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TOKYO, JAPAN

August 2, 2011

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Attention: Mr. Jeffrey A. Ciocco

Docket No. 52-021
MHI Ref: UAP-HF-11252

Subject: Revised Responses to US-APWR DCD RAI No.744-5668 Revision 2 (SRP 19.0)

References: 1) Letter MHI Ref:UAP-HF-11155 from Y. Ogata to U.S. NRC "MHI's Responses to US-APWR DCD RAI No.744-5668 Revision 2 (SRP 19.0)" dated May 27, 2011

With this letter, Mitsubishi Heavy Industries, Ltd. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") a document entitled "Revised Responses to Request for Additional Information No. 744-5668 Revision 2".

Enclosed is the amendment of the responses to RAI contained within Reference 1.

As indicated in the enclosed materials, this submittal contains information that MHI considers proprietary, and therefore should be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4) as trade secrets and commercial or financial information which is privileged or confidential. A non-proprietary version of the document is also being submitted with the information identified as proprietary redacted and replaced by the designation "[ ]".

This letter includes a copy of the proprietary version (Enclosure 2), a copy of the non-proprietary version (Enclosure 3), and the Affidavit of Atsushi Kumaki (Enclosure 1) which identifies the reasons MHI respectfully requests that all materials designated as "Proprietary" in Enclosure 2 be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4).

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

Sincerely,

Y. Ogata

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

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NRO

Enclosure:

1. Affidavit of Yoshiki Ogata
2. Revised Responses to Request for Additional Information No. 744-5668 Revision 2 (Proprietary Version)
3. Revised Responses to Request for Additional Information No. 744-5668 Revision 2

(Non-Proprietary Version)

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

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Enclosure 1

Docket No. 52-021
MHI Ref: UAP-HF-11252

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) 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 "Revised Response to Request for Additional Information No. 744-5668, Revision 2", and have determined that portions of the document contain proprietary information that should be withheld from public disclosure. Those pages contain proprietary information are identified with the label "Proprietary" on the top of the page, and the proprietary information has been bracketed with an open and closed bracket as shown here "[ ]". The first page of the document indicates that all information identified as "Proprietary" should be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4).

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

4. The basis for holding the referenced information confidential is that it describes the unique technique of the hydrogen burning analysis results related to the US-APWR severe accident analytical models developed by MHI.

5. The referenced information is being furnished to the Nuclear Regulatory Commission ("NRC") in confidence and solely for the purpose of information to the NRC staff.

6. The referenced information is not available in public sources and could not be gathered readily from other publicly available information. Other than through the provisions in paragraph 3 above, MHI knows of no way the information could be lawfully acquired by organizations or individuals outside of MHI.

7. Public disclosure of the referenced information would assist competitors of MHI in their design of new nuclear power plants without incurring the costs or risks associated with the design of the subject systems. Therefore, disclosure of the information contained in the referenced document would have the following negative impacts on the competitive position of MHI in the U.S. nuclear plant market:

   A. Loss of competitive advantage due to the costs associated with the development of the methodology related to the analysis.

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

I declare under penalty of perjury that the foregoing affidavit and the matters stated therein are true and correct to the best of my knowledge, information and belief.

Executed on this 2nd day of August 2011.

Y. Ogata

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

Enclosure 3

UAP-HF-11252
Docket Number 52-021

Revised Responses to Request for Additional Information
No.744-5668
Revision 2

August, 2011
(Non-Proprietary)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

08/02/2011

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.:	**NO. 744-5668 REVISION 2**
SRP SECTION:	**19 – Probabilistic Risk Assessment and Severe Accident Evaluation**
APPLICATION SECTION:	**19.1**
DATE OF RAI ISSUE:	**04/27/2011**

QUESTION NO. : 19-502

(Follow-up to Question 19-459) The staff observes that the US-APWR internal fire and internal flooding risk assessments have been performed with conservative assumptions and thus removal of these conservatisms from the models could lower the estimated risks. However, it is not appropriate to conclude that removing the conservatisms would also lower the conditional containment failure probability (CCFP) accordingly as discussed in the response. A quantitative assessment of the CCFP may be necessary. For example, if the dominating scenario of switchyard fire as mentioned in the response is removed from the models due to its conservatism, the total fire CDF and LRF are appropriately reduced to 1.1E-6/yr and 1.73E-7/yr respectively. This yields a CCFP of 0.16, which is higher than the original estimate of 0.15.
The above consideration is also applicable to the internal flooding risk assessment. Removal of the most significant flood scenario of "Operator Failure to Open the Valve of the EFW Pit Discharge Cross-tie Line" mentioned in the response would not necessarily result in a reduction in CCFP.
Please justify the CCFP safety goal exceedance.

ANSWER:

DCD Revision 3 updated the internal fire PRA and the internal flooding PRA with design specific information and internal events PRA model in accordance with the response to RAI #641-5045 19-483 and RAI #640-5051 19-472.

The following are updated CDFs, LRFs and CCFPs.

