Proceedings of the Korean Nuclear Society Spring Meeting
Cheju, Korea, May 1996

Procedure of Pressure/Temperature Curves Generation for Brittle Fracture Prevention of Reactor Vessel

M. K. Park, Y. J. Kim, J. M. Kim, J. H. Jheon, I. K. Kim
Korea Atomic Energy Research Institute

Abstract

The purpose of this study is to estabilish the pressure/temperature curves of Reactor Coolant System for brittle fracture prevention. The pressure/temperature curve is the basis to select RC Pump and limits to operate the plant. Based on the plant operation experience, this curve should be re-generated periodically in order to ensure the structural integrity using data from the test of reactor vessel surveilance materials to compensate for the irradiation effects. This study provides the procedure of pressure/temperature curve generation in term of brittle fracture prevention of reactor vessel. Using the UCN 3&4 data, the sample pressure/temperature curve was generated, and it was compared with those of YGN 3&4 based on the stress and RT_{NDT} value.

1.0 Introduction

All components in the Reactor Coolant System(RCS) are designed to withstand the effects of cyclic loads due to RCS pressure and temperature changes. These cyclic loads occur throughout the life time of the RCS and are introduced by normal operation load transients, startup and shutdown operations, and reactor trip occurances. The maximum allowable RCS pressure at any temperature is based upon the stress limitations developed for brittle fracture prevention due to the severity and velocity of the fracture propagation. During startup and shutdown the rates of temperature and pressure changes are limited so the maximum heatup and cooldown rates satisfy the above stress limitations. The pressure/temperature(P/T) curve is basis to select the Pump and guide line for operation of the plant. Based on the plant operation experience, this curve should be re-generated periodically during plant operation using the data from the test of reactor vessel surveilance materials to compensate for the irradiation effects. This study provides the procedure of P/T curve generation in term of brittle fracture prevention of reactor vessel. And using the UCN 3&4 data, the sample P/T curve was generated, and it was compared with those of YGN 3&4 based on the stress and RT_{NDT} value.

2.0 Method of analysis

The principle of linear elastic fracture mechanics(LEFM) is used to determine the safe operational pressure and temperature conditions for light water nuclear power reactors. The basic parameter of LEFM is the stress intensity factor, K_I, which is a function of stress state and flaw configuration. The basic premise of linear elastic fracture mechanics is that unstable propagation of an existing flaw will occur when the value of K_I attains a critical value designated as K_{IC}. So, the purpose of this analysis is to show that the stress intensity factors for various parts of the pressure vessel at various operating and testing conditions will be always within the limit of the fracture toughness of the vessel material at various temperature in order to ensure the integrity of the pressure vessel.

For the Westinghouse case, only the beltline region is considered for P/T curve generation. But in this study the most limiting parts of reactor vessel are considered, i.e., beltline, nozzle, flange part. And, the P/T curve was calculated at various heatup/cooldown rates, i.e., 10°F/Hr, 20°F/Hr, 40°F/Hr, 60°F/Hr, 100°F/Hr.

Fracture Toughness

The K_{IC} is generally referred to as the fracture toughness of the material or simply the critical stress intensity factor and is depend on temperature and on the loading rate imposed on the flaws. For the nuclear power reactor coolant pressure boundary materials and components, the reference stress intensity factor, K_{IR}, is used in accordance to the requirements of ASME Code[1].

$$K_{IR} = 26.777 + 1.223 \exp\{ 0.0145 [T-(RT_{NDT} - 160)]\} \qquad (1)$$

where, T is temperature, RT_{NDT} is reference nil ductility temperature.

Since the fracture toughness of reactor vessel materials is sensitive to irradiation, reference nil ductility temperature, RT_{NDT}, of the material will increase over core life. The shift of RT_{NDT} by the irradiation is predicted by Reg. 1.99[2] and the shift value is generally dependent on the copper and nickel contents of vessel material and initial RT_{NDT}. For UCN 3&4, the initial RT_{NDT} of beltline region is the same as that of YGN 3&4, but the copper content is less than that of YGN 3&4. And, the 10 year and the end of life P/T curves were generated based on the RT_{NDT} shift.

