold leg piping if all safety injection systems failed (Alternate core cooling operation). Alternate core cooling operation may be required under conditions where containment protection signal is valid. In such cases, alternate core cooling operation is prioritized over containment spray, because prevention of core damage would have higher priority than prevention of containment vessel rupture.	EOP/ERG
6. When 3 EFW pumps fail, with those pumps flow rate, operator remotely open EFW pump outlet tie-line stop valve in order to ensure feedwater to multiple SG.	EOP/ERG
7. When any two EFW pumps that commonly utilize at EFW pit have failed, operators supply water to operating EFW pumps from alternate EFW pit or demineralized water storage pit in order to ensure the water source.	EOP/ERG

Key Insights and Assumptions	Dispositions
8. In the case of fail to isolate failed SG, but success to depress enough RCS by secondary side cooling and Safety depressurization valve in SGTR event, operators do RCS pressure control in order to prepare to early RHR cooling in order to ensure long term heat removal. (RCS pressure control means stopping SI safety injection and starting charging pump. RCS pressure under SI injection keeps higher for connecting RHR system. Charging pump is back up for failure RHR cooling after stopping SI injection.)	EOP/ERG
9. In the case of above, if operators fail to move RHR cooling after SI injection control, operators start to bleed and feed operation. Operators open safety depressurization valve and start the safety injection pump in order to ensure long term heat removal.	EOP/ERG
10. When the main steam isolation valve fail to close in SGTR event, with status signal of this valve, operators try to close this valve in order to stop leakage of RCS coolant from the failed SG.	EOP/ERG
11. When the main steam isolation valve fail to close in SGTR event, with SG pressure indication after above operation, operators close turbine bypass stop valves in order to stop leakage of RCS coolant from the failed SG.	EOP/ERG
12. In the case of loss of failed SG isolation function in SGTR event, with SG pressure indication after above operation, operators open main steam depressurization valve of intact SG loop in order to promote SG heat removal and to depressurize RCS and move to cool down and recirculation operation.	EOP/ERG
13. In the case of loss of secondary side cooling function by emergency feedwater system in transient events including turbine trip, load loss event etc., with emergency feedwater pump flow rate, operators start to recover main feedwater system in order to maintain secondary side cooling.	EOP/ERG
14. In the case of loss of SI injection function entirely in LOCA event, with SI flow rate and RCS temperature indication, operators provide secondary side cooling to reduce RCS pressure and temperature by opening the main steam depressurization valves manually and supplying water from the emergency feedwater system in order to enable low pressure injection with containment spray system / residual heat removal system.	EOP/ERG
15. In the case of loss of contain spray system function, with contain spray pump flow rate and CV pressure-high signal, operators provide preparation for CV natural recirculation cooling operation in order to remove heat from CV. This preparation contains CCW pressurization with N2 gas, disconnection heat load of non-safety chiller and CRDM etc. and connection to containment fan cooler units.	EOP/ERG
16. In the case of leakage of the RWSP water from HHIS piping, CSS/RHR piping or refueling water storage system piping, with drain sump water level – abnormally high, operators close the RWSP suction isolation valves respectively in order to prevent leakage of RWSP water from failed piping.	EOP/ERG

Key Insights and Assumptions	Dispositions
17. When the CV isolation signal fail to automatically actuate, with CV pressure abnormally high signal, operators manually actuate the CV isolation signal in order to remove heat from the containment vessel.	EOP/ERG
18. When the ESF actuation signal fail to automatically actuate, with pressurier pressure-abnormally low signal or CV pressure abnormally high signal or ECCS actuation failure alarm, operators manually actuate the ESF actuation signal in order to recover the ESF actuation signal.	EOP/ERG
19. When the CCW header tie-line isolation valves fail to automatically close with specific signals which contain SI signal plus UV signal, P signal, and surge tank level low signal, operators manually close these valves in order to separate CCW header.	EOP/ERG
Operator actions during LPSD events and assumptions	19.1.6
1. When the RCS is under atmospheric pressure, gravity injection from SFP is effective. Operator will perform the gravity injection by opening the injection flow path from SFP to RCS cold legs, and supplying water from RWSP to SFP.	EOP/ERG
2. When station blackout occurs, with emergency bus voltage – low after connecting EGTs, operators connect the alternative ac power with alternate gas turbines to class 1E bus in order to recover emergency ac power.	EOP/ERG
3. In the case of loss of CCW/ESW, operator will perform alternate charging pump cooling in order to maintain RCS injection by establishing 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.	EOP/ERG
4. In the case of loss of CCW cooling function, with CCW flow rate – low or CCW pressure – low, operators connect the fire suppression system to the CCWS and start the fire suppression pump in order to cool the charging pump and maintain injection to RCS.	EOP/ERG
5. When RCS makeup is required during charging pump being standby, with RCS water level – low, operators start the charging pump in order to recover water level in the RCS.	EOP/ERG
6. In case LOCA occurs in RHR line, operator will perform isolation of the RHR hot legs suction isolation valves.	EOP/ERG
7. When LOCA occurs, with RCS water level – low, operators close the RHR hot legs suction isolation valves in order to stop leakage of RCS coolant from RHRS where LOCA occurs.	EOP/ERG
8. In case the RCS water level decreases during mid-loop operation and the failure of automatic isolation valve occurs, operator will perform the manual isolation of low-pressure letdown line.	EOP/ERG
9. When over-draining occurs and the automatic isolation valve fails, with RCS water level – low, operators close the valve on the letdown line in order to stop draining.	EOP/ERG

Key Insights and Assumptions	Dispositions
10. In the case of loss of decay heat removal functions by RHRS and SGs, and loss of injection by SI pump, with RCS temperature – high or RCS water level – low, and RWSAT water level – low, operators open the injection path from RWSP to RWSAT and start the refueling water recirculation pump in order to maintain RCS water level.	EOP/ERG
11. In the case of loss of decay heat removal functions by RHRS and SGs, with RCS temperature – high or RCS water level – low, operators start the safety injection pump in order to maintain RCS water level.	EOP/ERG
12. In the case of failure of running RHRS, with RHR flow rate – low, operators open the valves on the standby RHR suction line and discharge line and start the standby RHR pump in order to maintain RHR operating.	EOP/ERG
13. In the case of leakage of the RWSP water from HHIS piping, CSS/RHR piping or refueling water storage system piping, with drain sump water level – abnormally high, operators close the RWSP suction isolation valves respectively in order to prevent leakage of RWSP water from failed piping.	EOP/ERG
14. In the case of failure of running CCWS, with CCW flow rate – low, operators start the standby CCW pump in order to maintain CCWS operating.	EOP/ERG
15. In the case of failure of running ESWS, with CCW flow rate – low, operators start the standby ESW pump in order to maintain ESWS operating.	EOP/ERG
16. When ESW strainer plugs up, with ESW pump pressure – normal, ESW flow rate – low and differential pressure – significant, operators switch from plugged strainer to standby strainer in order to maintain ESWS operating.	EOP/ERG
17. In the case of loss of decay heat removal functions from RHRS and SGs, and loss of injection by SI pump and charging pump, with RCS temperature – high or RCS water level – low, and SFP water level – low, operators open flow path from RWSP to SFP and the gravity injection path from SFP to RCS cold leg, then start the refueling water recirculation pump and supply water to RCS in order to maintain RCS water level.	EOP/ERG
18. In the case of loss of decay heat removal functions from RHR, with RCS temperature – high or RCS water level – low, operators feed water to SGs by motor-driven EFW pump and open safety depressurization valve in order to remove decay heat from RCS.	EOP/ERG
19. In the case of failure of feed or steam line associated with available motor-driven EFW pump during secondary side cooling, operators open the EFW tie-line valves in order to feed water to multiple SGs.	EOP/ERG
20. Revised and updated evaluations of the identified operator actions and human error probabilities can be performed as more specific US-APWR design information becomes available.	19.1.2.4

