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A discussion about P-T limit curves and PTS evaluation

Cruz, J.R.B., Mattar Neto, M., Andrade,A.H.P.
CNEN/SP-IPEN - Comissão Nacional de Energia Nuclear, São Paulo, SP, Brazil

ABSTRACT: A discussion about two important issues related to reactor pressure vessel integrity is presented. These issues are the definition of Pressure-Temperature Limit Curves for Heatup and Cooldown, and Pressurized Thermal Shock evaluation. The paper is restricted to the approach used in USA, and tries to make a correlation between the several documents that have to do with these problems, starting from the basic requirements of 10CFR50 (Code of Federal Regulation, Title 10, Part 50) and going through ASME Code requirements, and USNRC Standards and Regulatory Guides. The aspects related to the radiation effects on fracture toughness of the vessel material are discussed. Also, some comments about the conservatisms and non-conservatisms of the American approach are presented.

1 INTRODUCTION

Regulatory requirements for fracture prevention in reactor pressure vessels exist in all countries operating and constructing power plants. These requirements have one important element in common in that all employ fracture mechanics concepts to assure the retention of adequate fracture prevention margins throughout the life of the nuclear power plant. However, specific approaches used to achieve this objective differ significantly from country to country. In Brazil, CNEN (Brazilian Nuclear Energy Commission) follows closely the requirements of USNRC (Nuclear Regulatory Commission), the nuclear regulatory agency in the USA. Two relevant topics concerning fracture prevention of reactor pressure vessel are treated in this paper. They are the definition of Pressure-Temperature (P-T) limit curves, and Pressurized Thermal Shock (PTS) evaluation. Some important aspects related to these issues are presented, but the primary purpose of the paper is to point out the main documents containing both the regulatory requirements and the procedures needed to show compliance with them.

Regarding the design, the original concernment to guarantee the integrity of the reactor pressure vessel is expressed in the General Design Criterion 31 - GDC 31, "Fracture Prevention of Reactor Coolant Pressure Vessel" of Appendix A, "General Design Criteria for Nuclear Power Plants" to 10CFR50 (USNRC 1993). GDC 31 requires, in part, that the reactor coolant pressure boundary be designed with sufficient margin to assure that, when stressed under operating, maintenance, testing, and postulated accident conditions, (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. It also requires that the design reflect the uncertainties in determining the effects of irradiation on material properties. Two other appendices to 10CFR50 (USNRC 1993) implement, in part, these requirements: Appendix G, "Fracture Toughness Requirements," and Appendix H, "Reactor Vessel Material Surveillance Program Requirements".

Appendix G requires the effects of neutron radiation to be predicted from the results of pertinent radiation effects studies. In addition, it describes the basis for setting the upper limit for pressure as a function of temperature during heatup and cooldown for a given service period in terms of the predicted value of the adjusted reference temperature at the end of the service period.

Appendix H presents the requirements for the reactor vessel material surveillance program. It requires that the materials to be placed in surveillance be those that limit operation of the reactor during its lifetime, i.e., those expected to have the highest adjusted reference temperature or the lowest Charpy upper-shelf energy at the end of life. Both measures of radiation embrittlement must be considered.

The integrity of the reactor pressure vessel is of such importance that the USNRC Standard Review Plan (USNRC 1981) has a special section (5.3.3 - Reactor Vessel Integrity) devoted to make a summary review of all the factors relating to the vessel integrity, even though most of these factors be reviewed separately in other SRP sections.

2 NEUTRON RADIATION EMBRITTLEMENT

The reduction in the fracture toughness is caused by the high fast neutron fluence and concentrations of copper and nickel in the vessel wall. The so-called beltline region is the critical portion of the vessel because it is directly opposite the core, where there is a high neutron fluence rate, and it is adjacent to the coolant downcommer, a region with great potential for thermal shock. The occurrence of high tensile stresses due to the thermal shock where the radiation-induced reduction in the fracture toughness is the greatest introduces the possibility of propagation of surface flaws. A coolant leakage in the beltline area would tend to uncover the core with serious consequences.

