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

July 3, 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-09342

Subject: MHI's Response to US-APWR DCD RAI No. 303-2329 Revision 2

Mitsubishi Heavy Industries, Ltd. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") the document entitled "MHI's Response to US-APWR DCD RAI No. 303-2329 Revision 2". The material in Enclosure 1 provides MHI's response to the NRC's "Request for Additional Information (RAI) 303-2329 Revision 2," dated May 4, 2009.

Please contact Dr. C. Keith Paulson, Senior Technical Manager, 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
General Manager- APWR Promoting Department
Mitsubishi Heavy Industries, Ltd.

Enclosures:

1. MHI's Response to US-APWR DCD RAI No. 303-2329 Revision 2 (non-proprietary)

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

Contact Information

C. Keith Paulson, Senior Technical Manager
Mitsubishi Nuclear Energy Systems, Inc.
300 Oxford Drive, Suite 301
Monroeville, PA 15146
E-mail: ck_paulson@mnes-us.com
Telephone: (412) 373-6466

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ENCLOSURE 1

UAP-HF-09342
Docket No. 52-021

MHI's Response to US-APWR DCD RAI No. 303-2329 Revision 2

July 2009

(Non-Proprietary)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

7/03/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 303-2329 REVISION 2**
SRP SECTION:	**15.02.01 – 15.02.05 – LOSS OF EXTERNAL LOAD; TURBINE TRIP; LOSS OF CONDENSER VACUUM; CLOSURE OF MAIN STEAM ISOLATION VALVE (BWR); AND STEAM PRESSURE REGULATOR FAILURE (CLOSED)**
APPLICATION SECTION:	**15.2.1 – 15.2.5**
DATE OF RAI ISSUE:	**5/04/2009**

QUESTION NO.: 15.2-1

SRP Sections15.2.1-5 and 15.2.7 state, "For new applications, LOOP should not be considered a single failure; [all AOOs] should be analyzed with and without LOOP in combination with a single active failure." In compliance with this requirement, provide the results of calculations that include the occurrence of a LOOP unless otherwise exempted by the NRC staff. If exempted by the NRC, provide documentation of the exemption.

ANSWER:

A sensitivity analysis concerning the US-APWR LOOP assumptions and their supporting bases is described in detail in the response that is being submitted for Question 15.0.0-3 of RAI 297-2287. That response includes a comparison of the DCD Subsection 15.2.1 (loss of external load) analysis with and without LOOP.

Figure 15.2-1.1 below provides the transient DNBR curve for the loss of normal feedwater event in DCD Subsection 15.2.7 considering a LOOP, in which the reactor coolant pump coastdown is delayed 3 seconds after turbine trip (turbine trip is assumed to occur at the same time as reactor trip). For comparison, the curve without LOOP is provided on the same figure. For the DNBR figures shown in this response, the results are generated using the MARVEL-M/VIPRE-01M methodology rather than the MARVEL-M lookup table methodology utilized in the DCD. Both of these methodologies are described in detail in the Non-LOCA Methodology Topical Report (MUAP-07010). Since it was necessary to use the MARVEL-M/VIPRE-01M methodology for the LOOP case due to the flow coastdown, the same methodology was used for the without LOOP (i.e., DCD) case for consistency.