Key Insights and Assumptions	Dispositions
LPSD other assumptions	
1. Freeze plug may not use for US-APWR because the isolation valves are installed considering maintenance and CCWS has been separated individual trains. Therefore, the freeze plug failure is excluded from the potential initiator.	19.1.2.4
2. For manual operation, one hour is conservatively assumed to be the allowable time until the exposure of reactor core. This allowable time is determined from previous PRA studies and experience which mid-loop operation.	EOP/ERG
3. When the RCS is mid-loop operation, it is assumed that the reflux cooling with the SGs is effective.	19.1.2.4
4. The success criteria of LPSD system are determined based on the success criteria of the Level 1 internal events PRA at power. The success criteria of the LPSD PRA are determined for each POS and each system.	19.1.2.4
5. Various equipments will be possible temporary in the containment during LPSD operation for maintenance. However, there are few possibilities that these materials fall into the sump because the debris interceptor is installed on the sump of US-APWR. Therefore, potential plugging of the suction strainers due to debris is excluded from the PRA modeling.	(RAI19-11) 6.2.2
6. 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. There are no features that automate the response to loss of RHR.	(RAI19-27) 19.1.2.4
• The time when loss of RHR occur were set to be 12 hours after plant trip, which is the time POS 4 (mid-loop operation) is entered after plant trip, since this condition gives the most severe condition for mid-loop operation from a decay heat perspective. The pressurizer spray-line vent line with 3/4 inch diameter is assumed to be open at the initial condition. One hour after loss of RHR function, the operator is assumed to perform the following actions: - Close pressurizer spray line vent, - Start emergency feed water (EFW) pump, and - Open main steam depressurization valve.	(RAI19-45) 19.1.2.4
7. Nitrogen will not be injected in the SG tubes to speed draining in the US-APWR design. The SG tubes will be filled with air during midloop operation.	(RAI19-45) 19.1.2.4
8. Insight that vent must be closed in certain conditions to prevent core damage.	RAI19-45) 19.1.2.4

Key Insights and Assumptions	Dispositions
9. The shutdown response guideline will be developed making sure NUMARC 91-06 is satisfied.	(RAI19-26) EOP/ERG &SAMG
10. The reactivity insertion event due to boron dilution has been judged to be insignificant to risk because of the following factors: - Strict administrative controls are in place to prevent boron dilution - Boron dilution events are highly recoverable - The consequences of re-criticality are minor unless they continue for very long.	(RAI19-63) 15.4.6.2
11. The controls that ensure the indication are available during shutdown.	(RAI19-73) COL 13.5
Operator actions during severe accidents and assumptions	
1. Operators manually initiate active severe accident mitigation systems except for the containment isolation system and combustible gas control system which start up automatically with signals.	SAMG
2. In the loss of support system sequences, operators will attempt to recover CCW/ESW or ac power while suppressing containment overpressure with firewater injection into spray header.	SAMG
Seismic risk assumptions	19.1.5.1.2
1. It is assumed that the seismic event would result in a LOOP, since offsite power equipment is not seismic Category I. (The insulators on the offsite power feed lines can fail in a seismic event such that a LOOP occurs.)	19.1.2.4
2. No credit is taken for non-safety-related systems. They are assumed in the model to have failed or to be non-functional due to the seismic event.	19.1.2.4
3. In the SMA system fault trees, the operator actions in the random failure cutsets from the internal events PRA are assumed as having a failure probability of 1.0. Thus, no credit is taken for the operator actions.	19.1.2.4
4. As a conservative assumption, if one component fails due to the seismic event, the same type components of the system will fail as well.	19.1.2.4
5. Failure of the reactor trip signal is not modeled since the control rod motor generator sets would be de-energized following a LOOP due to a seismic event and succeed in the release of control rods into the core even if the reactor trip function fails. However, if the core assembly or the control rod system fails to insert into the core, these equipment failures are addressed in the event, which leads to core damage.	19.1.2.4

Key Insights and Assumptions	Dispositions
6. It is assumed that piping will fail prior to failure of associated pressure boundary valves. Therefore, valves that are not required to change positions are not included. Also, orifices are not included. Valves that change position, such as motor-operated valves or check valves are assumed to fail the function at the HCLPFs.	19.1.2.4
7. Failure of the RHRS isolation valves is not included in the analysis, because the pipe sections are assumed to fail before the valves fail and these valves are normally closed. Also, the US-APWR design has provided further protection against interfacing system LOCA by upgrading design pressure. Therefore, interfacing system LOCA is not modeled.	19.1.2.4
8. Identified pipe segments in the same system are modeled as failing at the same acceleration level at the same time.	19.1.2.4
9. Failure of buildings that are not seismic Category I (i.e., turbine building, auxiliary building and access building) does not impact SSCs designed to be seismic Category I. Seismic spatial interactions between SSCs design to be seismic Category I and any other buildings will be avoided by proper equipment layout and design. The following seismic Category I buildings and structures are identified as buildings and structures that involve safety-related SSCs to prevent core damage. - Reactor building - Safety power source buildings - Essential service water intake structure - Essential service water pipe tunnel	19.1.2.4
10. Relay chatter does not occur or does not affect safety functions during and after seismic event.	19.1.2.4
Internal fire risk assumption (PRA Report 23.2.1)	19.1.5.2
1. All fire doors serving as fire barriers between redundant safety train fire compartments are normally closed.	(RAI19-51) Fire Protection Program & 19.1.2.4
(PRA Report 23.3.1)	
2. Damage to any control circuit associated with a PRA component is assumed to (i.e., probability 1.0) lead to spurious operation of the component.	19.1.2.4
3. Reactor transient event may occur if no other initiating event can be identified associated with that compartment.	19.1.2.4
(PRA Report 23.7.1)	
4. For transient combustibles, "three Airline trash bags" has been assumed in each fire compartment.	Fire Protection program & 19.1.2.4
5. Fire frequency of Gas Turbine system has been assumed to be the same as that of Diesel Generator system.	

Key Insights and Assumptions	Dispositions
6. The frequencies provided in NUREG/CR-6850 are based on fire event experience in the U.S. NPPs prior to December 2000. It is assumed that those frequencies apply to the US-APWR. It is noted that while it is not an assumption, it is a fact that all cables in US-APWR are qualified cables.	19.1.2.4
(PRA Report 23.8.1)	
7. The effects of fire -generated heat and smoke may damage all PRA equipment and cables in the fire compartment in which the fire takes place. (No credit is given to the fire suppression systems of the fire compartment)	19.1.2.4
8. No Credit is given to potential mitigation functions of the Main Feed Water System.	19.1.2.4
(PRA Report 23.9.1)	
9. Open circuit or short to ground failure mode probability of a hardwired cable is assumed to be 1.0.	19.1.2.4
10. Hot-short probability of a hardwired cable has been assumed to be the followings:	19.1.2.4