Computation of Stress intensity factor

The stress computed by the finite element method or the interaction analysis can be expressed as primary membrane strees, σ_m, primary bending stress, σ_b, secondary membrane strees, σ_{sm}, secondary bending stress, σ_{sb} and used to compute the stress intensity factor. For YGN 3&4, these stress was calculated by the interaction analysis, whereas for UCN 3&4 the stress was calculated by the finite element method. The summation of calculated stress intensity factor, K_I values is then compared to an allowed K_{IR}. A safety factor of 2.0 is used for K_I values caused by primary stress for normal operation and 1.5 for inservice test[1].

$$K_{IR} = \{\sum_i (2K_i)\}_{primary} + \{\sum_i (K_i)\}_{secondary} \qquad (3)$$

$$K_{IR}(T, RT_{NDT}) = \sum_i \{n_i K_{Ii} (\sigma_m(P), \sigma_b(P, \delta T))\}$$

The primary stress intensity factor for the beltline transition region is composed of membrane stress only. For the beltline transition region, primary part of Eq. (3) becomes....

$$\{\sum (2K_i)\}_{primary} = \{2 K_{m,\ pressure}\}_{primary} \qquad (4)$$

The secondary stress intensity factors for the beltline transition region are composed of membrane stress and bending stress contributions. The secondary membrane stresses are due to pressure and the secondary bending stresses that are considered are thermal and those due to pressure.

$$\{\sum(K_i)\}_{secondary} = (K_{m,pressure} + K_{b,pressure} + K_t)_{secondary} \qquad (5)$$

For the nozzles, the primary stress intensity factor is computed based on the work in WRC Bullutin[3]. The primary stress intensity factor in Eq. (3) is composed only of a membrane stress contribution. That is...

$$\sum_i (2K_i)_{primary} = 2K_{Im} \qquad (6)$$

The secondary stress intensity factor for nozzles is composed only of a thermal stress contribution. Therefore, the secondary part of Eq. (3) is...

$$\sum_i (K_i)_{secondary} = K_{Ib} \qquad (7)$$

For the flange, the primary part of Eq. (3) becomes...

$$\{\sum(2K_i)\}_{primary} = \{2(K_{m,pressure} + K_{m,boltup} + K_{b,boltup})\}_{primary} \qquad (8)$$

The secondary stress intensity factors for the flange are composed of only bending stress contributions. The secondary bending stresses that are considered are thermal and those due to pressure.

$$\{\sum(K_i)\}_{secondary} = (K_{b,pressure} + K_t)_{secondary} \qquad (9)$$

3.0 Generation of P/T Limits

The types of P/T limits include the heatup and cooldown curve of various parts, minimum reactor vessel head boltup temperature, lowest service temperature(LST), maximum pressure when below the LST, isothermal, inservice leak & hydrostatic test, core critical curve. According to the above method, the P/T curve of UCN 3&4 was generated. As the results, the pressure of UCN 3&4 is higher than those of YGN 3&4, as shown in Fig. 1, which considered only the calculated stresses of YGN 3&4 and

UCN 3&4 to show the effect of stress at the pre-core condition. Fig. 2 through Fig. 6 show P/T curve of various parts at 8 Effective Full Power Year(EFPY) and 32 EFPY. And Fig. 7 and Fig. 8 are the final P/T curves of UCN 3&4. For the P/T curve of YGN 3&4 was presented in Fig. 9 & Fig. 10.

4.0 Conclusion

This study presents theoretical and technical bases for P/T curve and provides calculation procedure based on requirements. At the comparison of P/T curve of UCN 3&4 and YGN 3&4, the curves are shifted by the heatup/cooldown rates and RT_{NDT} value, so, more limited to operation of the plant. The UCN 3&4 P/T curve have more the margin than those of YGN 3&4 to determine the shutdown cooling system setpoint and operate the plant due to the less RT_{NDT} shift and the more stress value.

References

1. ASME Code, Section III, Division 1, Appendix G, Protection Against Nonductile Failure, 1989 Edition.
2. NRC R.G. 1.99, Rev.02, Radiation Embrittlement of Reactor Vessel Materials, 1988.
3. WRC Bulletin No. 175, PVRC Recommendations on Toughness Requirements for Ferritic Materials, August 1972.