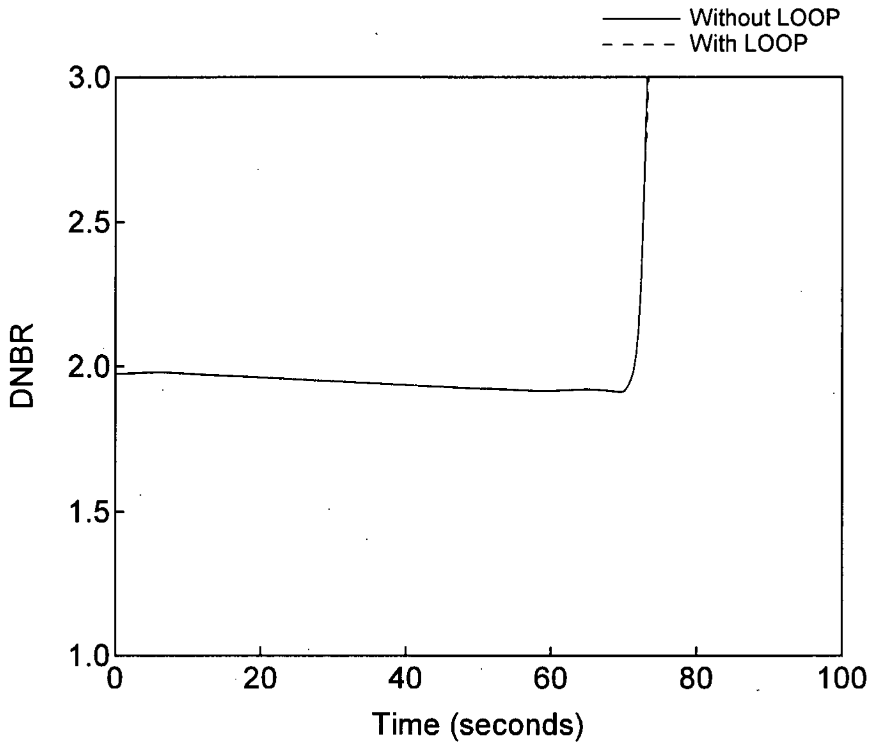

Figure 15.2-1.1 **DNBR versus Time with and without LOOP Loss of Normal Feedwater Flow**

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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APPLICATION SECTION: **15.2.1 – 15.2.5**

DATE OF RAI ISSUE: **5/04/2009**

QUESTION NO.: 15.2-2

In DCD Section 15.2.1, Loss of External Load, the direct reactor trip on turbine trip is not credited in the analysis. Instead, the reactor trip is initiated at 8.5 seconds after turbine trip for the DNBR analysis and 8.7 seconds after turbine trip for the RCS analysis, following the receipt of the high pressurizer pressure signal. Minimum DNBR is calculated to occur one second later in both analyses. It appears from the results presented that the RCPs continue to operate for the duration of the transient. Explain the logic as to why the turbine trip does not result in a trip and coast down of the RCPs. What would be the results if the RCPs coasted down following the turbine trip as proposed in DCD Section 15.0.0.7?

ANSWER:

As described in DCD Subsection 15.2.1.1, the loss of external load may result from an abnormal grid frequency, a trip of the generator and / or turbine, or spurious closure of the main turbine stop or control valves or main steamline isolation valves. In any of these cases, the effect on the plant will be a sudden reduction in main steam flow beginning at time zero. The use of the term "turbine trip" in DCD Tables 15.2.1-1 and 15.2.1-2 is therefore confusing and the DCD will be revised to use the term "loss of main steam flow", as shown in the Impact on DCD section below, in order to better describe the analysis performed in DCD Subsection 15.2.1. The turbine trip may not occur until a later time and may be caused by other protective signals, such as the reactor trip, as is the case in DCD Subsection 15.2.1.

For the analysis in DCD Subsection 15.2.1, loss of external load, the turbine trip is assumed to be caused by the reactor trip and is assumed to occur at that same time. A LOOP may occur following turbine trip from reactor trip. The sensitivity of the loss of external load analysis to the LOOP timing assumptions is described as part of the sensitivity analysis in the response to Question 15.0.0-3 of RAI 297-2287.

