US-APWR REACTOR VESSEL
Pressure and Temperature Limits Report

June 2009

Prepared:

Hiroyuki Sakamoto, Engineer
Component Designing Section
Nuclear Plant Component Designing Department

Jun 1, 2009
Date

Reviewed:

Takatoshi Hirota, Acting Manager
Component Designing Section
Nuclear Plant Component Designing Department

6/1/09
Date

Approved:

Yurugi Kanzaki, Manager
Component Designing Section
Nuclear Plant Component Designing Department

6/1/09
Date

Approved:

Hidehito Mimaki, Deputy Manager
Nuclear Plant Component Designing Department

Jun. 1. '09
Date

Approved:

Hitoshi Kaguchi, Manager
Nuclear Plant Component Designing Department

June 1, 2009
Date

Revision History

Revision	Page	Description
0	All	Original Issue

MITSUBISHI HEAVY INDUSTRIES, LTD.
16-5, Konan 2-chome, Minato-ku
Tokyo 108-8215 Japan

Abstract

This report contains the generic pressure and temperature limit curves applicable to the US-APWR standard design and is submitted as part of the US-APWR Design Certifiicate.

The evaluations are based on the bounding properties of the reactor vessel beltline region materials and projected fluence.

This report is prepared following NRC Generic Letter 96-03.

Tables of Contents

List of Tables

List of Figures

List of Acronyms

The following list defines the acronyms used in this document.

CS/RHR	Containment Spray/Residual Heat Removal
DCD	Design Control Document
EFPY	Effective Full Power Years
EOL	End-of-Life
LCO	Limiting Condition of Operation
LTOP	Low Temperature Overpressure Protection
P-T	Pressure-Temperature
PTLR	Pressure and Temperature Limits Report
PTS	Pressurized Thermal Shock
RT_{PTS}	Pressurized Thermal Shock Reference Temperature
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RHRS	Residual Heat Removal System
RV	Reactor Vessel
RT_{NDT}	Reference Nil Ductility Temperature
USE	Upper-Shelf Energy

1.0 INTRODUCTION

This report presents the Reactor Coolant System (RCS) Pressure-Temperature (P-T) limits for the US-APWR standard design in accordance with the requirements of Technical Specification 5.6.4. In addition, the requirements of the reactor vessel material surveillance program are discussed.

The following Technical Specification Limiting Condition of Operation (LCO) is addressed in this report:

LCO 3.4.3 RCS Pressure and Temperature (P-T) Limits

The methodologies used to determine the RCS P-T limits are described in US-APWR Design Control Document (DCD) Subsection 5.3.2.1.

This report covers the US-APWR operation until the plant end-of-life (EOL) of 60 Effective Full Power Years (EFPY).

2.0 OPERATING LIMITS

RCS P-T Limits

The RCS P-T limits presented in this report consist of the RCS (except the pressurizer) temperature rate-of-change limits and P-T limits during heatup, cooldown, inservice leak and hydrostatic testing, and criticality. The P-T limits for the US-APWR are based on the methodology presented in US-APWR DCD Subsection 5.3.2.1.

The RCS P-T limits are developed based on the evaluations of the beltline region material. The material properties applied are the bounding material properties specified for the standard US-APWR reactor vessel and are those described in US-APWR DCD Subsection 5.3.1.1.

The RCS P-T limits calculated for selected heatup and cooldown rates for the US-APWR are the same as the curves described in US-APWR DCD Subsection 5.3.2.1.

LTOP System

The LTOP System acts as a backup to the reactor operators to mitigate RCS pressurization transients at low temperatures so that the integrity of reactor coolant pressure boundary (RCPB) is not compromised by violating the pressure and temperature limits of 10 CFR 50 Appendix G. The reactor vessel is the limiting RCPB component for demonstrating such protection.

The LTOP system for US-APWR consists of the containment spray/residual heat removal (CS/RHR) pump suction relief valves, which are installed in the RHR system (RHRS). The valves provide low temperature overpressure protection for RCS components when the RHRS is aligned to the RCS to remove decay heat during plant shutdown and startup operations. The LTOP system is described in US-APWR DCD Subsection 5.2.2.