	CDF	LRF	CCFP
Internal events at power:	1.0E-06/RY	1.1E-07/RY	0.11
Internal fire:	8.6E-07/RY	1.9E-07/RY	0.22
Internal flood:	8.9E-07/RY	1.6E-07/RY	0.18
Total	2.8E-06/RY	4.6E-07/RY	0.16

In DCD Rev.3, CCFP is 0.16 and does not satisfy 0.10 of safety goal. The cause of the CCFP value exceeding the NRC safety goal is due to use of highly conservative assumptions as discussed in RAI Question 19-459. The safety goal will be satisfied if these conservative assumptions are eliminated.

(1) There are two significant conservative assumptions commonly considered in the PRA models as follows:

(a) The PRA models consider impractical maintenance configurations which cause a maintenance outage of SSCs for two trains simultaneously. (For example, simultaneous maintenance outage of C-CS/RHR pump and D-ESW pump, which causes C and D train CS/RHR systems unavailable.)

(b) The PRA models do not consider manual activation of the containment spray system in the case of failure of CS/RHR system automatic start-up due to common cause failure of the basic software.(a)The PRA model considers impractical maintenance configurations which cause a maintenance outage of SSCs for two trains simultaneously. (For example, simultaneous maintenance outage of C-CS/RHR pump and D-ESW pump causes C and D train CS/RHR systems to be unavailable.)

If the PRA model is updated to eliminate these unnecessary conservative assumptions, which are applicable to internal events, internal fire and internal flood, the probabilities of containment failure sequences will be reduced and the total CCFP will be approximately 0.14 as follows.

	CDF	LRF	CCFP
Internal events	1.02E-06	1.03E-07	0.10
Internal fire	8.37E-07	1.64E-07	0.19
Internal flood	8.62E-07	1.26E-07	0.15
Total	2.72E-06	3.93E-07	0.14

(2) The internal fire risk assessment described in the DCD does not consider the fire suppression effect.

The internal fire risk assessment described in the DCD does not consider the fire suppression effect. However, generally, the fire risk assessment would consider the fire suppression effect.

A sensitivity analysis using a fire suppression probability 0.1/d (arbitrary value) except for fires in the containment was performed, and the CDF and LRF were reduced to 1.2E-07/RY and 4.4E-08/RY, respectively, in DCD Revision 3. This result shows that the internal fire risk contribution to total risk can be reduced by considering fire suppression effects.

(3) The internal flood risk assessment described in the DCD does not consider the effect of the in-service inspection (ISI) program, and isolation of the flooding source.

The internal flood risk assessment described in the DCD does not consider the effect of the in-service inspection (ISI) program, and isolation of the flooding source. However, generally, the internal flooding risk assessment would consider the ISI program and isolation of flood sources.

The pipe rupture frequencies for the internal flooding PRA used for the base case are based on EPRI-TR-1013141 Revision1 Table 6-1 (In-Service Inspection for Wall Thinning: None, Leak Inspection Interval: Weekly). In a sensitivity analysis, Case 7a (In-Service Inspection for Wall Thinning: 10 year ISI Interval – Limited Coverage, Leak Inspection Interval: Weekly) in Table 4-27 of the EPRI report, the corresponding inspection effectiveness factors are assumed in evaluating the pipe rupture frequencies, except for piping of the fire protection water supply system and sampling system. The total internal flooding CDF and LRF are 5.6E-07/RY and 1.0E-07/RY, respectively, for the Case 7a analysis. This result shows that an adequate inspection program would be effective in reducing the flood risk contribution to total risk. Class 1 and 2 piping would be inspected every 120 months in accordance with 10 CFR 50.55a.

In addition, internal fire and flood Level 2 PRA result shows that the dominant release category is RC 3, containment failure before core damage, for both fire and flood as shown below. This table is a summary of the frequency of release categories for the DCD Rev. 3.

Release Category	Internal fire		Internal flood	
	Freq.(/RY)	%	Freq.(/RY)	%
RC1 (Containment Bypass)	2.7E-08	3.2%	4.0E-09	0.5%
RC2 (Containment Isolation Failure)	6.2E-09	0.7%	4.0E-09	0.5%
RC3 (Containment Failure before Core Damage)	9.2E-08	10.7%	1.0E-07	11.2%
RC4 (Early Containment Failure)	3.6E-08	4.1%	1.6E-08	1.8%
RC5 (Late Containment Failure)	2.5E-08	2.9%	3.2E-08	3.6%
RC6 (Intact Containment)	6.7E-07	78.3%	7.4E-07	82.5%
Total CDF	8.6E-07	100.0%	8.9E-07	100.0%
LRF	1.9E-07	-	1.6E-07	-
CCFP	0.22		0.18	

The scenario of containment failure before core damage is assumed as CCFP of one and this scenario is the main contributor of the total LRF for both internal fire and flood. Therefore the CCFP of internal fire and flood is evaluated to be high. For the internal fire and flood model, various operators' recovery actions are not credited conservatively, although lots of operators' recovery actions may be expected achievable because sufficient time is available before containment failure and the environmental condition has not been significantly degraded because core is still intact. These operators' recovery actions include following:

· Local manual operation of mitigation systems when signal or power cable is damaged by fire.