	Intra Cable Short	Inter Cable Short
Tray	1.0	1.0
Conduit	1.0	1.0

11. For an optical cable, only open circuit is possible with probability of 1.0.	19.1.2.4
(PRA Report 23.10.3)	
12. Analysis of the Fire Scenario Impacting Multiple Compartments – Only one fire barrier failure at any given time is assumed because the probability of coincident failure of multiple fire barriers is very low. – Given large amount of oil in the Non-safety Gas Turbine room, the possibility of cascading fire passing through two doors affecting the corridor and from the Safety Chiller rooms is considered in the analysis. – Where the fire resistant level of walls or floors is not available, barrier failure probability is estimated to be 1.0.	19.1.2.4
(PRA Report 23.10.4)	
13. Transient combustibles with total heat release capacity of 93,000 Btu (obtained from NUREG/CR-6850, "AppendixG-table-7LBL-Von Volkinburg, Rubbish Bag" Test results) is assumed for Fire ignition source within Containment Vessel.	19.1.2.4
14. The Heat Release Rate of various items as specified in Chapter-11 of NUREG/CR-6850 is used.	19.1.2.4
15. Damage temperature of thermoplastic cables as shown in Appendix-H of NUREG/CR-6850 is used as the target damage temperature.	19.1.2.4
(PRA Report 23.11.1)	
16. Operators are well trained in responding to s fire event.	19.1.2.4
17. In case of MCR evacuation, none of the operators take any actions in the MCR.	19.1.2.4

Key Insights and Assumptions	Dispositions
Internal flood risk assumption (At power operation : PRA Report 22.2.2)	19.1.5.3.1
1. Drain systems are designed to compensate with flood having flow rate below 100 gpm. Flood with flow rate below 100 gpm will not propagate to other areas due to the drain systems.	19.1.2.4
2. R/B is separated in two divisions (i.e. east area and west area). This design is prevents loss of all safety systems though postulated major floods that leak water over the capacities of flood mitigation systems. East side and west side of reactor building (R/B) are physically separated by flood propagation preventive equipments such as water tight doors. Therefore, flood propagation between east side and west side in the reactor building is not considered.	COL13.5(1) COL13.5(7)
3. The first floor of the electrical equipment room of T/B is designed to be water proof. And the first floor of T/B is equipped with relief panels. These measures prevent loss of offsite power due to flood in the T/B.	19.1.2.4
4. Watertight doors are provided for the boundaries between R/B and A/B in the bottom floor and between R/B and T/B in flood area 1F. This measure prevents flood propagation from non-safety building to R/B.	19.1.2.4
5. Flooding of ESW system can to be isolated within 15 minutes~~and flooding of fire protection system can be isolated within 30 minutes.~~	(RAI 19-101) 19.1.2.4
6. Flood propagation from the flood areas which enclosed by water tight doors are considered if the flood water is much and high water level in the area.	19.1.2.4
7. Four trains of ESW system have physical separations and flooding in one train does not propagate to other trains.	19.1.2.4
8. Flooding resulting from pipe and tank ruptures is considered. However, concurrent sprays or flooding from different sources are not considered.	19.1.2.4
9. The loss of functions of electric equipment such as motors, electrical cabinets, solenoid valves and terminal boxes by spraying or flooding is assumed.	19.1.2.4
10. Components such as check valves, pipes and tanks are not vulnerable to effects of flooding.	19.1.2.4
11. The components that are environmentally qualified are considered impregnable to spraying or submerge effects. Also component failure by flooding will not result in the loss of an electrical bus.	19.1.2.4
12. Same models used for internal PRA models are used for internal flooding PRA, such as event trees, fault trees of mitigating systems to prevent core damage.	19.1.2.4
13. It is assumed that the operators in the control room can not mitigate flood outside of the control room during the flood.	19.1.2.4

Key Insights and Assumptions	Dispositions
14. Flooding inside of containment is not included in the internal flooding PRA because inside of containment vessel are designed and evaluated for LOCA events.	19.1.2.4
15. Walls are assumed to remain intact against flooding events since they are designed to withstand anticipated maximum flood loading. Flood propagation from the flood areas which enclosed by water tight doors are considered if the flood water is much and high water level in the area.	19.1.2.4
16. Fire protection doors are considered as flood propagation paths, but the propagation through penetrations is not considered since fire protection seals are provided for walls, floors and ceilings, which compose the fire area boundaries.	19.1.2.4
17. Penetrations within the boundaries between the restricted area and non-restricted area are sealed and doors or dikes are provided for openings. Therefore, flood propagation, except for major flood events is not considered.	19.1.2.4
18. Flood areas are provided in the same way as fire areas because of the following characteristics of the US-APWR. • Fire areas are divided in fire zones which are divided by walls. Boundaries of fire areas consist of fire walls which maintain integrity for three hours. The walls are also effective to mitigate the effects of sprays. • Fire protection seals for penetrations or fire protection doors are effective to mitigate the impact of flood.	19.1.2.4
(during LPSD, PRA Report 22.8.3)	19.1.6.3.3
19. Assumed most risk dominant POS: POS 8-1 (mid-loop operation, 55.5hours).	19.1.2.4
20. Initiating event frequencies for LPSD flood initiating events are assumed as the flood frequencies of each flood mode (spray, flood, and major flood) at power.	19.1.2.4
21. The impacts to LPSD mitigation systems are estimated assuming the worst scenario (boundary conditions of event trees.)	19.1.2.4
22. The administrative controlled flood barriers that separated the reactor building between the east side and the west side are effective. The other water tight doors may be opened during maintenance.	(RAI19-50) COL13.5(1) COL13.5(7)
23. The outage states of mitigation systems are important for LPSD risk. From the insight of flooding risk, one train of mitigation system on each side in R/B should be available. So that assumed the available safety injection pumps trains A and C are available during POS 8-1. B and D pumps are assumed out of services.	19.1.2.4
24. The impacts to LPSD mitigation systems are estimated assuming the worst scenario (boundary conditions of event trees).	19.1.2.4
25. LRFs are assumed to the same as CDFs because the condition large release frequency is assumed to be 1.0.	19.1.2.4

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

3/10/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: **NO.138-1704 REVISION 1**

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

APPLICATION SECTION: **19.1.6.1**

DATE OF RAI ISSUE: **1/9/2009**

QUESTION NO. : 19-208

(Follow-up to Question 19-72) The response to Question 19-72 indicates that the DCD and PRA will be modified to revise the shutdown success criteria for safety injection (SI) and charging to match the low temperature overpressure (LTOP) technical specifications (TS) requirements in TS 3.4.12. The staff needs additional information on this ongoing effort. Specifically:

a. The staff notes that the bases for TS 3.4.12 state that "[i]f conditions require the use of more than two SI pumps or one charging pump for makeup in the event of loss of inventory, then pumps can be made available through manual actions." This statement calls into question how many pumps can be considered available to mitigate accidents during shutdown. However, no additional discussion of these manual actions could be found in the DCD. Clearly state and justify the success criteria for SI and charging injection during shutdown in all POS, including when the pressurizer is solid and LTOP is a more serious concern. Discuss whether the operators would be expected to start SI or charging pumps, contrary to the stated TS requirement, following a loss of inventory at shutdown. Would additional pumps also be started following other initiating events for which injection is credited in the PRA? What guidance is provided to the operators on when to start additional pumps, balancing LTOP concerns with mitigation of accidents?

b. Describe the modifications that were made to the PRA in response to Question 19-72, including a list of all fault trees that were changed in both the initiating event assessment (FLML, etc.) and systems analysis.

c. Provide marked-up pages of an interim revision so the staff can review the proposed changes.

Answer:

<u>Answer to the question a.</u>

The number of operable SI pumps and charging pumps are 2 and 1 respectively in the PRA. Actuation of the additional pumps is not credited in the PRA.

As responses to question 19-248, it is considered that the success criteria for charging pumps and SI pumps require only one pump for all POSs are adequate.

