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

December 25, 2008

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-08303

**Subject:** **MHI's Responses to US-APWR DCD RAI No.103-1448 Revision 0**

**Reference:** 1) "Request for Additional Information No.103-1448 Revision 0, SRP Section: 05.02.02 – Overpressure Protection Application Section: 5.2.2, QUESTIONS for Reactor System, Nuclear Performance and Code Review (SRSB)" dated November 20, 2008.

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.103-1448 Revision 0."

Enclosed are the responses to Questions 05.02.02-1 through 05.02.02-8 that are 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. Responses to Request for Additional Information No.103 Revision 0

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

D081
NRO

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

# Enclosure 1

UAP-HF-08303
Docket No. 52-021

# Responses to Request for Additional Information No.103-1448
Revision 0

December 2008

## RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

12/25/2008

**US-APWR Design Certification**

**Mitsubishi Heavy Industries**

**Docket No. 52-021**

**RAI NO.:** **NO. 103-1448 REVISION 0**

**SRP SECTION:** **05.02.02 – Overpressure Protection**

**APPLICATION SECTION:** **5.2.2**

**DATE OF RAI ISSUE:** **11/20/2008**

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**QUESTION NO.: 05.02.02-1**

RAI 5.2.2-1
In section 5.2.2, Overpressure Protection, the applicant stated that the "Combinations of these systems provide compliance with the overpressure protection requirements of the ASME Boiler and Pressure Vessel Code, Section III, Paragraphs NB-7200, Overpressure Protection Report for pressurized water reactor systems." The staff review, of the overpressure protection systems of DCD sections 5.2.2 and 5.4.11 with regard to compliance with ASME III NB-7200, concluded that additional documentation is required to support applicant's positions of the following ASME III NB-7200 requirements: "(c) the range of operating conditions, including the effect of discharge piping back pressure; (k) consideration of set pressure and blowdown limitations, taking into account opening pressure tolerances and overpressure of the pressure relief device; and (l) consideration of burst pressure tolerance and manufacturing design range of the rupture disk device." Provide the references or documentation that supports compliance to the NB-7200 requirements discussed above.

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**ANSWER:**

Backpressure compensation is used so that the pressurizer safety valve is independent of backpressure. The pressurizer safety valve is adjusted to close after blowing down to a pressure not lower than 95 % of the set pressure in accordance with ASME NB-7500.
The description about operating conditions, set pressure, opening pressure tolerances and overpressure of the pressure relief device assumed in the analysis is described in the answer to the question No. 05.02.02-2.

Specification of burst pressure tolerance and manufacturing design range of the rupture disk device are as follows:

Burst pressure tolerance : ± 5%
Manufacturing design range : 0%

**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.: 05.02.02-2**

RAI 5.2.2-2
In section 5.2.2, the applicant stated that the "RCS and main steam system overpressure protection during power operation are provided by the pressurizer safety valves and the main steam safety valves, ..... Combinations of these systems provide compliance with the overpressure protection requirements of the ASME Boiler and Pressure Vessel Code...." Provide the references or documentation for the methodology, analysis, and results that support the applicant's position that the ASME criteria are satisfied as stated.

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**ANSWER:**

Per DCD Section 5.2.2.1.2, the sizing of the pressurizer safety valves is based on an analysis of a complete loss of steam flow to the turbine with the reactor operating at 102% of the design nuclear steam supply system (NSSS) thermal power. In the analysis, the reactor is maintained at full power by not taking credit for the first reactor trip signal and conservatively ignoring the second reactor trip signal during the short duration of the transient. Steam relief through the main steam safety valves is considered. Additionally, feedwater flow is also assumed to be lost, and no credit is taken for operation of the following control systems:

- Pressurizer level control system
- Pressurizer pressure control system
- Rod control system
- Turbine bypass system
- Main steam relief valves

The MARVEL-M plant transient analysis code is used to calculate transient responses for reactor power, RCS pressure, and reactor coolant temperature. The MARVEL-M model is described in the US-APWR DCD Section 15.0.2.2.1. Additional details regarding the MARVEL-M code are provided in the Non-LOCA Methodology Topical Report, MUAP-07010-P (Proprietary) and MUAP-07010-NP (Non-Proprietary), dated July 2007.

