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TOKYO, JAPAN

February 6, 2009

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-09046

Subject: MHI's Responses to US-APWR DCD RAI No.150-1635 REVISION 1

Reference: 1) "Request for Additional Information No. 150-1635 Revision 1, SRP Section: 17.04 - Reliability Assurance Program (RAP), Application Section: 17.4 Reliability Assurance Program," dated January 9, 2009.

With this letter, Mitsubishi Heavy Industries, Ltd. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") a document entitled "Responses to Request for Additional Information No. 150-1635 Revision 1".

Enclosed are the responses to the RAIs contained within Reference 1. Of these RAIs, questions #17-04-19, #17-04-20, #17-04-23, #17-04-24 and #17-04-30 will not be answered within this package. These questions require additional time for analyses and internal discussions, and will be answered by 10th of March 2009.

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.

Enclosure:

1. Responses to Request for Additional Information No.150-1635 Revision 1.

CC: J. A. Ciocco
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Docket No. 52-021
MHI Ref: UAP-HF-09046

Enclosure 1

UAP-HF-09046
Docket Number 52-021

Responses to Request for Additional Information No. 150-1635
Revision 1

February 2009

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

2/6/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. No.52-021

RAI NO.: **NO.150-1635 REVISION 1**

SRP SECTION: **17 04 - Reliability Assurance Program (RAP)**

APPLICATION SECTION: **17.4 Reliability Assurance Program**

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

QUESTION NO. : 17.04-21

The applicant did not include in Table 17.4-1 of the US-APWR DCD, Revision 1, the Emergency Feedwater System (EFWS) pit water level indicators/sensors (i.e., LA3760, LA3761, LA3762, LA3763). The evidence presented below suggests these water level indicators are risk-significant and should be included in D-RAP:

a) HRA event EFWOO04LAAA ("Miscalibration of EFW Pit A Water Level Sensors LA3760, LA3761") has a RAW of 41.9 and FV of 9.11E-03 for internal flooding at power (Table 22.6-6 of US-APWR Probabilistic Risk Assessment, MUAP-07030(R0)). The common cause failure of these indicators would have a similar RAW value.

b) HRA event EFWOO04LAAA has a RAW of 2.3 for internal events at power (Table18.6-2 of US-APWR Probabilistic Risk Assessment, MUAP-07030(R0)). The common cause failure of these indicators would have a similar RAW value.

c) HRA event EFWOO04LAAA has a RAW of 2.4 for internal fire at power (Table 23R-10 of US-APWR Probabilistic Risk Assessment, MUAP-07030(R0)). The common cause failure of these indicators would have a similar RAW value.

The staff requests that the applicant include in Table 17.4-1 of the US-APWR DCD the EFWS pit water level indicators. Otherwise, provide the basis for not including these water level indicators in Table 17.4-1 of the US-APWR DCD (include in the basis a discussion of the associated risk importance measures from the various PRA models, consideration of deterministic methods, e.g., defense-in-depth, consideration of seismic margins analysis, and the expert panel's deliberation for not including these SSCs in D-RAP).

ANSWER:

In EFW pit, water level is checked every 12 hours and failure of a sensor can be detected by recognizing inconsistent output signals (water levels) from the two sensors, and are likely to be repaired within a short period of time if failure has been detected. Common cause failure of two sensors of an EFW pit that result in same incorrect output signal may not be detected and left faulted until the initiating event occurs, but such kind of failure is considered to be rare. For this reason failure of water level sensors that occur prior to the initiating event is not modeled.

Failure of sensors during the 24 hour mission time is modeled in the PRA. When the two sensors associated to an EFW pit both fail, the operator cannot detect the timing to change over the water source and the EFW pumps that are supplied water from the pit may fail. CCF may occur among the sensors and therefore the PRA will be revised to take into account of the CCF of sensors, during the next update. The probability of CCF of sensors during the mission time is much lower than the human error probability of operators to change over from the low level EFW pit, which is the order of 1.0E-2. Therefore, the change to the PRA model will not impact the PRA result.

Incorporating the results of the revised PRA model and the discussion of expert panel, Table 17.4-1 will be revised (See the Attachment to this RAI response, page 17.4-16). This will be done by the next revision of the US-APWR DCD.

Impact on DCD

List of risk significant SSCs will be revised as noted above considering the result of revised PRA model and the discussion of expert panel in the response to RAI 17.04-21.

Impact on COLA

There is no impact on COLA from this RAI.

Impact on PRA

There is no impact on PRA from this RAI.

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US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: **NO.150-1635 REVISION 1**

SRP SECTION: **17.04 – Reliability Assurance Program (RAP)**

APPLICATION SECTION: **17.4 Reliability Assurance Program**

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

QUESTION NO. : 17-04-22

The applicant did not include in Table 17.4-1 of the US-APWR DCD, Revision 1, the High Head Safety Injection (HPI) motor-operated valves (MOV) 014A, B, C, D (i.e., MOV-8810A, B, C, D of the US-APWR Probabilistic Risk Assessment, MUAP-07030(R0)). The evidence presented below suggests these MOVs are risk-significant and should be included in D-RAP:

a) Basic event HPIMVIL8810C ("M/V 8810C INTERNAL LEAK L") has a RAW of 10.9 for internal flooding at power (Table 22.6-7 of US-APWR Probabilistic Risk Assessment, MUAP-07030(R0)).

b) Basic event HPIMVEL8810C ("M/V 8810C EXTERNAL LEAK L") has a RAW of 10.9 for internal flooding at power (Table 22.6-7 of US-APWR Probabilistic Risk Assessment, MUAP-07030(R0)).

c) Basic event HPIMVOM8810C ("M/V 8810C MIS-OPENING") has a RAW of 10.9 for internal flooding at power (Table 22.6-7 of US-APWR Probabilistic Risk Assessment,MUAP-07030(R0)).

d) Basic event HPIMVIL8810D ("M/V 8810D INTERNAL LEAK L") has a RAW of 5.5 for internal flooding at power (Table 22.6-7 of US-APWR Probabilistic Risk Assessment, MUAP-07030(R0)).

e) Basic event HPIMVEL8810D ("M/V 8810D EXTERNAL LEAK L") has a RAW of 5.5 for internal flooding at power (Table 22.6-7 of US-APWR Probabilistic Risk Assessment, MUAP-07030(R0)).

The staff requests that the applicant include in Table 17.4-1 of the US-APWR DCD MOV-014A, B, C, D. Otherwise, provide the basis for not including these MOVs in Table 17.4-1 of the US-APWR DCD (include in the basis a discussion of the associated risk importance measures from the various PRA models, consideration of deterministic methods, e.g., defense-in-depth, consideration of seismic margins analysis, and the expert panel's deliberation for not including these SSCs in D-RAP).

ANSWER:

Based on Table 22.6-7 of US-APWR Probabilistic Risk Assessment, MUAP-07030(R0), RAWs for M/V 8810C and D, which correspond to MOV- 014C and D in Table 17.4-1 of the US-APWR DCD, Revision 1, are at least 5.5E+0.0 and exceed the RAW criteria (i.e. RAW>2.0).

And M/V 8810A and B (MOV- 014A and B) are considered to be equal to M/V 8810C and D (MOV-014C and D) in their functions and failure behavior.

Therefore, these 4 motor valves will be included as "A (B, C, D)-Hot leg recirculation line isolation valves [MOV- 014 A (B, C, D)]" in Table 17.4-1 incorporating the discussion of expert panel (See the Attachment to this RAI response, page 17.4-23). This will be done by the next revision of the US-APWR DCD.

Impact on DCD

List of risk significant SSCs will be revised as noted above considering the discussion of expert panel in the response to RAI 17.04-22.

Impact on COLA

There is no impact on COLA from this RAI.

Impact on PRA

There is no impact on PRA from this RAI.

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US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. No. 52-021

RAI NO.: **NO.150-1635 REVISION 1**

SRP SECTION: **17 04 - Reliability Assurance Program (RAP)**

APPLICATION SECTION: **17.4 Reliability Assurance Program**

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

QUESTION NO. : 17.04-25

The applicant did not include in Table 17.4-1 of the US-APWR DCD, Revision 1, the following SSCs of the Refueling Water Storage System (RWS) (as shown in Figure6A.14.3-1 of the US-APWR Probabilistic Risk Assessment (PRA), MUAP-07030(R0)): Manual Valve (X/V)-027, X/V-028, Orifice 025, and RWS Auxiliary Tank. Based on Table 20.11-3 of the US-APWR PRA, MUAP-07030(R0), these SSCs have a RAW of at least 3.8 for the internal events low-power and shutdown risk assessment, which exceeds the RAW criteria for risk significance (i.e., RAW > 2).

Also, RWS valves VLV-051, VLV-052 and AOV-022 are considered risk-significant in Table 17.4-1 of the US-APWR DCD, Revision 1. However, it is not clear as to which valve numbers these valves correspond to in the US-APWR PRA, MUAP-07030(R0). For example, does VLV-051 in Table 17.4-1 correspond to VLV-027, VLV-028 or another valve in Figure 6A.14.3-1 of the US-APWR PRA.

The staff requests that the applicant:

a) Include in Table 17.4-1 of the US-APWR DCD the X/V-027, X/V-028, Orifice 025, and RWS Auxiliary Tank. Otherwise, provide the basis for not including these SSCs in Table 17.4-1 of the US-APWR DCD (include in the basis a discussion of the associated risk importance measures from the various PRA models, consideration of deterministic methods, e.g., defense-in-depth, consideration of seismic margins analysis, and the expert panel's deliberation for not including these SSCs in D-RAP).

b) For RWS valves VLV-051, VLV-052, and AOV-022 identify the corresponding valve numbers used in the US-APWR PRA, MUAP-07030(R0).

ANSWER:

a) Based on Table 20.11-3 of the US-APWR PRA, MUAP-07030(R0), Manual Valve X/V-027, X/V-028, Orifice 025 and RWS Auxiliary Tank have RAWs which exceed the RAW criteria (i.e., RAW > 2).

Of these SSCs, Manual Valve X/V-027 corresponds to "Refueling water auxiliary tank suction line manual valves [VLV-051]" and X/V-028 corresponds to "Refueling water auxiliary tank inlet line manual valve [VLV-052]". These SSCs are already included as Item 13 and Item 11 respectively of "22.Refueling water storage system (RWS)" in Table 17.4-1 of the US-APWR DCD, Revision 1.

On the other hand, Orifice 025 and RWS Auxiliary Tank are not included in Table 17.4-1. Therefore, these SSCs will be included as "Refueling water storage auxiliary tank [RWS-OTK-002]" and "RWSAT line orifice [TBD : downstream side of VLV-021]" in Table 17.4-1 incorporating the discussion of expert panel (See the Attachment to this RAI response, page 17.4-34). This will be done by the next revision of the US-APWR DCD.

b) As noted above, Manual Valve X/V-027 corresponds to VLV-051 and X/V-028 corresponds to VLV-052. And AOV-022 corresponds to "RWS03" on the RWSP suction line of Figure 6A.14.3-1 of the US-APWR PRA, MUAP-07030(R0).

<u>Impact on DCD</u>

List of risk significant SSCs will be revised as noted above considering the discussion of expert panel in the response to RAI 17.04-25.

<u>Impact on COLA</u>

There is no impact on COLA from this RAI.

<u>Impact on PRA</u>

There is no impact on PRA from this RAI.

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Docket No.52-021

RAI NO.: **NO.150-1635 REVISION 1**

SRP SECTION: **17.04 – Reliability Assurance Program (RAP)**

APPLICATION SECTION: **17.4 Reliability Assurance Program**

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

QUESTION NO. : 17-04-26

The applicant did not include in Table 17.4-1 of the US-APWR DCD, Revision 1, the Essential Service Water System (ESWS) strainers ST02A, B, C, D as shown in Figure 6A.9-2 of the US-APWR Probabilistic Risk Assessment (PRA), MUAP-07030(R0).