Although it is not credited in PRA, operators are expected to start the additional pumps when SI or charging pumps have failed following a loss of inventory at shutdown. This operator action will be incorporated in the maintenance procedure. Development of the maintenance procedure is one of the COL items identified in Chapter 13, and will include shutdown responses as part of the procedure to incorporate the discussions given in NUMARC 91-06.
In addition, when a pump (or pumps), which has been counted to be operable to meet the LCO, has failed and identified to be inoperable, initiation of the additional pump will not violate the TS requirement.

Answer to the question b.

Mitigating system fault trees (i.e. top event "MC", "CV" and "SC") related to the charging injection were changed to the condition that the charging pump B is unavailable and re-quantified. The sub gate in the fault trees of the charging pump B is changed to "True" so that the charging pump B is unavailable. There are no other changes in the structure of the fault trees. The modifications are listed in Table 19-208-1.

Table 19-208-1 The list of modification for the fault trees related to charging injection

Top Event Name	FT ID	Page of the PRA Technical Report	Sub Gate ID set to "True"
MC	CHI21	20A.3.B-19	CHI-04B
CV	CHI	20A.3.B-1	CHI-04B
SC	ACW	20A.6.B-21	@ACW-CHI-2
	ACW-LO	20A.6.B-46	@ACW-CHILO-2
	ACW-LON	20A.6.B-69	@ACW-CHILON-2

The number of operable charging pumps in POS8-2 and POS8-3 for FLML event is changed from two to one.

Answer to the question c.

Revision to reflect response to this RAI will be made with other modification of PRA model. Additionally, the modifications that were made to the PRA in response to Question 19-72 will be provided in revised PRA.

Impact on DCD
The DCD will be revised reflecting this response to this RAI as this additional information is requested in order to consist with TS requirement.

Impact on COLA
There is no impact on COLA.

Impact on PRA
There is an impact on PRA from this RAI, but it is very small as shown in the response to the Question 19-8.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

3/10/2009

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Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: **NO.138-1704 REVISION 1**

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

APPLICATION SECTION: **19.1.6.1**

DATE OF RAI ISSUE: **1/9/2009**

QUESTION NO. : 19-209

(Follow-up to Question 19-25) The response to Question 19-25 uses the frequency of manual reactor trips obtained from NUREG/CR-5750, combined with conditional probabilities that these shutdowns need drained maintenance or fuel removal, to obtain the frequency of Type A (non-drained), B (drained), and C (refueling) outages. However, page 3 of NUREG/CR-5750 states that, to be included in the study, events needed to include an unplanned reactor trip (not a scheduled reactor trip on the daily operations schedule). NUREG-1350, Volume 20, states that the average capacity factor for U.S. nuclear power plants was 92 percent in 2007. In comparison, the POS durations and frequencies from DCD Table 19.1-79 and the response to Question 19-25, respectively, indicate that the US-APWR would be shut down only 3.6 percent of the year. Therefore, it appears that the US-APWR shutdown PRA does not account for maintenance outages (planned or unplanned, although unplanned reactor trips are addressed by the sensitivity study provided in the response to Question 19-25). The staff needs additional information on the shutdown schedule to resolve this issue. Specifically:

a. Discuss how the POS durations listed in Table 19.1-79 were developed.

b. Discuss, with support from operating experience as needed, how often planned maintenance outages (not unplanned reactor trips) would be expected to occur for the US-APWR. Discuss whether these outages are expected to be drained or non-drained.

c. Provide the projected capacity factor for the US-APWR and a justification for this value.

d. Discuss how the POS durations and frequencies currently assumed in the PRA (not those provided in response to Question 19-25) compare to the projected capacity factor.

e. Revise the PRA as needed to ensure that shutdown risk is not underestimated because of low POS frequencies.

ANSWER:

(1) It seems that the following indication was interpreted as the unplanned shutdown not being included in the value of NUREG/CR-5750.

> "However, page 3 of NUREG/CR-5750 states that, to be included in the study, events needed to include an unplanned reactor trip (not a scheduled reactor trip on the daily operations schedule)."

However, it is a statement about the definition of including an unplanned shutdown as an event counted as a generating track record by NUREG/CR-5750. Therefore, the sensitivity analysis of 19-25 and 19-71 was estimated including unplanned shutdown appropriately. Incidentally, the planned shutdown which is not included in NUREG/CR-5750 is also taken into consideration as 0.5 events of Type C.

(2)It's right that assumption of the shutdown period of 3.6% per year is too short. However, it is because the shortest process was assumed during fuel extraction, and generally, the shutdown period depends on the duration of fuel extraction period greatly. Since the risk during fuel extraction is evaluated as negligible in LPSD PRA, there is no significant influence on the evaluation result by this assumption. Incidentally, the availability goal of US-APWR is 95% with the assumption that there is also the shutdown period of such long fuel extraction period.

(3)In addition, since safety systems of 4 trains of US-APWR is the design with easy maintaining, without shutdown, it is hard to consider the large value as shutdown frequency by the unplanned maintenance. However, based on the operating experience in the U.S., large uncertainty is also assumed by the frequency of an unplanned shutdown (Type-A) of a short period without drain. Sensitivity of 0.5/yr about POS3 and POS11 of Type-A increases to 0.84/yr was evaluated in 19-25 and 19-71. As excessive case, the CDF is 3.7E-07/yr when this frequency is set as 2/yr shown in Table 19-209-1. This is the increase in about 85% of a base case, and the influence of this uncertainty is small.

(4)The answers about specifically issues are shown below.

a. Discuss how the POS durations listed in Table 19.1-79 was developed.

Shorter periodic test was set up and POS durations which can carry out equipment maintenance in periodic test were developed in consideration of experience in Japan.

b. Discuss, with support from operating experience as needed, how often planned maintenance outages (not unplanned reactor trips) would be expected to occur for the US-APWR. Discuss whether these outages are expected to be drained or non-drained.

See (3).
Since safety systems of 4 trains of US-APWR is the design with easy maintaining, without shutdown, it is hard to consider the large value as shutdown frequency by the unplanned maintenance.
However, large uncertainty is also assumed by the frequency of an unplanned shutdown (Type-A) of a short period without drain.

c. Provide the projected capacity factor for the US-APWR and a justification for this value.

See (2).
The availability goal of US-APWR is 95% with the assumption that there is also the shutdown period of long fuel extraction period.

d. Discuss how the POS durations and frequencies currently assumed in the PRA (not those provided in response to Question 19-25) compare to the projected capacity factor.

See (2).
The availability goal of US-APWR is 95%. This differs from that assumption of the shutdown period of 3.6% per year. The key factor of this difference is because the shortest process was assumed during fuel extraction, and generally, the length of shutdown period depends on the duration of a fuel extraction

period greatly. Since the risk under fuel extraction is evaluated as negligible in LPSD PRA, there is no significant influence on the evaluation result by this assumption.

e. Revise the PRA as needed to ensure that shutdown risk is not underestimated because of low POS frequencies.

See (3).
The influence of uncertainty of the frequency of an unplanned shutdown (Type-A) of a short period without drain is small. Therefore LPSD PRA will not be revised.

Table 19-209-1 Yearly frequency of each POS for Sensitivity Analysis of LPSD PRA

POS	Type			Yearly frequency
	A	B	C	
POS 3	×	×	×	2.00
POS 4-1	N/A	×	×	0.55
POS 4-2	N/A	×	×	0.55
POS 4-3	N/A	N/A	×	0.50
POS 8-1	N/A	N/A	×	0.50
POS 8-2	N/A	N/A	×	0.50
POS 8-3	N/A	N/A	×	0.50
POS 9	N/A	×	×	0.55
POS 11	×	×	×	2.00

Impact on DCD
There is no impact on DCD.