· Manual activation of the containment spray system when auto activation by signal is failed.

(4) Any recovery actions are not considered for the sequences of containment failure before core damage.

The sensitivity analysis result of (4) (failure probability of recovery for the sequences of containment failure before core damage is assumed 0.1) is follows:

	CDF	LRF	CCFP
Internal events	1.02E-06	8.61E-08	0.08
Internal fire	8.00E-07	1.27E-07	0.16
Internal flood	8.04E-07	6.90E-08	0.09
Total	2.62E-06	2.82E-07	0.11

(5) The failure fraction for containment failure by the rocket-mode RV failure is conservatively assumed.

Containment failure by the rocket-mode RV failure is also conservatively assumed when neither TI-SGTR nor hot leg rupture occurs, in the high RCS pressure scenario. Therefore, CCFP of internal fire is high since all/some of SDV/DV is unavailable in many of dominant fire scenarios.

The sensitivity analysis result of (5) (failure probability of rocket-mode RV failure is assumed 0.1) is follows:

	CDF	LRF	CCFP
Internal events	1.03E-06	1.05E-07	0.10
Internal fire	8.60E-07	1.66E-07	0.19
Internal flood	8.91E-07	1.53E-07	0.17
Total	2.78E-06	4.24E-07	0.15

In addition, the conservative assumptions of (1) to (5) are independent of each other. Therefore, these conservative assumptions can be eliminated simultaneously. The sensitivity analysis results of having eliminated these conservative assumptions of (1) to (5) with several combinations are shown below.

The sensitivity analysis result of eliminating (1) and (4) is follows:

	CDF	LRF	CCFP
Internal events	1.02E-06	8.58E-08	0.084
Internal fire	7.74E-07	1.01E-07	0.13
Internal flood	8.01E-07	6.60E-08	0.082
Total	2.59E-06	2.53E-07	0.098

The sensitivity analysis result of eliminating (1), (4), and (5) is follows:

	CDF	LRF	CCFP
Internal events	1.02E-06	8.44E-08	0.083
Internal fire	7.74E-07	8.26E-08	0.11
Internal flood	8.01E-07	6.30E-08	0.079
Total	2.59E-06	2.30E-07	0.089

The sensitivity analysis result of eliminating (2) and (3) is follows:

	CDF	LRF	CCFP
Internal events	1.03E-06	1.07E-07	0.10
Internal fire	1.20E-07	4.40E-08	0.37
Internal flood	5.62E-07	9.99E-08	0.18
Total	1.71E-06	2.51E-07	0.15

The sensitivity analysis result of eliminating (1), (2), (3), (4), and (5) is follows:

	CDF	LRF	CCFP
Internal events	1.02E-06	8.44E-08	0.083
Internal fire	1.00E-07	2.36E-08	0.24
Internal flood	5.04E-07	4.20E-08	0.083
Total	1.62E-06	1.50E-07	0.092

As described above, the CCFP is evaluated to be less than 0.1 by eliminating the conservative assumptions of (1), (4), and (5).

However, elimination of the conservative assumptions of (2) and (3) affect the CCFP slightly since these conservative assumptions affect both CDF and LRF.

The internal fire CCFP is increased by eliminating the conservative assumption of (2). This is because this sensibility analysis does not consider the fire suppression in the containment. Therefore fire scenario of loss of RCS depressurization function via SDV/DV may not be mitigated. The CCFP of fire scenarios due to loss of SDV/DV is relatively high so that CCFP increases if this conservative assumption is eliminated.

Two operator actions are addressed in these sensitivity analyses, i.e. fire suppression and recovery of containment heat removal when core cooling is successful.

(Fire suppression)

In the base case fire suppression systems are not taken into consideration explicitly. However, most fires in the existing plants are extinguished by automatic fire suppression system or manual actions before they have the ability to spread and cause considerable damage. A sensitivity analysis has therefore been performed using a failure probability of 0.1/Demand to determine the effects of fire suppression system. Credit for fire suppression system for in-containment fire scenarios is not taken because the containment is closed during operation and is inaccessible for manual fire fighting. In this sensitivity case, the fire risk reduced significantly; CDF reduced by a factor of 0.1. Therefore CDF of this fire PRA remains conservative because almost all fire may be extinguished before fire induced plant damage has been caused.

(Recovery of containment heat removal when core cooling is available)

In the base case, various operators' recovery actions are not credited conservatively, although lots of operators' recovery actions may be expected achievable because sufficient time is available before containment failure and the environmental condition has not been significantly degraded because core is still intact. These operators' recovery actions include following:

- Local manual operation of mitigation systems when signal or power cable is damaged by fire.
- Manual activation of the containment spray system when auto activation by signal is failed.
- Repair components which are failed. (This is added from previous response)

A sensitivity analysis has been performed using a failure probability of 0.1 to evaluate the effects of recovery of containment heat removal when core cooling is available. In the result of this sensitivity analysis, the CCFP is reduced significantly; total CCFP is reduced to 0.11 from base case 0.16, and

CCFP of internal events and internal flood are evaluated less than 0.10.

