MITSUBISHI HEAVY INDUSTRIES, LTD.
16-5, KONAN 2-CHOME, MINATO-KU
TOKYO, JAPAN

August 2, 2013

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-13201

Subject: MHI's Response to US-APWR DCD RAI No. 1037-7045 (SRP Section 07.08)

Reference: 1) "Request for Additional Information No. 1037-7045, SRP Section: 07.08 - Diverse Instrumentation and Control Systems, Application Section: MUAP-07014 (Rev 5) - D3 Coping Analysis," dated May 20, 2013.

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. 1037-7045."

Enclosed are the responses to four RAI questions contained within Reference 1.

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

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

Please contact Mr. Joseph Tapia, General Manager of Licensing Department, Mitsubishi Nuclear Energy Systems, Inc. if the NRC has questions concerning any aspect of this submittal. His contact information is provided below.

Sincerely,

Y. Ogata

Yoshiki Ogata,
Executive Vice President
Mitsubishi Nuclear Energy Systems, Inc.
On behalf of Mitsubishi Heavy Industries, LTD.

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NRO

Enclosures:

1. Affidavit of Yoshiki Ogata
2. Response to Request for Additional Information No. 1037-7045 (Proprietary version)
3. Response to Request for Additional Information No. 1037-7045 (Non-proprietary version)

CC: J. A. Ciocco
J. Tapia

Contact Information
Joseph Tapia, General Manager of Licensing Department
Mitsubishi Nuclear Energy Systems, Inc.
1001 19th Street North, Suite 710
Arlington, VA 22209
E-mail: joseph_tapia@mnes-us.com
Telephone: (703) 908 – 8055

ENCLOSURE 1

Docket No. 52-021
MHI Ref: UAP-HF-13201

MITSUBISHI HEAVY INDUSTRIES, LTD.
AFFIDAVIT

I, Yoshiki Ogata, state as follows:

1. I am Executive Vice President of Mitsubishi Nuclear Energy Systems, Inc., and have been delegated the function of reviewing Mitsubishi Heavy Industries, Ltd's (MHI) US-APWR documentation to determine whether it contains information that should be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4) as trade secrets and commercial or financial information which is privileged or confidential.

2. In accordance with my responsibilities, I have reviewed the enclosed document entitled "Response to Request for Additional Information No. 1037-7045" dated August 2013, and have determined that portions of the document contain proprietary information that should be withheld from public disclosure. Those pages containing proprietary information are identified with the label "Proprietary" on the top of the page and the proprietary information has been bracketed with an open and closed bracket as shown here "[ ]." The first page of the document indicates that all information identified as "Proprietary" should be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4).

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

4. The basis for holding the referenced information confidential is that it describes the unique design approach for the Defense-in-Depth ("D3") of the Instrumentation and Control ("I&C") system and the unique post-accident response of the US-APWR crediting design features for D3, developed by MHI and not used in the exact form by any of MHI's competitors. This information was developed at significant cost to MHI, since it required the performance of Research and Development and detailed design for its software and hardware extending over several years.

5. The referenced information is being furnished to the Nuclear Regulatory Commission (NRC) in confidence and solely for the purpose of supporting the NRC staff's review of MHI's application for certification of its US-APWR Standard Plant Design.

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

7. Public disclosure of the referenced information would assist competitors of MHI in their design of new nuclear power plants without incurring the costs or risks associated with

the design of the subject systems. Therefore, disclosure of the information identified as proprietary would have the following negative impacts on the competitive position of MHI in the U.S. nuclear plant market:

A. Loss of competitive advantage due to the costs associated with development of the D3 approach for the I&C system and the post-accident plant response of the US-APWR to the design features for D3. Providing public access to such information permits competitors to duplicate or mimic the D3 design approach without incurring the associated costs.

B. Loss of competitive advantage of the US-APWR created by benefits of enhanced plant safety, and reduced operation and maintenance costs associated with the D3 approach.

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

Executed on this 2nd day of August, 2013.

Y. Ogata

Yoshiki Ogata,
Executive Vice President
Mitsubishi Nuclear Energy Systems, Inc.