The results for the pressurizer safety valve sizing analysis are provided in Figure 5.2.2-2.1. Per DCD Section 5.2.2.1.1, the total pressurizer safety valve capacity is required to be at least as large as the maximum surge rate into the pressurizer during the analyzed transient. Figure 5.2.2-2.1 provides the calculated volumetric flow rate of the surge line for the loss of external load / turbine trip event assuming no reactor trip. Conservative assumptions are made in this analysis to maximize the pressurizer insurge.

This sizing methodology results in a pressurizer safety valve capacity in excess of the capacity required to prevent exceeding 110% of the RCS design pressure, ignoring the first reactor trip in accordance with ASME Pressure Vessel Code Paragraph NB-7311. This is done by assuring for the most limiting anticipated operational occurrence (loss of load) that the pressurizer steam space volume does not exceed 103% of design pressure corresponding to the pressure when the safety valve is fully open. This pressurizer steam space pressure increase also considers piping downstream of the safety valve discharge. In addition, the pressure difference between the safety valve inlet and the highest pressure in the RCS is considered. This includes the pressure drop around the RCS piping, pressurizer surge line, and elevation heads in the RCS. The highest RCS pressure is at the reactor coolant pump discharge when the pumps are operating.

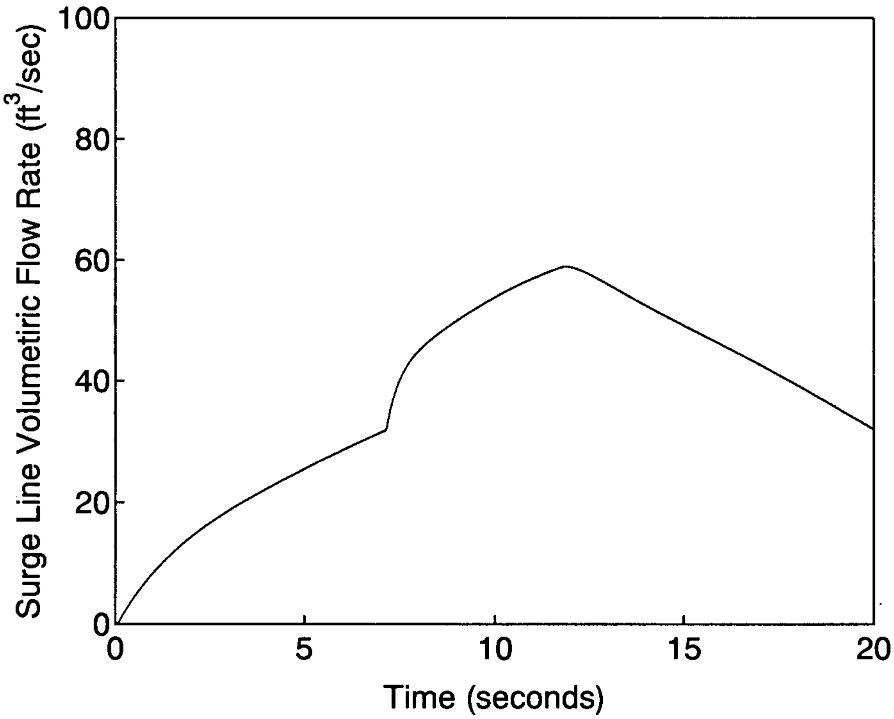


**Figure 5.2.2-2.1 Surge-Line Volumetric Flow Rate versus Time Loss of External Load/Turbine Trip Transient - without Reactor Trip**

**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.: 05.02.02-3**

RAI 5.2.2-3
In Subsection 5.2.2.1.2, the applicant states that "the LTOP is designed to meet the following requirements" of Branch Technical Position (BTP) 5-2 which requires that the "...system is designed and installed to prevent exceeding the applicable technical specifications and Appendix G limits for the RCS while operating at low temperatures. The system is capable of relieving pressure during all anticipated over-pressurization events at a rate sufficient to satisfy the technical specification limits, particularly while the RCS is in a water-solid condition." Provide the references related to the analysis or supporting documentation which demonstrates how the above requirements are satisfied.