Based on Table 20.11-3 of the US-APWR PRA, MUAP-07030(R0), these strainers have a RAW of at least 5 for the internal events low-power and shutdown risk assessment, which exceeds the RAW criteria for risk significance (i.e., RAW > 2). Also, the description of the ESWS strainers listed under Item 4 of ESWS in Table 17.4-1 of the US-APWR DCD, Revision 1, is unclear and appears to be duplicated under Item 5 of ESWS in Table 17.4-1 (page 17.4-36).

The staff requests that the applicant:

a) Include in Table 17.4-1 of the US-APWR DCD the ESWS strainers ST02A, B, C, D. Otherwise, provide the basis for not including these strainers in Table 17.4-1 of the US-APWR DCD (include in the basis a discussion of the associated risk importance measures from the various PRA models, consideration of deterministic methods, e.g., defense-in-depth, consideration of seismic margins analysis, and the expert panel's deliberation for not including these strainers in D-RAP).

b) Clarify in Table 17.4-1 of the US-APWR DCD the description of the ESWS strainers listed under Items 4 and 5 of ESWS in Table 17.4-1 (page 17.4-36).

ANSWER:

As mentioned in QUESTION, ST02 A, B and C have the RAWs which exceed the RAW criteria (i.e., RAW > 2) in the Table 20.11-3 of the US-APWR PRA, MUAP-07030(R0). ST02D is considered to be equal to the ST02A, B and C in its function and failure behavior.

Therefore, these 4 strainers will be included as "A (B,C,D) -CCW heat exchanger inlet strainers[TBD]" under Item 4 of ESWS in Table 17.4-1 incorporating the discussion of expert panel (See the Attachment to this RAI response, page 17.4-36). This will be done by the next revision of the US-APWR DCD.

Items 4 and 5 of ESWS in Table 17.4-1 contain 8 strainers (i.e. EWS-OSR-001A (B,C,D) and EWS-OSR-002A (B,C,D)). EWS-OSR-001A (B,C,D) correspond to ST01, 03, 05, 07 in Figure 6A.9-2 of the US-APWR PRA, MUAP-07030(R0), and EWS-OSR-002A (B,C,D) correspond to ST02, 04, 06, 08 in the same figure.

The FVs and RAWs for these 8 strainers are checked in the following tables of the US-APWR PRA, MUAP-07030(R0):
Table 18.2-1 Basic Events (Hardware Failure, Human Error) FV Importance for CDF
Table 18.2-2 Basic Events (Hardware Failure, Human Error) RAW Importance for CDF
Table 20.11-2 Basic Events (Hardware Failure and Human Error) FV Importance of POS 8-1 for LPSD PRA
Table 20.11-3 Basic Events (Hardware Failure and Human Error) RAW of POS 8-1 for LPSD PRA

And it is confirmed that RAWs for ST03 and ST05 [EWS-OSR-001 B and C] exceed the criteria in Table 18.2-2. ST01 and ST07 [EWS-OSR-001 A and D] is also considered to be equal to the ST03 and ST05 in their functions and failure behavior. Therefore, these 4 strainers will be included as "A1~D1 ESWS sump outlet strainers 1 [EWS-OSR-001A (B,C,D)]" under Item 5 of ESWS in Table 17.4-1 incorporating the discussion of expert panel (See the Attachment to this RAI response, page 17.4-36). This will be also done by the next revision of the US-APWR DCD.

Impact on DCD

List of risk significant SSCs will be revised as noted above considering the discussion of expert panel in the response to RAI 17.04-26.

Impact on COLA

There is no impact on COLA from this RAI.

Impact on PRA

There is no impact on PRA from this RAI.

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US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. No. 52-021

RAI NO.: **NO.150-1635 REVISION 1**

SRP SECTION: **17 04 - Reliability Assurance Program (RAP)**

APPLICATION SECTION: **17.4 Reliability Assurance Program**

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

QUESTION NO. : 17.04-27

Under Item 8 of Essential Service Water System (ESWS) in Table 17.4-1 of the US-APWR DCD, Revision 1 (page 17.4-37), the applicant did not include the flow meters located in ESWS pump motor cooling lines A and D (i.e., FT-2060, 2063). The evidence presented below suggests that FT-2060 and 2063 are risk-significant and should be included in D-RAP:

a) Based on Table 20.11-3 of the US-APWR Probabilistic Risk Assessment (PRA), MUAP-07030(R0), FT-2060 has a RAW of 5.4 for the internal events low-power and shutdown risk assessment, which exceeds the RAW criteria for risk significance (i.e., RAW > 2).

b) Based on Items 6 and 7 of ESWS in Table 17.4-1 of the US-APWR DCD, Revision 1 (page 17.4-36), the ESWS pump motor cooling lines A, B, C, and D are considered risk-significant.

Also, Item 8 of ESWS in Table 17.4-1 of the US-APWR DCD, Revision 1 (page 17.4-37), should include orifices ESS0003A, B, C, D as shown in Figure 6A.9-2 of the US-APWR PRA, MUAP-07030(R0) since these orifices would have RAW values similar to FT-2060, 2061, 2062, and 2063.

The staff requests that the applicant provide the basis for not including FT-2060, FT-2063 and orifices ESS0003A, B, C, D in Table 17.4-1 of the US-APWR DCD (include in the basis a discussion of the associated risk importance measures from the various PRA models, consideration of deterministic methods, e.g., defense-in-depth, consideration of seismic margins analysis, and the expert panel's deliberation for not including these SSCs in D-RAP).

ANSWER:

As mentioned in QUESTION, FT-2060, 2061 and 2062 (FM 2055A,B and C in Figure 6A.9-2 of the US-APWR PRA, MUAP-07030(R0)) have the RAWs which exceed the RAW criteria in Table 20.11-3 of the US-APWR PRA, MUAP-07030(R0). And FT-2064 (FM 2055D) is considered to be equal to the FT-2060, 2061 and 2062 (FM 2055A, B and C) in its failure behavior. Therefore, these 4 flow transmitters will be included as "ESW pump motor cooling line transmitters [FT-2060, 2061, 2062 and 2063]" under Item 8 of ESWS in Table 17.4-1 incorporating the discussion of expert panel (See the Attachment to this RAI response, page 17.4-37). This will be done by the next revision of the US-APWR DCD.

Based on Table 20.11-3 of the US-APWR Probabilistic Risk Assessment (PRA), MUAP-07030(R0), ESS0003A ,B and C in Figure 6A.9-2 of the US-APWR PRA, MUAP-07030(R0) have RAW of at least >5.4 for the internal events low-power and shutdown risk assessment, which exceed the RAW criteria for risk significance(i.e., RAW > 2). And ESS0003D is considered to be equal to the ESS0003A, B and C in their functions and failure behavior. Therefore, these 4 orifices will be included as "ESW pump motor cooling line orifices [TBD]" in Table 17.4-1 incorporating the discussion of expert panel (See the Attachment to this RAI response, page 17.4-37). This will be also done by the next revision of the US-APWR DCD.

<u>Impact on DCD</u>

List of risk significant SSCs will be revised as noted above considering the discussion of expert panel in the response to RAI 17.04-27.

<u>Impact on COLA</u>

There is no impact on COLA from this RAI.

<u>Impact on PRA</u>

There is no impact on PRA from this RAI.

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Docket No. No. 52-021

RAI NO.: **NO.150-1635 REVISION 1**

SRP SECTION: **17 04 - Reliability Assurance Program (RAP)**

APPLICATION SECTION: **17.4 Reliability Assurance Program**

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

QUESTION NO. : 17.04-28

Under Item 9 of Essential Service Water System (ESWS) in Table 17.4-1 of the US-APWR DCD, Revision 1 (page 17.4-37), the applicant states "Main Piping Orifices of Train B and D [FE2025 , FE2026]." The applicant's statement should refer to main piping orifices of Trains B and C since FE2026 is associated with Train C.

The staff requests that the applicant amend the text description, based on the above discussion, under Item 9 of ESWS in Table 17.4-1 of the US-APWR DCD (page 17.4-37).

ANSWER:

This is a typographic error.

As mentioned in QUESTION, the description under Item 9 of ESWS in Table 17.4-1 of the US-APWR DCD will be amended to "Main piping orifices of train B and C" (See the Attachment to this RAI response, page 17.4-37).

Impact on DCD

List of risk significant SSCs will be revised as noted above considering the discussion of expert panel in the response to RAI 17.04-28.

Impact on COLA

There is no impact on COLA from this RAI.

Impact on PRA

There is no impact on PRA from this RAI.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

2/6/2009

US-APWR Design Certification

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Docket No.52-021

RAI NO.: **NO.150-1635 REVISION 1**

SRP SECTION: **17.04 – Reliability Assurance Program (RAP)**

APPLICATION SECTION: **17.4 Reliability Assurance Program**

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

QUESTION NO. : 17-04-29

Alternate containment cooling by the containment fan cooler system is developed in Chapter 6A.14.1 of the US-APWR Probabilistic Risk Assessment (PRA), MUAP-07030(R0). The modeling of alternate containment cooling includes Human Reliability Analysis (HRA) event NCCOO02CCW ("Operator Fails to Line Up for Alternative Containment Cooling"). Actions associated with HRA event NCCOO02CCW is provided in Chapter 9 (page 9-28, 29) of the US-APWR PRA, MUAP-07030(R0), which include manipulating various valves (for example, opening the nitrogen supply valve 890, closing the non-safety chiller isolation valves CH-1 and CH-3, opening the CV recirculation unit inlet valves CH-5 through CH-8, and so on). Based on Table 18.6-1 of the US-APWR PRA, MUAP-07030(R0), HRA event NCCOO02CCW is considered risk-significant since it has a RAW of 6.4 and FV of 0.14 for large release frequency (LRF) internal events at power, which exceeds the RAW and FV criteria for risk significance (i.e., RAW > 2 or FV>0.005). Therefore, the SSCs that are manipulated/operated during execution of HRA event NCCOO02CCW could be considered risk-significant also.

The staff requests that the applicant provide the basis for not including in Table 17.4-1 of the US-APWR DCD the SSCs that are manipulated/operated during execution of HRA event NCCOO02CCW.

ANSWER:

Concerning the alternative containment cooling, it was considered that the failure caused by human error will be the dominant factor rather than hardware failures, therefore hardware failures were not modeled.

As mentioned in QUESTION, the SSCs that are manipulated/operated during execution of

NCCOO02CCW could be also risk-significant. Therefore the PRA will be revised to include the hardware failures of alternate containment cooling system to evaluate the importance of the hardware adequately.

Incorporating the results of the revised PRA model and the discussion of expert panel, list of risk significant SSCs, Table 17.4-1, will be updated by the next revision of the US-APWR DCD.

Impact on DCD

List of risk significant SSCs will be revised as noted above considering the result of revised PRA model and the discussion of expert panel in the response to RAI 17.04-29.

Impact on COLA

There is no impact on COLA from this RAI.

Impact on PRA

There is no impact on PRA from this RAI.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

2/6/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. No. 52-021

RAI NO.: **NO.150-1635 REVISION 1**

SRP SECTION: **17 04 - Reliability Assurance Program (RAP)**

APPLICATION SECTION: **17.4 Reliability Assurance Program**

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

QUESTION NO. : 17.04-31

The applicant did not include in Table 17.4-1 of the US-APWR DCD, Revision 1, charging injection filter CVC10 from Figure 6A.4-1 of the US-APWR Probabilistic Risk Assessment (PRA), MUAP-07030(R0). Since the valves in series with filter CVC10 are considered risk-significant in Table 17.4-1 of the US-APWR DCD, Revision 1, it would suggest that filter CVC10 may also be risk-significant (e.g., CVC10 fails due to plugging).

The staff requests that the applicant provide the basis for not including filter CVC10 in Table 17.4-1 of the US-APWR DCD (include in the basis a discussion of the associated risk importance measures from the various PRA models, consideration of deterministic methods, e.g., defense-in-depth, consideration of seismic margins analysis, and the expert panel's deliberation for not including filter CVC10 in D-RAP).