Impact on COLA
There is no impact on COLA.

Impact on PRA
There is no impact on PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

3/10/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: **NO.138-1704 REVISION 1**

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

APPLICATION SECTION: 19.1.6.1

DATE OF RAI ISSUE: 1/9/2009

QUESTION NO. : 19-214

Table 19.1-80 of the DCD indicates that no outages are planned for the electrical buses, transformers, or essential service water (ESW) system during shutdown. Discuss when maintenance on these structures, systems, and components (SSC) is expected to be performed. Discuss how this maintenance is modeled in the at-power or shutdown PRA.

ANSWER:

The maintenance of essential service water (ESW) system will be performed at the same time with component cooling water (CCW) system and the shutdown PRA model is based on this condition. Table 19.1-80 of the DCD will be amended to show the outage of ESW system. On the other hand, maintenance of electrical buses and transformers is assumed to be performed when the fuel is removed from the core (POS6) and this POS is screened out from the shutdown PRA.

Table19.1-80 of the DCD revision 1 will be revised to Table 19-214-1 in the DCD revision 2.

Table 19-214-1 Planned Maintenance Schedule for LPSD PRA

System \ POS	(1) Low power operation	(2) Hot standby	(3) Hot and cold shutdown (RCS is filled with coolant)	(4)-1 Cold shutdown (Mid-loop operation) (RCS closed)	(4)-2 Cold shutdown (Mid-loop operation) (RCS opened)	(4)-3 Cold shutdown (Mid-loop operation) (SG isolated)	(5) Refueling cavity is filled with water	(6) No fuels in the core	(7) Refueling cavity is filled with water	(8)-1 Cold shutdown (Mid-loop operation) (SG isolated)	(8)-2 Cold shutdown (Mid-loop operation) (RCS opened)	(8)-3 Cold shutdown (Mid-loop operation) (RCS closed)	(9) Cold shutdown (RCS is filled with coolant)	(10) RCS leakage test (RHRS isolated from RCS)	(11) Cold and hot shutdown (RCS is filled with coolant)	(12) Hot standby	(13) Low power operation
A safety 6.9kV bus	N/A	N/A	△	△	△	△	N/A	N/A	N/A	△	△	△	△	N/A	△	N/A	N/A
B safety 6.9kV bus	N/A	N/A	△	△	△	△	N/A	N/A	N/A	△	△	△	△	N/A	△	N/A	N/A
C safety 6.9kV bus	N/A	N/A	△	△	△	△	N/A	N/A	N/A	△	△	△	△	N/A	△	N/A	N/A
D safety 6.9kV bus	N/A	N/A	△	△	△	△	N/A	N/A	N/A	△	△	△	△	N/A	△	N/A	N/A
A safety 480V bus	N/A	N/A	△	△	△	△	N/A	N/A	N/A	△	△	△	△	N/A	△	N/A	N/A
B safety 480V bus	N/A	N/A	△	△	△	△	N/A	N/A	N/A	△	△	△	△	N/A	△	N/A	N/A
C safety 480V bus	N/A	N/A	△	△	△	△	N/A	N/A	N/A	△	△	△	△	N/A	△	N/A	N/A
D safety 480V bus	N/A	N/A	△	△	△	△	N/A	N/A	N/A	△	△	△	△	N/A	△	N/A	N/A
A safety 480V motor control center	N/A	N/A	△	△	△	△	N/A	N/A	N/A	△	△	△	△	N/A	△	N/A	N/A
B safety 480V motor control center	N/A	N/A	△	△	△	△	N/A	N/A	N/A	△	△	△	△	N/A	△	N/A	N/A
C safety 480V motor control center	N/A	N/A	△	△	△	△	N/A	N/A	N/A	△	△	△	△	N/A	△	N/A	N/A
D safety 480V motor control center	N/A	N/A	△	△	△	△	N/A	N/A	N/A	△	△	△	△	N/A	△	N/A	N/A
Offsite power main transformer	N/A	N/A	△	△	△	△	N/A	N/A	N/A	△	△	△	△	N/A	△	N/A	N/A
Offsite power reserve transformer	N/A	N/A	△	△	△	△	N/A	N/A	N/A	△	△	△	△	N/A	△	N/A	N/A
A emergency generator	N/A	N/A	△	×	×	×	N/A	N/A	N/A	△	△	△	△	N/A	△	N/A	N/A
B emergency generator	N/A	N/A	△	△	△	△	N/A	N/A	N/A	△	△	△	△	N/A	△	N/A	N/A
C emergency generator	N/A	N/A	△	△	△	△	N/A	N/A	N/A	△	△	△	△	N/A	△	N/A	N/A
D emergency generator	N/A	N/A	△	△	△	△	N/A	N/A	N/A	×	×	×	×	N/A	△	N/A	N/A
A essential service water pump	N/A	N/A	×	×	×	×	N/A	N/A	N/A	○	○	○	○	N/A	○	N/A	N/A
B essential service water pump	N/A	N/A	○	○	○	○	N/A	N/A	N/A	○	○	○	○	N/A	○	N/A	N/A
C essential service water pump	N/A	N/A	○	○	○	○	N/A	N/A	N/A	○	△	△	△	N/A	○	N/A	N/A
D essential service water pump	N/A	N/A	○	○	○	○	N/A	N/A	N/A	×	△	△	△	N/A	○	N/A	N/A
A essential service water header	N/A	N/A	×	×	×	×	N/A	N/A	N/A	○	○	○	○	N/A	○	N/A	N/A
B essential service water header	N/A	N/A	○	○	○	○	N/A	N/A	N/A	○	○	○	○	N/A	○	N/A	N/A
C essential service water header	N/A	N/A	○	○	○	○	N/A	N/A	N/A	○	△	△	△	N/A	○	N/A	N/A
D essential service water header	N/A	N/A	○	○	○	○	N/A	N/A	N/A	×	△	△	△	N/A	○	N/A	N/A
A component cooling water pump	N/A	N/A	×	×	×	×	N/A	N/A	N/A	○	○	○	○	N/A	○	N/A	N/A
B component cooling water pump	N/A	N/A	○	○	○	○	N/A	N/A	N/A	○	○	○	○	N/A	○	N/A	N/A
C component cooling water pump	N/A	N/A	○	○	○	○	N/A	N/A	N/A	○	△	△	△	N/A	○	N/A	N/A
D component cooling water pump	N/A	N/A	○	○	○	○	N/A	N/A	N/A	×	△	△	△	N/A	○	N/A	N/A
A component cooling water header	N/A	N/A	×	×	×	×	N/A	N/A	N/A	○	○	○	○	N/A	○	N/A	N/A
B component cooling water header	N/A	N/A	○	○	○	○	N/A	N/A	N/A	○	○	○	○	N/A	○	N/A	N/A
C component cooling water header	N/A	N/A	○	○	○	○	N/A	N/A	N/A	○	△	△	△	N/A	○	N/A	N/A
D component cooling water header	N/A	N/A	○	○	○	○	N/A	N/A	N/A	×	△	△	△	N/A	○	N/A	N/A
A CS/RHR pump	N/A	N/A	×	×	×	×	N/A	N/A	N/A	○	○	○	○	N/A	○	N/A	N/A
B CS/RHR pump	N/A	N/A	○	△	△	△	N/A	N/A	N/A	○	○	○	○	N/A	○	N/A	N/A
C CS/RHR pump	N/A	N/A	○	○	○	○	N/A	N/A	N/A	△	△	△	△	N/A	○	N/A	N/A
D CS/RHR pump	N/A	N/A	○	○	○	○	N/A	N/A	N/A	×	△	△	△	N/A	○	N/A	N/A
A Safety injection pump	N/A	N/A	×	×	×	×	N/A	N/A	N/A	△	△	△	△	N/A	△	N/A	N/A
B Safety injection pump	N/A	N/A	×	×	×	×	N/A	N/A	N/A	△	△	△	△	N/A	△	N/A	N/A
C Safety injection pump	N/A	N/A	△	△	△	△	N/A	N/A	N/A	×	×	×	△	N/A	△	N/A	N/A
D Safety injection pump	N/A	N/A	△	△	△	△	N/A	N/A	N/A	×	×	×	△	N/A	△	N/A	N/A
A Charging pump	N/A	N/A	×	×	×	×	N/A	N/A	N/A	△	○	○	○	N/A	○	N/A	N/A
B Charging pump	N/A	N/A	○	○	○	○	N/A	N/A	N/A	△	△	△	△	N/A	△	N/A	N/A
B Motor-driven emergency feed water pump	N/A	N/A	×	×	×	×	N/A	N/A	N/A	×	×	△	△	N/A	△	N/A	N/A
C Motor-driven emergency feed water pump	N/A	N/A	△	△	△	△	N/A	N/A	N/A	△	△	×	×	N/A	×	N/A	N/A
A main steam relief valve	N/A	N/A	×	×	×	×	N/A	N/A	N/A	△	△	△	△	N/A	△	N/A	N/A
B main steam relief valve	N/A	N/A	×	×	×	×	N/A	N/A	N/A	△	△	△	△	N/A	△	N/A	N/A
C main steam relief valve	N/A	N/A	△	△	×	×	N/A	N/A	N/A	×	×	△	△	N/A	△	N/A	N/A
D main steam relief valve	N/A	N/A	△	△	×	×	N/A	N/A	N/A	×	×	△	△	N/A	△	N/A	N/A
RWSP	N/A	N/A	△	△	△	△	N/A	N/A	N/A	△	△	△	△	N/A	△	N/A	N/A
RWSAT	N/A	N/A	△	△	△	△	N/A	N/A	N/A	△	△	△	△	N/A	△	N/A	N/A