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**ANSWER:**

The summary of analysis for LTOP is shown in Attachment A. Please confirm this attachment.

**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.: 05.02.02-4**

RAI 5.2.2-4
In Subsection 5.2.2.1.2, the applicant states that "the LTOP is designed to meet the following requirements" of Branch Technical Position (BTP) 5-2 which requires that, "The system should be able to perform its function assuming any single active component failure. Analyses using appropriate calculational techniques must demonstrate that the system will provide the required pressure relief capacity assuming the most limiting single active failure. The cause for initiation of the event (e.g., operator error, component malfunction) should not be considered as the single active failure. The analyses should assume the most limiting allowable operating conditions and systems configuration at the time of the postulated cause of the overpressure event." Provide the references or supporting documentation in regard to the analyses which demonstrates how the above requirements are satisfied.

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**ANSWER:**

The LTOP system for US-APWR is consisting of CS/RHR pump suction relief valves, which are spring-loaded relief valves. As spring-loaded relief valve is passive component, the LTOP system is passive system. Therefore, any single active failure does not affect the LTOP system.

The summary of analysis for LTOP is shown in Attachment A. Please confirm this attachment.

**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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**SRP SECTION:** 05.02.02 – Overpressure Protection

**APPLICATION SECTION:** 5.2.2

**DATE OF RAI ISSUE:** 11/20/2008

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**QUESTION NO.: 05.02.02-5**

RAI 5.2.2-5
In Subsection 5.2.2.1.2, the applicant states that the system satisfies the BTP 5-2 requirement that, "The design of the system should use Institute of Electrical and Electronics Engineers (IEEE) Standard 603 as guidance." In which the system may be manually enabled; however, an alarm should be provided to alert the operator to enable the system at the correct plant condition during cooldown. Positive indication should be provided to indicate when the system is enabled. An alarm should activate when the protective action is initiated. Identify the plant (condition) parameter that is monitored during cooldown to alert the operator to enable the system and, if applicable, the technical specifications (TS) related to the parameter.

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**ANSWER:**

The LTOP system for US-APWR is consisting of CS/RHR pump suction relief valves, which is spring-loaded relief valve. 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 automatically and the valve position alarm alerts the operator.

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

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

**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.: 05.02.02-6**

RAI 5.2.2-6
From subsection 5.2.2.1.2, the applicant states that the BTP 5-2 requirements: (1) the overpressure protection system does not depend on the availability of offsite power to perform its function and (2), pressure relief is from a low-pressure system not normally connected to the primary system, ensure that the interlocks that would isolate the low pressure system from the primary coolant system do not defeat the overpressure protection function are satisfied. From the information presented, the staff was unable to confirm the requirements were satisfied. Provide references or documents of detailed system description and P&IDs to support the statement that the above referenced BTP 5-2 requirements are satisfied.

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**ANSWER:**

CS/RHR pump suction relief valves (RHS-VLV-003A, B, C, D), which are spring-loaded relief valves, are adopted for the LTOP system of US-APWR. These valves are shown in P&ID of Residual Heat Removal (RHR) System (Figure 5.4.7-2).

These valves are spring-loaded relief valves and can actuate without electrical power. Therefore, LTOP system does not depend on the availability of offsite power to perform its function, meeting requirement (1).

