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Comparative Study of P-T Limit Curves between 1998 ASME and 2017 ASME Code Applied to Typical OPR1000 Reactors

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ABSTRACT

The integrity of the Reactor Pressure Vessel (RPV) is affected by the neutrons bombarding the vessel wall leading to embrittlement. This irradiation-induced embrittlement leads to reduction in the fracture toughness of RPV materials. This paper presents a comparative study of typical Optimized Power Reactor (OPR)1000 reactor pressure-temperature (P-T) limit curves using the pre-2006 American Society of Mechanical Engineers (ASME) editions used in the power plant and the current ASME edition of 2010. The current ASME Code utilizes critical reference stress intensity factor based on the lower bound of static, while the Pre-2006 ASME editions are based the critical reference stress intensity factor based on the lower bound of static, dynamic and crack arrest. Model-Based Systems Engineering approach was used to evaluate ASME Code Section XI Appendix G for generating the P-T limit curves. The results obtained from this analysis indicate decrease in conservatism in P-T limit curves constructed using the current 2017 ASME code, which can potentially increase operational flexibility and plant safety. Hence it is recommended to use ASME code edition after 2006 be used in all operating nuclear power plants (NPPs) to establish P-T limit curve.

Key Words : Typical OPR1000 Unit, Model-Based Systems Engineering, curve, curve, P-T limit curve

Nomenclature

ART	:	Adjusted reference temperature
RT_{NDT}	:	Reference temperature for nil-ductility transition
CF	:	Chemistry factor
FF	:	Fluence factor
IRT_{NDT}	:	Reference temperature for un-irradiated material.
K_{IC}	:	Reference critical stress intensity factor based on lower bound of static fracture toughness
K_{IR}	:	Reference critical stress intensity factor based on lower bound of all static, dynamic and crack arrest toughness test
K_{IM}	:	Stress intensity factor for membrane stress due to pressure and is defined in G-2214.1 of ASME Sec. XI, Appendix G(1)
K_{IT}	:	Radial thermal stress intensity factor
M_m	:	Membrane correction
P	:	Internal RPV pressure
r	:	Internal radius of RPV
t	:	Reactor vessel wall thickness
CR	:	cooldown rate
HU	:	Heat-up rate

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1. Introduction

Systems engineering approach can be used to evaluate the structural integrity of RPV through the application of ASME Section XI Appendix G requirements. These requirements comply with the US Nuclear Regulatory

Commission (USNRC) 10CFR50 Appendix G[1] which gives the fracture toughness requirements for the ferritic material used in the reactor coolant pressure boundary of light water nuclear power reactors to provide adequate margins of safety during Level A and B Service Conditions. In order to ensure an acceptable safety margin against the brittle failure of RPV, plant operators are required to adhere to P-T limit curves to maintain plant operation within an acceptable operational window. Any operation outside P-T limit curves will cause undesirable high tensile stresses due to thermal gradient on the inner surface during cooldown and the outer surface during heat-up of RPV. To avoid this phenomenon, the utility designers have installed an inadvertent relief actuation valve and established a Low-Temperature Overpressure Protection (LTOP)[2] system which must be set to preclude excursions beyond the code allowable pressure for the reactor vessel.

The P-T limit curves are calculated and constructed based on a predicted or a calculated level of embrittlement at a Specified Effective Full Power Years (EFPY). This implies that P-T limits[3] should be adjusted periodically based on the measured shift in Reference Temperature for Nil-Ductility Transition (RT_{NDT}) by testing the reactor vessel surveillance material samples. These updates are necessary for the continued operation of NPP, power uprate, life extension, or when the existing P-T limit curves expire.

2. Project Planning and Control

2.1 Scope

The scope of this study includes collecting material properties of typical OPR1000 nuclear reactor, operation fluence levels, operating temperature, and developing heat up and cooldown profiles. The regulatory guide 1.99 Revision 2 is used to predict the adjusted reference temperature (*ART*). Using applicable the current 2017 ASME code and the Pre-2006 ASME code, P-T limit curves are generated to evaluate the integrity of the reactor vessel.

2.2 Objectives

The objective of this paper is to apply Systems Engineering approach to carry out a comparative study of the P-T limit curves of typical OPR1000 unit. Using the current ASME 2017 Code and that of the utility owner which based on earlier editions of ASME Code prior to 2006, different P-T limit curves will be generation for comparison.