Therefore, the US-APWR will satisfy the safety goal if these conservative assumptions are eliminated.

Impact on DCD

There is no impact on the DCD.

Impact on R-COLA

There is no impact on the R-COLA.

Impact on S-COLA

There is no impact on the S-COLA.

Impact on PRA

There is no impact on the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

08/02/2011

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.:	**NO. 744-5668 REVISION 2**
SRP SECTION:	**19 – Probabilistic Risk Assessment and Severe Accident Evaluation**
APPLICATION SECTION:	**19.1**
DATE OF RAI ISSUE:	**04/27/2011**

QUESTION NO. : 19-503

(Follow-up to Question 464) Generally, in performing fire risk assessment, the human error probabilities (HEPs) modeled in the internal events (IEs) PRA as provided in Table 23.3-2 "Human Actions Considered in the Internal Events PRA Model" of US-APWR PRA Chapter 23 must reflect equipment/indication losses, fire induced stress, communications difficulties, availability of lighting, potential impacts from smoke and heat, etc. Please describe the method used to examine and modify the IEs PRA HEPs to account for the potential impacts of fire events.

ANSWER:

The following were assumed to estimate the human error probabilities of post-fire operator actions.

- Fire brigade is always provided based on the fire protection program which will be developed later to meet with the requirements in regulatory guide 1.189. In case that fire occurs and manual fire suppression is required, fire brigade will participate in fire fighting and it allows the plant operators to concentrate their attention on plant safe-shutdown operation.
- Stress levels of human actions of post-fire plant safe shutdown do not become higher due to a fire. The reasons are as follows.
  - The communication system will remain active during a fire because the redundant plant communication systems are installed with a minimum of two verbal communication paths between all plant locations.
  - The lighting system will remain active during a fire because emergency lighting system with redundant power sources is provided to every fire area.
  - Since the probability of heat and smoke propagation to adjacent fire compartment could be assumed to be extremely low by the fire dampers which are installed in series to all HVAC ducts which pass through the fire compartment, it is possible to ignore the stress increase due to their effects.
- Any recovery of equipment damaged by a fire is not expected. Also if a fire has potential failing instrumentation equipment for monitoring plant condition, relevant operator actions are not expected.

In the fire PRA, the HEPs of Human Failure Events (HFEs) estimated in the internal events PRA have been examined and modified based on the following assumptions.

- Recovery of equipment damaged by a fire has not been postulated, therefore, no credit has been taken for the operation of any equipment damaged by a fire.
- Fire induced heat and smoke effects have the potential to jeopardize communication between the area of the fire and the MCR. Therefore, no credit has been taken for operation of the equipment in the fire compartments affected by the fire.
- The probability of heat and smoke propagation to the adjacent fire compartment could be assumed to be very low because the fire dampers have been installed in series to all HVAC ducts in the fire compartment. Therefore it is not necessary to consider the heat and smoke propagation to the adjacent fire compartment through the duct. The stress level of operators in the adjacent fire compartment would not increase.
- For the case where a fire has the potential to affect the function of instrumentation required for recognition of the plant condition, the human error probability of any operation associated with responding to information from the affected instrumentation is set to 1.0.
- For the case where a fire in the MCR has the potential to jeopardize the ability of operations that would be performed in the MCR, affect MCR habitability or affect communication functions in the MCR, due to the effects of heat and smoke generated by a fire, the operators will abandon the MCR and evacuate to the remote shutdown room (RSR) to perform required operations from the remote shutdown console (RSC). The operating environment of RSR is worse than MCR because annunciators in RSR are not alarming and not blinking, and supervisor console and shift technical advisor console for plant monitoring are not installed in RSR. In this scenario, the HEP for operations at the RSC has been set to 0.1.

As a bounding sensitivity study for human error probabilities, it is assumed that the HEPs of all human actions outside control room are multiplied ten times. This value (ten times) is selected as an extreme case from the screening value for high stress HEPs in NUREG-1921. Though this conservative sensitivity study, total CDF is result in 3.5E-06/ry, only about four times of the base case.

DCD Revision 3 page 19.1-937, sixth assumption "6. Operators are well trained in responding to fire event" Table 19.1-119 will be revised to include above discussion as follows.

~~6. Operators are well trained in responding to fire event.~~

6. Human error probabilities of post-fire operator actions are assumes as follows.

- *No credit has been taken for the operator actions of any equipment in the fire compartment affected by fire.*
- *Fire brigade is always provided based on the fire protection program which will be developed later to meet with the requirements in regulatory guide 1.189. Higher stress levels of human actions of post-fire are not assumed.*
- *The HEP for operations at the remote shutdown console is assumed as 0.1.*

Impact on DCD

DCD Table 19.1-119 will be revised as attached mark-up. (Attachment 1)

Impact on R-COLA

There is no impact on the R-COLA.