For the case where the loss of main steam flow is the result of a turbine trip that occurs at time zero, the event could be mitigated by crediting the direct reactor trip on turbine trip. A LOOP may occur following the turbine trip, causing all RCPs to coast down 3 seconds later. If the direct reactor trip on turbine trip is conservatively not assumed, then the DNBR response for this event is very similar to and is bounded by the complete loss of forced reactor flow event in DCD Subsection 15.3.1.2. This scenario, with LOOP, was analyzed for the RCS pressure, main steam pressure, and DNBR responses. Figures 15.2-2.1 and 15.2-2.2 below show the RCP outlet and steam generator pressure responses, respectively, and Table 15.2-1.1 lists the key events and times at which they occur, relative to the initiation of the transient. For the RCS and main steam pressure response, the 3 second delay between turbine trip and LOOP/RCP coastdown is conservatively not credited (see the response to Question 15.0.0-3 of RAI 297-2287 for a full sensitivity analysis to all LOOP assumptions). The peak RCP outlet and steam generator pressures remain below 110% of their respective design pressures, and are bounded by the existing analysis in DCD Subsection 15.2.1, loss of external load, as shown in the figures. Figure 15.2-2.3 below shows the DNBR response and Table 15.2-1.2 lists the key events and times at which they occur, relative to the initiation of the transient. The minimum DNBR remains above the 95/95 DNBR design limit and is bounded by the existing analysis in DCD Subsection 15.3.1.2, complete loss of forced reactor flow, as shown in the figure.

In summary, the LOOP cases following turbine trip have been provided and confirm the explanation in DCD Subsection 15.0.0.7 that the LOOP portion of these events is not limiting with respect to peak RCS pressure or DNBR.

Table 15.2-2.1
Time Sequence of Events for Turbine Trip (Assuming LOOP) –
RCS & Main Steam Pressure Analysis

Event	Time (sec)
Turbine Trip	0.0
RCP coastdown begins due to loss of offsite power	0.0
Low reactor coolant pump speed reactor trip analysis value reached	0.5
Reactor trip initiated (rod motion begins)	1.1
Peak RCP outlet pressure occurs	6.0
Peak main steam system pressure occurs	18.2

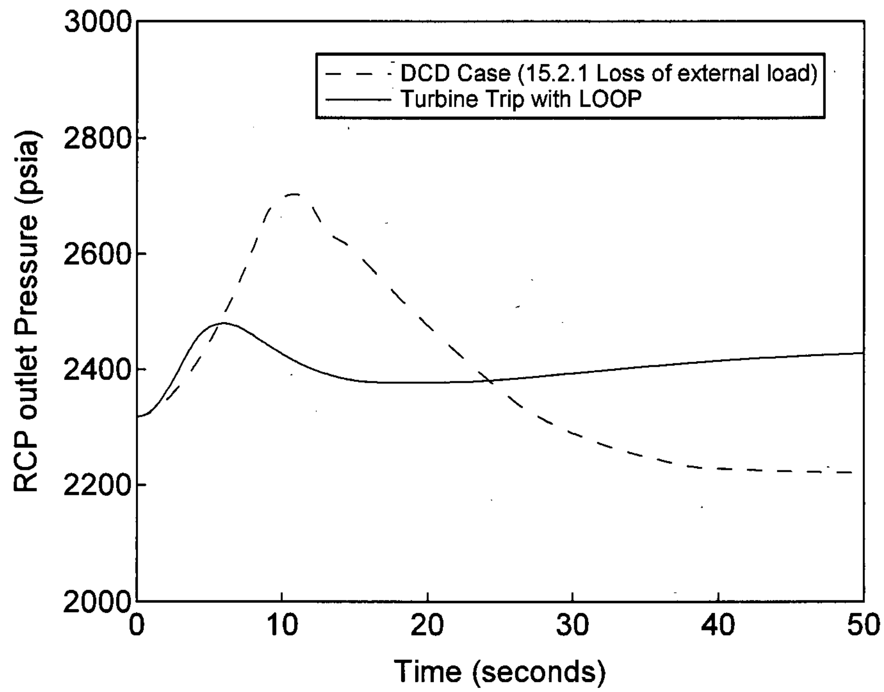

Figure 15.2-2.1 **RCP Outlet Pressure versus Time Comparison of Turbine Trip with LOOP to DCD 15.2.1 Case (Loss of External Load)**