ANSWER:

As shown in Figure 6A.4-1 of the US-APWR Probabilistic Risk Assessment (PRA), MUAP-07030(R0), on the seal water line where CVC10 are located, there are valves CV02, CV06, CV09 and CV11 which correspond to VLV-164, VLV-166, VLV-170B and VLV-171B under Item 30 of charging injection system in Table 17.4-1 of the US-APWR DCD, Revision 1.

Based on the Table 18.2-2 of the US-APWR Probabilistic Risk Assessment (PRA), MUAP-07030(R0), RAWs of failures of these valves due to plugging are 2.4E+01. And as mentioned in QUESTION, it would be suggested that RAW for the failure of CVC10 due to plugging is also around 2.4E+01.

Nevertheless, according to the latest design plan for charging injection system, parallel to the original seal water line which contains CVC09, CVC10 and CVC11 (VLV-170B,KFT-003B and VLV-171B), there is a redundant line containing VLV-170A, KFT-003A and VLV-171A.

In case of plugging of one filter (e.g., KFT-003A), the line will be changed to the other (e.g., KFT-003B).

It can be said that the risk impact of this filter plugging is smaller than that of valves of upstream and downstream on this line. Consequently filter CVC 10 (KFT-003A) will not be included in Table 17.4-1.

Impact on DCD

There is no impact on DCD from this RAI.

Impact on COLA

There is no impact on COLA from this RAI.

Impact on PRA

There is no impact on PRA from this RAI.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

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US-APWR Design Certification

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Docket No.52-021

RAI NO.: **NO.150-1635 REVISION 1**

SRP SECTION: **17.04 – Reliability Assurance Program (RAP)**

APPLICATION SECTION: **17.4 Reliability Assurance Program**

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

QUESTION NO. : 17-04-32

The applicant did not include in Table 17.4-1 of the US-APWR DCD, Revision 1, charging injection motor-operated valves (MOV) 121D and 121E from Figure 6A.4-1 of the US-APWR Probabilistic Risk Assessment (PRA), MUAP-07030(R0). The downstream check valve 169 has a RAW of about 32 (for large release frequency of internal events at power, see Table 18.6-2 of the US-APWR PRA, MUAP-07030(R0)) and is considered risk-significant. Though the RAWs for MOV-121D, E are likely to be less than 32 since the two valves are in parallel, the staff was not able to confirm that the RAWs for MOV-121D, E are less than 2.

The staff requests that the applicant provide the basis for not including MOV-121D, E in Table 17.4-1 of the US-APWR DCD (include in the basis a discussion of the associated risk importance measures from the various PRA models, consideration of deterministic methods, e.g., defense-in-depth, consideration of seismic margins analysis, and the expert panel's deliberation for not including these MOVs in D-RAP).

ANSWER:

As pointed out in QUESTION, the failure of MOV 121D and 121E, especially the common cause failure of MOV 121D and 121E, will have significant risk impact.

Therefore, these 2 motor-operated valves will be included as "RWS refueling water auxiliary tank discharge line change valves [LCV-121D(E)]" in Table 17.4-1 incorporating the discussion of expert panel (See the Attachment to this RAI response, page 17.4-7). This will be done by the next revision of the US-APWR DCD.

Impact on DCD

List of risk significant SSCs will be revised as noted above considering the discussion of expert panel in the response to RAI 17.04-32.

Impact on COLA

There is no impact on COLA from this RAI.

Impact on PRA

There is no impact on PRA from this RAI.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

2/6/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. No. 52-021

RAI NO.: **NO.150-1635 REVISION 1**

SRP SECTION: **17 04 - Reliability Assurance Program (RAP)**

APPLICATION SECTION: **17.4 Reliability Assurance Program**

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

QUESTION NO. : 17.04-33

Under Item 3 of High Head Safety Injection System (HPI) in Table 17.4-1 of the US-APWR DCD, Revision 1 (page 17.4-22), the text description "Minimum flow line orifices 3 C(D)" does not match the listed component identification numbers (i.e., FE972(973, 974, 975)). The component identification numbers appear to be correct.

The staff requests that the applicant make consistent the text description and component identification numbers under Item 3 of HPI in Table 17.4-1 of the US-APWR DCD, Revision 1.

ANSWER:

As mentioned in QUESTION, RAWs for FE862C and FE862D, corresponding to FE974 and FE975 in Table 17.4-1 of the US-APWR DCD, Revision 1, exceed the RAW criteria (i.e. RAW>2) in the Table 22.6-7 of the US-APWR Probabilistic Risk Assessment (PRA) , MUAP-07030(R0).

And FE862A and FE862B (FE972 and FE973) are considered to be equal to FE862C and FE862D (FE974 and FE975) in their functions and failure behavior. Therefore, these 4 orifices will be included as "Minimum flow line orifices 3 A(B,C,D) [FE972(973,974,975)]" in Table 17.4-1 incorporating the discussion of expert panel (See the Attachment to this RAI response, page 17.4-22). This will be done by the next revision of the US-APWR DCD.

Impact on DCD

List of risk significant SSCs will be revised as noted above considering the discussion of expert panel in the response to RAI 17.04-33.

Impact on COLA

There is no impact on COLA from this RAI.

Impact on PRA

There is no impact on PRA from this RAI.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

2/6/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. No. 52-021

RAI NO.: **NO.150-1635 REVISION 1**

SRP SECTION: **17 04 - Reliability Assurance Program (RAP)**

APPLICATION SECTION: **17.4 Reliability Assurance Program**

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

QUESTION NO. : 17.04-34

Under Items 5 and 6 of High Head Safety Injection System (HPI) in Table 17.4-1 of the US-APWR DCD, Revision 1 (page 17.4-22), MOV-011A(B,C,D) and MOV-0011A(B,C,D) are listed. It is unclear if these are the same valves or different valves. Also under Items 5 and 11 it is unclear if these are the same valves since both item descriptions are "Containment isolation motor operated valves" but in item 5 the valves listed are MOV-011A(B,C,D) while in item 11 the valves listed are MOV-001A(B,C,D).

The staff requests that the applicant clarify the text descriptions and component identification numbers under Items 5, 6, and 11 of HPI in Table 17.4-1 of the US-APWR DCD, Revision 1.

ANSWER:

The description of Items 5 and 6 of High Head Safety Injection System (HPI) in Table 17.4-1 of the US-APWR DCD, Revision 1 are as follows:

#	Systems, Structures and Components (SSCs)
5	Containment isolation motor operated valves [MOV-011 A(B,C,D)]
6	RV injection line orifices (between VLV-012 A(B,C,D) and MOV-0011 A(B,C,D))

The description of Item 5 is deferent from that of Item 6. SSCs under Item 6 of HPI in Table 17.4-1 correspond to the RV injection line orifices and these are located between VLV-012 A(B,C,D) and MOV-0011 A(B,C,D). This orifice of A-train can be identified as "OR003A" in Figure6A.1-1 of the US-APWR Probabilistic Risk Assessment (PRA), MUAP-07030(R0).

As mentioned in QUESTION, the description of SSCs of Item 11 of HPI in Table 17.4-1 is inappropriate.

Therefore, the description of Item 11 is amended to "Safety injection pump suction isolation valves". incorporating the discussion of expert panel (See the Attachment to this RAI response, page 17.4-23). This will be done by the next revision of the US-APWR DCD.

Impact on DCD

List of risk significant SSCs will be revised as noted above considering the discussion of expert panel in the response to RAI 17.04-34.

Impact on COLA

There is no impact on COLA from this RAI.

Impact on PRA

There is no impact on PRA from this RAI.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

2/6/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: **NO.150-1635 REVISION 1**
SRP SECTION: **17.04 – Reliability Assurance Program (RAP)**
APPLICATION SECTION: **17.4 Reliability Assurance Program**
DATE OF RAI ISSUE: **1/9/2009**

QUESTION NO. : 17-04-35

In Table 17.4-1 ("Risk Significant SSCs") of the US-APWR DCD, Revision 1, B,C Emergency feedwater pump room fans are listed as risk-significant SSCs while in Revision 0, A,D-Emergency feedwater pump room fans were listed twice as risk-signficiant SSCs. It is unclear as to why the A,D-Emergency feedwater pump room fans were listed twice in Revision 0 while only the B,C-Emergency feedwater pump room fans are listed in Revision 1.

The staff requests that the applicant clarify why the A,D-Emergency feedwater pump room fans were listed twice in Revision 0 of the US-APWR DCD while only the B,C Emergency feedwater pump room fans are listed in Revision 1 of the US-APWR DCD.

ANSWER:

EFW pumps for A, D-train are turbine-driven and for B,C-train are motor-driven. And pump room cooling is required for motor-driven EFW pumps' operations. Accordingly, the description of "B,C-Emergency feedwater pump room fans" is appropriate.

The description of "A,D-Emergency feedwater pump room fans" in the US-APWR DCD, Revision 0 was not correct, therefore this was revised to "B,C-Emergency feedwater pump room fans" in Revision 1.

<u>Impact on DCD</u>

There is no impact on DCD from this RAI.

<u>Impact on COLA</u>

There is no impact on COLA from this RAI.

Impact on PRA

There is no impact on PRA from this RAI.

CONTENTS

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TABLES

ACRONYMS AND ABBREVIATIONS

AAC	alternative AC
ac	alternating current
CAP	corrective action program
CCF	common cause failure
CCW	component cooling water
CCWS	component cooling water system
CDF	core damage frequency
CFR	Code of Federal Regulations
COL	Combined License
COLA	Combined License Application
CS	containment spray
CSS	containment spray system
CVCS	chemical volume control system
DAS	diverse actuation system
dc	direct current
DCD	Design Control Document
D-RAP	design reliability assurance program
DVI	direct vessel injection
ECCS	emergency core cooling system
EFW	emergency feedwater
EFWP	emergency feedwater pit
EFWS	emergency feedwater system
EJ	engineering judge
EP	expert panel
EPS	emergency power source
ESF	engineered safety features
ESW	essential service water
ESWS	essential service water system
FIRE	FIRE event
FLOOD	FLOOD event
FSS	fire suppression systems
FV	Fussell Vesely
FVW	Fussell Vesely worth
HSIS	human-system interface system
HVAC	heating, ventilation, and air conditioning

ACRONYMS AND ABBREVIATIONS

I&C	instrumentation and control
ITAAC	inspection, test, analyses, and acceptance criteria
kV	kilovolt
LOCA	loss-of-coolant accident
LOOP	loss of offsite power
LPSD	low power and shut down operation
M/D	motor driven
MCC	motor control center
MFWS	main feedwater system
MHI	Mitsubishi Heavy Industries, Ltd.
MOV	motor operated valve
MSS	main steam supply system
NESH	Nuclear Energy Systems Headquarters
NRC	U.S. Nuclear Regulatory Commission
O-RAP	operational reliability assurance program
PAM	postaccident monitoring
PCMS	plant control and monitoring system
PRA	probabilistic risk assessment
QA	quality assurance
QAP	quality assurance program
QAPD	quality assurance program description
RAP	reliability assurance program
RAW	risk achievement worth
RCP	reactor coolant pump
RCS	reactor coolant system
RG	Regulatory Guide
RHR	residual heat removal
RHRS	residual heat removal system
RPS	reactor protection system
RRW	risk reduction worth
RTNSS	regulatory treatment of non-safety-related systems
RWAT	refueling water auxiliary tank
RWS	refueling water storage
RWSP	refueling water storage pit
RWSS	refueling water storage system

ACRONYMS AND ABBREVIATIONS (Continued)

SBO	station blackout
SDV	safety depressurization valve
SFP	spent fuel pit
SFPCS	spent fuel pit cooling and purification system
SG	steam generator
SGTR	steam generator tube rupture
SIS	safety injection system
SRP	Standard Review Plan
SSC	structure, system, and component
T/D	turbine driven
VCT	volume control tank
VWS	chilled water system
WMS	waste management system

17.0 QUALITY ASSURANCE AND RELIABILITY ASSURANCE

Quality Assurance Program Description (QAPD) as described in Sections 17.1, 17.2, 17.3 and 17.5 of this chapter of DCD is applicable for Quality Assurance (QA) during design certification.