Impact on S-COLA

There is no impact on the S-COLA.

Impact on PRA

There is no impact on the PRA.

Table 19.1-119 Key Insights and Assumptions (Sheet 40 of 46)

Key Insights and Assumptions	Dispositions
Internal fire assumption	
1. All fire doors serving as fire barriers between redundant safety train fire compartments are normally closed.	9.5.1.2.1 COL 9.5(1)
2. For transient combustibles, "three Airline trash bags" has been assumed in each fire compartment.	9.5.1.2.1 COL 9.5(1)
3. 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.	9.5.1.2.1 COL 9.5(1)
4. The Heat Release Rate of various items as specified in Chapter-11 of NUREG/CR-6850 is used.	9.5.1.2.1 COL 9.5(1)
5. Damage temperature of thermoplastic cables as shown in Appendix-H of NUREG/CR-6850 is used as the target damage temperature.	9.5.1.2.1 COL 9.5(1)
~~6. Operators are well trained in responding to fire event.~~ 6. Human error probabilities of post-fire operator actions are assumes as follows. - No credit has been taken for the operator actions of any equipment in the fire compartment affected by fire. - Fire brigade is always provided based on the fire protection program will be developed later to meet with the requirements in regulatory guide 1.189. Higher stress levels of human actions of post-fire are not assumed. - The HEP for operations at the remote shutdown console is assumed as 0.1.	9.5.1.2.1 COL 9.5(1)
7. One of RCS letdown isolation valves and one of RCS vent line isolation valves are locked close by administrative controls	COL 13.5(1) COL 13.5(7)
8. Each yard transformer is separated by a fire barrier.	19.1.5.2.1

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

08/02/2011

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.:	**NO. 744-5668 REVISION 2**
SRP SECTION:	**19 – Probabilistic Risk Assessment and Severe Accident Evaluation**
APPLICATION SECTION:	**19.1**
DATE OF RAI ISSUE:	**04/27/2011**

QUESTION NO. : 19-504

(Follow-up To Question 19-469) Part (3) of the response to Question 469 provides the flooding frequencies for EFWS, MFWS, CWS, and MSS. Please explain why these frequencies do not match the corresponding failure rates provided in Table 22.5-2 "Mean Pipe Failure Rates for US-APWR Internal Flood PRA" of the US-APWR PRA Chapter 22.
Part (5) of the response identifies the most significant human actions modeled in the flooding assessment. Clarify whether any human actions will be conducted outside the control room. If there are any, describe the potential impacts on these actions due to a postulated flooding event.

ANSWER:

Follow-up to RAI 19-469 response, part (3)

Table 22.5-2 "Mean Pipe Failure Rates" of the PRA report shows flood frequency per reactor operating year - linear foot of each system. The flood frequency described in the answer to RAI 19-469 part (3) is total flood frequency (per reactor year) of each system. The total flood frequency of each system is calculated by multiplying flooding frequency of each system in Table 22.5-2 (unit is per reactor operating year - linear foot) by the total pipe lengths (unit is per reactor operating year – total feet).

For example,

Frequency of spray of EFWS (8.1E-02) in Table 1

= total length of EFW pipe (length of EFW pipe (a) + length of EFW pipe (b) + length of EFW pipe (c) + …) * (3.2E-06: Mean Pipe Failure Rates of EFWS from Table 22.5-2)

= 8.1E-02 /ry.

The length of each pipe is assumed as "Length + Width + Height" of each flooding area for flooding risk assessment in DCD Revision 2 conservatively.

In DCD Revision 3, to reduce the conservative assumptions, design specific pipe lengths of flood areas are used to calculate flood frequencies instead of the conservative piping lengths in DCD Revision 2.

Table 1 Total flood frequency of each significant system (DCD Revision 2)

System		Frequency (/ry)			
		Spray	Flood	Major Flood	Total
EFWS	Emergency Feed Water System	8.1E-02	9.1E-03	2.3E-02	1.1E-01
MFWS	Main Feed Water System	2.8E-02	0.0E+00	9.5E-03	3.7E-02
CWS	Circulatiing Water System (Piping)	2.7E-02	0.0E+00	9.6E-03	3.7E-02
MSS	Main Steam System	5.2E-03	3.9E-04	4.8E-03	1.0E-02

Table 2 presents the results of the flood frequencies in DCD Revision 3.

Table 2 Total flood frequency of each significant system (DCD Revision 3)

System		Frequency (/ry)			
		Spray	Flood	Major Flood	Total
EFWS	Emergency Feed Water System	1.8E-02	1.4E-03	5.5E-03	2.5E-02
MFWS	Main Feed Water System	1.0E-02	0.0E+00	3.5E-03	1.4E-02
CWS	Circulatiing Water System (Piping)	2.2E-02	0.0E+00	7.7E-03	2.9E-02
MSS	Main Steam System	8.5E-04	4.8E-07	8.5E-04	1.7E-03

Follow-up to RAI 19-469 response, part (5)

The operator action outside the MCR which is a significant contributor for internal flooding risk is the failure of switching over the water source to the intact EFW pump lines for long term secondary cooling. It would be successful if one of three water supply lines is switched, in accordance with the developing emergency response guideline.