Reactor Vessel Material Surveillance Program

Reduction in ductility from neutron radiation results in an increase in the reference nil ductility temperature (RT_{NDT}) and a reduction of the upper-shelf energy (USE) for the reactor vessel beltline materials, including welds. For the US-APWR, these quantities are predicted at EOL (60 EFPY) using the methods described in Reference 1. Forecast properties of the limiting material are used to establish P-T limits for heatup and cooldown curves and LTOP setpoint.

The guide baskets for the surveillance capsules are located so that the lead factors (ratio of the neutron flux at the location of the capsule to that at the reactor vessel inner surface at the peak fluence location) of the capsules are between 2 and 3.

The above range for the lead factors is specified to monitor the embrittlement properties of the reactor vessel materials in the future, and takes into consideration the recommendations of Reference 2. The calculated lead factors of the capsules for the US-APWR reactor vessel are 2.7 for two (2) capsules (Type A), 2.4 for two (2) capsules (Type B) and 2.1 for two (2) capsules (Type C).

2.1 RCS Temperature Rate-of-Change Limits (LCO 3.4.3)

2.1.1 Maximum Heatup Rate

The RCS heatup rate limit is 50°F in any 1-hour period.

2.1.2 Maximum Cooldown Rate

The RCS cooldown rate limit is 100°F in any 1-hour period.

2.1.3 Maximum Temperature Change During Inservice Leak and Hydrostatic Testing

During inservice leak and hydrostatic testing operations above the heatup and cooldown limit curves, constant RCS temperature is assumed.

2.2 P-T Limits for Heatup, Cooldown, Inservice Leak and Hydrostatic Testing, and Criticality (LCO 3.4.3)

The limiting materials and adjusted reference temperatures at the 1/4t and 3/4t locations for the US-APWR reactor vessel are presented in Table 2-1. Adjusted reference temperatures are determined based on the chemical composition limits for the US-APWR reactor vessel, described in the US-APWR DCD Section 5.3.

The pressurized thermal shock reference temperature (RT_{PTS}) values for the US-APWR reactor vessel are calculated in accordance with Reference 4 and shown in Table 2-1. The results of the material surveillance program will be used to verify the validity of ΔRT_{NDT} that is the basis for the P-T limit curves. As the surveillance capsule results become available, the projected fluence and the RT_{NDT} calculations will be adjusted, if necessary. The development of new P-T curves may become necessary from these adjustments.

For the standard US-APWR, the RT_{PTS} values at EOL are not expected to exceed the pressurized thermal shock (PTS) screening criteria of Reference 4.

2.2.1 Calculation of Chemistry Factors using Surveillance Capsule Test Results

The bounding material specification for the copper and nickel weight percent values for the US-APWR reactor vessel beltline region materials are used to calculate the chemistry factors in accordance with Reference 1. The calculation results of the Chemistry Factors for the US-APWR reactor vessel are summarized in Table 2-1. Once two or more surveillance data sets are available after capsules have been removed from the US-APWR reactor vessel; this data will be used to calculate chemistry factor values per Position 2.1 of the Reference 1.

2.2.2 P-T Limits for Heatup, Inservice Leak & Hydrostatic Testing, and Criticality

The P-T limit curves for heatup, inservice leak and hydrostatic testing, and criticality, based on the limiting material properties for the US-APWR reactor vessel, are specified in Figure 2-1.

2.2.3 P-T Limits for Cooldown

The P-T limit curves for cooldown, based on the limiting material properties for the US-APWR reactor vessel, are specified in Figure 2-2.

2.3 LTOP System Setpoint

The LTOP system setpoint is 470 psig as shown in Table 5.2.2-2, DCD Subsection 5.2.2. The validity of this system is verified in Attachment A of Reference 6. The setpoint is to be updated when revised P-T limits conflict with the LTOP system limits.

2.4 Reactor Vessel Material Surveillance Program

The reactor vessel material surveillance capsule withdrawal schedule for the US-APWR is provided in Table 2-2.