○ : Run
△ : Standby
× : Outage
N/A : Not applicable

Impact on DCD
Revise the Table 19.1-80 of the DCD revision 1 by the next revision.

Impact on COLA
There is no impact on COLA.

Impact on PRA
There is no impact on PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

3/10/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: NO.138-1704 REVISION 1

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

APPLICATION SECTION: 19.1.6.1

DATE OF RAI ISSUE: 1/9/2009

QUESTION NO. : 19-218

Discuss how the shutdown PRA addresses the dependence between human errors that cause or contribute to initiating events and subsequent operator actions needed for mitigating system function. Address in the response both the detailed models for POS 8-1 and their extension to other POS, in which the number of human errors is counted to determine the correction factor.

Answer:

The human errors that cause or contribute to initiating events are the following four items.

(1) Operation error that occurs during RCS coolant drain operation to prepare for the mid-loop operation. This error will cause over-drain initiating event.
(2) Accidentally opening of the RCS isolation valve. This error will cause LOCA initiating event.
(3) Failure of manual operation to start the standby RHR pump when the running RHR pump fails to run. This error will cause the loss of RHR initiating event.
(4) Failure of manual operation to start the standby CCW/ESW pump when the running CCW/ESW pump fails to run. This error will cause the loss of CCWS/ESWS initiating event.

Dependency idea for above the human errors (1) and (2)

Both RCS coolant drain operation and RCS isolation valve open operation themselves are routine tasks for normal shutdown state.

There are a lot of deference between the condition under the routine task and the condition under the accident in the routine task caused by human errors (1), (2). For example, the procedure for routine task is deferent from the procedure for mitigating action after initiating event. And also plant condition and stress level of operators are deferent between pre- initiating event and post-initiating event.

So MHI judged that the dependence between human errors regarding to (1) and (2) that cause or contribute to initiating events and subsequent operator actions needed for mitigating system function is vanishingly low.

Dependency idea for above the human errors (3) and (4)

Both manual operation to start the standby RHR pump and manual operation to start the standby CCW/ESW pump are backup operation after the failure of normal operated pump.

There are not a lot of deference between the condition under the backup operation task and the condition under the accident cased by human errors (3), (4) during the operation for the routine task.
They can be categorized in the backup operation after the failure of normal operation.

As NRC staff pointed out, there is some dependency between human errors regarding to (3) and (4) that cause or contribute to initiating events and subsequent operator actions needed for mitigating system function. However, the contribution of the human errors (3) and (4) to the initiating event frequency is less than 1%, and also the contribution of the human errors (3) and (4) to the total CDF is negligible small.

So MHI judges it is needless to model the dependence between human errors regarding to (3) and (4) that cause or contribute to initiating events and subsequent operator actions needed for mitigating system function.

Impact on DCD
There is no impact on DCD.

Impact on COLA
There is no impact on COLA.

Impact on PRA
There is no impact on PRA.

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US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: **NO.138-1704 REVISION 1**

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

APPLICATION SECTION: **19.1.6.1**

DATE OF RAI ISSUE: **1/9/2009**

QUESTION NO. : 19-222

Page 9-70 of the PRA indicates that "if the crew uses different control panel (window) to perform next action, then [the location of] these actions are defined as 'different.'" For the dependent operator actions modeled in the shutdown PRA, provide additional justification for the assumption that different control panels are used and that this difference is equivalent to a different location. The discussion of the control room design in MUAP-07007 appears to describe a single operational visual display unit (VDU) for each operator.

Answer:

It is planed to use a single operational visual display unit (VDU) for the operation of safety related mitigation system. Operator uses the specific touch panel screen to operate the specific mitigation system. Changing screen is necessary for operating a different mitigation system.

"Location (distance)" is one of dependency factor in human reliability evaluation. NUREG/CR-6883 (SPAR-H) and NUREG/CR-1278 (THERP) describes that changing location (distance) degrades dependence level. MHI judged that the reason for above is that the surrounding condition of operator changes due to change location. For example, changing location may promote to calm down operator or may bring new effective information etc.

MHI judged that the changing screen can also promote to calm down operator or may bring new effective information etc.

Impact on DCD
There is no impact on DCD.

Impact on COLA
There is no impact on COLA.

Impact on PRA

There is no impact on PRA.

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Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: **NO.138-1704 REVISION 1**

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

APPLICATION SECTION: **19.1.6.1**

DATE OF RAI ISSUE: **1/9/2009**

QUESTION NO. : 19-230

Justify why a lower failure rate of 2.8E-6 per hour (/hr) was assumed for running component cooling water (CCW) pumps (e.g., CWSPCYRCCWPA-CG3) compared to running motor-driven pumps, for which a failure rate of 5.0E-6/hr is presented in DCD Table 19.1-14. The text above Table A.2.27-8 of NUREG/CR-6928, which is cited in DCD Table 19.1-14, states that "[b]ecause some system and failure mode data sets are limited (few or only one failure and/or limited demands or hours), the results should be viewed with caution." Discuss the impact of using two different pump failure rates on the shutdown PRA results and insights.

ANSWER:

The failure rate of motor driven pumps reported in NUREG/CR-6928 includes 213 component cooling water pumps as listed in Table A.2.27-2 in the report. This number of components is relatively large and is approximately 30% of the number of components used to estimate the failure rate of Running/Alternating motor driven pumps. Moreover, each component cooling water pumps experience a long running time since they are alternately running. For this reason, failure rate for component cooling water pumps reported in Table A.2.27-8 of NUREG/CR-6928 is considered to be reliable and applicable to the PRA. The use of a component specific failure rates for components that have large impact on the PRA results will enhance the accuracy of the PRA insights and hence preferable.