- open two manual valves between demineralized tank and EFW pump suction lines, EFS-VLV-004 and EFS-VLV-006A (or EFS-VLV-006B) to supply water from the demineralized water tank to A and B EFW pumps (or C and D EFW pumps). EFS-VLV-004 and EFS-VLV-006A are located on B1F east side of reactor building and EFS-VLV-006B is located on B1F west side of reactor building.
- open two manual valves EFS-VLV-006A and EFS-VLV-006B to line up EFW pit A and B to accommodate water source into intact EFW pump lines.
- open a manual valve EFS-VLV-001A (or B) which is located on 4F east (or west) side of reactor building to supply water from the demineralized water tank to EFW pit A (or B).

These actions are required approximately eight hours later after the initiating event occurred when a water level of EFW pit is decreased. In the case of internal flooding, operator couldn't access to the valve areas which is submerged, for a while. However, operator could afford the time to perform actions in the one of above accessible valve areas. Therefore, at least one of three operations would succeed.

Other operator actions outside the MCR such as the isolation of flood sources are conservatively not considered for the internal flooding assessment.

Question at the public meeting on June 9, 2011

- Check the latest version of pipe rupture frequency data for internal flooding PRA.

Flooding risk assessment of the DCD Revision 3 is performed based on EPRI-1013141, "Pipe Rupture Frequencies for Internal Flooding PRAs, Revision 1" March 2006. This report was updated as EPRI-1021086, "Pipe Rupture Frequencies for Internal Flooding Probabilistic Risk Assessments Revision 2" October 2010. These failure rates are used in conjunction with EPRI guidelines for performing a PRA internal flooding analysis, EPRI-1019194, "Guidelines for Performance of Internal Flooding Probabilistic Risk Assessment", December 2009.

Table 1 is pipe failure data for each system used for US-APWR DCD Revision3 internal flooding risk assessment. This table is described in Table 22.5-2 of US-APWR PRA technical report MUAP-07030(R3).

Table 2 is updated pipe failure data for each system used for this sensitivity study. In this sensitivity study, pipe failure data of following systems is modified.

- No. 8b Main Steam System (T/D-EFW Pump Steam Line Break): Applied failure data for main steam system which has come to be available in EPRI-1021086.
- No. 4 Emergency Feed Water System: Applied data for safety related CCW and CST (systems that take suction from the condensate storage tank – for PWRs primarily the emergency feedwater system) from the failure data for non safety related main feedwater system.

The CDF and LRF of this sensitivity study reduced to about half of total CDF and LRF of the DCD Revision 3. Major internal flooding scenarios will not be changed significantly.

Flood Category	DCD R3 (EPRI 1013141) CDF (/ry)		Sensitivity Study (EPRI 1021086) CDF (/ry)		Ratio (Sensitivity Study / DCD R3)
Spray	CDF: 3.3E-08	4%	CDF: 6.4E-08	15%	2.0
	LRF: 3.5E-09	2%	LRF: 1.7E-08	19%	5.0
Flood	CDF: 2.6E-07	30%	CDF: 1.1E-07	25%	0.4
	LRF: 4.2E-08	27%	LRF: 2.7E-08	29%	0.7
Major Flood	CDF: 5.9E-07	67%	CDF: 2.6E-07	61%	0.4
	LRF: 1.1E-07	71%	LRF: 4.9E-08	53%	0.5
Total	**CDF: 8.9E-07**	100%	**CDF: 4.4E-07**	100%	**CDF: 0.5**
	LRF: 1.6E-07		**LRF: 9.4E-08**		**LRF: 0.6**

Followings are dominant flood areas for CDF and LRF.

DCD Revision 3 (EPRI-1010141)				Sensitivity study (EPRI 1021086)			
	Area	Category	CDF		Area	Category	CDF
1	FA2-321-01	Major Flood	1.6E-07	1	FA2-321-01	Major Flood	5.5E-08
2	FA2-320-01	Major Flood	1.5E-07	2	FA2-320-01	Major Flood	4.5E-08
3	FA2-507-02	Flood	7.1E-08	3	FA2-321-01	Flood	2.8E-08
4	FA2-321-01	Flood	5.7E-08	4	FA2-414-01	Spray	2.5E-08
5	FA2-320-01	Flood	4.4E-08	5	FA2-320-01	Flood	2.2E-08
6	FA2-102-01	Major Flood	3.2E-08	6	FA2-414-01	Major Flood	1.7E-08
7	FA2-111-01	Major Flood	3.0E-08	7	FA2-112-01	Major Flood	1.7E-08
8	FA2-108-01	Major Flood	2.9E-08	8	FA2-415-01	Major Flood	1.6E-08
9	FA2-420-01	Flood	2.3E-08	9	FA2-101-01	Major Flood	1.5E-08
10	FA2-109-01	Major Flood	2.3E-08	10	FA2-108-01	Major Flood	1.2E-08