Enclosure 3

UAP-HF-13201
Docket No. 52-021

Response to Request for Additional Information
No. 1037-7045

August 2013

(Non-Proprietary)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

08/02/2013

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 1037-7045**
SRP SECTION:	**07.08 - DIVERSE INSTRUMENTATION AND CONTROL SYSTEMS**
APPLICATION SECTION:	**MUAP-07014 (REV 5) - D3 COPING ANALYSIS**
DATE OF RAI ISSUE:	**05/20/2013**

QUESTION NO. : 07.08-31

D3 Coping Analysis criteria

In Section 5.0 of the MUAP-07014-P, revision 5, it states that the criteria used in the D3 coping analysis is based on: 1) pressure boundary (PB) integrity, 2) coolability, and 3) the dose not to exceed 10 percent of the 10 CFR 100 guidelines. Further, it states that dose evaluations are not necessary if coolability is maintained except for the events which lead to release of primary coolant from RCS outside the containment vessel (CV). Please explain:

a) The basis for not requiring dose evaluations, if the coolability is maintained. Also, explain why the dose is within 10 percent, if the coolability is maintained.

b) Clarify the exception for the events which lead to release of primary coolant from RCS outside the CV.

ANSWER:

a) If the minimum departure from nucleate boiling ratio (DNBR) remains above the 95/95 limit, the coolability is maintained and the fuel integrity is not degraded. If the fuel is not damaged, then fission products are not released due to the event. With no release of fission products, it is not necessary to perform a dose evaluation. This is endorsed by the acceptance criteria for anticipated operational occurrences (AOOs) in the SRP 15.0 of NUREG-0800.

   Therefore, for events which do not violate the DNB analytical limit, it follows that the dose is within 10 percent of the 10 CFR 100 guidelines for these events. The exception is the cases where primary coolant is released outside the CV as described in part b below.

b) The events which lead to release of primary coolant from the RCS to outside the CV are the failure of small lines carrying primary coolant outside containment and the steam

generator tube ruptures. These events are described in Sections 5.6.2 and 5.6.3, respectively, in MUAP-07014-P, Revision 5. The Section 5.6.2, Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment describes the time available to terminate the leakage by local control which meets the 10 CFR 100 criteria (100% for PA), from event initiation to termination of the leakage is 180 minutes. The Section 5.6.3, Radiological Consequences of Steam Generator Tube Failure describes that the DAS and appropriate manual actions provide an event termination time that is similar to the DCD evaluation. Therefore, the 10 CFR 100 criteria are met (100% for PA).

Impact on DCD
There is no impact on the DCD.

Impact on R-COLA
There is no impact on the R-COLA.

Impact on PRA
There is no impact on the PRA.

Impact on Technical / Topical Reports
There is no impact on the technical / topical reports.

This completes MHI's response to the NRC's question.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

08/02/2013

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 1037-7045**
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DATE OF RAI ISSUE:	**05/20/2013**

QUESTION NO. : 07.08-32

Event Evaluation Methods

In MUAP-07014-P, revision 5, Section 4.6, "Event Evaluation Methods," it states that each Chapter 15 event is evaluated based on one of the following methods:

1) Equivalent protection, 2) Expertly judged, and 3) Analyzed

The staff requests the basis for this categorization, and more importantly provide justification as most of them are not analyzed or expertly judged. For example: In Section 5.2, "Decrease in Heat Removal by the Secondary System," only "Loss of External Load" is analyzed. The other events, such as turbine trip, loss of condenser vacuum, closure of main steam isolation valve, and steam pressure regulator failure are not specifically analyzed or shown how they are similar or different to the event with which they are compared. The staff needs a specific and brief description of all such events in order to make a clear distinction between those events which are expertly judged versus analyzed. Further, describe how they are bounded by those categories that are analyzed and/or expertly judged.

ANSWER:

MUAP-07014-P, Revision 5, Section 4.6 outlines the analysis methodology. The justifications requested by this RAI are provided in each event section (Sections 5.1.1 through 5.6.5).

The following table provides the categorization. Other events which are not listed in the table is categorized 3) Analyzed.