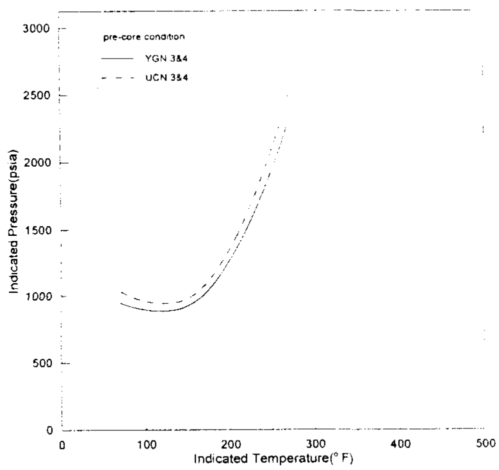

Fig. 1 Comparison of Beltline Heatup Curves

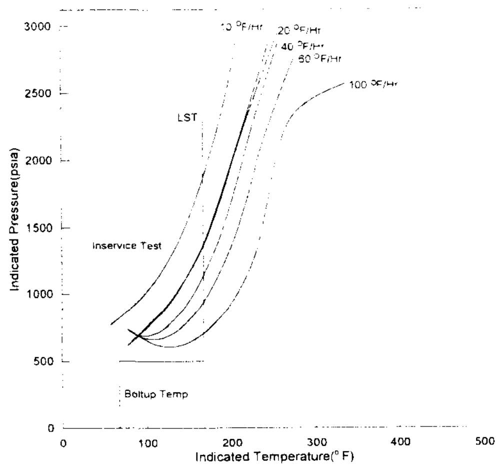

Fig. 2 Beltline Heatup Curves(8EFPY)

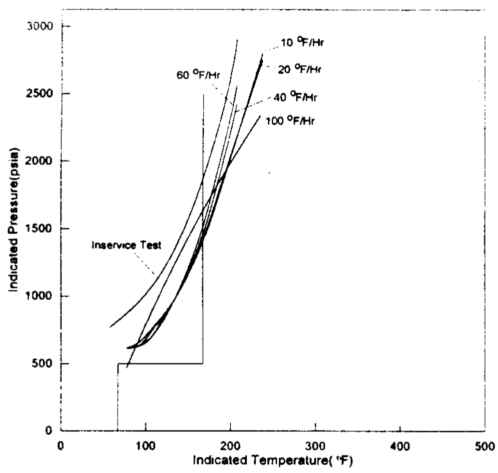

Fig. 3 Beltline Cooldown Curves(8EFPY)

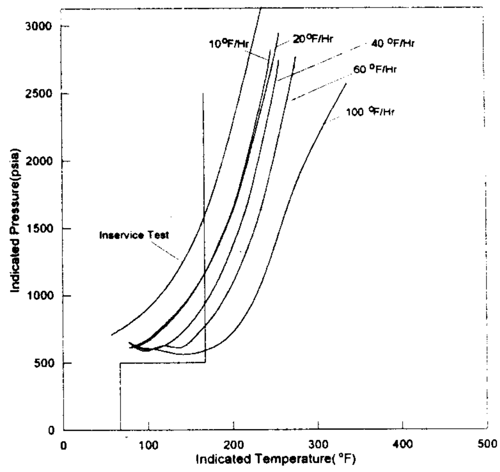

Fig. 4 Beltline Heatup Curves(32EFPY)