17.1 Quality Assurance During the Design Phase

For quality assurance during the design certification phase, see Section 17.5.

The Combined License (COL) Applicant is responsible for development a Quality Assurance Program applicable to its activities during design other than the Design Certification.

17.2 Quality Assurance During the Construction and Operations Phase

The COL Applicant is responsible for development of the construction and operational phase Quality Assurance Program.

17.3 Quality Assurance Program

The General Manager of Nuclear Energy Systems Headquarters (NESH) is responsible for the Design Certification Activities of US-APWR. The major design activities are performed by the Nuclear Energy Systems Engineering Center engineers. QA Program controls governing the activities are specified in QAPD (PQD-HD-19005 Rev.1) (Ref 17.5.5-4).

Subcontractors of the Nuclear Energy Systems Engineering Center performing design activities in support of the US-APWR are also required to follow QAPD (PQD-HD-19005 Rev.1).

For the quality assurance program description during the design certification phase, see Section 17.5.

The COL applicant is responsible for development a Quality Assurance Program Description during design other than the Design Certification, construction and operation phase.

17.4 Reliability Assurance Program

This section presents the US-APWR reliability assurance program (RAP).

17.4.1 New Section 17.4 in the Standard Review Plan

As noted in Item E of SECY 95-132 (Ref. 17.4-1), an applicant for design certification should establish the scope, purpose, objective, and essential elements of an effective D-RAP and would implement those portions of the D-RAP that apply to design certification. A COL Applicant is responsible for augmenting and completing the remainder of the D-RAP to include any site-specific design information and identify the risk-significant SSCs. Once the site-specific D-RAP is established and the risk-significant SSCs are identified, the procurement, fabrication, construction, and preoperational testing can be implemented in accordance with the COL holder's D-RAP or other programs and would be verified using the inspections, test, analyses and acceptance criteria (ITAAC) process.

17.4.2 Introduction

The purposes of the US-APWR RAP are to provide reasonable assurance that: 1) the US-APWR is designed, constructed, and operated in a manner that is consistent with the assumptions and risk insights for the risk-significant SSCs, 2) the risk-significant SSCs do not degrade to an unacceptable level during plant operations, 3) the frequency of transients that challenge SSCs is minimized, and 4) the SSCs function reliably when challenged. An additional goal is to facilitate communication between the probabilistic risk assessment (PRA), the design, and the ultimate COL activity. SPLA 1474-012

The PRA evaluates the US-APWR design response to a spectrum of initiating events to ensure that plant damage has a very low probability and that risk to the public is minimized. Risk significant SSCs for the US-APWR design control document (DCD) are identified and made available to the design organization.

The US-APWR D-RAP process is implemented in several phases. Phase I, the Design Certification phase, collects system information and develops a system model. This system information and model is used as input to the design phase PRA, an operating experience review, and a review for external events. The goal of the RAP during this stage is to ensure that the reactor design meets the purposes above, through the design, procurement, fabrication, construction and preoperational testing activities and programs. The results of each of these activities are provided to an expert panel (EP) which identifies risk significant items using probabilistic, deterministic, and other methods for inclusion in the program. Phase II, the site-specific phase, introduces the plant's site-specific information to the D-RAP process. During Phase II, the site-specific SSCs are combined with the US-APWR design SSCs into a list for the specific plant. Phase III, the last phase of the D-RAP, implements the procurement, fabrication, construction, and preoperational testing. The site-specific list of SSCs is also provided as an input to the operational phase of RAP (O-RAP) which addresses the specific plant operation and maintenance activities. The designer, MHI, is responsible for Phase I of the D-RAP. The objective during this stage is to ensure that the reliability for the SSCs within the scope of the RAP is maintained during plant operations. Phases II and III of the D-RAP and the O-RAP are the responsibility of the COL Applicant. The COL Applicant will specify the policy and implement procedures to address the specific plant operation and maintenance activities associated with the risk-significant SSCs identified by the D-RAP.

17.4.3 Scope

The US-APWR D-RAP identifies risk-significant SSCs and provides risk insights and reliability assumptions for aspects of plant operation, maintenance, and performance monitoring to be addressed to ensure safe, reliable plant operation or mitigate plant transients or other events that could present a risk to the public. The risk-significant SSCs are identified using PRA, deterministic, or other methods of analysis, including industry experience, and EPs.

17.4.4 Quality Controls

a. Organization

The MHI is responsible for Phase I of the D-RAP.

General Manager, US-APWR project: The General Manager, US-APWR project is overall responsible for the establishment of and implementation of the US-APWR D-RAP. In this regard, the General Manager or his designated representative is responsible to assure all affected organizations are aware of the D-RAP, its purpose, and the requirements herein.

General Manager, Reactor and Plant Safety: The General Manager, Reactor and Plant Safety, is responsible for the use of the PRA results and risk insights for the EP, and for the conduct and coordination of the EP. The Reactor and Plant Safety organization includes the risk and reliability organization.

General Manager, QA: The General Manager, QA is responsible to assure proper implementation of QA program elements. This includes design control, procedures and instructions, records, corrective actions and audits pertaining to the D-RAP.

General Managers, Design Engineering: The General Managers, Design Engineering, are responsible to implement this D-RAP and specifically to assure that the US-APWR is designed consistent with the reliability assumptions and insights of the PRA for risk-significant SSCs.

The risk and reliability organization is responsible to ask the related design engineering sections to review key assumptions and to feed back their comments to ensure key assumptions are realistic and achievable.

The risk and reliability organization is responsible to provide the RAP related inputs in the design process by participating in the design change process.

The risk and reliability organization is also responsible to involve in the design review.

b. Design Control

The list of risk-significant SSCs for the D-RAP and its key assumptions shall be maintained by the risk and reliability organization. The list and changes thereof shall be approved by the EP and be provided to design engineering and QA staff working on the US-APWR project.

The risk and reliability organization shall ensure that the design engineers are provided the list of risk-significant SSCs for the D-RAP and its key assumption. The design engineers shall take into account the list of the risk-significant SSCs for the D-RAP and its key assumptions in their design activities and give some feedback to the risk and reliability organization in order to ensure that the key assumptions are realistic and achievable, if necessary.

c. Procedures and Instructions

General Manager, US-APWR project or his designated representative has prepared the procedures and instructions used in implementation of the D-RAP. General Manager, US-APWR project is responsible for development and verification of implementation of the D-RAP, and for assuring all affected MHI organizations are aware of the D-RAP.

d. Records

Records related to the D-RAP which are required to be maintained include the following:

- List of Risk-Significant SSCs
- EP meeting minutes/summaries
- Other quality assurance program records in accordance with the US-APWR QAPD for design certification.

e. Corrective action

Deficiencies identified where design documents address SSC reliability assumptions which are not compatible with the reliability assumptions of the PRA, or are not achievable or are unrealistic shall be entered into the corrective action program (CAP) system and addressed appropriately. The CAP utilized to support the QAPD can be used to implement the corrective actions related to the RAP.

f. Audit

Audit plans shall include for consideration, sampling the effectiveness of implementation of RAP implementation procedure. Audits shall consider several key aspects of the RAP including the identification of risk-significant SSCs, whether design and procurement information is consistent with the risk insights from the PRA, and whether assumed equipment reliability is determined to be practicable or achievable.

17.4.5 Integration into Existing Operational Programs

The US-APWR D-RAP is a source to other administrative and operational programs. Certain risk-significant SSCs identified in the D-RAP are included in existing operational programs such as the technical specifications surveillance requirements and provide assurance that the reliability values assumed in the PRA will be maintained throughout the plant life. The O-RAP implements the measures that yield the significant improvements in the PRA through the plant's existing programs for maintenance or QA. Implementation of the Maintenance Rule requirements contained in 10CFR50.65 (Ref. 17.4-2) is an example of how the plant could address the enhanced treatment of certain SSCs in the O-RAP. Per SECY 95-132, the COL Applicant may meet most of the objectives of the O-RAP via existing programs such as maintenance rule, in-service testing, and QA. The COL Applicant must address non-safety risk significant SSCs.

17.4.6 Operating Experience

Consideration and use of operating experience is vital to the overall objective of the D-RAP. Operating experience is considered along with various PRA analytical and importance measures when developing a comprehensive risk analysis. The EP considers component operating history and industry operating experience when it can be applied to assessing risk significance. For example, operating experience indicates that motor driven and turbine driven pumps may have different reliability.

The review of operating experience investigates situations where previous failures of components in similar design applications have led to functional failures of SSCs. The review of operating experiences is not limited to hardware failure but also extends to situations where human performance led to functional failures of SSCs of a similar system design. As an example, the US-APWR design improves reliability and eliminates required operator actions to switch over from injection to recirculation typical in conventional PWRs.

17.4.7 D-RAP

As discussed in Section 17.4.2, Phase I of the D-RAP includes the initial identification of SSCs to be included in the program, implementation of the aspects applicable to design efforts, and definition of the scope, requirements, and implementation options to be included in the later phases.

17.4.7.1 SSCs Identification

During the US-APWR design phase, risk significant SSCs are identified for inclusion in the scope of the D-RAP. A list of risk significant SSCs is developed and controlled as a design input for consideration during the design phase. The list of risk significant SSCs is initially based on the results of the PRA and the EP. For further discussion on PRA, refer to Chapter 19, Section 19.1, of this DCD. The PRA is used to identify risk significant SSCs based on risk achievement worth (RAW) and Fussell-Vesely Worth (FVW). For further information, see Chapter 19, Section 19.1.7.4 of this DCD. The list of risk significant SSCs identified during the design phase is updated when the plant-specific PRA is developed. In addition to the PRA input, information from operating experience of Japanese design plants, as well as US industry experience is considered for identification of risk significant SSCs. A third source in the D-RAP process for identifying risk significant SSCs is the use of an EP consisting of representatives from Design Engineering, PRA, as well as other highly qualified individuals with operations, and maintenance experience who are independent of the PRA Section. The EP also SPLA 1474-006 reviews the categorization of SSCs determined to be not risk significant (NRS) from quantified PRA results (e.g., technical adequacy of the basis used in the categorization, review of defense-in-depth implications, review of safety margin implications). As part of the D-RAP process, the PRA analytical results, operating experience, and an EP process are combined to develop a comprehensive list of risk significant SSCs.

17.4.7.2 Expert Panel

An EP, consisting of highly qualified representatives of Reliability and PRA Engineering, as well as representatives independent of the PRA process from Design and Plant Engineering at least one person with design engineering experience, at least one person with PRA experience, at least one person with operations and maintenance experience, and at least one person with quality assurance experience, is responsible for the final selection of the SSCs included in the D RAP. Industry operating experience when it can be applied to assessing risk significance, and engineering judgment are employed in considering the addition of SSCs to the D-RAP. Industry operating experience and use of the Expert Panel are used as the part of deterministic approach and other processes, and engineering judgment are employed in considering the addition of SSCs to the D-RAP. Each voting member of the RAP EP should have the level of education and experience defined by the RAP. The level of education and experience of voting member of the RAP EP is defined in the Expert Panel Implementing Procedure for US-APWR Reliability Assurance Program as follows: SPLA 1474-007 SPLA 1474-008 SPLA 1474-014

- A person who has graduated science and technology university or who has identical educational background, and who has more than 10 years of experience in the specific area of Nuclear Power Plant, such as design, or has identical experience.

or

- A person who has graduated high school or who has identical educational background, and who has more than 15 years of experience in the specific area of Nuclear Power Plant, such as design, or has identical experience.

17.4.7.3 Phase I D-RAP Implementation and SSCs included

The implementation of the Phase I D-RAP is the responsibility of MHI as it applies to the reactor design process. The SSCs included in this phase are listed in Table 17.4-1.