Table 2-1 Calculation of RT_{NDT} / RT_{PTS} at EOL (60EFPY)

Location		$RT_{NDT(U)}$ (1) (°F)	f (2)	FF (3)	CF (4) (°F)	ΔRT_{NDT} (5) (°F)	σU (6) (°F)	$\sigma\Delta$ (7) (°F)	M (8) (°F)	RT_{NDT} (9) (°F)
Beltline Region Forgings	ID	0	0.98	0.99	31	30.8	17	15.4	45.9	76.7
	1/4-T		0.52	0.82		25.4	17	12.7	42.4	67.8
	3/4-T		0.15	0.50		15.6	17	7.8	37.4	53.0
Beltline Region Weld	ID	-20	0.85	0.95	108	103.1	17	28	65.5	148.6
	1/4-T		0.45	0.78		84.3	17	28	65.5	129.8
	3/4-T		0.13	0.47		50.9	17	25.5	61.2	92.1

Notes:

(1) Initial RT_{NDT} value.
(2) Fluence f (10^{19}n/cm^2, E > 1 MeV) at a depth of a (inch) based on the fluence fo (10^{19}n/cm^2, E > 1 MeV) at the ID is calculated by;
f = fo exp(-0.24 × a), where fo is the fluence of ID from Reference 3.
(3) FF = Fluence Factor = $f^{(0.28 - 0.10 \cdot \log F)}$
(4) Values from Table 1 and Table 2 of 10 CFR 50.61 for Cu = 0.05 wt% and Ni = 1.0 wt% for the forgings, and Cu = 0.08 wt% and Ni = 0.95 wt% for the weld material.
(5) ΔRT_{NDT} = CF × FF
(6) Standard deviation for $RT_{NDT(U)}$. 17°F selected from Table-P Footnote (5) of Reference 5.
(7) Standard deviation for ΔRT_{NDT}. $\sigma\Delta$ = smaller of 17 or 0.5 × ΔRT_{NDT} for the forgings, and smaller of 28 or 0.5 × ΔRT_{NDT} for the weld material
(8) Margin determined by M = $\sqrt{\sigma U^2 + \sigma\Delta^2}$
(9) RT_{NDT} at EOL (RT_{PTS}) calculated by;
$RT_{NDT} = RT_{NDT(U)} + \Delta RT_{NDT} + M$

Table 2-2 US-APWR Reactor Vessel Material Surveillance Program - Withdrawal Schedule

Sequence	Capsule Type	Withdrawal Schedule	Note
1st	A	Approx. 3 EFPY	At the time when the predicted ΔRT_{NDT} of the capsule material is approximately 50°F.
2nd	A	Approx. 12 EFPY	At the time when the accumulated neutron fluence of the capsule corresponds to the approximate 32EFPY fluence at the reactor vessel inner wall location.
3rd	C	Approx. 29 EFPY	At the time when the accumulated neutron fluence of the capsule corresponds to the peak fluence 60EFPY (not less than once or greater than twice the 32EFPY) at the reactor vessel inner wall location.
4th	B or C	Standby	Supplemental
5th	B or C	Standby	Supplemental
6th	B or C	Standby	Supplemental