Category	Section	Event
Equivalent protection	5.1.1	Decrease in Feedwater Temperature as a Result of Feedwater System Malfunctions
	5.1.2	Increase in Feedwater Flow as a Result of Feedwater System Malfunctions
	5.1.3	Increase in Steam Flow as a Result of Steam Pressure Regulator Malfunction
	5.1.5	Steam System Piping Failures Inside and Outside of Containment
	5.2.6	Loss of Non-Emergency AC Power to the Station Auxiliaries
	5.3.1.2	Complete Loss of Forced Reactor Coolant Flow
	5.4.3	Control Rod Misoperation (System Malfunction or Operator Error)
	5.5.2	Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory
	5.6.2	Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment
Expertly judged	5.4.6	Inadvertent Decrease in Boron Concentration in the Reactor Coolant System
	5.6.3	Radiological Consequences of Steam Generator Tube Failure
	5.6.5.2	Small Break Loss-of-Coolant Accident (SBLOCA)

For the event in Section 5.2.1, the loss of load event is modeled by assuming an instantaneous step load decrease in both steam flow and feedwater flow from their full value (100%) to zero at the beginning of the transient. This assumption bounds all credible loss of load scenarios in the event group, such as loss of external load, turbine trip, loss of condenser vacuum, closure of main steam isolation valve. This assumption is the same as the DCD Chapter 15 safety analysis. This is consistent with the response to RAI No. 303-2329, Question No. 15.2-4, for DCD Section 15.2.1 which was submitted by MHI letter UAP-HF-09342 dated July 3, 2009. Note that in the response to RAI 789-5920, Question No. 15.02.01-15.02.05-9, submitted by MHI letter UAP-HF-11331 dated September 30, 2011, MHI added an additional turbine trip case that included the assumption of a LOOP following the turbine trip. The assumption of a LOOP as an additional failure is not required as part of the D3 coping analysis in MUAP-07014-P. Therefore, the conclusion in MUAP-07014-P Section 5.2.1 that the loss of load event is modeled as a bounding case remains valid.

Impact on DCD
There is no impact on the DCD.

Impact on R-COLA
There is no impact on the R-COLA.

Impact on PRA
There is no impact on the PRA.

Impact on Technical / Topical Reports
There is no impact on the technical / topical reports.

This completes MHI's response to the NRC's question.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

08/02/2013

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 1037-7045**
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DATE OF RAI ISSUE:	**05/20/2013**

QUESTION NO. : 07.08-33

ATWS acceptance criteria (MUAP-07014 R5, section 4.3)

Explain why the 3200 psig pressure acceptance criteria is applicability to the US-APWR design when the reactor coolant hydrostatic pressure boundary integrity test uses a lower acceptance pressure.

ANSWER:

This question involves the same subject as that in Question 07.08-27 issued to MHI in RAI 988-7021 dated January 29, 2013. As described in UAP-HF-13137 submitted on June 17, 2013, MHI responded to the issue contained in Question 07.08-27 by pointing to this response. Therefore, those questions posed in RAI 988-7021, Question 07.08-27 are repeated and answered here.

QUESTION a) Revision 3 of DCD Chapter 5, Table 5.4.10.3 states that the US-APWR reactor coolant system hydrostatic test pressure is 3106 psig. Justify why 3200 psig, as identified in MUAP-07014, Revision 5, section 4.3, is an acceptable pressure acceptance criteria or provide a new pressure acceptance criteria.

ANSWER a):
SRP 15.8 states that the RCS pressure shall not exceed ASME Service Level C limits (approximately 22 MPa or 3200 psig) as the acceptance criteria for ATWS. The SRP 15.8 also states that; Appendix A to WASH-1270 states that in evaluating the reactor coolant system boundary for ATWS events, "the calculated reactor coolant system transient pressure should be limited such that the maximum primary stress anywhere in the system boundary is less than that of the 'emergency conditions' as defined in the ASME Nuclear Power Plant Components Code, Section III." The acceptance criteria for reactor coolant pressure, based upon the ASME Service Level C limits, are approximately 10.3 MPa (1500 psig) for BWRs and approximately 22MPa (3200 psig) for PWRs.