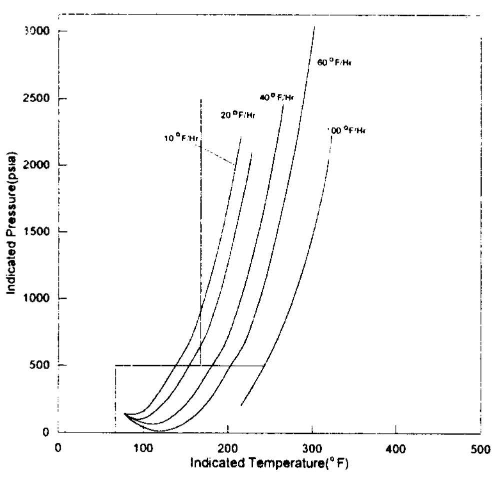

Fig. 5 Flange Heatup Curves

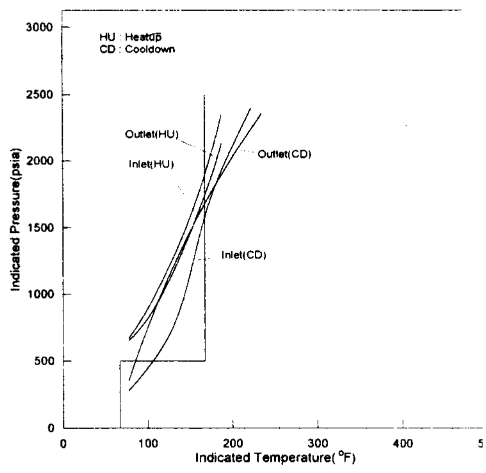

Fig. 6 Inlet & Outlet Nozzle Curves-100°F/hr

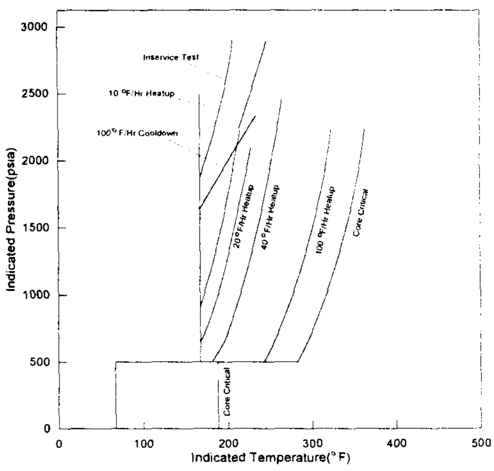

Fig. 7 RCS Pressure/Temperature Limits(8EFPY)

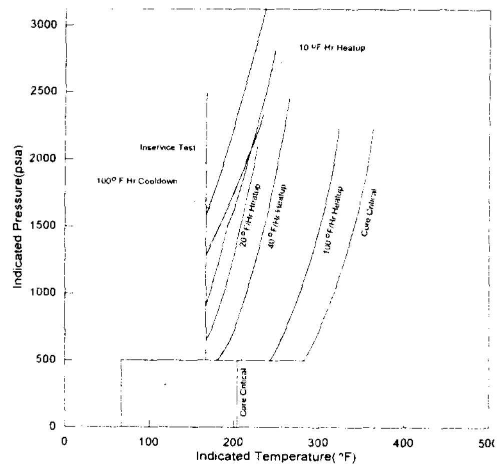

Fig. 8 RCS Pressure/Temperature-Limits(32EFPY)

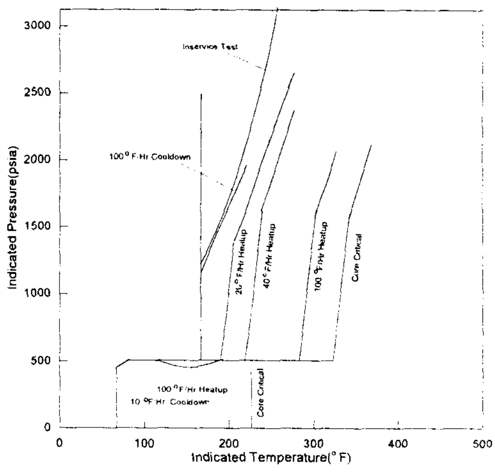

Fig. 9 YGN 3 & 4 RCS Pressure/Temperature Limits(8EFPY)

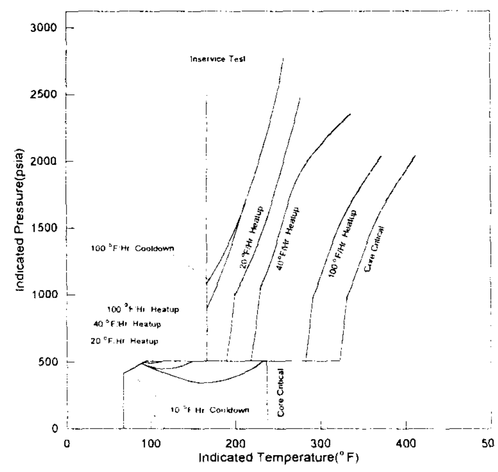

Fig. 10 YGN 3 & 4 RCS Pressure/Temperature Limits(32EFPY)