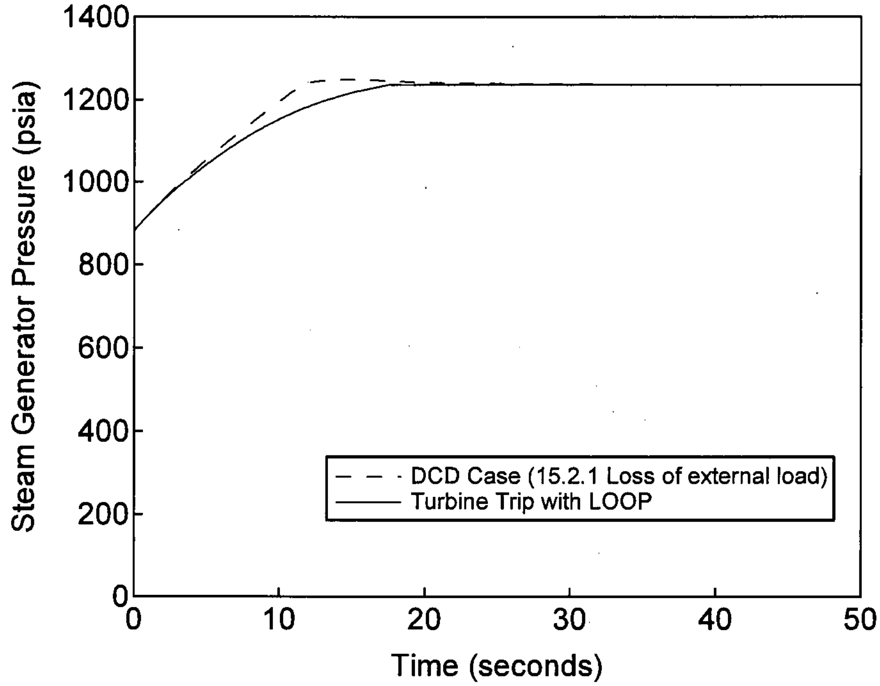

Figure 15.2-2.2 **Steam Generator Pressure versus Time Comparison of Turbine Trip with LOOP to DCD 15.2.1 Case (Loss of External Load)**

Table 15.2-2.2
Time Sequence of Events for Turbine Trip (Assuming LOOP) – DNBR Analysis

Event	Time (sec)
Turbine Trip	0.0
RCP coastdown begins due to loss of offsite power	3.0
Low reactor coolant pump speed reactor trip analysis value reached	3.5
Reactor trip initiated (rod motion begins)	4.1
Minimum DNBR occurs	6.2