CS/RHR pump suction relief valves are installed in the RHR system and not normally connected to the primary system. There are CS/RHR Pump Hot Leg Isolation Valves (RHS-MOV-001A, B, C, D and RHS-MOV-002A, B, C, D), which are motor-operated valves, between the primary system and these relief valves. These valves have interlocks not to open unless the reactor coolant pressure is less than the pressure for RHR operation for overpressure protection of RHR system. However, there is no interlock to close automatically. This interlock of these valves is described in Subsection 7.6.1 of the DCD and is shown in Figure 7.6-1. Additionally, these valves are locked open with operator power removed during LTOP system is required. This is periodically confirmed in accordance with SR 3.4.12.7 of Technical Specification of DCD Chapter 16. Therefore, the LTOP system meets the requirement of (2).

**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.: 05.02.02-7**

RAI 5.2.2-7
In subsection 5.2.2.2.2.1, the applicant discussed low temperature transient evaluation during water solid conditions. The applicant stated that the most limiting mass input transient is an inadvertent safety injection, and the most limiting heat input transient is an inadvertent reactor coolant pump startup in a loop where the steam generator temperature is 50°F higher than the other temperatures in the loop. The range of RCS temperatures was 70°F to 280°F. And the anticipated mass and heat input transients are evaluated to demonstrate conformance with ASME III, Appendix G. Provide the references or documentation that demonstrates compliance with ASME III, Appendix G. Also, the analysis was based on a range of RCS temperatures from 70°F to 280°F, whereas the LTOP operation starts below approximately 350°F. Include an explanation of why the analysis was not based on a range of RCS temperatures from 70°F to 350°F.

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**ANSWER:**

The summary of analysis for LTOP is shown in Attachment A. Please confirm this attachment.

The range of RCS temperature from 70°F to 280°F is typographical error. It should be from 70°F to 350°F. DCD will be revised.

**Impact on DCD**

DCD subsection 5.2.2.2.2.1, last sentence will be revised as follows.

The range of RCS temperatures was 70°F to ~~280~~350°F with a corresponding range of steam generator temperatures from 120°F to ~~330~~400°F.

**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.: 05.02.02-8**

RAI 5.2.2-8
The applicant designated RCS-MOV-118 and RCS-MOV-119 as depressurization valvesin Tier 1, Table 2.4.2-2; however, only RCS-MOV-119 is listed as a component in Tier 1,Table 2.4.2-1. Explain the omission of RCS-MOV-118 in Table 2.4.2-1. Should RCSMOV-118 be designated as a depressurization block valve?

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**ANSWER:**

RCS-MOV-118 is located upstream of RCS-MOV-119, thus is not appeared in the description in Table 2.4.2-1 "Pressurizer piping upstream of and including the pressurizer safety valves RCS-VLV-120,121,122,123, safety depressurization valves RCS-MOV-117A,B, and depressurization valves RCS-MOV-119", as well as RCS-MOV-116A,B which are located upstream of RCS-MOV-117A,B.
Unlike SDV block valve and pressurizer spray block valve which are normally open valves and block a line if necessary, RCS-MOV-118 is not designated as the block valve because of the normally closed position.

**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.

**Attachment A**

## Summary of LTOP analysis for US-APWR

### 1. Introduction

LTOP is designed to meet the requirements of BTP5-2 which requires that the LTOP system is designed and installed to prevent exceeding the applicable technical specifications and Appendix G limits for the RCS while operating at low temperature.

Following is the summary of the analysis which demonstrates how the above requirements are satisfied.

### 2. Analyses

Overpressure transient in the RCS at low temperature can be caused by either of two types of events; i.e., mass input or heat input.

For mass input case, inadvertent actuation of two high head injection pumps is assumed as a initiating event and for heat input case, inadvertent start of one reactor coolant pump is assumed as a initiating event.

The MARVEL-M plant transient analysis code is used to calculate transient responses of primary coolant pressure for cold over pressure event. This MARVEL-M code is described in US-APWR DCD Section 15.0.2.2.1. Additional details regarding the MARVEL-M code are provided in the Non-LOCA Methodology Topical Report, MUAP-07010-P (Proprietary) and MUAP- 07010-NP (Non-Proprietary), dated July 2007.

