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TOKYO, JAPAN

October 26, 2010

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-10291

Subject: **MHI's Response to US-APWR DCD RAI No. 638-5032 Revision 0 (SRP Section 07.06)**

Reference: 1) "Request for Additional Information No. 638-5032 Revision 0, SRP Section 07.06 – Interlock Systems Important to Safety - Application Section: 7.6 dated 09 23, 2010.

With this letter, Mitsubishi Heavy Industries, Ltd. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") a document entitled "Response to Request for Additional Information No. 638-5032 Revision 0."

Enclosed is the response to a question contained within Reference 1.

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. Response to Request for Additional Information No. 638-5032 Revision 0

CC: J. A. Ciocco
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Docket No. 52-021
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Enclosure 1

UAP-HF-10291
Docket No. 52-021

Response to Request for Additional Information No. 638-5032
Revision 0

October 2010

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

10/26/2010

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 638-5032 REVISION 0**
SRP SECTION:	**07.06 - INTERLOCK SYSTEMS IMPORTANT TO SAFETY**
APPLICATION SECTION:	**07.06 - INTERLOCK SYSTEMS IMPORTANT TO SAFETY**
DATE OF RAI ISSUE:	**09/23/2010**

QUESTION NO. : 07.06-17

MHI is requested to clarify whether figure numbers referenced in third paragraph of Subsection 7.6.1.5 for MOV-020 and MOV-007 should be read as "Figure 9.2.2-1 (Sheet 1 of 9)", and "Figure 9.2.2-1 (Sheet 2 of 9)" respectively. They currently read as (Sheet 1 of 7) and (2 of 7) respectively, and do not match up with the figure titles in Chapter 9.

ANSWER:
DCD Chapter 7, Subsection 7.6.1.5 states, "For NCS-MOV-020A, B, C, D the piping diagrams for these valves are shown in Figure 9.2.2-1 (Sheet 1 of 7) in Chapter 9, and for NCS-MOV-007A, B, C, D in Figure 9.2.2-1 (Sheet 2 of 7) of Chapter 9."

The third paragraph of Subsection 7.6.1.5 will be revised with corrected figure references to read: "For NCS-MOV-007A, B, and NCS-MOV-020A, B the piping diagrams for these valves are shown in Figure 9.2.2-1 (Sheet 1 of 9), and for NCS-MOV-007C, D, and NCS-MOV-020C, D in Figure 9.2.2-1 (Sheet 2 of 9)."

Impact on DCD

The DCD Chapter 7, third paragraph of Subsection 7.6.1.5 will be revised as indicated above.

Impact on COLA

There is no impact on the COLA

Impact on PRA

There is no impact on the PRA

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QUESTION NO. : 07.06-18

MHI is requested to resolve the inconsistency between Table 7.1-2 and Subsection 7.6.2 with regard to regulatory requirements applicability for PCMS and Section 7.6. A list of regulation requirements applicability was added into Subsection 7.6.2 as the result of RAI 07.06-5, but this list is not consistent with Table 7.1-2. The following regulations need to be included in Subsection 7.6.2 to be consistent with Table 7.1-2:

- GDC 10, "Reactor Design"
- GDC 15, "Reactor Coolant System Design"
- GDC 16, "Containment Design"
- GDC 28, "Reactivity Limits"
- GDC 29, "Protection Against Anticipated Operational Occurrences"
- GDC 33, "Reactor Coolant Makeup"
- GDC 34, "Residual Heat Removal"
- GDC 35, "Emergency Core Cooling"
- GDC 38, "Containment Heat Removal"
- GDC 41, "Containment Atmosphere Cleanup"
- GDC 44, "Cooling Water"

ANSWER:

The inconsistency between Table 7.1-2 and Subsection 7.6.2 is editorial error. The following regulations will be added in Subsection 7.6.2 to be consistent with Table 7.1-2.

GDC 10 "Reactor Design"
GDC 15 "Reactor Coolant System Design"
GDC 16 "Containment Design"
GDC 20 "Protection System Functions"
GDC 21 "Protection Systems Reliability and Testability"
GDC 22 "Protection System Independence"
GDC 23 "Protection System Failure Modes"
GDC 25 "Protection System Requirements for Reactivity Control Malfunctions"
GDC 28 "Reactivity Limits"
GDC 29 "Protection Against AOOs"
GDC 33 "Reactor Coolant Makeup"
GDC 34 "Residual Heat Removal"
GDC 35 "Emergency Core Cooling"

GDC 38 "Containment Heat Removal"
GDC 41 "Containment Atmosphere Cleanup"
GDC 44 "Cooling Water"

It is noted that the all interlocks important to safety are implemented in the PSMS as described in Subsection 7.6.2.5.