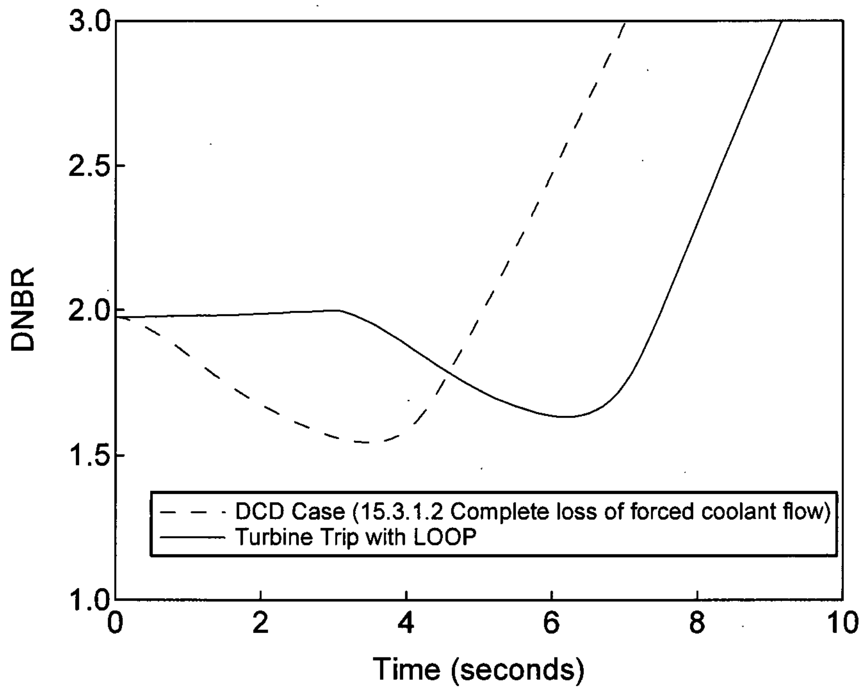

Figure 15.2-2.3 **DNBR versus Time**
Comparison of Turbine Trip with LOOP to
DCD 15.3.1.2 Case (Complete Loss of Forced Coolant Flow)

Impact on DCD

Tables 15.2.1-1 and 15.2.1-2 in DCD Subsection 15.2.1 will be modified as indicated below.

Table 15.2.1-1
Time Sequence of Events for
Loss of External Load/Turbine Trip Transient - DNBR Analysis

Event	Time (sec)
Loss of main steam flow ~~Turbine trip~~, loss of main feedwater flow	0.0
High pressurizer pressure analytical limit reached	6.7
Reactor trip initiated (rod motion begins)	8.5
Pressurizer safety valves open	8.6
Minimum DNBR occurs	9.5
Main steam safety valves open	9.7
Peak RCP outlet pressure occurs	10.3
Peak main steam system pressure occurs	14.3

Table 15.2.1-2
Time Sequence of Events for Loss of External Load/Turbine Trip Transient
- RCS & Main Steam Pressure Analysis

Event	Time (sec)
Loss of main steam flow ~~Turbine trip~~, loss of main feedwater flow	0.0
High pressurizer pressure analytical limit reached	6.9
Pressurizer safety valves open	8.6
Reactor trip initiated (rod motion begins)	8.7
Peak RCP outlet pressure occurs	10.9
Main steam safety valves open	11.5
Peak main steam system pressure occurs	14.9

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

7/03/2009

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APPLICATION SECTION:	**15.2.1 – 15.2.5**
DATE OF RAI ISSUE:	**5/04/2009**

QUESTION NO.: 15.2-3

In DCD Section 15.2.1, Loss of External Load, explain why a LOOP does not occur three seconds after the turbine trip?

ANSWER:

See the response to Question 15.2-2 of this RAI.

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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APPLICATION SECTION:	**15.2.1 – 15.2.5**
DATE OF RAI ISSUE:	**5/04/2009**

QUESTION NO.: 15.2-4

In DCD Section 15.2.2, Turbine Trip, the argument is made that this event is similar to the loss of external load transient and is bounded by the results for DCD Section 15.2.1, except that the steam flow following a turbine trip transient would be isolated by closure of the main turbine stop valves rather than the main turbine control valves (as in the case for DCD Section 15.2.1). Provide quantitative justification for why DCD Section 15.2.2-15.2.4 events are bounded by the analysis presented in DCD Section 15.2.1.

ANSWER:

The single initiator for all of the events in DCD Subsections 15.2.1 to 15.2.5 is the sudden reduction of main steam flow (including the sudden reduction of main feedwater flow). For all SRP Section 15.2 AOOs, this reduction in steam flow combined with the time delay until reactor trip signals are reached, causes a RCS heatup that can challenge the acceptance criteria. Therefore, these events are very similar, regardless of their specific cause or timing of the reduction in steam flow. For this reason, conservative assumptions were made in DCD Subsection 15.2.1 to assure that the analysis would bound all these events. SRP 15.2.1 to 15.2.5 discusses the differences between these events due to valve closure time and the status of feedwater pumps. However, for the DCD, the bounding condition assumed in the analysis is that both main steam flow and main feedwater flow instantaneously decrease from their initial conditions to zero at time zero. This is the most severe case possible, since all steam flow ceases at time zero, rather than allowing for some additional steam flow early in the event. As indicated in Figure 15.2-4.1, the triangular area under the dashed line represents additional steam flow that can occur, thus mitigating the RCS heatup. By assuming instantaneous valve closure, this additional steam flow is not credited. Therefore, the analysis in DCD Subsection 15.2.1 bounds all of the events in DCD Subsections 15.2.1 to 15.2.5.

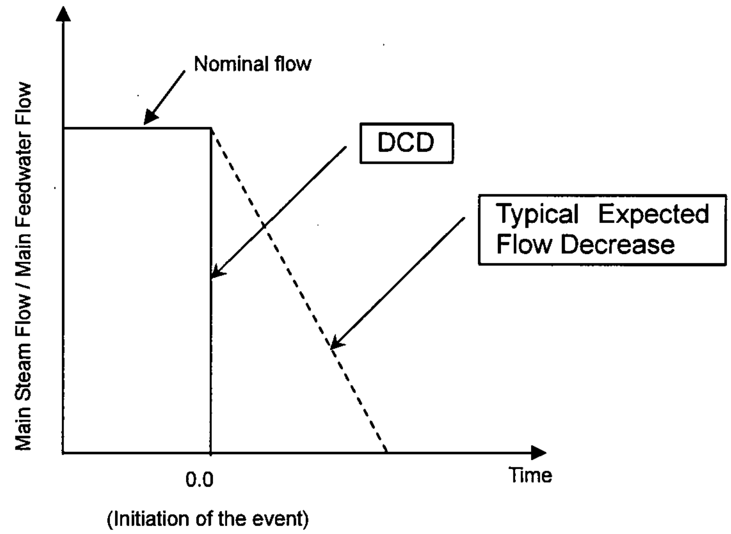

Figure 15.2-4.1 Assumption of Main Steam/Feedwater Flow in DCD

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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APPLICATION SECTION:	**15.2.1 – 15.2.5**
DATE OF RAI ISSUE:	**5/04/2009**

QUESTION NO.: 15.2-5

Provide a comparative sensitivity analysis for the minimum DNBR for a limiting AOO, varying the assumed time delay between turbine trip and RCP coast down following the turbine trip from zero to three seconds. Also provide confirmation that the minimum DNBR occurs during this 3-second time window for all AOOs, prior to the coast down of the RCPs.

ANSWER:

A sensitivity analysis concerning the US-APWR LOOP assumptions and their supporting bases is described in detail in the response that is being submitted for Question 15.0.0-3 of RAI 297-2287. This sensitivity analysis includes the results of varying the time delay between turbine trip and RCP coast down following the turbine trip from zero to three seconds.

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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APPLICATION SECTION:	**15.2.1 – 15.2.5**
DATE OF RAI ISSUE:	**5/04/2009**

QUESTION NO.: 15.2-6

Question has been deleted.

ANSWER:

NA

Impact on DCD

NA

Impact on COLA

NA

Impact on PRA

NA

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APPLICATION SECTION:	**15.2.1 – 15.2.5**
DATE OF RAI ISSUE:	**5/04/2009**

QUESTION NO.: 15.2-7

Question has been deleted.

ANSWER:

NA

Impact on DCD

NA

Impact on COLA

NA

Impact on PRA

NA

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APPLICATION SECTION:	**15.2.1 – 15.2.5**
DATE OF RAI ISSUE:	**5/04/2009**

QUESTION NO.: 15.2-8

Section 15.2.1 indicates that the radiological consequences of the loss of load event are bounded by the main steam line break analysis described in Section 15.1.5. The approach of comparing one event to another in a different event type (heat up transients vs. cooldown transients) requires more justification since the assumptions, initial conditions and other plant conditions for different types of events will not provide the same base line for comparisons. For each event type, identify the most limiting case (with respect to all acceptance criteria) within the group of transients in the same event type.