Table 17.4-1 Risk significant SSCs (sheet 1 of 34)

#	Systems, Structures and Components (SSCs)	Rationale[(1)]	Insights and Assumptions
1	Accumulator injection system		
1	Discharge line secondary isolation check valves train A through D [VLV-102A (B,C,D)]	RAW/CCF	The accumulator provides safety injection function for refill and re-flooding of the reactor vessel following a loss of coolant accident (LOCA). Also provides negative reactivity to shutdown the reactor. Single failure of any SSCs listed here has potential to cause failure of its dedicated train to inject coolant to RCS.
2	Boundary check valves train A through D (Discharge line) [VLV-103A (B,C,D)]	RAW/CCF	
3	Discharge line isolation motor operated valves train A through D [VLV-101A (B,C,D)]	RAW	
4	Discharge line orifices train A through D [R006A (B,C,D)]	RAW	
5	Piping of discharge lines train A through D [TBD]	RAW	
6	A~D-Accumulators [SIS-CTK-001A (B,C,D)]	EJ	

Table 17.4-1 Risk significant SSCs (sheet 2 of 34)

#	Systems, Structures and Components (SSCs)	Rationale(1)	Insights and Assumptions
2	Charging injection system		
1	Charging line air operated valves [AOV-146] [FCV-138] [AOV-159]	RAW/LPSD	The chemical volume control system (CVCS) maintains appropriate volume and quality of reactor coolant for the primary reactor coolant system, adjusts boron concentration for the chemical shim control, and supplies seal water to the reactor coolant pump seals, and disposes borated water discharged from the primary reactor coolant system. RCP seal water injection provided by the CVCS is an essential function to prevent RCP seal LOCA under loss of CCW conditions. When loss of CCW occurs, either the fire suppression system or the non-essential chilled water system is connected to the charging pump cooling line. Thus, the RCP seal water injection is maintained under loss of CCW conditions. Since CVCS is not completely separated in trains, large external leak from SSCs that result in loss of inventory is assumed to result in degradation or failure of the system. Accordingly, SSCs that has the potential of large leak are risk significant. SSCs that have potential to cause common cause failures among multiple trains are also important. Such common cause failure results in loss of redundant SSCs.
2	RCP seal cooling injection line air operated valves [FCV-140] [AOV-165]	RAW	
3	Auxiliary spray injection line air operated valve [AOV-155]	RAW/LPSD	
4	A,B-Charging pumps [CVS-RPP-001A (B)]	RAW/CCF/LPSD	
5	Volume control tank discharge line check valve [VLV-125]	RAW/LPSD	
6	Volume control tank discharge line motor operated valves [LCV-121B] [LCV-121C]	RAW/LPSD	
7	RWS refueling water auxiliary tank discharge line change valves [LCV-121D(E)]	RAW(L2)	
~~7~~8	RWS refueling water auxiliary tank discharge line check valve [VLV-595]	RAW/LPSD	
~~8~~9	RWS refueling water auxiliary tank discharge line manual valve [VLV-591]	RAW/LPSD	

RAI 17.04-32 (This will be revised incorporating the discussion of expert panel.)

Table 17.4-1 Risk significant SSCs (sheet 3 of 34)

#	Systems, Structures and Components (SSCs)	Rationale(1)	Insights and Assumptions
~~9~~10	Charging pump minimum flow line check valves [VLV-129A (B)]	RAW/LPSD	During low power and shutdown operation, CVCS provides RCS make up function. On low VCT level, suction is switched from the VCT to the refueling water auxiliary tank, which is supplied by the refueling water storage pit. Low-pressure letdown line isolation valves are automatically closed and the CVCS is isolated from the RHRS with receiving the RCS loop low-level signal to prevent loss of RCS inventory at mid-loop operation. When these valves are not closed, loss of a RCS inventory is prevented by manually closing the air-operated valve at the downstream of these valves.
~~10~~11	Charging pump discharge line check valves [VLV-131A (B)]	RAW/LPSD	
~~11~~12	Charging line containment isolation check valve [VLV-153]	RAW/LPSD	
~~12~~13	Charging line isolation check valve [VLV-160]	RAW/LPSD	
~~13~~14	Charging line boundary isolation check valve [VLV-161]	RAW/LPSD	
~~14~~15	RCP seal water injection line boundary isolation check valves [VLV-182A (B,C,D)]	RAW	
~~15~~16	RCP seal water injection line secondary isolation check valves [VLV-181A (B,C,D)]	RAW	
~~16~~17	RCP seal water injection line third isolation check valves [VLV-179A (B,C,D)]	RAW	
~~17~~18	Charging line containment isolation motor operated valve [MOV-152]	RAW/LPSD	

RAI 17.04-32
(This will be revised incorporating the discussion of expert panel.)

Table 17.4-1 Risk significant SSCs (sheet 4 of 34)

#	Systems, Structures and Components (SSCs)	Rationale[1]	Insights and Assumptions
18	Charging line containment isolation motor operated valve [MOV-151]	RAW/LPSD	The "Insights and Assumptions" for these SSCs are described on the previous page.
19	RCP seal water injection line containment isolation motor operated valves [MOV-178A (B,C,D)]	RAW	
20	Charging line orifice [FE-138]	RAW/LPSD	
21	Charging flow control orifice [TBD]	RAW/LPSD	
22	RCP seal water injection line orifices [FE-160A (B,C,D)]	RAW	
23	Regenerative heat exchanger [CHX-001]	RAW/LPSD	
24	Charging pump minimum flow line manual valves [VLV-130A (B)]	RAW/LPSD	
25	Charging pump discharge line manual valves [VLV-132A (B)]	RAW/LPSD	
26	Charging pump discharge line cross tie-line manual valve [VLV-133]	RAW/LPSD	
27	Charging pump suction line manual valves [VLV-126A (B)]	RAW/LPSD	

Table 17.4-1 Risk significant SSCs (sheet 5 of 34)

#	Systems, Structures and Components (SSCs)	Rationale[1]	Insights and Assumptions
28	Charging line manual valves [VLV-145] [VLV-147]	RAW/LPSD	The “Insights and Assumptions” for these SSCs are described on the previous page.
29	Charging line by-pass line manual valve [VLV-144]	RAW/LPSD	
30	RCP seal water injection line manual valves [VLV-164] [VLV-166] [VLV-168] [VLV-170B] [VLV-171B] [VLV-173]	RAW	
31	RCP seal water injection by-pass line manual valve [VLV-163]	RAW	
32	RCP seal water injection line manual valves [VLV-180A (B,C,D)]	RAW	
33	RCP seal water injection line needle valves [VLV-177A (B,C,D)]	RAW	
34	Low-pressure letdown line air operated valve [HCV-102]	LPSD	

Table 17.4-1 Risk significant SSCs (sheet 6 of 34)

<table>
<tr><th>#</th><th>Systems, Structures and Components (SSCs)</th><th>Rationale[(1)]</th><th>Insights and Assumptions</th></tr>
<tr><td>3</td><td colspan="3">Component cooling water system (CCWS)</td></tr>
<tr><td>1</td><td>CCW pump discharge line check valves
[VLV-016A (B,C,D))</td><td>RAW/CCF/LPSD</td><td rowspan="7">The component cooling water system (CCWS) transfer heat from plant safety-related components to the essential service water system (ESWS). This system supports various safety and non-safety mitigation systems. Accordingly, reliability of CCWS emergency feedwater system (EFWS) has significant impact on risk.

CCWS has four trains, each having a component cooling water pump and a component cooling water heat exchanger. Two trains compose a subsystem, which shares a supply / return header and a surge tank.

SSCs that have either of the following characteristics are risk significant.
- SSCs that have potential to cause common cause failures among multiple trains. Common cause failure of such system will result in loss of multiple trains.
- SSCs that have potential to cause large external leak are risk significant. Since the two trains that compose a subsystem are not physically isolated, large external leak from SSCs that result in loss of inventory is assumed to result in degradation or failure of two trains.</td></tr>
<tr><td>2</td><td>A~D-Component cooling water pumps
[NCS-RPP-001A (B,C,D)]</td><td>FV/RAW/CCF
/LPSD</td></tr>
<tr><td>3</td><td>A~D-Component cooling water heat exchangers
[NCS-RHX-001A (B,C,D)]</td><td>RAW/CCF/LPSD</td></tr>
<tr><td>4</td><td>CCW pump discharge cross tie-line motor operated valves
[MOV-020A (B,C,D)]</td><td>RAW/CCF/LPSD</td></tr>
<tr><td>5</td><td>CCW pump suction line cross tie-line motor operated valves
[MOV-007A (B,C,D)]</td><td>RAW/CCF/LPSD</td></tr>
<tr><td>6</td><td>SSCs that compose CCW boundary</td><td>RAW/EJ/LPSD</td></tr>
<tr><td>7</td><td>CS/RHR heat exchanger discharge line motor operated valves
[MOV-145A (B,C,D)]</td><td>FV/RAW/CCF
/LPSD</td></tr>
</table>

Table 17.4-1 Risk significant SSCs (sheet 7 of 34)

#	Systems, Structures and Components (SSCs)	Rationale[1]	Insights and Assumptions
8	Charging injection Pump Cooling Line Check Valves [TBD]	RAW/CCF/LPSD	These valves are used (opened) to provide alternative CCW from the fire suppression system or the non-essential chilled water system to the charging pump cooling line under loss of CCW events. These are important SSCs at loss of CCW events to prevent RCP seal LOCA.
9	Charging injection pump cooling discharge line motor operated valves [TBD]	RAW/CCF/LPSD	
10	CCWS - fire suppression system boundary motor operated valves [TBD]	RAW/CCF/LPSD	
11	CCWS - RWSP line boundary check valves [VLV-065A (B)]	RAW/LPSD	
12	CCWS - RWSP line boundary manual valves [VLV-066A (B)]	RAW/LPSD	

Table 17.4-1 Risk significant SSCs (sheet 8 of 34)

#	Systems, Structures and Components (SSCs)	Rationale(1)	Insights and Assumptions
4	Containment system		
1	Containment vessel [TBD]	EJ	The containment vessel is designed to completely enclose the reactor and reactor coolant system and to ensure that essentially no leakage of radioactive materials to the environment would result even if a major failure of the reactor coolant system were to occur. Hydrogen ignition system are provided for protection against possible detonation following a core damage accident to meet the requirement of 10CFR50.34(f) and 10CFR50.44(c).
2	Hydrogen ignition system [TBD]	EJ	
5	Containment isolation system		
1	Instrument air system check valve [VLV-003]	RAW(L2)	In the case of core damage accident, the containment isolation valve is important to prevent radionuclide releases to the environment.

Table 17.4-1 Risk significant SSCs (sheet 9 of 34)

#	Systems, Structures and Components (SSCs)	Rationale(1)	Insights and Assumptions
6	Emergency feedwater system (EFWS)		
1	EFW pit discharge line check valves [VLV-008A (B)]	RAW/CCF/LPSD	The emergency feedwater system (EFWS) supplies feedwater to the steam generators in order to remove reactor decay heat and RCS residual. This system is required after all initiating events exceeding large and medium LOCA. Accordingly, reliability of EFW system has significant impact on risk. Two trains share one emergency feedwater pit, which has 50% capacity to perform cold shutdown. Large leak from SSCs or failure that result in degradation of water supply from EFW pit will lead to lack of EFW. In this case manual action to supply feedwater from Secondary Demineralizer Water Tank is required. SSCs that have either of the following characteristics are risk significant. - SSCs that have potential to cause common cause failures among multiple trains. Common cause failure of such system will result in loss of multiple trains. - SSCs that have potential to cause large leak or failure that result in degradation of water supply from EFW pit will lead are risk important. If such failure occurs, manual action to supply feedwater from secondary demineralizer water tank will be required.
2	A(D)-emergency feedwater pump actuation valves [EFS-MOV-103A(D)]	RAW/LPSD	
3	B,C-Emergency feedwater pumps [EFS-RPP-001B (C)]	RAW/CCF/LPSD	
4	A,D-Emergency feedwater pumps [EFS-RPP-001A (D)]	FV/RAW/CCF/LPSD	
5	Feedwater line check valves [VLV-018A (B,C,D)]	RAW/CCF/LPSD	
6	EFW pump discharge line check valves [VLV-012A (B,C,D)]	RAW/CCF/LPSD	
7	Minimum/Full flow line check valves [VLV-020A (B,C,D)] [VLV-022A (B,C,D)]	RAW/LPSD	
8	Minimum/Full flow line manual valves [VLV-021A (B,C,D)] [VLV-023A (B,C,D)]	RAW/LPSD	
9	A~D-emergency feedwater control valves [EFS-MOV-017A (B,C,D))	RAW/LPSD	
10	A~D-emergency feedwater isolation valves [EFS-MOV-019A (B,C,D)]	RAW	