2.3 Deliverables

The deliverable of the study include:

i. Temperature gradient for reactor start ups
ii. Temperature profile for the reactor shutdown
iii. Current fluence levels for ¾ and ¼ of RPV
iv. Adjusted reference temperature for ¾ t and ¼ t of RPV
v. Stress intensity factors K_{IC} or K_{IR}, K_{IT}, K_{IM} values
vi. P-T limit curves corresponding to both K_{IC} and K_{IR} Curves

2.4 Operational Concept

The P-T limit curve defines the operational concept of reactor vessel for both service level A and B conditions. The operating envelope reduces progressively because of periodic adjustment to accommodate the effects of irradiation embrittlement of the RPV material as shown in Figure 1 below.

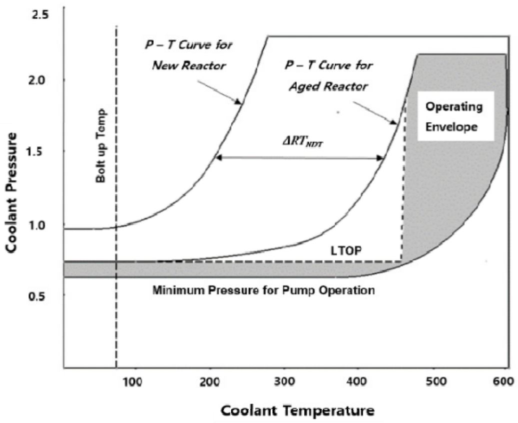

Fig. 1 P-T[3] operating envelop for RPV

2.5 Work Breakdown Structure

This study has been broken down into finer work packages which make it easier to comprehend the ASME

Code requirements. The Figure 2 below illustrates this decomposition process.

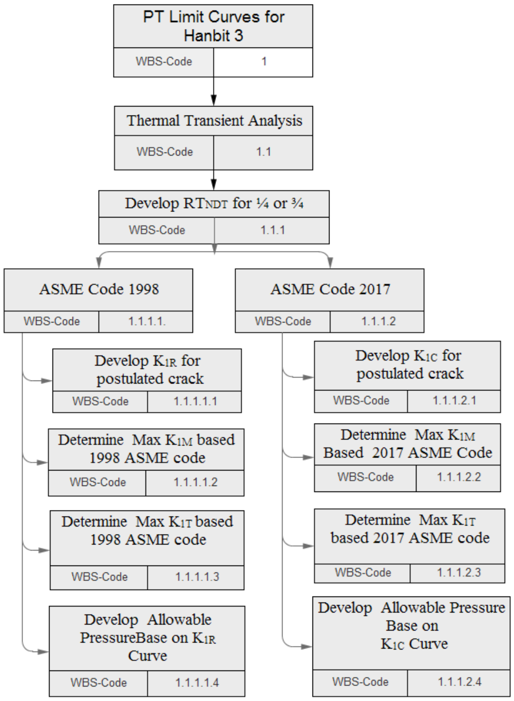

Fig. 2 Work breakdown structure

2.6 Risks and mitigation measures

The major risk in this study is the unavailability of credible data for typical OPR1000 RPV. These include reactor dimensions, material properties including the restricted elements and the operating fluence levels. To ensure credible result is achieved, primary data from typical OPR1000 final safety analysis report (FSAR) and Korea Atomic Energy Research Institute (KAERI) data have been used in this study. Also, correct mesh quality is adopted during modeling to obtain correct temperature gradient during heat and cooldown events.

3. System Engineering Processes

The general design process of pressurized pressure following Systems Engineering approach[(4),(5)] is shown in figure 3 below.