The ASME Service Level C limit was developed based on a typical PWR with normal operating pressures of 2250 psia. Since the US-APWR normal operating pressure is 2250 psia, it is consistent with most operating plants. Thus, the ASME Service Level C limit (and 3200 psig criteria) is applicable to the US-APWR. To verify this, MHI has performed stress analyses. For major reactor coolant system components / locations (reactor vessel and reactor coolant piping) which maintain core cooling under emergency core cooling system operation, the maximum allowable RCS pressures which satisfy the primary stress limit based on the Level C service limit in Section III (NB-3224 (vessel) or NB-3655 (piping)) of the ASME Code at the saturation temperature (700F) are summarized in Table 07.08.33-1 below. The results show that core cooling based on the RCS pressure boundary integrity is maintained at the pressure (3200psig) and the saturation temperature (700F). Therefore, the 3200 psig pressure acceptance criteria is applicable to the US-APWR for beyond design basis events such as digital I&C CCF.

Note that although MUAP-07014-P uses 3200 psig as the pressure acceptance criterion, the RCS pressures are actually maintained well below this criterion. As described in the response to Question 07.08-34 of this RAI, the DAS reactor trip and pressurizer safety valves provide protection to ensure that the RCS pressure boundary integrity is maintained. With these design features, the results in MUAP-07014-P show that the RCS pressure can be maintained below 110% of the system design pressure (i.e., much less than 3200 psig).

Table 07.08.33-1 Maximum Allowable RCS Pressure for Major Reactor Coolant System Components

Component	Location	Maximum Allowable RCS Pressure (psig)
Reactor Vessel	Lower Shell	
	Inlet / Outlet Nozzle	
Reactor Coolant Piping	Main Piping	

QUESTION b) It is not clear how reactor coolant pressure integrity is demonstrated. What is the pressurizer high-pressure setpoint to trip the reactor? Is the trip setpoint adequate to protect the 3200 psig limit or does it need actuation of the relief valves (RVs) to relieve the pressure?

ANSWER b):
Technical Report MUAP-07014-P (R5) Table 4.4-1 shows that the pressurizer high-pressure reactor trip setpoint is 2440 psia for DAS.

A sensitivity analysis was performed to show that the DAS reactor trip setpoints are adequate to protect against the SRP 15.8 limit (approximately 22 MPa or 3200 psig). Unless specifically listed below, the assumptions, input parameters, and initial conditions assumed in the D3 coping analysis are the same as the DCD Chapter 15 safety analysis.

- Any reactor trip actuation by the RTS is ignored.

- The DAS reactor trip analytical setpoints and delay times from Table 4.4-1 in MUAP-07014 (R5) are used. This analysis assumes the high pressurizer pressure reactor trip by the DAS. In addition to the signal time delay listed in Table 4.4-1, the time delay between when the MG-set power is cut and rod motion is assumed to be 5 seconds. Therefore, the total DAS reactor trip delay is 16.8 seconds.

- Two turbine-driven emergency feed pumps are assumed to be actuated by the DAS.

- The analysis in DCD Section 15.2.6, Loss of Non-Emergency AC Power to the Station Auxiliaries, assumes a combination of initial condition uncertainties in order to maximize peak pressurizer water volume. The sensitivity analysis in this response assumes a combination of initial condition uncertainties in order to maximize RCS pressure. The initial power level is taken as 102% of the licensed core thermal power level with the initial reactor coolant temperature 4°F above the nominal value and the pressurizer pressure 30 psi below the nominal value.

Figure 7.8-33.1 shows that the RCS pressure limit is not challenged. Note that this analysis assumes a loss of all feedwater at time zero as a precondition to the loss of offsite power which is consistent with the DCD Chapter 15 analysis. The loss of offsite power is then assumed to occur when the DAS reactor trip occurs. This results in the behavior shown in the first [ ] which is when the high pressurizer pressure DAS reactor trip occurs in Figure 7.8-33.1, where the RCS (and RCP outlet) pressure slowly increases due to the degrading secondary heat transfer as the SG water levels decrease due to the loss of feedwater flow and continued high reactor power as shown in Figure 7.8-33.2. Then the RCS pressure increases rapidly after the loss of offsite power and reactor trip occurs. The peak RCP outlet pressure is [ ] During this pressure peak, the pressurizer safety valves open as shown in Figure 7.8-33.3. The opening of the pressurizer safety valves along with the decrease in reactor power results in a decrease in RCS pressure. However, the loss of offsite power results in a delay in the start of emergency feedwater, such that RCS pressure increases again to the pressurizer safety valve setpoint. The RCS pressure then plateaus at a pressure of 2525 psia, which corresponds to the safety valve setpoint. At this point, only a small relief through the safety valves is required to maintain a constant pressure, as shown in Figures 7.8-33.1 and 7.8-33.3. Then once EFW flow begins, the RCS pressure finally begins to decrease. Note that the peak pressure in this analysis is higher than the DCD Chapter 15 analysis due to the different assumptions described in the bullets above. In addition, despite the fact that the RCPs trip due to the LOOP, MHI conservatively assumes a constant pressure difference between the RCP outlet and RCS pressures, as indicated in Figure 7.8-33.1.