Impact on DCD
There is no impact on DCD.

Impact on COLA
There is no impact on COLA.

Impact on PRA
There is no impact on PRA.

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Docket No.52-021

RAI NO.: **NO.138-1704 REVISION 1**

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

APPLICATION SECTION: **19.1.6.1**

DATE OF RAI ISSUE: **1/9/2009**

QUESTION NO. : 19-234

The CCF probabilities of the RHR pumps (both the group of two running pumps and the group of all three available pumps) and RHR heat exchangers listed in Section 20B.2.1 of the PRA are lower than the values expected based on the information in Tables 7.1-1 and 8.5-3 of the PRA. Discuss how the RHR CCF probabilities used in the shutdown
LORH model were developed.

Answer:

The CCF probability of the RHR heat exchangers is calculated from following formula, using the information in Tables 7.1-1 and 8.5-3 of the PRA report.

- Triple CCF probability (2.2E-7) =
  Failure probability of shell & tube heat exchanger (6.0E-7/hr)
  $\times \beta(0.39)$
  $\times \gamma(0.92)$
  $\times 1(\mathrm{hr})$
- Fourfold CCF probability (2.0E-7) =
  Failure probability of shell & tube heat exchanger (6.0E-7/hr)
  $\times \beta(0.39)$
  $\times \gamma(0.92)$
  $\times \delta(0.97)$
  $\times 1(\mathrm{hr})$

(β : beta factor γ : gamma factor δ : delta factor)

Above CCF probabilities are same as listed in Section 20B.2.1 of the PRA.

Beta for the CCF probability of the RHR pump is set to beta prime (β') =0.001 described in page 8-16 of the PRA instead of that in Tables 8.5-3 of the PRA. The reason is same to the case of CCW and ESW pumps CCF probability. These are normally running components. MHI applied to β' for the RHR pumps.

In addition, even if standard MGL parameter based on NUREG/CR-5497 is applied, the total CDF will increase only about 1%.

Impact on DCD
There is no impact on DCD.

Impact on COLA
There is no impact on COLA.

Impact on PRA
There is no impact on PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

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Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: **NO.138-1704 REVISION 1**

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

APPLICATION SECTION: **19.1.6.1**

DATE OF RAI ISSUE: **1/9/2009**

QUESTION NO. : 19-236

(Follow-up to Question 19-45) Chapter 20 of the PRA indicates that the SGs are credited for heat removal in POS 3, 4-1, 8-3, 9, and 11. According to DCD Tables 19.1-76 to 19.1-78, RCS level is at the nozzle center in POS 4 and 8 and the RCS is full in POS 3, 9, and 11. However, the analysis of reflux cooling performed in response to Question 19-45 addresses RCS levels only at the top, center, and bottom of the main coolant piping. NUREG-1410, cited in the response to Question 19-6, showed different results when the RCS was initially full (see Section 8.3.8 and Figures 8.6 to 8.8). Therefore, provide a description and results of a design-specific analysis demonstrating the effectiveness of reflux cooling in the US-APWR when the RCS is initially full, as in POS 3, 9, and 11. Identify, as in the response to Question 19-45, whether any operator actions (e.g., closing a vent) are critical to avoid core damage. Discuss the mechanism for communicating these critical actions to the developers of procedures and training.

ANSWER:

If RCS is full and natural circulation in the RCS can be achieved, decay heat is removed via steam generator (SG) with the depressurization of secondary system and feed water to SG. In order to confirm successful decay heat removal under such condition, the case with secondary system depressurization and feed water to one SG during POS 3 is analyzed utilizing MAAP 4.0.6. The analysis condition is shown in Table 19-236-1, and the results are shown in Figure 19-236-1, Figure 19-236-2 and Figure 19-236-3. The results show that the temperature of primary system falls gradually, and decay heat can be removed by only one SG.

Operator actions of decay heat removal by SGs will be listed in key assumption in section 19.1.7 of the DCD and will be incorporated in the emergency response guideline.

Table 19-236-1 Analysis Condition

Initiating Event / Accident Sequence	Loss of RHR
Break Location	-
Break Size	-
High Head Injection	0 / 4
Alternative Core Injection	0 / 4
Accumulators	0 / 4
Containment Spray	0 / 4
Heat Exchanger	Disable
Emergency Feed Water	1 / 4 to 1SG
RCS Depressurization	Disable
SG Depressurization	1 / 4
Firewater Injection to Cavity	Disable
Firewater injection to Spray Header	Disable
Alternative Containment Cooling	Disable
Containment Opening Area	$0.0m^2$
Reactor Scram	0.0sec
Condition Resulting in Reactor Scram	-
High Head Injection Started	-
Alternative Core Injection Started	-
Accumulators Started	-
Accumulators Stopped	-
Containment Spray Started	-
RCS Depressurization Started	-
Firewater Injection to Cavity Started	-
Firewater injection to Spray Header Started	-
Alternative Containment Cooling Started	-
SG Relief Valve Opened	1.00hr
Emargency Feed Water Started	1.00hr