DCD Revision 3 (EPRI-1010141)				Sensitivity study (EPRI 1021086)			
	Area	Category	LRF		Area	Category	LRF
1	FA2-321-01	Major Flood	3.9E-08	1	FA2-321-01	Major Flood	1.4E-08
2	FA2-320-01	Flood	1.4E-08	2	FA2-321-01	Flood	6.9E-09
3	FA2-320-01	Major Flood	1.2E-08	3	FA2-112-01	Major Flood	4.2E-09
4	FA2-507-02	Flood	7.5E-09	4	FA2-320-01	Major Flood	3.8E-09
5	FA2-108-01	Major Flood	7.3E-09	5	FA2-108-01	Flood	2.9E-09
6	FA2-507-01	Major Flood	6.5E-09	6	FA3-112-01	Major Flood	2.6E-09
7	FA2-109-01	Major Flood	5.7E-09	7	FA2-206-01	Major Flood	2.0E-09
8	FA2-108-01	Flood	5.6E-09	8	FA2-108-01	Major Flood	1.9E-09
9	FA2-112-01	Major Flood	5.4E-09	9	FA2-320-01	Flood	1.8E-09
10	FA3-114-01	Major Flood	5.3E-09	10	FA2-507-01	Major Flood	1.5E-09

Impact on DCD

There is no impact on the DCD.

Impact on R-COLA

There is no impact on the R-COLA.

Impact on S-COLA

There is no impact on the S-COLA.

Impact on PRA

There is no impact on the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

08/02/2011

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: NO. 744-5668 REVISION 2

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

APPLICATION SECTION: 19.1

DATE OF RAI ISSUE: 04/27/2011

QUESTION NO. : 19-505

(Follow-up to Question 19-462) Please (a) describe in more detail and provide the failure probability of top event "EVA - Evacuation to RSP from MCR," (b) provide the accident frequencies of main control room sequences shown in Figure 1, and (c) explain why both success and failure paths of top event EVA are identical.

ANSWER:

(a) Description of top event "EVA - Evacuation to RSP from MCR"

Top event heading "EVA" means fraction of evacuations from the MCR to the RSC room for MCR fires.

"Challenging fire" for MCR shown in Table C-4 of appendix C of NUREG/CR-6850 will cause adverse effect to environment in MCR. The MCR "Challenging fire" may force the operators to evacuate from MCR to RSC room.

The fraction of EVA has been estimated by using the MCR fire event data presented in NUREG/CR-6850, Appendix –C, Table C-4. The probability of EVA has been estimated as follows.

- Fires of category "challenging fire" of the MCR fire events in Table C-4 are assumed as the fires that cause evacuation from MCR.
- The probability of EVA has been estimated by the following equation.

EVA = Number of MCR challenging fire events / Total number of MCR fire events
= 5 events / 28 events
≈ 0.2

Most of plant control systems of US-APWR are composed of low current digital systems. Although sufficient fire event data of digital control system have not been recorded, fire frequency of low current digital system will not greater than fire frequencies of conventional nuclear plants with analog control circuits. In this analysis, the fraction of MCR abandonment is set to same fraction of above conventional plant MCR conservatively.

The failure of transferred operation from the MCR to the RSC room is not considered in the top event "EVA". The reasons are as follows.

- The transferring to the RSC from the OC (Operator Console) in the MCR is a simple operation, requiring only the operation of two switches. Therefore, the possibility of failure of this action would be very low.
- MCR/RSC transfers switches are installed on panels are different from those on the OC and the RSC. No switches, except the MCR/RSC transfer switches, are installed on the panels. Therefore, the possibility of selecting switches other than the transfer switches would be very unlikely.
- Even if operators forget to operate the MCR/RSC transfer switches, operators would recognize this easily because the operators would not be able to perform any action or monitor any plant parameters from the RSC under such a condition. Therefore, the possibility of an omission error of these actions (to operate the transfer switches) which would go undetected would be very unlikely.

(b) Accident frequencies of main control room sequences shown in Figure 1

Each sequence of the MCR event tree in Figure-1 is as follows. The accident frequencies are results from the internal fire PRA in DCD Revision 3. The fire PRA model is updated according to the internal event PRA model.

Accident Sequence Number of MCR-ET	CDF(/RY)
4	3.8E-13
6	5.5E-12
7	3.4E-10
8	2.7E-14
9	3.9E-15
10	9.9E-13
11	4.2E-12
16	7.8E-13
18	5.0E-12
19	8.1E-09
20	6.8E-13
21	1.1E-13
22	1.1E-11
23	8.5E-12

(c) Reason why both success and failure paths of top event EVA are identical

Figure 1 MCR event tree has been developed to quantify sequences for both operations at the MCR and at the RSC. The sequences of the event tree after the upper branch of top event "EVA" are for the MCR and the sequences after the lower branch of "EVA" are for the RSC. The success and failure paths of top event EVA are identical because the RSC has the same functions and capabilities for the safety VDUs at the operator console (OC) in the MCR, and all equipment that is operated from the safety VDU of the OC in the MCR can be operated from the RSC.