BTP5-2 requires that the system should be able to perform its function assuming any single active component failure. The LTOP system for US-APWR is consisting of CS/RHR pump suction relief valves, which are spring-loaded relief valves. As spring-loaded relief valve is passive component, the LTOP system is passive system. Therefore, any single active failure does not affect the LTOP system.

### 2.1 Mass input case

#### 2.1.1 Analysis condition

Initial primary coolant temperature: 100 F
Initial primary coolant pressure: 400 psig
Solid RCS condition
Two RHR relief valves operable
Inadvertent start of two high head injection pumps

#### 2.1.2 Results

Figure 1 is the plot of primary coolant pressure variation versus time for mass input case. Maximum primary coolant pressure variation is 120 psi.

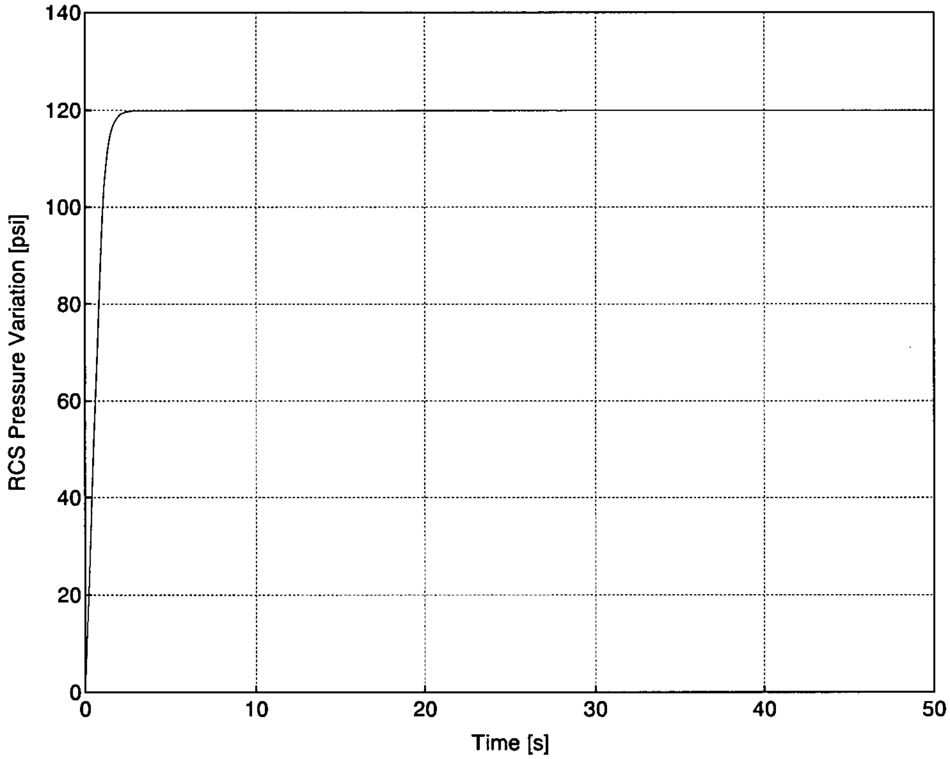


**Figure 1 Primary coolant pressure variation (mass input case)**

## 2.2 Heat input case

### 2.2.1 Analysis condition

Initial primary coolant temperature: 100 F

Initial secondary coolant temperature: 150 F

Initial primary coolant pressure: 400 psig

Solid RCS condition

Two RHR relief valves operable

Inadvertent start of one reactor coolant pump

### 2.2.2 Results

Figure 2 is the plot of primary coolant pressure variation versus time for mass input case. Maximum primary coolant pressure variation is 95.1 psi.