Two measures of radiation embrittlement are obtained from the results of the Charpy V-notch impact test. The first measure is the adjustment of the reference temperature, ΔRT_{NDT}, defined as the temperature shift in the Charpy curve for the irradiated material relative to that for the unirradiated one measured at the 30-foot pound energy level. Appendix G to 10CFR50 (USNRC 1993) requires that a full curve of absorbed energy versus temperature be obtained through the ductile-to-brittle transition temperature region (see Figure 1). The second measure of radiation embrittlement is the decrease in the Charpy upper-shelf energy level (see Figure 1), which is defined in ASTM E 185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels" (ASTM 1982).

The general procedures for calculating the two above mentioned measures of neutron radiation embrittlement of the low-alloy steels currently used for light-water-cooled reactor vessels are described in the Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials" (USNRC 1988).

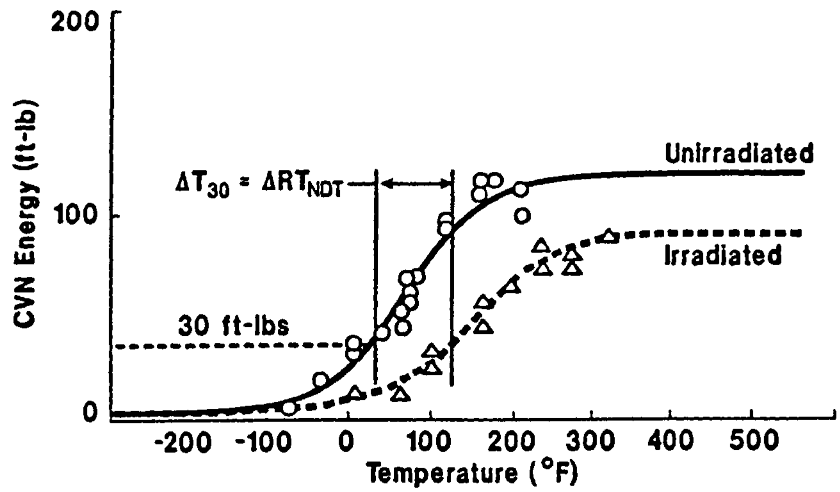

Figure 1- Shift in Charpy Curve with Irradiation (EPRI 1991)

3 PRESSURE-TEMPERATURE (P-T) LIMITS

One of the main requirements for safe operation of reactor pressure vessels is the definition of the pressure and temperature regimes where catastrophic failure of the vessel is not likely to occur. The vessel may be subjected to brittle fracture during heatup (caused by the propagation of outer flaws) and cooldown (due to the propagation of inner flaws). The allowable pressure-temperature limits are defined by Appendix G, "Protection Against Nonductile Failure" of Section III, Division 1 (or Appendix G, "Fracture Toughness Criteria for Protection Against Failure" of Section XI, Division 1) of ASME Boiler and Pressure Vessel Code (ASME 1992), and by Section 5.3.2 (Pressure-Temperature Limits) of the USNRC Standard Review Plan (USNRC 1981). Guidelines for P-T violation are given in Subsection IWB-3700, "Analytical Evaluation of Plant Operating Events," and Appendix E, "Evaluation of Anticipated Operating Events" of Division 1 of ASME (1992).

Appendix G (to Section III or Section XI) contains a recommended procedure for specifying allowable loading and temperatures in pressure retaining ferritic steel components, to provide adequate protection against nonductile fracture. It utilizes the principles of Linear Elastic Fracture Mechanics (LEFM) to determine the stress intensity factor (K_I) due to each specified loading of the component, assuming a conservative, postulated surface flaw (the one-quarter thickness flaw) at the location under evaluation. The sum of K_I values for each load (typically pressure + thermal) is compared to a reference value, K_{IR}, which is the lower bound fracture toughness adjusted for the vessel-specific level of embrittlement. Reference (Kuo et. al. 1990) describes a computer program for constructing P-T limit curves based on the Appendix G recommendations, and includes some illustrative examples.