ANSWER:

For the event in DCD Subsection 15.2.8 (feedwater system pipe break) secondary water and steam are released through the secondary piping break as well as through the main steam safety valves. There are no fuel failures predicted for this event, but in the case where there is primary-to-secondary leakage from normal plant operations, this is an available path for radiation to be released to the environment. As a result, this scenario is very similar to the radiological release path for the main steam system piping failure described in DCD Subsection 15.1.5. Both of these events result in the complete blowdown of the affected loop to the atmosphere. Decay heat and sensible heat removal from the intact loops secondary safety and relief valves is also assumed to be released to the atmosphere. The primary source term and the primary-to-secondary leakage assumptions would be the same for both of these events. However, the main steam system piping failure results in a larger steam release to the environment because for this event the assumed initial steam generator inventory at hot zero power is much larger than the hot full power inventory of the feedwater system piping failure. Therefore, the main steam system piping failure results in more severe radiological consequences than the feedwater system piping failure. For this reason, the DCD states that the radiological consequences of the feedwater system piping failure event in Subsection 15.2.8 is bounded by the radiological consequences of the main steam system piping failure event in Subsection 15.1.5, even though the events are of different types. The DCD will be revised to justify why the radiological consequences of Subsection 15.2.8 are compared to those of Subsection 15.1.5.

However, the feedwater system piping failure event in DCD Subsection 15.2.8 results in a larger steam release to the environment and thus more severe radiological consequences than the events in DCD Subsection 15.2.1 to 15.2.7. DCD Subsections 15.2.1 to 15.2.7 will be revised to describe that these events are bounded by the radiological consequences of the feedwater system piping failure event, which is the same type of event.

Impact on DCD

DCD Subsections 15.2.1.5, 15.2.2.5, 15.2.3.5, 15.2.4.5, 15.2.6.5, 15.2.7.5, and 15.2.8.5 will be revised as follows:

15.2.1.5 Radiological Consequences

The radiological consequences of this event are bounded by the radiological consequences of the ~~main steam line break accident~~ feedwater system piping failure event evaluated in Section ~~15.1.5~~15.2.8.

15.2.2.5 Radiological Consequences

The radiological consequences of this event are bounded by the radiological consequences of the ~~main steam line break accident~~ feedwater system piping failure event evaluated in Section ~~15.1.5~~15.2.8.

15.2.3.5 Radiological Consequences

The radiological consequences of this event are bounded by the radiological consequences of the ~~main steam line break accident~~ feedwater system piping failure event evaluated in Section ~~15.1.5~~15.2.8.

15.2.4.5 Radiological Consequences

The radiological consequences of this event are bounded by the radiological consequences of the ~~main steam line break accident~~ feedwater system piping failure event evaluated in Section ~~15.1.5~~15.2.8.

15.2.6.5 Radiological Consequences

The radiological consequences of this event are bounded by the radiological consequences of the ~~main steam line break accident~~ feedwater system piping failure event evaluated in Section ~~15.1.5~~15.2.8.

15.2.7.5 Radiological Consequences

The radiological consequences of this event are bounded by the radiological consequences of the ~~main steam line break accident~~ feedwater system piping failure event evaluated in Section ~~15.1.5~~15.2.8.

15.2.8.5 Radiological Consequences

No fuel failures are predicted for this event but radiological releases are possible due to the secondary system piping failure in the presence of primary-to-secondary leakage from normal plant operations. The radiological consequences of this event are bounded by the radiological consequences of the main steam line break accident evaluated in Section 15.1.5. Both the feedwater system pipe break and the main steam line break events are secondary system piping failures for which no event-specific fuel failures are predicted.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.