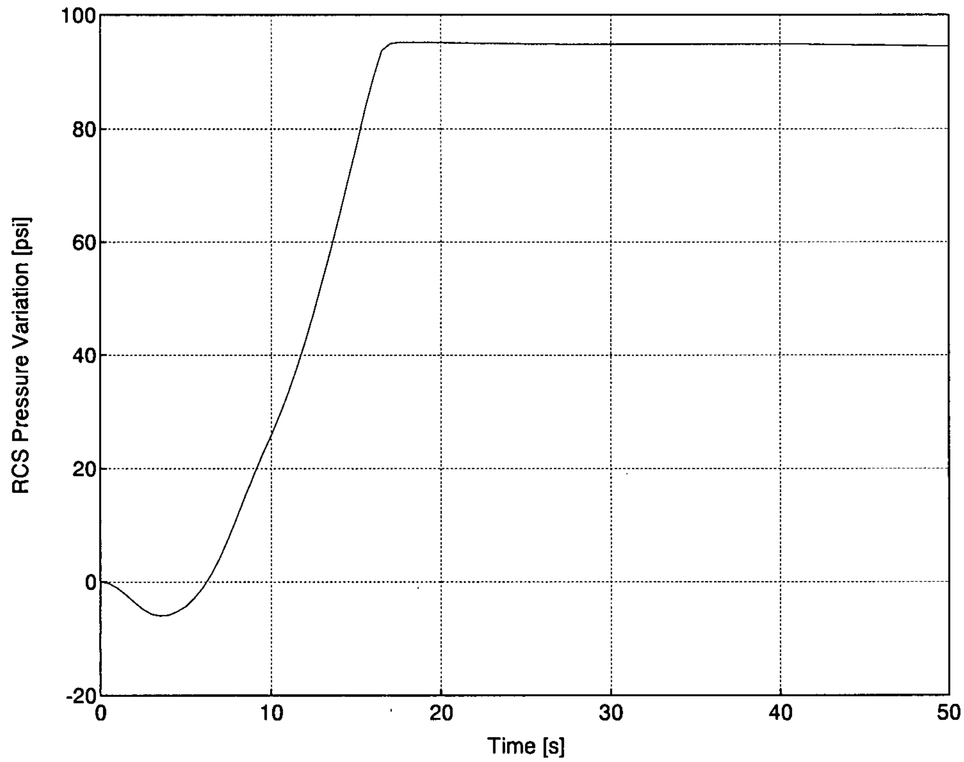


**Figure 2 Primary coolant pressure variation (heat input case)**

### 3. Evaluation

Figure 3 and 4 show representative P-T Limit Curves for heatup and cooldown, respectively (Same as Figure 5.3-2 and 5.3-3 of DCD Chapter 5). In low temperature range (below approx. 125 deg F), the limit is flat and minimized. In this range, the upper limit is approx. 620 psi.

From the results of analyses, the largest pressure variation is 120 psi (mass input case). As initial pressure condition is 400 psi, so the maximum pressure is 520 psi. This pressure is lower than the upper limit, approx 620 psi. therefore, LTOP system meets the Appendix G limits.

In P-T Limit Curve for cooldown, at lower temperature (below 95 deg F) the upper limit is lower than flat line. As this pressure limit is too close to RHR operating pressure (400 psi), it is difficult to prevent from overpressure by some system. Therefore, when RCS temperature is in this lower temperature range, sufficient open area should be provided to ensure not to exceed this pressure limit (e.g, Reactor vessel head should be removal).