With increasing levels of embrittlement, the allowable region for operation can become prohibitively small so that start-up and shutdown operations are difficult if not impossible without violating the administrative limits (see Figure 2). Experience along the years has been showing that some conservatisms exist in Appendix G (to Section III or Section XI). The postulated flaw size was chosen as the quarter thickness flaw, and no credit is given for non-destructive examination of the vessel, even though USNRC requires periodical inspection of the vessel and, currently, is implementing requirements to make the inspections more effective. Therefore, now it should be considered a vessel-specific reference flaw size, that would be determined, at least in part, by the periodical inspection program. However, if smaller flaw sizes are to be considered, fracture mechanics analysis methods must be more rigorous, taking in consideration the effects of the stainless steel cladding, and residual stresses. It is also important to make sure that the embrittlement estimates are appropriate without being unnecessary conservative (Shao 1993).

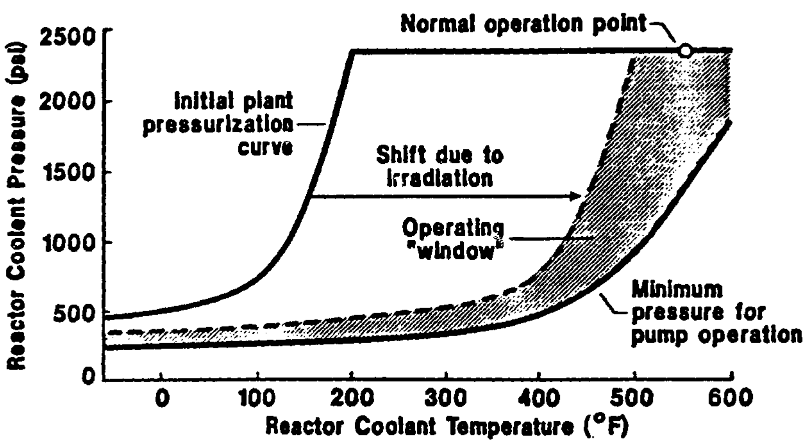

Figure 2- Pressure-Temperature Limitation Curves (EPRI 1991)

4 PRESSURIZED THERMAL SHOCK (PTS)

Fracture mechanics analyses have been used to evaluate reactor vessel integrity under severe transient conditions. Also, relevant material properties values are used. The best and most reliable way to validate the assessment methods is to perform simulated tests on pressure vessels in conditions as close to the reality as possible.

A limiting condition on reactor vessel integrity known as pressurized thermal shock (PTS) may occur during a severe system transient such as a loss-of-coolant accident (LOCA) or a steam line break. Such transients could challenge the integrity of a pressurized water reactor vessel under the following conditions: (a) severe overcooling of the inside surface of the vessel wall followed by high pressurization; (b) significant degradation of vessel material toughness caused by radiation embrittlement; (c) the presence of a critical-size defect in the vessel wall (EPRI 1991).

The PTS problem has been studied since 1967 by the nuclear industry and as part of the Heavy-Section Steel Technology (HSST) Program (USNRC 1986), which is sponsored by the USNRC. At the end of the 70's, integrity studies related to postulated and actual PWR PTS transients, indicated that if such transients occurred late in the life of a high copper vessel containing appropriate flaws, the chances of vessel failure could be high. However, these analyses assumed a combination of a very severe PTS gradient, high concentrations of copper in the vessel, lower-bound fracture-toughness data, and the existence of flaws of appropriate size. The analyses were of a deterministic nature and were believed to be quite conservative. Therefore there was a generally thought that the probability of vessel failure was actually very small (Cheverton and Selby 1992).

To obtain a better understanding of the nature and magnitude of the problem, the USNRC proposed the development of a comprehensive probabilistic approach and in May 1981 established the Integrated-Pressurized-Thermal-Shock (IPTS) Program. In July 1985 the USNRC published the PTS Rule, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events" - §50.61 of 10CFR50 (USNRC 1993). This regulation specifies criteria for the maximum permissible level of vessel embrittlement.