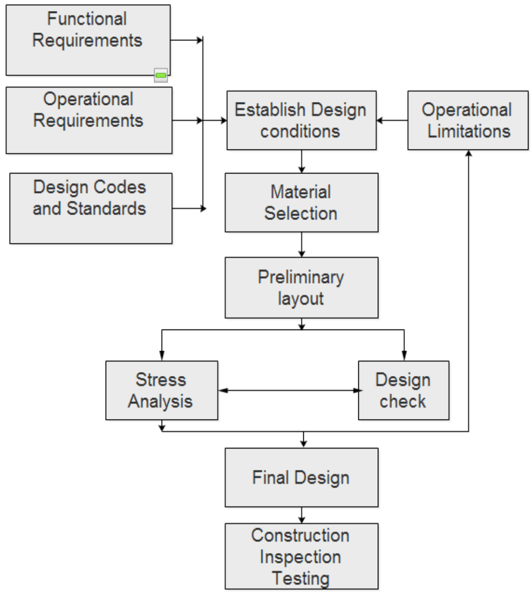

Fig. 3 Reactor pressure vessel design procedure

3.1 Design requirements

The General Design Criterion (GDC) 31 of the 10CFR50 Appendix A[(6)] requires RPV as a pressure boundary component to be designed with adequate margin to assure that when stressed under Service Level A, Level B or postulated accident conditions. The pressure boundary shall behave in a non-brittle manner with low likelihood of a rapidly propagating crack. The design shall also be consistent with the change of material properties due to differences in service temperatures, effects of radiation embrittlement and distribution flaw sizes in RPV. The initial Charpy upper-shelf energy must not be less than 75 ft-lb and not less than 50 ft-lb throughout the life of the reactor vessel[(7)].

The maximum postulated crack defect is a sharp-edged surface crack defect oriented axially for plates, forgings and axial welds, and circumferentially for circumferential welds. The depth of crack is 1/4 of the vessel thickness exist at the inner and outer surface of the vessel with a length equal to 1.5 of the vessel thickness range thickness range between 4 inches to 12 inches[(2)]. The reference critical stress intensity factor applicable by current ASME Code is based on K_{IC} curve while the pre-2006 ASME Codeis based curve.

3.2 Functional requirements

RPV provides an environment for fission reaction to take place to produce the thermal heat necessary for the generation of electricity. It accommodates the system temperature and pressure whilst containing radiation products inside the core.

3.3 Operational requirements

The operational requirements include:

a. **The minimum bolt-up temperature requirements**. It should take into account any effects of radiation.

$$T_{MIN} - BOLTUP = \text{Initial } RT_{NDT} + \text{effects of radiation}$$

The effect of irradiation embrittlement is negligible and = the refuelling water temperature) = 70°F

b. **Low-Temperature Overpressure Protection requirements.** It is defined as 20% of the preoperational hydrostatic test pressure.

$$20\% \text{ of } 2500 \text{ psi} = 625 \text{ psi}$$

c. **Lowest Service Temperature requirements (LST).** It is defined to be not lower than RT_{NDT} + 100°F

d. **Normal operation requirements**. Sets the maximum heat up and cooldown rate of 100°F/hour.

e. **Pump seal requirement requirements**. The pump is affected by narrow operation window of P-T Curves.

4. Verification and validation process

The V-model[(4),(5)] has been used for verification and validation process because of its effectiveness to break down the project requirements on the left side and verification criteria on the right side of the model in all stages of the project as shown in figure 4 below.

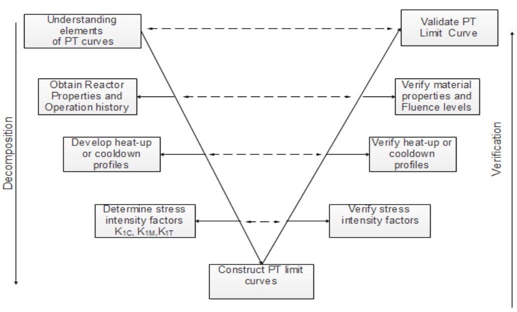

Fig. 4 Verification and validation process

5. OPR1000 Reactor and Material Properties

The RPV consists of vessel flange, reactor closure head, forged rings of the upper, intermediate and lower sections, and a hemispherical bottom head. The vessel flange is first forged and then ledge machined on the inner surface to support core support barrel. The reactor closure head is fabricated and bolted to the RPV. The doom and the flange are welded together to form reactor closure head. The three forged ring, reactor closure head, and bottom hemispherical head are joined together by welding as shown in figure 5 below.

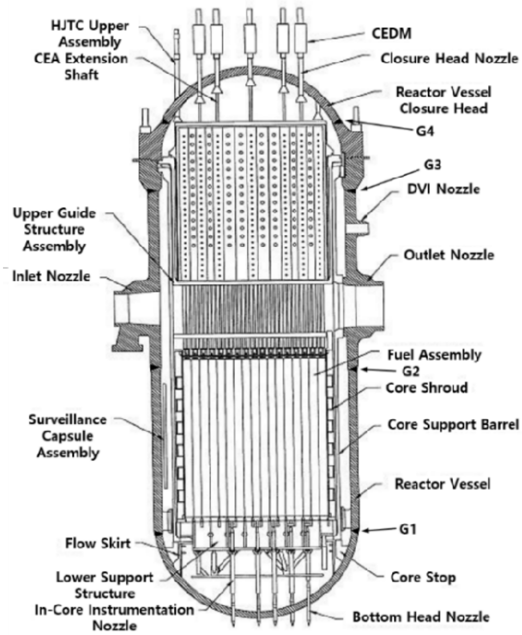

Fig. 5 Typical OPR1000 RPV weld regions (indicated by G1, G2, G3, G4)

5.1 Reactor Vessel Material

The material for RPV is SA-508 Grade 3 Class 1 with material properties defined in the ASME BPVC[(8)]. In order to reduce the effects neutron irradiation, the amount Copper, Nickel, Phosphorous and Manganese in RPV fabrication are restricted. The percentages of Copper and Nickel for typical OPR1000 are given in 1 below.