Table 17.4-1 Risk significant SSCs (sheet 10 of 34)

#	Systems, Structures and Components (SSCs)	Rationale[1]	Insights and Assumptions
11	A~D-emergency feedwater line orifices [FE3716,3726,3736,3746]	RAW (FLOOD)	The "Insights and Assumptions" for these SSCs are described on the previous page.
12	A~D-emergency feedwater line tie-line valves [EFS-MOV-014A (B,C,D)]	RAW/CCF(FLOOD)	
13	EFW pit discharge line piping [TBD]	RAW/LPSD	
14	EFW pit discharge line tie-line piping [TBD]	RAW(FLOOD)	
15	A~D-emergency feedwater line A(B,C,D) piping [TBD]	RAW(FLOOD)	
16	T/D pump steam supply line piping [TBD]	RAW/LPSD	
17	Minimum/Full flow line piping [TBD]	RAW/LPSD	
18	A,B-Emergency feedwater pits [EFS-RPT-001A(B)]	RAW/LPSD	
19	Minimum/Full flow line manual valves [VLV-026A (B)]	RAW/LPSD	
20	EFW pump suction line manual valves [VLV-009A (B,C,D)]	RAW/LPSD	
21	EFW pump discharge line manual valves [VLV-013A (B,C,D)]	RAW/LPSD	

Table 17.4-1 Risk significant SSCs (sheet 11 of 34)

#	Systems, Structures and Components (SSCs)	Rationale[1]	Insights and Assumptions
22	EFW pit discharge line manual valves [VLV-007A (B)]	RAW/LPSD	The "Insights and Assumptions" for these SSCs are described on the previous page.
23	Secondary demineralizer water tank discharge line manual valves [VLV-006A (B)]	RAW/LPSD	
24	Secondary demineralizer water tank discharge line check valve [VLV-005]	RAW(FLOOD)	
25	EFW pit water level transmitter 1(2, 3, 4) [EFS-LT-3760, 3761, 3770, 3771]	—	

RAI 17.04-21 (This will be revised incorporating the results of the revised PRA model and the discussion of expert panel.)

Table 17.4-1 Risk significant SSCs (sheet 12 of 34)

#	Systems, Structures and Components (SSCs)	Rationale[1]	Insights and Assumptions
7	Emergency power source (EPS)		
1	480V AC motor control center (MCC) buses [TBD]	RAW/LPSD	The EPS consists of four separate trains. Each safety train consists of one 6.9kV AC medium voltage bus and 480V AC low voltage buses (Load Centers, Motor Control Centers). Each AC medium voltage bus connects to class 1E gas turbine generator. This system supports various safety mitigation systems and therefore, reliability of the EPS system has significant impact on risk. Since the EPS consists of four separate trains, single failure in trains not significantly impact risk. However, failure of multiple trains is have significant impact on risk. Accordingly, SSCs that have potential to cause common cause failures among multiple trains are risk significant
2	480V AC load center buses [TBD]		
3	6.9kV buses [TBD]	RAW/EJ/LPSD	
4	125V DC buses train A and D [TBD]	RAW/LPSD	
5	125V DC buses train B and C [TBD]	RAW(L2)	
6	120V buses train A-D [TBD]	RAW(L2/ FIRE)	
7	Swing MCC incomer circuit breakers [TBD]	RAW/CCF/LPSD	
8	Batteries [TBD]	RAW/CCF/LPSD	
9	6.9kV AC bus incomer circuit breakers [TBD]	FV/RAW/CCF/LPSD	
10	Gas turbine discharge circuit breakers [TBD]	RAW/CCF/LPSD FV/CCF(FIRE)	

Table 17.4-1 Risk significant SSCs (sheet 13 of 34)

#	Systems, Structures and Components (SSCs)	Rationale(1)	Insights and Assumptions
11	Circuit breakers between 6.9kV bus and 6.9kV/480V safety power transformers [TBD]	RAW/CCF/LPSD	The "Insights and Assumptions" for these SSCs are described on the previous page.
12	MCC bus incomer circuit breakers [TBD]	RAW/CCF/LPSD	
13	Circuit breakers between 125V DC bus and Inverter [TBD]	RAW/CCF/LPSD	
14	Class 1E gas turbine generators [TBD]	FV/RAW/CCF /LPSD)	
15	Gas turbines generator sequencers [TBD]	RAW/CCF/LPSD FV(FIRE)	
16	Inverters [TBD]	RAW/CCF/LPSD	
17	Main transformers [TBD]	RAW(L2)	
18	6.9kV/480V safety power transformers [TBD]	RAW/LPSD	

Table 17.4-1 Risk significant SSCs (sheet 14 of 34)

#	Systems, Structures and Components (SSCs)	Rationale(1)	Insights and Assumptions
8	Alternative AC power sources (Permanent bus)		
1	Non-class 1E gas turbine generators [TBD]	FV/RAW/CCF /LPSD	Two non-safety buses called “Permanent bus”, which is connected to Alternative AC (AAC), which consists of non-class 1E gas turbine generators respectively. Each non-class 1E gas turbine generators is manually connected to two safety medium voltage buses via selector circuit under the occurrence of loss of safety AC power. The AAC is a countermeasure against station blackout events. SSCs that have potential to cause failures that degrade the availability to supply AAC power to safety medium voltage are risk significant. Systems for the mitigation of core damage accident are connected to permanent bus.
2	480V permanent buses [TBD]	RAW(L2)	
3	6.9kV permanent buses [TBD]	RAW(L2)	
4	Circuit breakers between 6.9kV bus and 6.9kV/480V power transformer [TBD]	RAW(L2)	
5	Batteries [TBD]	RAW/CFF/LPSD	
6	Gas turbine generator discharge circuit breakers [TBD]	RAW/CCF/LPSD	
7	AAC selector circuit breakers [TBD]	RAW/CCF/LPSD	
8	Circuit breakers between 125V DC bus and Inverter [TBD]	RAW/CCF/LPSD	
9	Inverters [TBD]	RAW/CCF/LPSD	
10	Gas turbine generator sequencers	RAW/CCF/LPSD	
11	6.9kV/480V power transformers [TBD]	RAW/LPSD	

Table 17.4-1 Risk significant SSCs (sheet 15 of 34)

#	Systems, Structures and Components (SSCs)	Rationale(1)	Insights and Assumptions
9	Non-essential chilled water system		
1	Non-essential chilled water system - CCWS boundary motor operated valves [TBD]	RAW/LPSD	In the case of loss of component cooling water events, non-essential chilled water system or fire suppression system provides alternative component cooling water to charging pumps in order maintain RCP seal water injection. These SSCs are risk significant because large external leak from these valves result in loss of alternative component cooling water from both non-essential chilled water system and fire suppression system. On the other hand, failure of other SSCs of this system affects only the non-essential chilled water system itself.

Table 17.4-1 Risk significant SSCs (sheet 16 of 34)

#	Systems, Structures and Components (SSCs)	Rationale[1]	Insights and Assumptions
10	Fire suppression systems (FSS)		
1	FSS pump discharge motor operated valve [TBD]	FV(L2)/RAW(L2)	In the case of core damage accident, Fire Suppression Systems (FSS) injects water from Raw Water Tank into the reactor cavity via the direct injection line by the fire water pumps. The containment spray system and/or safety injection system perform the reactor cavity flooding through the drain line at loop compartment to prevent core-concrete interaction when the reactor vessel is failed. The Fire suppression system performs as alternative function for the reactor cavity flooding. In the case of loss of component cooling water events, fire suppression system or non-essential chilled water system provides alternative component cooling water to charging pumps in order maintain RCP seal water injection.
2	FSS pump discharge flow meter [TBD]	RAW(L2)	
3	Reactor cavity injection line orifice [TBD]	RAW(L2)	
4	FSS piping (from tank to tie line piping) [TBD]	RAW(L2)	
5	Raw water tank [TBD]	RAW(L2)	
6	FSS pump discharge manual valve [TBD]	RAW(L2)	
7	FSS - CCWS Boundary motor operated valves [TBD]	RAW/LPSD	Large external leak from these valves result in loss of alternative component cooling water from both non-essential chilled water system and fire suppression system. On the other hand, external leak from other SSCs degrade the fire suppression system but the non-essential chilled water system is still available for alternative component cooling. Therefore these valves are risk significant SSCs in preventing core damage.

Table 17.4-1 Risk significant SSCs (sheet 17 of 34)

#	Systems, Structures and Components (SSCs)	Rationale[(1)]	Insights and Assumptions
11	High head safety injection system		
1	Safety injection pump discharge check valves [VLV-004A (B,C,D)]	FV/RAW/CCF/LPSD	In the case of LOCA, high head safety injection system injects coolant from refueling water storage pit (RWSP) into the reactor vessel via the Direct Vessel Injection (DVI) line by the safety injection pumps. This system is also essential for bleed and feed operation. Since this system consists of four independent trains, failure of one train does not have significant impact on risk. However, failures of SSCs that impact multiple trains are risk significant. SSCs that have either of the following characteristics are risk significant. - SSCs that have potential to cause common cause failures among multiple trains. Common cause failure of such system will result in loss of multiple trains. - SSCs that have potential to cause loss of RWSP inventory out side the containment due to large external leaks. Loss of RWSP inventory impacts not only all four trains of high head safety injection system but also other systems that use RWSP as water source.
2	Safety injection pump outlet orifices 1A(B,C,D) [FE962(963,964,965)]	RAW(FLOOD)	
3	Minimum flow line orifices 3 A(B,C, ~~(~~D) [FE972(973,974,975)]	RAW(FLOOD)	
4	Containment isolation check valves [VLV-010A (B,C,D)]	RAW/CCF/LPSD	
5	Containment isolation motor operated valves [MOV-011 A(B,C,D)]	RAW(FLOOD) FV(FLOOD)	
6	RV injection line orifices (between VLV-012 A(B,C,D) and MOV-0011 A(B,C,D))	RAW(FLOOD)	
7	Injection line secondary isolation check valves [VLV-012A (B,C,D)]	RAW/CCF/LPSD	
8	Injection line boundary check valves [VLV-013A (B,C,D)]	RAW/CCF/LPSD	
9	A~D-Safety injection pumps [SIS-RPP-001A (B,C,D)]	FV/RAW/CCF/LPSD	
10	Containment isolation motor operated valves [MOV-009A (B,C,D)]	RAW FV(FLOOD)	

RAI 17.04-33
(This will be revised incorporating the discusion of the expert panel.)

Table 17.4-1 Risk significant SSCs (sheet 18 of 34)

#	Systems, Structures and Components (SSCs)	Rationale[1]	Insights and Assumptions
11	~~Containment isolation motor operated valves~~ Safety injection pump suction Isolation valves [MOV-001A(B,C,D)]	RAW/LPSD FV(FLOOD)	The "Insights and Assumptions" for these SSCs are described on the previous page.
12	Piping	RAW/LPSD	
13	Minimum flow line orifices (next to VLV-L023 A(B,C,D))	RAW(FLOOD)	
14	Minimum flow line manual valves [VLV-024 A(B,C,D)]	RAW(FLOOD)	
15	Minimum flow line manual valves [VLV-023 A(B,C,D)]	RAW(FLOOD)	
16	A(B,C,D)-Hot leg recirculation line isolation valves [MOV-014 A(B,C,D)]	RAW(FLOOD)	

For Item 16 RAI 17.04-22 (This will be revised incorporating the discussion of the expert panel.)

For Item 11 RAI 17.04-34 (This will be revised incorporating the discussion of the expert panel.)