The index of embrittlement used in the PTS Rule is the nil-ductile reference temperature RT_{NDT}. The rule imposes limits on the adjusted reference temperature considering the effects of neutron irradiation (1993 Edition of PTS Rule is consistent with Reg. Guide 1.99 Rev. 2, 1988). The limits on the adjusted reference temperature are referred to as screening criteria and for operation beyond these limits PTS Rule requires a plant-specific safety analysis. This safety analysis shall determine what, if any, modifications to equipment, systems, and operation are necessary to prevent potential failure of the reactor pressure vessel as a result of postulated PTS events. In the analysis the reactor vessel material properties may be determined based on available information, research results, and plant surveillance data, and probabilistic fracture mechanics techniques may be used.

Regulatory Guide 1.154, "Format and Content of Plant-Specific Pressurized Thermal Shock Safety Analysis Reports for Pressurized Water Reactors" (USNRC 1987), issued by USNRC in January 1987, recommends the content and format for the plant-specific integrated PTS analyses. The objective of performing the plant-specific PTS analysis is to calculate the probability of vessel failure due to PTS. Some probabilistic fracture mechanics computer codes are referenced by Reg. Guide 1.154 as acceptable tools for performing these analyses. The results of such analyses, when compared with acceptable failure probabilities, provide an estimation of the residual life of the reactor pressure vessel and can be used to evaluate the benefits of plant-specific mitigating actions in order to extend the vessel operating life (Dickson 1993).

5 CONCLUSIONS

Considering the importance of P-T limit curves and PTS evaluation to assure the integrity of the reactor pressure vessel, this work presented the major aspects related to these issues emphasizing the regulatory point of view. A historical evolution of PTS studies and some comments relative to the conservatisms and non-conservatisms involved in the procedures for defining P-T limits was also addressed. The authors' main objective was to provide a guide for those interested in getting a better understanding on the American approach with respect to these subjects.

REFERENCES

ASME 1992. ASME Boiler and Pressure Vessel Code. New York: ASME.
ASTM.1982. ASTM E 185-82 Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels. New York: ASTM.
Cheverton, R.D. & D.L.Selby 1992. A Probabilistic Approach to the Evaluation of the PTS Issue. *Transactions of ASME.* Vol. 114: 396-404. New York: ASME.
Dickson, T.L. 1993. FAVOR: A New Fracture Mechanics Code for Reactor Pressure Vessels Subjected to Pressurized Thermal Shock. *Proceedings of Pressure Vessel and Piping Conference.* Vol. 250: 3-9. New York: ASME.
Kuo, A.-Y., P.C. Riccardella & T. Griesbach 1990. Development and Usage of P-T Calculator - A PC-Based Computer Program for Constructing P-T Limit Curves. *Proceedings of Pressure Vessel and Piping Conference.* Vol. 195: 141-148. New York: ASME.
USNRC 1981. NUREG-800 Standard Review Plan for the Review Reports for Nuclear Power Plants, LWR Edition. Washington: USNRC.
USNRC 1988. Regulatory Guide 1.99 Revision 2 Radiation Embrittlement of Reactor Vessel Materials. Washington: USNRC.
USNRC 1987. Regulatory Guide 1.154 Format and Content of Plant-Specific Pressurized Thermal Shock Safety Analysis Reports for Pressurized Water Reactors. Washington: USNRC.
Shao L.C. 1993. Technical Issues and U.S. Nuclear Regulatory Commission's Research Related to the Primary Pressure Boundary. *Trans. of the 12th Int. Conf. on Structural Mec. in Reac. Technology.* Vol. A - Supplement: 21-37. Berlin: IASMiRT.
USNRC 1993.10CFR50 Code of Federal Regulation, Title 10, Part 50. Washington: USNRC
EPRI 1991. Wessel, E.T., W.L. Server & E.L.Kennedy. Primer: Fracture Mechanics in the Nuclear Power Industry. Palo Alto: EPRI.
USNRC 1986. Whitman, G.D. Historical Summary of Heavy-Section Steel Technology Program and Some Related Activities in Light-Water Reactor Pressure Vessel Safety Research. NUREG/CR-4489 (ORNL-6259) Washington: USNRC.