Additionally, GDC applicability to section 7.6 in the row q. of Table 7.1-2 (Sheet 3 of 8) will be corrected to read GDC 33 "Reactor Coolant Makeup" is not related to the section.

Impact on DCD

Table 7.1-2 (Sheet 3 of 8) will be revised as attachment 1.

The list of applicable codes in Subsection 7.6.2 will be revised as follows (Changes are underlined):

1. 10 CFR 50.55a(a)(1), "Quality Standards."
2. 10 CFR 50.55a(h), "Protection and Safety Systems,"
3. 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 1, "Quality Standards and Records."
4. GDC 2, "Design Bases for Protection Against Natural Phenomena."
5. GDC 4, "Environmental and Dynamic Effects Design Bases."
6. **GDC 10 "Reactor Design"**
7. GDC 13, "Instrumentation and Control."
8. **GDC 15 "Reactor Coolant System Design"**
9. **GDC 16 "Containment Design"**
10. GDC 19, "Control Room."
11. **GDC 20 "Protection System Functions"**
12. **GDC 21 "Protection Systems Reliability and Testability"**
13. **GDC 22 "Protection System Independence"**
14. **GDC 23 "Protection System Failure Modes"**
15. GDC 24, "Separation of Protection and Control Systems."
16. **GDC 25 "Protection System Requirements for Reactivity Control Malfunctions"**
17. **GDC 28 "Reactivity Limits"**
18. **GDC 29 "Protection Against AOOs"**
19. **GDC 34 "Residual Heat Removal"**
20. **GDC 35 "Emergency Core Cooling"**
21. **GDC 38 "Containment Heat Removal"**
22. **GDC 41 "Containment Atmosphere Cleanup"**
23. **GDC 44 "Cooling Water"**
24. 10 CFR 50.34(f)(2)(v), "Additional TMI-Related Requirements, Bypass and Inoperable Status Indication"

Impact on COLA

There is no impact on the COLA

Impact on PRA

There is no impact on the PRA

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DATE OF RAI ISSUE:	09/23/2010

QUESTION NO. : 07.06-19

MHI is requested to effectively demonstrate how independent and diverse interlock guidance in Position 2 of Section B in BTP 7-1, "Guidance on Isolation of Low Pressure Systems from the High Pressure Reactor Coolant Systems," is met.

BTP 7-1 states, in part, that "For system interfaces where both valves are motor operated, the valves should have independent and diverse interlocks to prevent both from opening unless the primary system pressure is below the subsystem design pressure."

Section 7.6 does not address diversity interlocks. Subsection 7.6.2.3, "Independence," states that "Redundancy and independent train assignments are specifically discussed for each interlock in the sections above." However, neither the interlock signal path's descriptions nor interlock figures demonstrate the interlock independence.

ANSWER:

As described in the response to Question No.07-06-15 of RAI No.239-2033, the RHR system is able to withstand normal operating RCS pressure without rupture when both CS/RHR Pump Hot Leg Isolation Valves are inadvertently opened since RHR components are designed with sufficient wall thickness. The RHR system is open to the refueling water storage pit (RWSP) to prevent radioactive release outside the containment. Power is normally removed from the CS/RHR Pump Hot Leg Isolation Valves to ensure they cannot open inadvertently.

Therefore, it is not necessary for the valves to have independent and diverse interlocks because the current design of RHR system has sufficiently high reliability against overpressurization or possible radioactive release as described above.

The design feature of the RHR system is described in Subsection 5.4.7 and 7.6.1.1, justification for the interlocks of the valves is not, however, clearly described in Subsection 7.6.1.1. Therefore Subsection 7.6.1.1 will be revised to justify the current design of interlocks of the valves.

Impact on DCD

Conformance status to BTP 7-1 on Table 1.9.2-7 will be revised as Attachment 2.

The following description will be also added to the end of Subsection 7.6.1.1.

CS/RHR pump hot leg isolation valves open permissive interlock does not have independency and diversity in each train, because the current design of RHR system has sufficiently high reliability against overpressurization or possible radioactive release. Moreover the valves cannot be opened inadvertently because power is normally removed as described above.