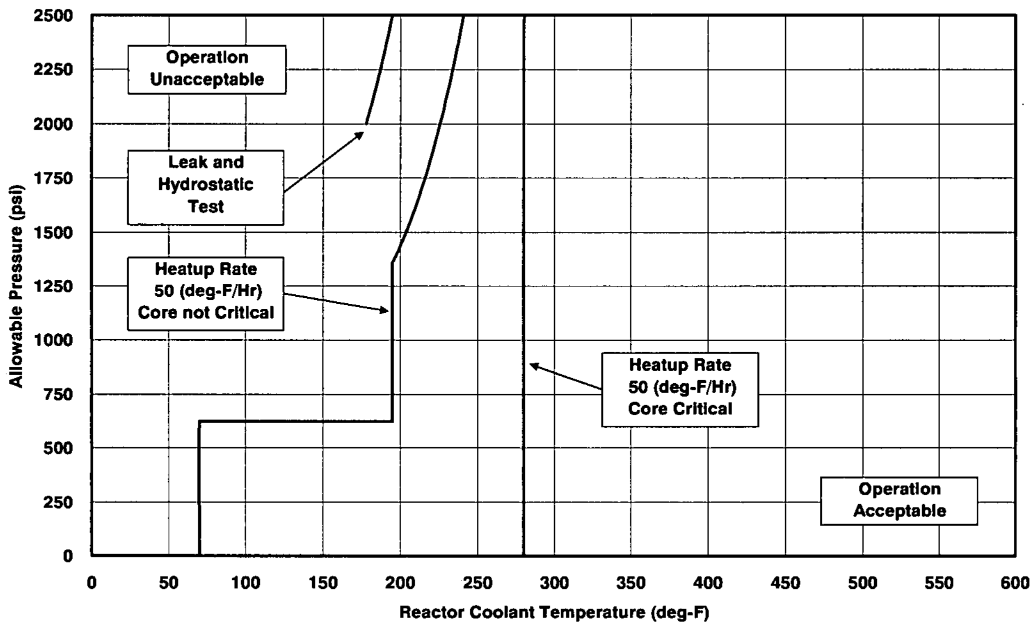


**Figure 3 Representative P-T Limit Curve for Heatup up to 60EFPY**

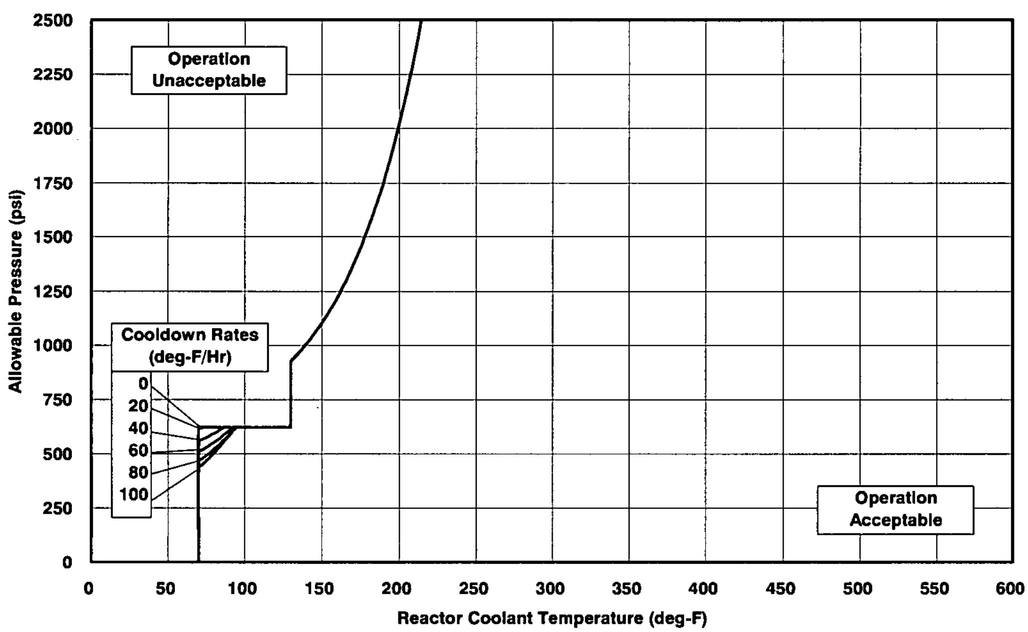


**Figure 4 Representative P-T Limit Curve for Cooldown up to 60EFPY**