Table 1 Average Cu and Ni Wt. % values-beltline materials

	Percentage (wt. %)	
Elements	Cu	Ni
Base metal	0.05	0.78
Weld metal	0.02	0.056

5.2 Initial RTNDT for OPR1000

The Initial RT_{NDT} of un-irradiated OPR1000 reactor vessel shell products and weld material are given in 2 below.

Table 2 Initial for RPV and surveillance specimen

Region	Initial
Vessel beltline (base metal)	10°F
Vessel beltline surveillance	10°F
Vessel weld material	-50°F
Vessel weld surveillance	-50°F

5.3 Operating Fluence Levels

It is believed that neutrons with threshold energy greater than 1*MeV* are responsible RPV embrittlement. The estimated peak fluence level for typical OPR1000 are tabulated in table 3 below.

Table 3 Peak fluence levels for typical OPR1000

Effective Full-Power Years	Maximum fluence level, 10^{19}n/cm^2, E>1MeV)
20	1.167
24	1.404
32	1.878

6. Modelling and Calculations

6.1 Methodology

The procedure for constructing deterministic P-T limit curves is summarized in Figure 6 below with defined input and output parameters as well as the process flow.

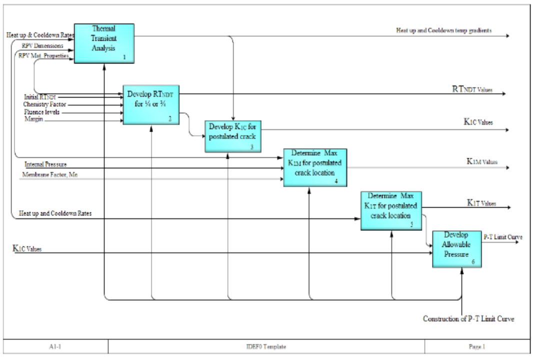

Fig. 6 Procedure for constructing of P-T limit curves

6.1.1 Determining allowable pressure using ASME 2017 Code

The determination of the allowable pressure to prevent brittle fracture is given by ASME Section XI Appendix G of 2017 edition[2].

$$K_{IC} > 2\,K_{IM} + K_{IT} \tag{1}$$

Where

$$K_{IM} = M_m * \frac{Pr}{t} \tag{2}$$

and

$$K_{IT} = 0.953 * 10^{-3} * CR * t^{2.5} \tag{3a}$$

$$K_{IT} = 0.753 * 10^{-3} * HU * t^{2.5} \tag{3b}$$

The K_{IT} defined by equations (3a) and (3b) corresponds to the maximum temperature difference between vessel inner-surface to outer-surface throughout heat-up and cooldown operation.

k_{IT} at any point is given by:

$$k_{IT} = \frac{Max\ K_{IT}}{\Delta T} \tag{4}$$

K_{IC} is defined by:

$$K_{IC} = 33.2 + 20.734e^{[0.02(T-RT_{NDT})]} \tag{5}$$

From equations (1) to (5), the pressure at any time is a function of operating pressure and is given by:

$$P = (K_{IC} - K_{IT}) * {}^{t}/_{r} * {}^{1}/_{2} * 1/M_m \tag{6}$$

6.1.2 Determining allowable pressure using ASME 1998 code

Since Korea Hydro Nuclear Power (KHNP) are still using pre-2006 ASME methodology, the governing equations are as follows[9]:

$$K_{IR} > 2\,K_{IM} + K_{IT} \tag{7}$$

Where

$$K_{IM} = M_m \times (\sigma_m) \tag{8}$$

and $\sigma_m = \Pr/t$, which is same as in (2)

$$K_{IR} = 26.78 + 1.223e^{[0.0145(T-RT_{NDT})+160]} \tag{9}$$

And

$$K_{IT} = M_t \times \Delta T \tag{10}$$

From equations (7) to (10), the pressure at any time is a function of operating pressure and is similarly obtained as before

$$P = (K_{IR} - K_{IT}) * {}^{t}/_{r} * {}^{1}/_{2} * 1/M_m \tag{11}$$

6.2 Setting up of RPV Model for Thermal Transient Analysis

To determine the RPV thermal gradient and maximum temperature gradient during heat-up and cooldown, a 2D axisymmetric model representing the active core region was created in ANSYS Workbench 19.0, a general purpose commercial FEM analysis software, and material properties specified for transient thermal analysis. Using the reactor vessel wall thickness for typical OPR1000, 8.06 inches and the active core height of 150 inches, the model was set as shown in figure7 below.