Table 17.4-1 Risk significant SSCs (sheet 19 of 34)

#	Systems, Structures and Components (SSCs)	Rationale(1)	Insights and Assumptions
12	Heating, ventilation, and air conditioning (HVAC) system		
1	B,C-Emergency feedwater pump room fans [VRS-RFN-401B,C]	RAW/CCF/LPSD FV(FLOOD)	EFW M/D pump room fans maintain room temperature when pumps are running. EFW M/D pumps are assumed to be unavailable within the mission time without room cooling due to high room temperature. HVAC systems of other rooms are considered not to be risk significant for the following reasons. - HVAC of emergency gas turbine room Gas turbine units itself has function to intake outer air to remove heat out to atmosphere. Accordingly, HVAC is considered not essential to maintain gas turbine function. - HVAC of ESF room (RHR/CSS pump, SI pump) According to room temperature analysis, room temperature will not exceeds limit of the system during the mission time regardless of availability of HVAC. - HVAC of class1E electric power room (Class 1E I&C, switch gear, battery, battery charger) This system is running during normal operation and continues to run after initiating events. Reliability of normally operating HVAC systems are considered to be high and failure of this system is unlikely to occur during the mission time. - HVAC of EFW T/D pump room Since T/D driven EFW pump room can operate under high room temperature conditions, they are assumed to be available regardless of room cooling during the mission time.

Table 17.4-1 Risk significant SSCs (sheet 20 of 34)

#	Systems, Structures and Components (SSCs)	Rationale(1)	Insights and Assumptions
13	Containment fan cooler system		
1	Containment fan cooler [VCS-CAH-001A (B,C,D)]	EP	Temperature control of Containment Vessel atmosphere is judged important by experts from a point of view of keeping function of safety components in Containment Vessel.
14	Main control room HVAC system		
1	Main control room air handling unit [VRS-RAH-101A (B,C,D)]	EP	Temperature control of main control room atmosphere is judged important by experts from the viewpoint of operator habitability during an accident.

Table 17.4-1 Risk significant SSCs (sheet 21 of 34)

#	Systems, Structures and Components (SSCs)	Rationale(1)	Insights and Assumptions
15	Instrumentation and control (I&C) system		
1	Permanent bus low voltage signal software	RAW/CCF	This software provides start signal to non-class 1E gas turbine generator. Under SBO, This software must operate in order to backup of the safety bus by AAC power source.
2	Component cooling water system train isolation signal software	RAW/CCF	SSCs that have potential to cause common cause failure of signals are risk significant since such failure may result in loss of total system function. EFW T/D pump start signals are risk significant since such failure results in loss of one of two available EFW pumps under, SBO and loss of EFW room cooling conditions. Reliability of signals other than “S signal” is assumed to have same reliability with “P signal”.
3	SG isolation signal software	RAW/CCF	
4	Engineered safety features actuation signal software (P,S)	RAW/CCF	
5	SG(EFW) isolation signals	RAW/CCF	
6	Main steam line isolation signal software	RAW/CCF	
7	Black out signal software	RAW/CCF	
8	CCW start signals	RAW(L2,FLOOD)	
9	Containment pressure sensors [TBD]	RAW(L2)/CCF(L2)	
10	A~D-Emergency feed water pump start signals	RAW	
11	EFW pump start signal software	RAW/CCF	
12	Diverse actuation system	EJ	The unreliability of this system is assumed to be 0.01.

Table 17.4-1 Risk significant SSCs (sheet 22 of 34)

#	Systems, Structures and Components (SSCs)	Rationale[(1)]	Insights and Assumptions
16	Waste management system (WMS)		
1	Refueling water storage (RWS) system – WMS line boundary check valve [VLV-037]	RAW	Large External leak of the boundary check valve results in loss of inventory from the RWS system. Systems that relies on the RWS as water source is affected by this failure mode.
17	Main feedwater system (MFWS)		
1	Main feedwater system	RAW	The Main feedwater system is credited as a function to secondary side cooling during general transients, which does not involve loss of main feedwater.

Table 17.4-1 Risk significant SSCs (sheet 23 of 34)

#	Systems, Structures and Components (SSCs)	Rationale(1)	Insights and Assumptions
18	Main steam supply system (MSS)		
1	A~D-Main steam isolation valves [NMS-AOV-515A (B,C,D)]	RAW/CCF FV/CCF(FIRE)	Main steam isolation valve isolates the ruptured Steam Generator (SG) at the Steam Generator Tube Rupture (SGTR). In case of secondary line break, main steam isolation is required to prevent unlimited steam release. Main steam line piping is required to be intact to isolate the ruptured SG at SGTR events.
2	A~D-Main steam bypass isolation valves [NMS-HCV-3615,3625,3635,3645]	RAW(L2)	
3	Main steam line piping	RAW	
4	Main steam line isolation check valve s A(B,C and D) [VLV-516A(B,C and D)]	RAW(FIRE)	
5	A1~A2-Main steam safety valves B1~B2-Main steam safety valves C1~C2-Main steam safety valves D1~D2-Main steam safety valves [NMS-VLV-509A (B,C,D)] [NMS-VLV-510A (B,C,D)]	RAW(L2)	
6	A,B,C,D,E,F,G,H,J,K,L,M,N,P,Q-Turbine bypass valves [NMS-TCV-500A(B,C,D,E,F,G,H,J,K,L,M,N, P,Q)]	RAW(L2)	

Table 17.4-1 Risk significant SSCs (sheet 24 of 34)

#	Systems, Structures and Components (SSCs)	Rationale[1]	Insights and Assumptions
19	Pressurizer pressure control system part of emergency core cooling system (ECCS)		
1	A(B)--Safety depressurization valves [RCS-MOV-117A(B)]	RAW/CCF FV/CCF(FLOOD,FIRE)	Safety Depressurization Valves (SDVs) are required to open during bleed and feed operation.
2	A(B) -Safety depressurization valves [RCS-MOV-116 A(B)]	RAW(FLOOD,FIRE)	Pressurizer safety valves releases RCS pressure in case of high RCS pressure. Failure of safety valves to re-close results in loss of primary coolant.
3	A~D-Pressurizer safety valves [RCS-VLV-120] [RCS-VLV-121] [RCS-VLV-122] [RCS-VLV-123]	RAW	
20	Depressurization system for severe accident		
1	Depressurization valves [RCS-MOV-118] [RCS-MOV-119]	FV(L2)	In the case of core damage accident, depressurization of the reactor coolant system is required to prevent high pressure melt ejection and direct containment heating.

Table 17.4-1 Risk significant SSCs (sheet 25 of 34)

#	Systems, Structures and Components (SSCs)	Rationale[1]	Insights and Assumptions
21	Containment spray / residual heat removal (CS/RHR) system		
1	Heat exchanger bypass valves [FCV-604] [FCV-636]	RAW/LPSD	The Containment Spray / Residual Heat Removal (CS/RHR) System consists of four independent trains. The CS/RHR System has the following four functions. a. Containment Spray b. Alternative Core Cooling c. RHR Operation during operating modes 4 , 5 and 6.. Since CS/RHR system consists of four independent trains, failure of one train does not have significant impact on risk. However, failures of SSCs that impact multiple trains are risk significant. SSCs that have either of the following characteristics are risk significant. - SSCs that have potential to cause common cause failures among multiple trains. Common cause failure of such system will result in loss of multiple trains. - SSCs that have potential to cause loss of RWSP inventory out side the containment due to large external leaks. Loss of RWSP inventory impacts not only all four trains of CS/RHR system but also other systems that use RWSP as water source.
2	RHR line heat exchanger discharge air operated valves [FCV-603] [FCV-633]	RAW/LPSD	
3	Pump suction line check valves [VLV-004A (B,C,D)]	RAW/CCF/LPSD	
4	RHR line containment isolation check valves [VLV-022A (B,C,D)]	RAW/CCF/LPSD	
5	RHR line containment isolation motor operated valves [MOV-021A (B,C,D)]	RAW/CCF/LPSD	
6	A~D-Containment spray/residual heat removal pumps [RHS-RPP-001A (B,C,D)]	RAW/CCF/LPSD FV(FLOOD)	
7	A~D-Containment spray/residual heat removal heat exchangers [RHS-RHX-001A (B,C,D)]	RAW/CCF/LPSD	
8	RHR line boundary check valves [VLV-028A (B,C,D)]	RAW/LPSD	
9	RWSP discharge line isolation valves [TBD]	RAW	

Table 17.4-1 Risk significant SSCs (sheet 26 of 34)

#	Systems, Structures and Components (SSCs)	Rationale(1)	Insights and Assumptions
10	CS line containment isolation motor operated valves [MOV-004A (B,C,D)]	RAW FV(FLOOD)	The "Insights and Assumptions" for these SSCs are described on the previous page.
11	A~D-CS line check valves [VLV-005A(B,C,D)]	RAW/CCF(FLOOD)	
12	Piping [TBD]	RAW	
13	CS line heat exchanger discharge manual valves [VLV-002A (B,C,D)]	RAW	
14	Minimum flow line manual valves [VLV-13A (B,C,D)]	RAW	
15	CS/RHR - spent fuel pit boundary manual valves (discharge line) [VLV-031A (D))	RAW	
16	From FSS to CSS tie line check valve [VLV-012]	RAW(L2)	These valves are required to open to perform firewater injection from FSS to the spray header.
17	From FSS to CSS tie line motor operated valve [CSS-MOV-011]	FV(L2)/RAW(L2)	
18	CS/RHR - spent fuel pit boundary manual valves (suction line) [VLV-034A (D)]	RAW/LPSD	These valves are required to open to perform gravitational injection from the spent fuel pit to the RCS when RCS is atmospheric pressure at LPSD operation.
19	CS/RHR - spent fuel pit boundary manual valves (suction line) [VLV-33A(D)]	LPSD	

Table 17.4-1 Risk significant SSCs (sheet 27 of 34)

#	Systems, Structures and Components (SSCs)	Rationale(1)	Insights and Assumptions
20	CS/RHR pump hot leg suction isolation valves [MOV-001A(B;C,D)] [MOV-002A(B,C,D)]	LPSD	Failure of these valves result in loss of RHR during LPSD
21	RCS cold leg injection line motor operated valves [MOV-026A(B,C,D)]	LPSD	
22	RCS cold leg injection line check valves [VLV-027A(B,C,D)] [VLV-028A(B,C,D)]	LPSD	

Table 17.4-1 Risk significant SSCs (sheet 28 of 34)

#	Systems, Structures and Components (SSCs)	Rationale[1]	Insights and Assumptions
22	Refueling water storage system (RWS)		
1	Refueling water storage pit (RWSP) sump strainers [TBD]	FV/RAW/CCF	The RWSP is the source of borated water for containment spray and safety injection. During LPSD operation, RWSS has the following functions. a. Refill refueling water auxiliary tank (RWAT) for RCS injection via charging pumps. b. Refill SFP for gravitational injection to RCS. SSCs that have either of the following characteristics are risk significant. - SSCs that have potential to cause common cause failures among multiple trains. Sump strainers have potential of sump screen, which may occur in multiple trains. - SSCs that have potential to cause resulting loss of RWSP inventory out side the containment due to large external leaks are risk significant, since such failure impacts all systems that use RWSP as water source. SSCs that have potential to cause failure to supply RWSP water to RWAT or SFP during LPSD operation are also considered risk significant.
2	Refueling water storage pit [RWS-CPT-001]	RAW	
3	Refueling water recirculation pump suction line manual valves [VLV-006A (B)]	RAW/LPSD	
4	Refueling water recirculation pump discharge line check valves [VLV-012A (B)]	RAW/LPSD	
5	Refueling water recirculation pump discharge line manual valves [VLV-013A (B)]	RAW/LPSD	
6	RWSP discharge line containment isolation motor operated valves [MOV-002] [MOV-004]	RAW/LPSD	
7	A,B-Refueling water recirculation pumps [RWS-RPP-001A (B)]	RAW/LPSD	
8	RWSP discharge line manual valve [VLV-001]	RAW/LPSD	
9	Refueling water recirculation pump suction cross tie line manual valve [VLV-005]	RAW/LPSD	

Table 17.4-1 Risk significant SSCs (sheet 29 of 34)

#	Systems, Structures and Components (SSCs)	Rationale(1)	Insights and Assumptions
10	Refueling water recirculation pump discharge cross tie line manual valve [VLV-014]	RAW/LPSD	The "Insights and Assumptions" for these SSCs are described on the previous page.
11	Refueling water storage auxiliary tank [RWS-OTK-002]	LPSD	
~~11~~12	Refueling water auxiliary tank inlet line manual valve [VLV-052]	RAW/LPSD	
~~12~~13	Refueling water auxiliary tank discharge line manual valve [VLV-101]	RAW/LPSD	
~~13~~14	Refueling water auxiliary tank suction line manual valves [VLV-021] [VLV-051]	LPSD	
15	RWSAT line orifice [TBD : downstream side of VLV-021]	LPSD	
~~14~~16	RWSP suction line containment isolation air operated valve [AOV-022]	LPSD	

RAI 17.04-25 (This will be revised incorporating the discussion of the expert panel.)