Impact on COLA

There is no impact on the COLA

Impact on PRA

There is no impact on the PRA

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QUESTION NO. : 07.06-20

MHI is requested to effectively demonstrate how the guidance in Position 4 of Section B in BTP 7-1, "Guidance on Isolation of Low Pressure Systems from the High Pressure Reactor Coolant Systems," is met.

BTP 7-1 states, in part, that "Suitable valve position indication should be provided in the control room for the interface valves." It is not clear how this guidance is met. MHI is requested to clarify how suitable valve position indication is provided in the control room for the interface valves, in accordance with the guidance of BTP 7-1, Section B, Position 4.

ANSWER:

The position indication of the interface valves required by Position 4 of Section B in BTP 7-1 is designed to display in the Main Control Room. The positions of the valves are designed to be monitored by the Visual Display Unit in the Main Control Room. The system description of the intended valves are shown in Section 5.4.7.2.2.3, A.

Impact on DCD

There is no impact on the DCD

Impact on COLA

There is no impact on the COLA

Impact on PRA

There is no impact on the PRA

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QUESTION NO. : 07.06-21

MHI is requested to effectively demonstrate how to conform to guidance RG 1.206, "Combined License Applications for Nuclear Power Plants," with regard to interlock to prevent over pressurization of the primary coolant system during low-temperature operations of the reactor vessel.

Subsection C.I.5.2.2.2 of RG 1.206 states that "Applicants should describe the design of overpressure protection during low-temperature operations, including the capability to relieve pressure during all overpressure events during startup and shutdown conditions at low temperatures, particularly during water-solid conditions. Applicants should provide the analysis that demonstrates how overpressure protection is achieved, assuming any single active component failure. This section should identify all overpressure events and, as a subset, identify the events that can be prevented by preventive interlocks or locking-out power. Applicants should describe how the overpressure protection system is enabled, the alarms and indications associated with the system, and the power source for the system."

Subsection 7.6.3 of the DCD states that "There are no interlocks necessary to prevent over pressurization of the RCS during low-temperature operations of the RV. Refer to Subsection 5.2.2."

Subsection 5.2.2 identifies overpressure events but instead of identifying the events that can be prevented by preventive interlocks or locking-out power as described in RG 1.206 above, this subsection further states that "An important aspect of RCS overpressure protection at low temperatures is the use of administrative controls which are discussed in paragraph 5.2.2.2.2.2, Administrative Controls. Although specific alarms do not exist to invoke specific administrative procedures, annunciation is provided to alert the operator to arm the cold overpressure mitigation system."

It is not clear how the guidance in RG 1.206 with regard to interlock to prevent overpressurization of the primary coolant system during low-temperature operations of the reactor vessel is met.

ANSWER:

The LTOP system for US-APWR consists of CS/RHR pump suction relief valves, which are spring-loaded relief valves. Therefore, preventive interlock to activate the LTOP system is not needed.

These valves are equipped with direct position indication in accordance with a requirement of Section II.D.3 of the TMI Action Plan. When LTOP event occurs, these relief valves operate reactor coolant pressure and a valve position alarm alerts the operator.

In order to ensure the LTOP system is operable status at the correct plant condition during cooldown, the technical specifications require surveillances of the following status (Reference DCD Chapter 16, SR 3.4.12.1 through 3.4.12.7.).

- Number of available Safety Injection (SI) pump
- Number of available Charging pump
- Accumulators are isolated
- RHR suction motor-operated valves are open
- RHR suction motor-operated valves are locked open with operator power removed

Please refer to responses provided to RAI No.103, UAP-HF-08303, which pertain to the LTOP system.

Therefore, since the LTOP system does not need preventive interlock, there is no interlock provided to prevent overpressurization of the primary coolant system during low-temperature operation of the reactor vessel.

Impact on DCD

There is no impact on the DCD

Impact on COLA

There is no impact on the COLA

Impact on PRA

There is no impact on the PRA

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QUESTION NO. : 07.06-22

MHI is requested to effectively demonstrate how to conform to guidance RG 1.206, "Combined License Applications for Nuclear Power Plants," with regard to interlock required to preclude inadvertent inter-ties between redundant or diverse safety systems.

Subsection 7.6.3 states that "There are no interlocks required to preclude inadvertent inter-ties in the USAPWR" without further justification.