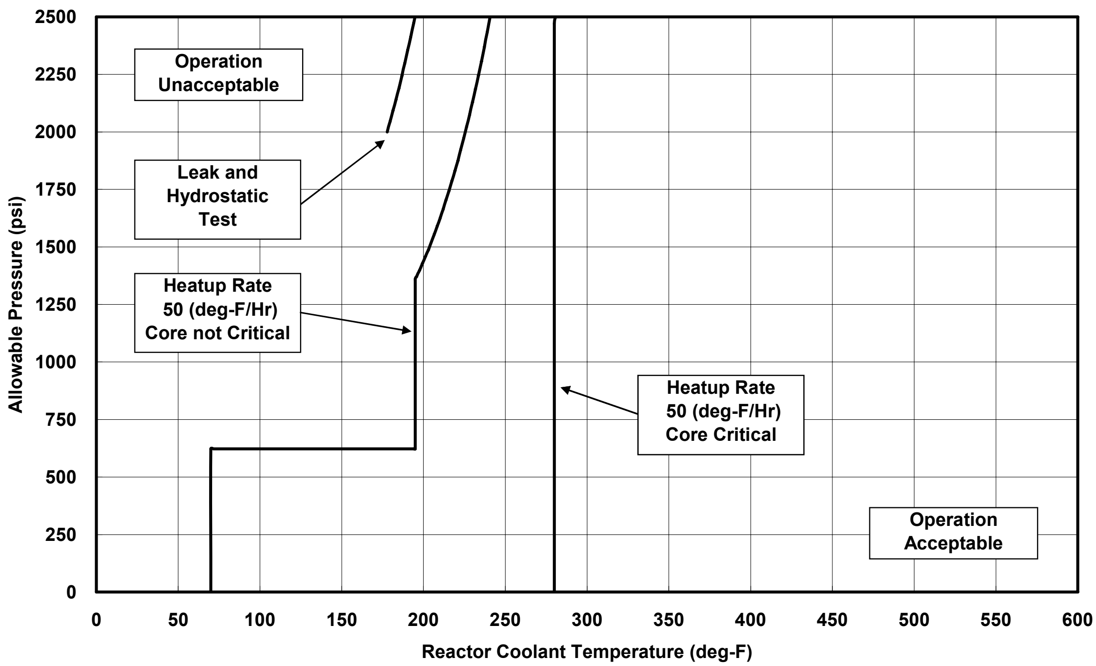

Figure 2-1 Generic P-T Limit Curve for Heatup up to 60EFPY

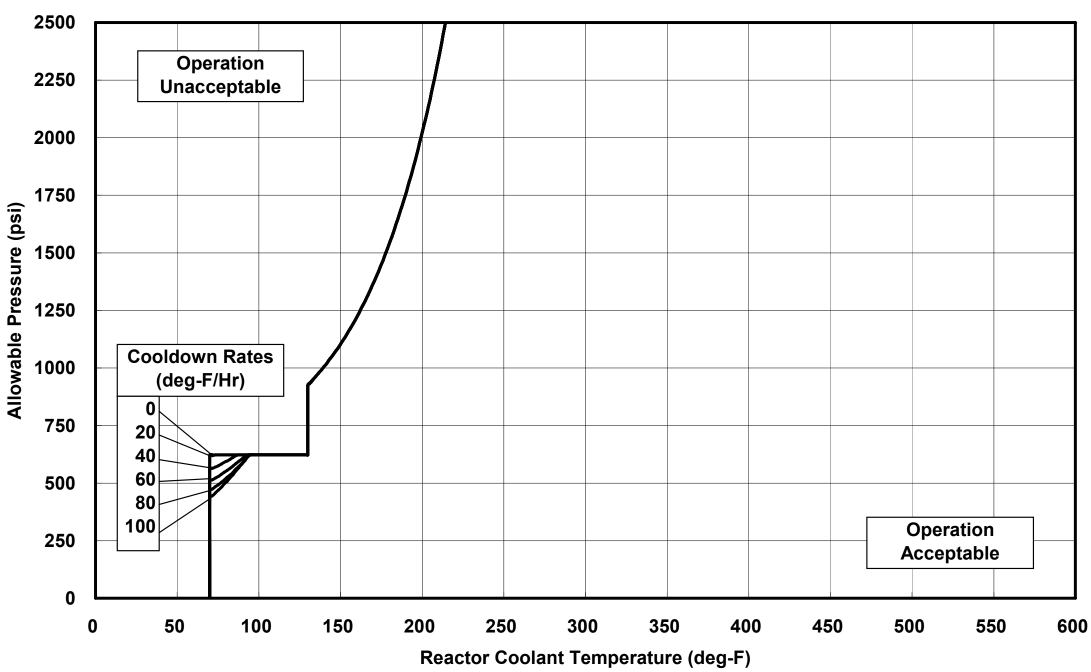

Figure 2-2 Generic P-T Limit Curve for Cooldown up to 60EFPY

3.0 REFERENCES

1. Radiation Embrittlement of Reactor Vessel Materials, Regulatory Guide 1.99, Rev.2, May 1988.

2. Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels, ASTM E-185-82.

3. Calculation Methodology for Reactor Vessel Neutron Flux and Fluence, MUAP-09018, July 2009.

4. Fracture toughness requirements for protection against pressurized thermal shock events, NRC Regulations Title 10, Code of Federal Regulations, 10 CFR 50.61.

5. Pressurized Thermal Shock (PTS), U.S.NRC, SECY-82-465.

6. MHI's Response to US-APWR DCD RAI No.103-1448 Revision 0, UAP-HF-08303, December 2008.