Figure 7.8-33.1 RCP Outlet and RCS Pressure versus Time
Loss of Non-Emergency AC Power to the Station Auxiliaries

Figure 7.8-33.2 Reactor Power versus Time
Loss of Non-Emergency AC Power to the Station Auxiliaries

Figure 7.8-33.3 Pressurizer Safety Valve Flow Rate versus Time
Loss of Non-Emergency AC Power to the Station Auxiliaries

QUESTION c) Revision 3 of DCD Section 15.8, section 15.8.2, identifies the pressure limit as 3200 psia. MUAP-07014, Revision 5, section 4.3, says the limit is 3200 psig. The units do not match. Identify the incorrect unit and correct it.

ANSWER c):
Revision 3 of DCD Section 15.8 contains the incorrect unit. MHI will revise DCD Section 15.8 to indicate that the limit is 3200 psig based on the SRP 15.8 acceptance criteria as described in Attachment-1.

QUESTION d) DAS trip is credited for coolability for this event. What is the heat sink and what is the source of water to the heat-sink, as the AFW is not available. Because, AFW pump is motor driven and loss of AC power results in loss of power supply for the MG set. What motive force is available to drive the AFW pump?

ANSWER d):
The two turbine-driven emergency feedwater pumps can be actuated by the DAS even if a LOOP has occurred. The motive force available to drive the pumps is steam and the pumps are supplied from the emergency feedwater pits. The sensitivity analysis in the response to part b) shows that the trip setpointis adequate to protect against the 3200 psig limit with actuation of the relief valves (RVs) to relieve the pressure.

Impact on DCD
DCD Section 15.8 will be revised as indicated in the attached markup in Attachment-1.

Impact on R-COLA
There is no impact on the R-COLA.

Impact on PRA
There is no impact on the PRA.

Impact on Technical / Topical Reports
There is no impact on the technical / topical reports.

This completes MHI's response to the NRC's question.

15.8 Anticipated Transients without Scram

15.8.1 Identification of Causes and Frequency Classification

An anticipated transient without scram (ATWS) is an anticipated operational occurrence (AOO) followed by the failure of the automatic reactor trip portion of the reactor trip system. Since the reactor trip system must satisfy the single-failure criterion, multiple failures or a common mode failure must occur to cause the assumed failure of the reactor trip.

The frequency of an AOO, in coincidence with multiple failures or a common mode failure, is much lower than any of the other events that are analyzed in the US-APWR DCD Chapter 15. Therefore, the ATWS event cannot be classified as either an AOO or design basis accident (postulated accident), and has been historically considered as a beyond-design-basis event.

15.8.2 ATWS Rule (10 CFR 50.62) Design Requirements

In the 1970s, analyses performed by both the PWR reactor vendors and NRC as part of the ATWS Rulemaking showed that although failure of the reactor trip system could transform a minor transient into a severe accident, consequences from an ATWS would be acceptable (maintain peak reactor coolant system (RCS) pressure below 3200 psi~~a~~g) provided that the moderator temperature coefficient of reactivity was sufficiently negative and that a turbine trip and automatic auxiliary (or emergency) feedwater flow are initiated in a timely manner. As a result, the final ATWS rule (10 CFR 50.62) required that each pressurized water reactor have equipment from sensor output to final actuation device, that is diverse from the reactor trip system, to automatically initiate the auxiliary (or emergency) feedwater system and initiate a turbine trip under conditions indicative of an ATWS. In the US, the plant equipment providing this capability has generally been referred to as the ATWS mitigation system actuation circuitry (AMSAC), which is described in Section 7.8. Therefore, the ATWS rule is met by the US-APWR. DCD_07.08-33

The goal of the ATWS rule is to reduce the core damage frequency contribution from ATWS to less than 10^{-5} per reactor year (Ref 15.8-1). The US-APWR core damage frequency is discussed in Chapter 19 of this DCD and shows that the contribution of ATWS to the total core damage frequency meets the safety goal.