Although the same operation could be taken by the operators in the MCR and in the RSC, each human error probability of the operation from the safety VDU in the MCR and in the RSC is different. Because Bypassed or Inoperable Status Indication (BISI) information is displayed on the Large Display Panel (LDP) in the MCR as an alarm, operators in MCR could easily recognize the plant abnormal condition and share the information. Additionally, because Supervisor Console and Shift Technical Advisor Console are also installed in MCR, senior reactor operators (SRO) could easily hold the plant condition and actions by other reactor operator. However, it is more difficult for operators to recognize plant abnormal condition and to achieve recovery action by SRO in RSC room than in MCR because LDP or associated consoles are not installed in RSC room. Therefore, it is assumed that HEP of the operation in RSC room is higher than that in MCR. The human error of 0.1 is assumed to the operations from RSC that covers alternate shutdown means.

Impact on DCD

There is no impact on the DCD.

Impact on R-COLA

There is no impact on the R-COLA.

Impact on S-COLA

There is no impact on the S-COLA.

Impact on PRA

There is no impact on the PRA.

19-505-4

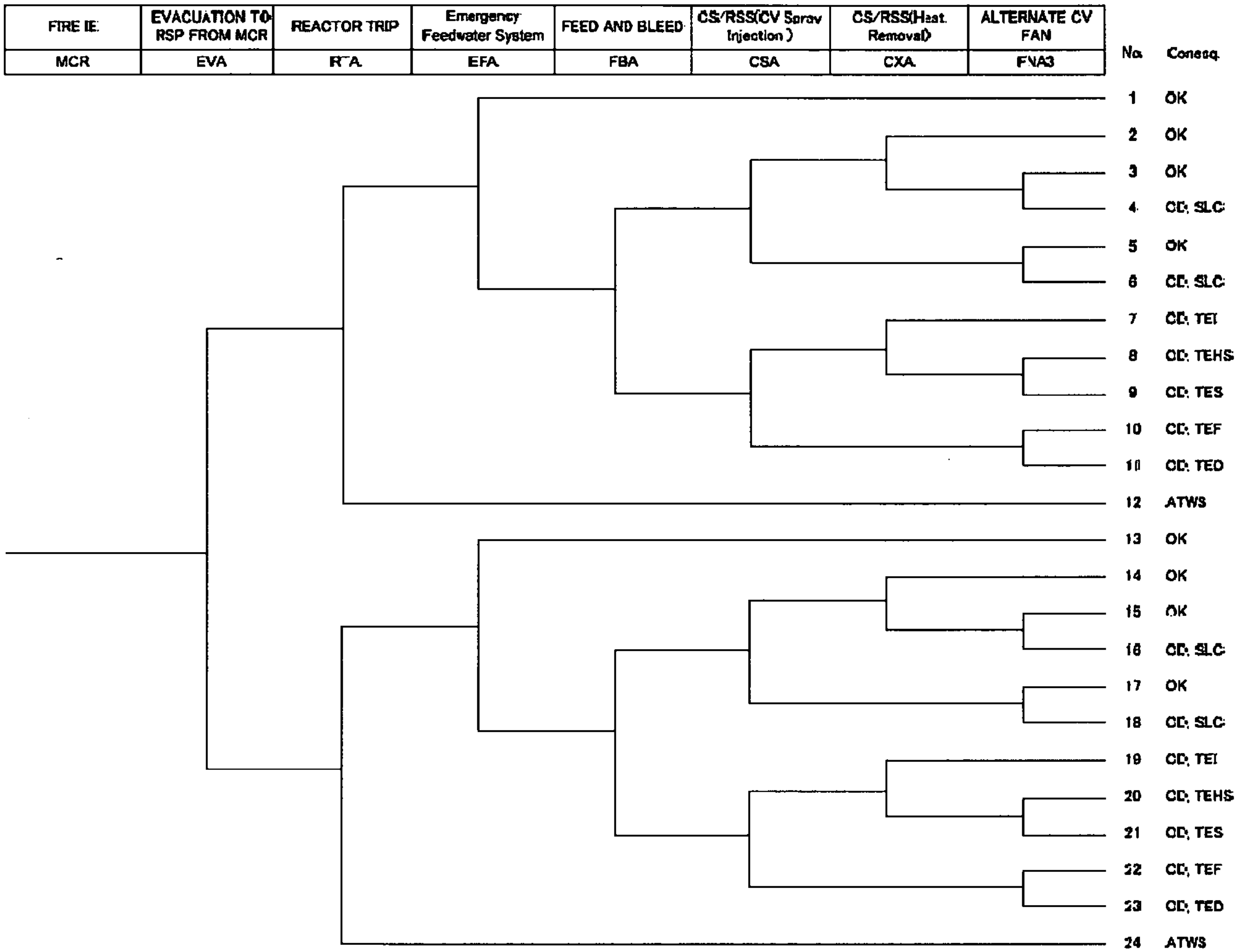

Figure 1 Revised MCR Event Tree