The maximum heat-up or cooldown rate of 100°F/hour was then defined in the analysis set-up and RPV beltline temperature profiles at the inner Surface, ¼ thickness, ¾ thickness and outer surface noted by setting the probes as illustrated in figure 8 below.

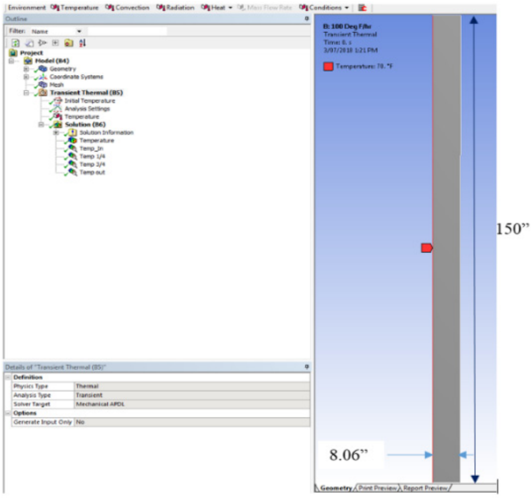

Fig. 7 Thermal transient boundary conditions

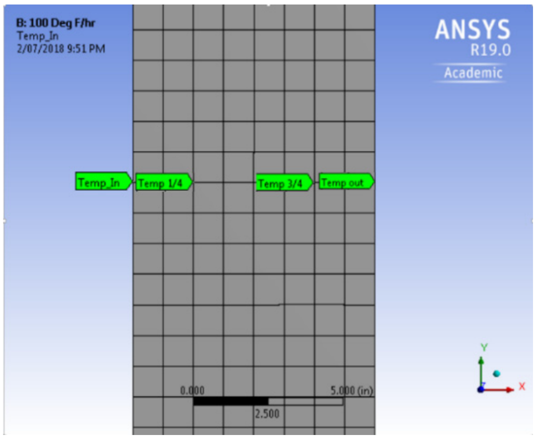

Fig. 8 Temperature probes at various RPV regions

6.3 Determination of ART Values

From the percentages of Copper and Nickel, chemistry factors were read directly from RG (Regulatory Guide) 1.99 Rev. 2[10] and results used in the computation of *ART*.

The Adjusted Reference Temperature (ART) is given as:

$$ART = IRT_{NDT} + \Delta RT_{NDT} + Margin \tag{12}$$

IRT_{NDT} is a reference temperature for the un-irradiated material.

$\triangle RT_{NDT}$ is a function of Fluence Factors and Chemistry Factors.

$$\Delta RT_{NDT} = CF \times FF \tag{13}$$

Where:

CF (°F): Chemistry Factor is function of Cu and Ni

$$FF = f^{(0.28-0.1 log f)} \tag{14}$$

Also;

$$FF = fluence\ factor\ (in\ 10^{19}\ n/\mathrm{cm}^2,\ E = 1MeV)$$

The neutron fluence, at any depth in the vessel wall is calculated using a method that conforms to the guideline of RG 1.99 Rev. 2.

$$f = f_{surf}(e^{-0.24x}) \tag{15}$$

"Margin" is the amount of temperature added to account for uncertainties in the evaluating the values of Initial RTNDT, Cu and Ni contents, fluence, and calculation procedures[10]. It is given by:

$$Margin = 2\sqrt{\sigma_1{}^2 + \sigma_\Delta{}^2} \tag{16}$$

Where

σ_1 = Standard deviation for the initial RT_{NDT}

σ_Δ= Standard deviation of ΔRT_{NDT}

Table 4 *ART* for the (typical OPR1000) at 32 EFPY

¼ t Region	Base metal	Weld metal
Chemistry Factor, CF($^\circ F$)	31.00	36.64
Surface Fluence, $\times 10^{19}$ cm^2	1.88	1.88
Fluence, f(1/4) $\times 10^{19}$ n/cm^2	1.16	1.16
IRT_{NDT}, $^\circ F$	10.00	-50.0
ΔRT_{NDT} at ¼ t, $^\circ F$	32.27	38.14
Margin at ¼ t, $^\circ F$	32.27	28.00
ART	74.54	16.14