ANSWER:
Safety systems, such as Safety Injection System, Residual Heat Removal System, Containment Spray System, and Essential Service Water System, are mechanically separated in each train, so interlocks to preclude inadvertent inter-ties are not needed. Component Cooling Water System has an interlock to preclude inadvertent inter-ties, and this interlock is described in Subsection 7.6.1.5. Therefore, the description in the DCD will be revised.

Impact on DCD

The description in Subsection 7.6.3 will be revised as follows (Changes are underlined);

"There are no interlocks required to preclude inadvertent inter-ties in the US-APWR **except in the CCWS, since safety systems, such as Safety Injection System, Residual Heat Removal System, Containment Spray System, and Essential Service Water System, are mechanically separated in each train**."

Impact on COLA

There is no impact on the COLA

Impact on PRA

There is no impact on the PRA

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QUESTION NO. : 07.06-23

MHI is requested to clarify the statement "There is no manual bypass capacity for interlocks important to safety" in Subsection 7.6.1. Subsections 7.6.1.4 and 7.6.1.5 state otherwise.

ANSWER:
There are some interlock systems important to safety which can be manually bypassed as mentioned in Subsection 7.6.1.4 and 7.6.1.5. The bypassed or inoperable status indication (BISI) provides the ability to manage properly the interlock systems important to safety.

DCD 7.6.1, however, describes "There is no manual bypass capacity for interlocks important to safety". This sentence should be deleted.

Impact on DCD

Subsection 7.6.1 will be revised as follows (Changes are underlined):

7.6.1 System Description

The PSMS provides the interlock systems important to safety for the plant, with the exception of electro-mechanical interlocks within the electrical distribution system.

Except as noted for specific interlocks described below:

~~**There is no manual bypass capability for interlocks important to safety.**~~

Interlocks are tested using the input test, self-test and output test methods as described in Subsections 7.1.3.10 and 7.1.3.11.

Bypassed and inoperable ~~**Abnormal**~~ status of all interlocks is provided via the BISI. The BISI is discussed in Section 7.5.

The following sections describe all interlock functions.

Impact on COLA

There is no impact on the COLA

Impact on PRA

There is no impact on the PRA

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QUESTION NO. : 07.06-24

MHI is requested to justify the statement in the first paragraph of Subsection 7.6.1.1 "The RHR system is able to withstand normal operating RCS pressure without rupture when both CS/RHR Pump Hot Leg Isolation Valves are inadvertently open, as explained in Subsection 5.4.7." This statement was added as the response to RAI 07.06-15. There is no information found in Subsection 5.4.7 supporting this statement. In the same paragraph of the above statement, Subsection 7.6.1.1 further states that "Overpressurization could damage and disable the RHRS, and could lead to inter-system loss of reactor coolant." The latter conflicts the first. Also, the RCS and the RHR design pressures do not support the concerned statement.

ANSWER:

A description regarding this issue is in Subsection 5.4.7.1, but to make it clear, detail description will be added in Subsection 5.4.7.1.

DCD Subsection 7.6.1.1 will be revised to make it clear.

Impact on DCD

Description in Subsection 5.4.7.1 will be revised as follows (Changes are underlined):

"The RHR system is designed to prevent an interfacing system LOCA by two motor operated valves in series on the suction line with power lockout capability **and design pressure with sufficient wall thickness to withstand the RCS pressure without rupture**. In the event that both these valves are opened, the RHR system is designed to withstand high pressure and discharge the RCS inventory to the in-containment RWSP. "

Descriptions in the first paragraph of Subsection 7.6.1.1 will be revised as follows:

~~"The RHR system is able to withstand normal operating RCS pressure without rupture when both CS/RHR Pump Hot Leg Isolation Valves are inadvertently open, as explained in Subsection 5.4.7.~~ **In general, c**~~C~~ompared to the RCS, RHRS is a low-pressure system. An inadvertent connection between the RCS and RHRS while rector coolant pressure is greater than RHR design pressure could lead to over pressurization of the RHRS. Over-pressurization could damage and disable the RHRS, and could lead to inter-system loss of reactor coolant. **In US-APWR, the RHR system is designed with sufficient wall thickness to withstand normal operating RCS pressure**

without rupture when both CS/RHR Pump Hot Leg Isolation Valves are inadvertently opened, as explained in Subsection 5.4.7.1."