Table 17.4-1 Risk significant SSCs (sheet 30 of 34)

#	Systems, Structures and Components (SSCs)	Rationale[1]	Insights and Assumptions
23	Reactor protection system (RPS)		
1	Reactor trip breakers [TBD]	RAW/CCF	These systems are necessary to provide negative reactivity for plan t trip.
2	Control rod (rod injection) [TBD]	FV/RAW/CCF	
24	Chilled water system (VWS)		
1	Chiller units train B and C [TBD]	FV/RAW/CCF/LPSD	The safety related water system supplies chilled water to safety related HVAC systems. SSCs that have potential to cause common cause failures among trains B and C are risk significant since such failures results in loss room cooling in M/D EWF pump area. SSCs that compose train A and D are not risk significant because the PRA assumes only the M/D EFW pumps to be dependent on room cooling during the mission time.
2	Pumps train B and C [TBD]	RAW/CCF/LPSD	

Table 17.4-1 Risk significant SSCs (sheet 31 of 34)

#	Systems, Structures and Components (SSCs)	Rationale[(1)]	Insights and Assumptions
25	Essential service water system (ESWS)		
1	Pump discharge line check valves [VLV-502A (B,C,D)]	RAW/CCF/LPSD	The essential service water system (ESWS) transfers heat from the CCW system as Ultimate Heat Sink (UHS). This system supports the CCW system, which supports various safety and non-safety mitigation systems. Accordingly, reliability of CCWS EFW system has significant impact on risk. Since ESWS consists of four independent trains, failure of one train does not have significant impact on risk. However, failures of SSCs that impact multiple trains have risk significant impact on risk. Accordingly, SSCs that have potential to cause common cause failures among multiple trains are risk significant.
2	Essential service water pump motor cooling line check valves [VLV-602A (B,C,D)]	RAW/CCF/LPSD	
3	A~D-Essential service water pump [EWS-OPP-001A (B,C,D)]	FV/RAW/CCF/LPSD	
4	~~A1,B1-Essential service water pump outlet strainers~~ ~~[EWS-OSR-001A (B)]~~ A (B,C,D) -CCW heat exchanger inlet strainers[TBD]	RAW/LPSD	
5	A1~D1-Essential service water pump outlet strainers ~~A2~D2-Essential service water pump outlet strainers~~ [EWS-OSR-001A (B,C,D)] ~~[EWS-OSR-002A (B,C,D)]~~	RAW/LPSD	
6	Valves located in essential service water pump motor cooling line of train B & C [VLV-601B (C)]	RAW/LPSD	
7	ESW pump motor cooling line valves of train A & D [VLV-601A (D)]	RAW(L2)	

RAI 17.04-26 (This will be revised incorporating the discussion of the expert panel.)

Table 17.4-1 Risk significant SSCs (sheet 32 of 34)

#	Systems, Structures and Components (SSCs)	Rationale[1]	Insights and Assumptions
8	~~Orifices located in essential service water~~ ESW pump motor cooling line transmitters of train A,B ~~&~~ ,C and D [FT-2060,2061,2062 and 2063~~(2)~~]		The "Insights and Assumptions" for these SSCs are described on the previous page.
9	ESW pump motor cooling line orifices of train A,B,C and D [TBD]	RAW/LPSD	
~~9~~10	Main piping orifices of train B and ~~D~~C [FE2025 , FE2026]	RAW/LPSD	
~~10~~11	Main piping orifices of train A and D [FE2024 , FE2027]	RAW(L2)	
~~11~~12	Main piping valves of train B and C [MOV-503B (C)] [VLV-506B (C)] [VLV-507B (C)] [VLV508B (C)] [VLV-509B (C)] [VLV-511B (C)] [VLV-514B (C)] [VLV-517B (C)] [VLV-520B (C)]	RAW/LPSD	
~~12~~13	Main piping valves of train A and D [MOV-503A (D)] [VLV506A (D)] [VLV-507A (D)] [VLV508A (D)] [VLV-509A (D)] [VLV-511A (D)] [VLV-514A (D)] [VLV-517A (D)] [VLV-520A (D)]	RAW(L2)	
~~13~~14	Piping of train B and C [TBD]	RAW/LPSD	
~~14~~15	Piping of train A and D [TBD]	RAW(L2)	

For Item 8&9 RAI 17.04-27 (This will be revised incorporating the discussion of the expert panel.)

For Item10 RAI 17.04-28

Table 17.4-1 Risk significant SSCs (sheet 33 of 34)

#	Systems, Structures and Components (SSCs)	Rationale[1]	Insights and Assumptions
26	Spent fuel pit cooling and purification system (SFPCS)		
1	RWS – SFP inlet line boundary check valves [VLV-027]	RAW/LPSD	Large External leak of valves that form boundary between RWS result in loss of inventory of the RWS system. Accordingly, systems that relies on the RWS as water source is affected by failure of these valves. During RCS is atmospheric pressure at LPSD operation, the spent fuel pit is used as water source of gravitational injection in case loss of decay heat removal function occurs. SSCs associated with gravitational injection line are considered to be risk significant.
2	RWS – SFP inlet line manual valve [VLV-028]	RAW/LPSD	
3	RWS – SFP demineralizer line boundary manual valves [VLV-103A (B)]	RAW	
4	RWS – SFP inlet line manual valves [VLV-029] [VLV-015] [VLV-017]	LPSD	
5	Spent fuel pit [RPT-001]	LPSD	
6	A~D-Spent fuel pit strainers [SFS-RSR-001A (B,C,D)]	LPSD	
7	Spent fuel pit discharge line manual valves [VLV-021A(D)]	LPSD	
8	Spent fuel pit discharge cross tie-line manual valve [VLV-022]	LPSD	

Table 17.4-1 Risk significant SSCs (sheet 34 of 34)

Notes:

1. Definition of Rationale Terms:
   CCF = Common Cause Failure
   FV = Fussell-Vesely
   RAW = Risk Achievement Worth
   FV(L2) = Fussell-Vesely for L2
   RAW(L2) = Risk Achievement Worth for L2
   CCF(L2) = Common Cause Failure for L2
   LPSD =Low Power and Shut Down Operation
   EJ = Engineering Judge
   FLOOD = FLOOD Event
   FIRE = FIRE Event
   EP = Expert Panel

17.4.8 ITAAC for the D-RAP

Tier 1 ITAAC are proposed to verify that the D-RAP provides reasonable assurance that the design of SSCs within the scope of the RAP is consistent with their assumed design reliability. The list of risk-significant SSCs for ITAAC will be prepared by introducing the plant's site-specific information to the list shown in Table 17.4-1 in the Phase II of the D-RAP. The ITAAC acceptance criteria are established to ensure that the estimated reliability of each as-built SSC is at least equal to the assumed design reliability and that industry experience including operations, maintenance, and monitoring activities were assessed in estimating the reliability of these SSCs.

17.4.9 Combined License Information

COL 17.4(1) *The COL Applicant shall be responsible for the development and implementation of the Phases II and III of the D-RAP. In the Phase II, the plant's site-specific information should be introduced to the D-RAP process and the site-specific risk-significant SSCs should be combined with the US-APWR design risk-significant SSCs into a list for the specific plant. In the Phase III, procurement, fabrication, construction, and test specifications for the SSCs within the scope of the RAP should ensure that significant assumptions, such as equipment reliability, are realistic and achievable. The QA requirements should be implemented during the procurement, fabrication, construction, and pre-operation testing of the SSCs within the scope of the RAP.* SPLA 1474-011

COL 17.4(2) *The COL Applicant shall be responsible for the development and implementation of the O-RAP, in which the RAP activities should be integrated into the existing operational program (i.e., Maintenance Rule, surveillance testing, in-service inspection, in-service testing, and QA). The O-RAP should also include the process for providing corrective actions for design and operational errors that degrade non-safety-related SSCs within the scope of the RAP.*

All SSCs identified as risk-significant within the scope of the D-RAP should be categorized as high-safety-significant (HSS) within the scope of initial Maintenance Rule. SPLA 1474-010

17.4.10 References

17.4-1 "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems (RTNSS) in Passive Plant Design," SECY 95-132, U.S. Nuclear Regulatory Commission, Washington, DC, May 1995.

17.4-2 'Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,' "Domestic Licensing of Production and Utilization Facilities," Energy. Title 10, Code of Federal Regulations, Part 50.65, U.S. Nuclear Regulatory Commission, Washington, DC.

17.5 Quality Assurance Program Description

For the Design Certification phase, the MHI-NESH US-APWR Project Quality Assurance Program (QAP) is the top-level policy document that establishes the quality assurance policy and assigns major functional responsibilities for plants designed by MHI-NESH. The QAP describes the methods and establishes QAP and administrative control requirements, described in "Quality Assurance Program (QAP) Description For Design Certification of the US-APWR (PQD-HD-19005 Rev.1)" (Ref 17.5.5-4), that meet 10 CFR Part 50, Appendix B and 10 CFR Part 52. The QAP is based on the requirements of ASME NQA–1-1994, "Quality Assurance Requirements for Nuclear Facility Applications," Parts I and II, as specified in Ref.17.5.5-4.

The MHI QAPD for the Design Certification Phase has been prepared on the basis of the NRC approved QAP template (NEI, 06-14A Rev.4 and earlier revisions) (Ref 17.5.5-3) prepared by the Nuclear Energy Institute and has been evaluated against the SRP. The MHI QAPD provides the QAP controls implemented. MHI performed the comparison of SRP (Mar. 2007) (Ref 17.5.5-2) and draft SRP (Sept. 2006) (Ref 17.5.5-1) which was used as a reference for the MHI QAPD and determined that there is no impact to the MHI QAPD.

Business policies of MHI-NESH establish high level responsibilities and authority for carrying out administrative functions which are outside the scope of the QAP.

Procedures establish practices for certain activities which are common to all MHI-NESH organizations performing those activities such that the activity is controlled and carried out in a manner that meets QAP requirements. Organization specific procedures establish detailed implementation requirements and methods, and may be used to implement the business policies of MHI-NESH or be unique to particular functions or work activities.

The COL applicant is responsible for development a Quality Assurance Program Description during design other than the Design Certification, construction and operation.

17.5.1 Combined License Information

COL 17.5(1) *The COL applicant shall develop and implement the design other than the Design Certification, construction and operational QAP that also covers the activities described in Section 17.5.*

17.5.2 References

17.5.5-1 "Draft Standard Review Plan (SRP) 17.5 dated September 22, 2006"

17.5.5-2 "Standard Review Plan (SRP) 17.5 March 2007"

17.5.5-3 "Quality Assurance Program Description (NEI 06-14A Rev.4 and earlier versions)"

17.5.5-4 "Quality Assurance Program (QAP) Description For Design Certification of the US-APWR (PQD-HD-19005 Rev.1)"

17.6 Description of the Applicant's Program for Implementation of 10 CFR 50.65, the Maintenance Rule

The COL Applicant is responsible for development of the program for implementation of 10 CFR 50.65, the Maintenance Rule.

17.6.1 Combined License Information

COL 17.6(1) *The COL applicant develops and implements* the program for implementation of 10 CFR 50.65, the Maintenance Rule.