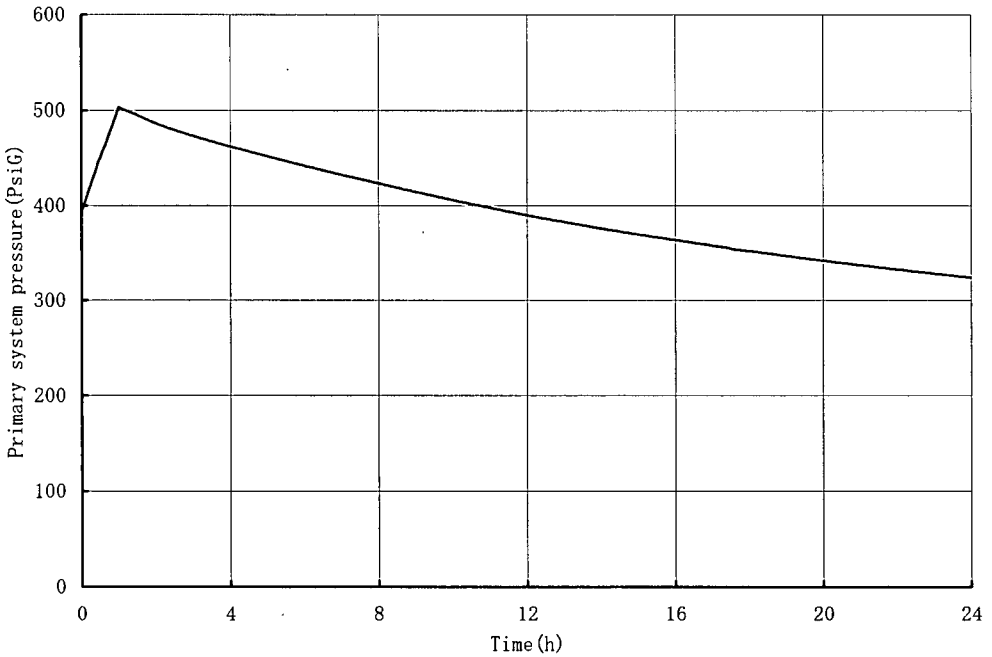

Figure 19-236-1 Primary System Pressure

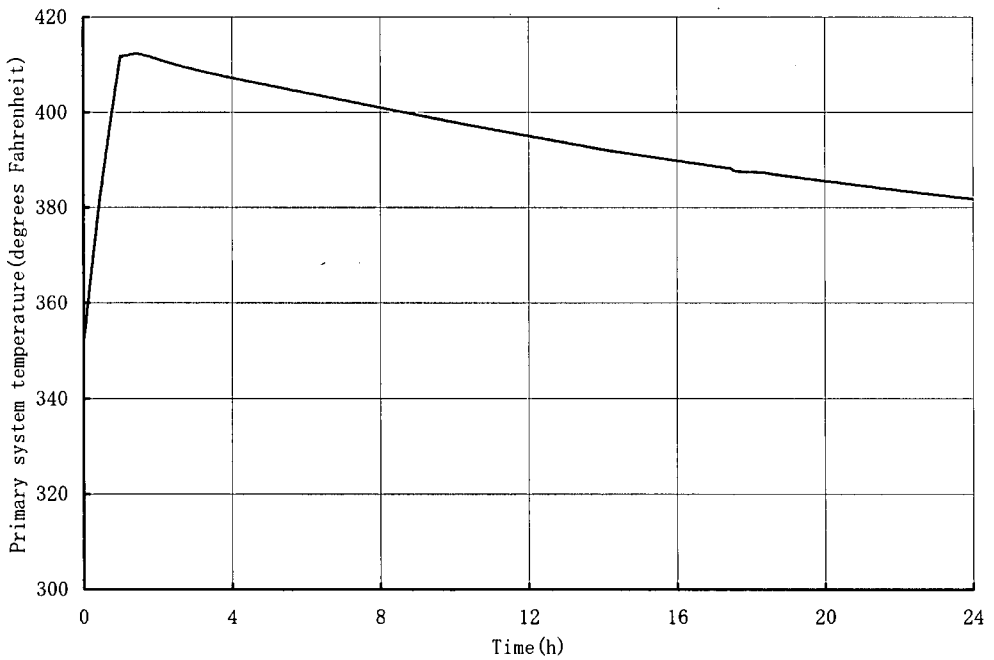

Figure 19-236-2 Primary System Temperature

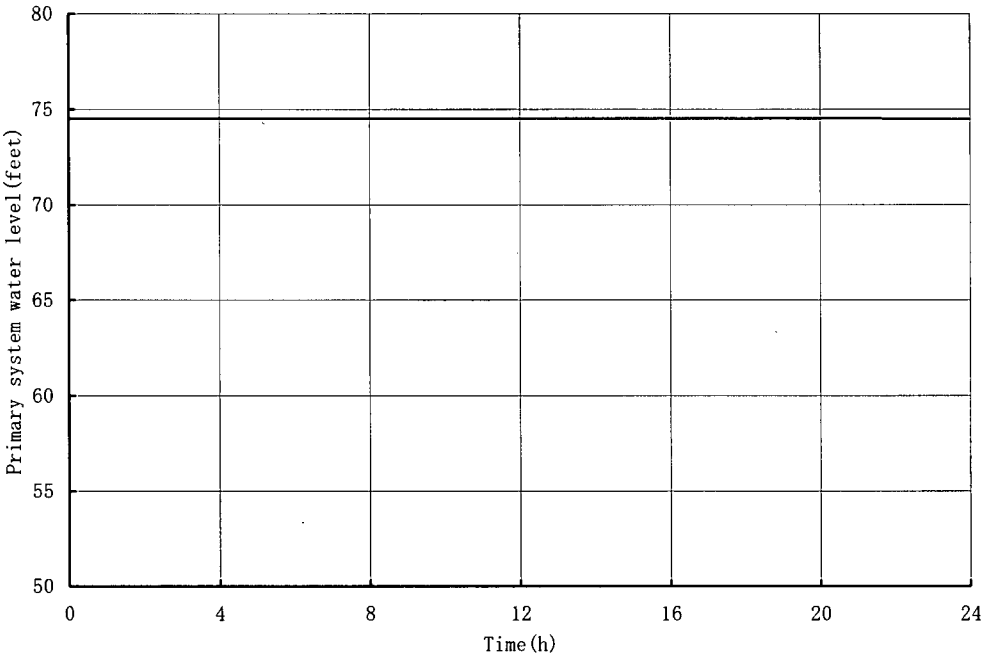

Figure 19-236-3 Primary System Water Levels

Impact on DCD

DCD will be further revised in accordance with the description of this RAI.

Impact on COLA

There is no impact on COLA.

Impact on PRA

There is no impact on PRA.

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Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: **NO.138-1704 REVISION 1**

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

APPLICATION SECTION: **19.1.6.1**

DATE OF RAI ISSUE: **1/9/2009**

QUESTION NO. : 19-247

Page 20A.3-2 of the PRA indicates that the operator action to provide makeup to the RWSAT depends on the operator detecting a "small leakage" by volume control tank (VCT) level decrease, charging flow rate increase, or makeup water increase. However, RWSAT makeup is expected when charging injection to the RCS following a LOCA or OVDR depletes the inventory of the RWSAT. Clarify how these "small leakage" cues relate to this scenario. Discuss whether the cue for starting the refueling water recirculation (RWR) pumps is low VCT level or low RWSAT level. If the cue were low VCT level, then operator might be directed to start the RWR pumps any time the charging system is used for injection (i.e., both MC and CV top events).

Answer:

RWSAT water makeup operation is assumed as below.

When a loss of RCS inventory event involving "small leakage" occurs, RCS water level is expected to be recovered by charging injection pump. The suction of this pump is VCT. When the level of VCT becomes low signal level, the suction of this pump automatically changes to RWSAT. At this timing, with the low VCT level signal, the operator begins to prepare RWSAT water makeup. The operator will open manual valve 026 and 028 to establish the flow path from RWSP to RWSAT. If the operator detects RWSAT low level signal, he remotely from the main control room begins to make up to RWSAT with the refueling water recirculation pumps.

Regarding to PRA model, CV top event requires long term injection which includes the failure of RWSAT water makeup. On the other hands, MC top event does not require long term injection which includes the failure of RWSAT water makeup. Because it is enough for restarting RHR pumps to make up from VCT and RWSAT.

<u>Impact on DCD</u>
There is no impact on DCD.

Impact on COLA
There is no impact on COLA.

Impact on PRA
There is no impact on PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

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Mitsubishi Heavy Industries

Docket No.52-021

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

APPLICATION SECTION: **19.1.6.1**

DATE OF RAI ISSUE: **1/9/2009**

QUESTION NO. : 19-249

Page 20A.3-8 of the PRA indicates that the VCT stop valves close and the RWSAT suction valves open when charging injection initiates. Are these valve manipulations automatic or manual? If they are automatic, discuss why related I&C failures are not modeled. If they are manual, discuss why failures to manipulate these valves are not included in the CHIOO02CV21 operator action.

Answer:

The VCT stop valves automatically close and also the RWSAT suction valves automatically open.

The reason related I&C failures are not modeled is that the impact of related I&C failures is very small against related human errors.

The order of failure probabilities of mitigating system (i.e. top event "MC", "CV" and "SC") related with charging injection is from 10^{-3} to 10^{-2}, and the major contributor to these probabilities is human error. The VCT stop valves and the RWSAT suction valves are automatically operated by VCT level decrease. The I&C related with VCT level is multiplexed. Therefore, the major factor in the loss of I&C function is multiple hardware failures. So the failure probability is expected to be much lower than the human error probability.

Impact on DCD
There is no impact on DCD.

Impact on COLA
There is no impact on COLA.

Impact on PRA
There is no impact on PRA.