Impact on COLA

There is no impact on the COLA

Impact on PRA

There is no impact on the PRA

Table 7.1-2 Regulatory Requirements Applicability Matrix
(per NUREG-0800 Standard Review Plan (SRP) Sec. 7.1 Rev. 5)
(Sheet 3 of 8)

Applicable Criteria		Title	I&C System							Related Section in US-APWR DCD
			RPS	ESFAS	SLS	Safety HSI	Safety DCS	PCMS	DAS	
p.	GDC 29	Protection Against AOOs	X	X	X	X	X	X		7.2 to 7.7, 7.9
q.	GDC 33	Reactor Coolant Makeup	X	X	X	X	X	X		7.3, ~~7.6,~~ Refer to Chapter 9
r.	GDC 34	Residual Heat Removal	X	X	X	X	X	X		7.3, 7.4, 7.6, Refer to Chapter 5
s.	GDC 35	Emergency Core Cooling	X	X	X	X	X	X		7.3, 7.4, 7.6, Refer to Chapter 6
t.	GDC 38	Containment Heat Removal	X	X	X	X	X	X		7.3, 7.4, 7.6, Refer to Chapter 6
u.	GDC 41	Containment Atmosphere Cleanup	X	X	X	X	X	X		7.3, 7.6, Refer to Chapter 6
v.	GDC 44	Cooling Water	X	X	X	X	X	X		7.3, 7.6, Refer to Chapter 9
		3. Staff Requirements Memoranda								
a.	SRM to SECY 93087 II.Q	Defense Against Common-Mode Failures in Digital I&C Systems	X	X	X	X	X	X	X	7.8
b.	SRM to SECY 93087 II.T	Control Room Annunciator (Alarm) Reliability						X		7.5, 7.9
		4. RGs								
a.	RG 1.22	Periodic Testing of Protection System Actuation Functions	X	X	X	X	X			7.2 to 7.6, 7.9
b.	RG 1.47	Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety System	X	X	X	X	X	X		7.2, 7.3, 7.5, 7.6, 7.9
c.	RG 1.53	Application of the Single-Failure Criterion to Safety Systems	X	X	X	X	X			7.2 to 7.6, 7.9
d.	RG 1.62	Manual Initiation of Protection Actions	X	X	X	X	X			7.2, 7.3
e.	RG 1.75	Independence of Electrical Safety Systems	X	X	X	X	X	X	X	7.2 to 7.9

Table 1.9.2-7 US-APWR Conformance with Standard Review Plan Chapter 7 Instrumentation and Controls (sheet 8 of 19)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
Branch Technical Position 7-1: Guidance on Isolation of Low-Pressure Systems From the High-Pressure Reactor Coolant System	The following measures should be incorporated in designs of the interfaces between low-pressure systems and the high pressure reactor coolant system: 1. At least two valves in series should be provided to isolate any subsystem whenever the primary system pressure is above the pressure rating of the subsystem. 2. For system interfaces where both valves are motor-operated, the valves should have independent and diverse interlocks to prevent both from opening unless the primary system pressure is below the subsystem design pressure. Also, the valve operators should receive a signal to close automatically whenever the primary system pressure exceeds the subsystem design pressure. 3. For those system interfaces where one check valve and one motor-operated valve are provided, the motor-operated valve should be interlocked to prevent the valve from opening whenever the primary pressure is above the subsystem design pressure, and to close automatically whenever the primary system pressure exceeds the subsystem design pressure. 4. Suitable valve position indication should be provided in the control room for the interface valves. 5. For those interfaces where the subsystem is required for emergency core cooling system operation, the above recommendations need not be implemented. System interfaces of this type should be evaluated on an individual basis, as discussed in GL 87-12 and GL 88-17.	Conformance with exceptions. Criteria 1 and 4-6: Conformance with no exceptions identified. Criterion 2: Conformance with exception (The CS/RHR pump hot leg isolation valves are interlocked so that they cannot be opened when the RCS pressure is above 400 psig. In US-APWR, CS/RHR pump suction relief valves provide the low-temperature over-pressure protection for RCS components. Therefore **CS/RHR pump hot leg isolation valves open permissive interlock does not have independency and diversity in each train, because the current design of RHR system has sufficiently high reliability against overpressurization or possible radioactive release, and** there is no interlock which automatically isolates RHRS from RCS when reactor coolant pressure exceeds the RHR design pressure to ensure performance of the low-temperature over-pressure protection function according to BTP 5-2.) Criterion 3: Not applicable to the US-APWR design certification (There are no such lines except for ECCS in the US-APWR.)	7.6.1