15.8.3 ATWS Design for the US-APWR

The features of the AMSAC described above have been incorporated into the US-APWR design as part of the diverse actuation system (DAS), which is described in Section 7.8. In addition to the AMSAC capability (diverse turbine trip and automatic emergency feedwater actuation), the US-APWR DAS also includes diverse reactor trip signals for the following trip functions that trip the motor-generator sets as diverse means of interrupting power to the reactor trip breakers in the event the ATWS is caused by a common mode failure of the reactor trip breakers:

- Low pressurizer pressure
- High pressurizer pressure

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

08/02/2013

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 1037-7045**
SRP SECTION:	**07.08 - DIVERSE INSTRUMENTATION AND CONTROL SYSTEMS**
APPLICATION SECTION:	**MUAP-07014 (REV 5) - D3 COPING ANALYSIS**
DATE OF RAI ISSUE:	**05/20/2013**

QUESTION NO. : 07.08-34

Since the pressurizer safeties have the capacity to limit RCS pressure below 3200 psig what role does the DAS low, low S/G setpoint play in ensuring pressure boundary integrity? Are there cases where the low, low S/G setpoint is credited and no credit is assumed for the pressurizer safeties?

ANSWER:

The sizing of the pressurizer safety valves is based on analysis of a complete loss of steam flow to the turbine with the reactor operating at 102% of the design NSSS thermal power. In this analysis, feedwater flow is also assumed to be lost, and no credit is taken for operation of the pressurizer level control system, pressurizer spray system, rod control system, turbine bypass system, or main steam relief valves. The reactor is maintained at full power (no credit for reactor trip), and steam relief through the main steam safety valves is considered. The total pressurizer safety valve capacity is required to be at least as large as the maximum surge rate into the pressurizer during this transient. Therefore, the pressurizer safety valves are sized such that the RCS pressure is maintained below 110% of the system design pressure during this scenario.

The pressurizer safety valves are not affected by a digital I&C CCF. Therefore, the D3 coping analysis assumes that the pressurizer safety valves open when the RCS pressure reaches the safety valve setpoint. There is no case where the pressurizer safety valve is not assumed to open when the RCS pressure reaches the valve setpoint. Although the safety valves can maintain RCS pressure below 110% of the system design pressure, reactor trip is still needed to move the plant to hot shutdown. The reactor trip may also prevent the pressure from increasing to the safety valve setpoint to begin with. The DAS provides the high pressurizer pressure automatic reactor trip. The reactor will be tripped when pressurizer pressure increases to the setpoint. Therefore, the RCS pressure is maintained below 110% of the system design pressure during Ch.15 events concurrent with digital I&C CCF due to the DAS reactor trip in conjunction with the pressurizer safety valves. For

example, the Loss of External Load event in Section 5.2.1 of MUAP-07014-P, Revision 5, takes credit for both the DAS high pressurizer pressure reactor trip and the safety valves to show that the pressure boundary integrity is maintained.

The DAS also provides the low steam generator (SG) water level reactor trip for events which result in a decrease in secondary system coolant inventory. In those events, the RCS pressure transient is mitigated by the low SG water level reactor trip signal in conjunction with the pressurizer safety valves. As demonstrated in Section 5.2.7 of MUAP-07014-P, Revision 5, the Loss of Normal Feedwater Flow event results in a DAS reactor trip by the low SG water level signal. The pressurizer safety valves also open such that RCS pressure transient is mitigated by both the reactor trip and safety valve opening. The same is true for the Feedwater System Pipe Break event in Section 5.2.8 of MUAP-07014-P, Revision 5. Therefore, there is no case where the pressurizer safety valves are not assumed to open when the RCS pressure reaches to the valve setpoint, regardless of whether or not the low SG water level reactor trip is assumed.

Impact on DCD
There is no impact on the DCD.

Impact on R-COLA
There is no impact on the R-COLA.

Impact on PRA
There is no impact on the PRA.

Impact on Technical / Topical Reports
There is no impact on the technical / topical reports.

This completes MHI's response to the NRC's question.