¾ t Region	Base metal	Weld metal
Chemistry Factor (CF), $^\circ F$	31.00	36.64
Surface Fluence, $\times 10^{19}$ cm^2	1.88	1.88
Fluence, f(3/4) $\times 10^{19}$ n/cm^2	0.44	0.44
IRT_{NDT}, $^\circ F$	10	-50.0
ΔRT_{NDT} at ¾ t, $^\circ F$	23.93	28.28
Margin at ¾ t, $^\circ F$	23.93	28.28
ART	57.85	6.56

The standard deviation for ΔRT_{NDT} (σ_Δ) is 28$^\circ F$ for welds and 17$^\circ F$ for base metal, and should not exceed 0.50 times the mean value of ΔRT_{NDT} [10].

Using the peak fluence for neutrons E>1.0 MeV on the pressure vessel/base metal interface at 32 EFPY, ART values were calculated for both ¼ and ¾ surfaces of the RPV using equations (11) to (16) above. The results obtained are summarized in table 3 below.

7. Results and discussions

The P-T limit curves for 100°F/hr. cooldown and heat-up for Current ASME 2017 and pre 2006 ASME Code were constructed as shown in Fig.9 and Fig. 10 below.

The results from the two figures indicate the 2017 ASME curves are less conservative for both heat up and cooldown processes. This is due to the use of static which is higher than. There is also a decrease in the membrane correction factor (Mm) for an inside axial surface flaw from 2.95 in the 1998 ASME code to 2.63 in the current 2017 ASME code. This signifies that ASME's efforts to setup P-T limit procedures without overly conservative result and this paper shows that the results of OPR1000 case confirms ASME's new procedure indeed gives more operational margin on reactor maneuvering capabilities.

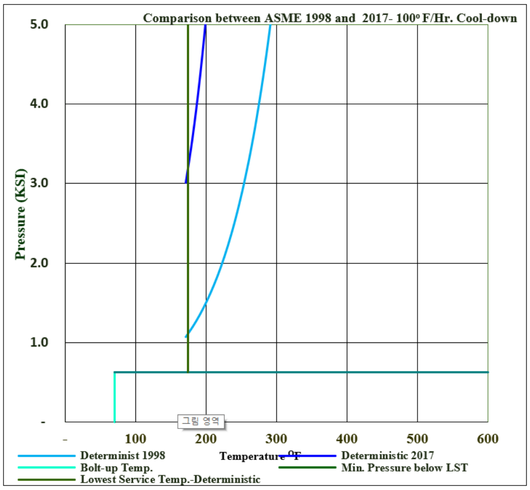

Fig. 9 Comparison of P-T limit curves prior to 2006 Code with 2017 edition – cooldown

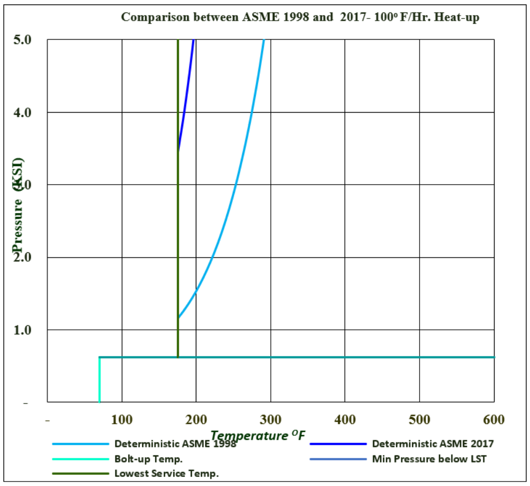

Fig. 10 Comparison of P-T limit curves prior to 2006 Code with 2017 edition – heat-up

8. Conclusions

The application of model-based systems engineering approach simplifies the application of ASME code for constructing P-T limit curves. This because of its ability to break down the project into deliverable work packages with clear input and output parameters as well as process flow. The results obtained indicate a considerable decrease in conservativeness in P-T limit curves constructed using the current 2017 ASME code. The adoption of the current fracture toughness requirements of the 2017 ASME code will increase operational flexibility and plant safety. This is because the less conservative P-T limit curves widen the operational window which in turn enhances the durability of the pump seal due to adequate cooling.

Acknowledgment

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