Review of Advanced Test Reactor Vessel Neutron Irradiation Affects

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INTRODUCTION

This assessment was conducted in response to recent questions regarding the affect of neutron irradiation on the Advanced Test Reactor (ATR), Reactor Vessel (RV).

The assessment was conducted by M. McAnulty and R. Boston. Mr. McAnulty is a Facility Representative at the Reactor Technology Complex (RTC). He has a bachelor's degree and is a licensed Professional Engineer in the field of Mechanical Engineering registered in the State of Idaho. Mr. Boston is the RTC Team Lead and has a master's degree in Nuclear Engineering.

It is common knowledge that gamma radiation causes only a heating affect on metals and does not result in embrittlement. Therefore, gamma irradiation of the RV was not investigated.

DISCUSSION OF THE CURRENT REACTOR VESSEL ANALYSIS

The ATR Upgraded Final Safety Analysis Report (SAR-153) [BEA, 2007a] section 4.3.2.8 cites a 1987 reference (Durney, PG-T-87-009) which validated that the RV could continue to operate at least through 2014. Engineering Design File EDF-TRA-ATR-1823 (BBWI, 2002) conducted further analysis of the fracture toughness of the Vessel Specimen Irradiation Program.

Inspection of ATR in-service components is governed by the ATR Inservice Inspection (ISI) Plan (PLN-859) [BEA, 2007b] which "implements a program for periodic inspection and testing of pressure boundary components and equipment in selected process systems. The Plan specifies the systems to be inspected, the types of inspections to be performed, the inspection schedules, and the standards for evaluation of inspection results. The Plan also specifies requirements for performing repairs, replacements, and modifications to the systems included in the Plan."

"The ATR ISI Plan is based on Section XI of the ASME Boiler and Pressure Vessel Code. Section XI was developed by the American Society of Mechanical Engineers for application to commercial nuclear power plants of the type currently in operation and is the industry standard for in-service inspection. The ATR ISI Plan was developed using Section XI as a model. However, the requirements of Section XI are adjusted as appropriate for applicability to the unique design and operational features of the ATR. Significant exceptions and modifications to Section XI requirements are denoted in the Plan."

The ATR primary systems were evaluated to determine susceptibility to intergranular stress corrosion cracking (IGSCC) and erosion/corrosion phenomena. The conclusion of the evaluations is that the primary systems are not susceptible to these phenomena because of the materials of construction (austenitic stainless steel), favorable water chemistry, and low operating temperatures. No corrosion, erosion or cracking have been detected during past ISI

plan weld inspections, and continued inspections will assure any erosion/corrosion or IGSCC is detected (Carlson, 2007).

The ATR primary systems have also been evaluated for fatigue. Based on an analysis of the highest stressed area of the reactor vessel it can conservatively tolerate 10^7 cycles per ASME design code. Since the primary system averages less than one cycle per week there is adequate safety margin against fatigue for the life of the facility. The highest stressed regions in the primary piping system are in welds that are included in the ISI plan. Fatigue problems are not expected because of ductile material properties and low stresses, and continued inspections will assure any cracking is detected before failure (Carlson, 2007).

The projected fluence through 2014 was based on ATR operations at 250 MW with an 80% reactor use, which projected a peak fast fluence at 2σ of 2.7x1018 nvt. Historically, the ATR operates well below 250 MW and 80% utilization which conclude that the existing assumption is still valid. Although more data is available to determine the current cumulative fluence on the reactor vessel and obtain a more accurate projected fluence, the current assumption is more conservative as projected fluence based on operating data (up to present) would be less than the projected fluence back in 1986.

The degradation of fracture toughness in weld material based on WGR-120-86, which was referenced in JEDw-25-86 (Dwight, 1986), changed from 135 ksi$\sqrt{\text{in}}$ unirradiated to 54 ksi$\sqrt{\text{in}}$ irradiated which is significant. However it should be noted that Figure 3 in WGR-120-86 draws a straight line between the two values on a semi-log graph to interpolate the fracture toughness at a fluence of $6\text{x}10^{17}$ to be 120 ksi$\sqrt{\text{in}}$. This is not appropriate since there is a break between 10^0 and 10^{17} on the graph and no explanation to the relationship of fracture toughness and neutron embrittlement that supports graphical interpolation. In addition, there are articles in the Journal of Nuclear Materials which suggest a fracture toughness dependence on irradiation temperature which may result in lower values of fracture toughness for the vessel than those of the test specimens which were irradiated at much higher temperatures.

As a result of the discrepancies noted in WGR-120-86, the fracture toughness values presented in WGR-120-86 are questionable.

Further research identified credible fracture toughness values provided in EDF-TRA-ATR-1823. This report documents the fracture toughness testing by B&W Nuclear Environmental Services, Inc. Lynchburg Technology Center and data normalization by McDermott Technology, Inc. (MDI). A normalization procedure was performed by MDI because the results failed the acceptance criteria outlined in ASTM E813-87. At the time of the report this normalization procedure was a recognized approach in the materials testing community but was not adopted by the American Society for Testing and Materials (ASTM) testing standards. This normalization procedure has since been accepted in ASTM E1820-06 annex 15. Additionally, the Vessel Specimen Irradiation Program documented that the specimens were irradiated in the south capsule irradiation tank which suggests the irradiation temperature is consistent with that of the ATR reactor vessel thus negating any irradiation temperature dependencies. In this analysis, data from the literature was used to estimate lower bound toughness for irradiated 308 stainless steel weld material (J-Integral = 464 in-lb/in^2). Using a conservative approach for predicting

fracture, the analysis concluded that the vessel would adequately perform its function at this lower bound toughness value.

Fracture toughness is relevant when discussing material failures associated with a pre-existing flaw. Initial crack size, applied loading, and fracture toughness determine the rate of crack growth and final crack length prior to failure. Margin of safety is typically thought of as a ratio of the applied stress to the maximum allowable stress neglecting the presence of a crack or the material's fracture toughness. However if the applied stress is considered constant and the fracture toughness is reduced through neutron embrittlement, a comparison could be made with the reduction in the critical crack length.

Give that critical crack size, $a_c = \frac{1}{\pi}\left(\frac{K_{Ic}}{S}\right)^2$, a ratio of critical crack size is,

$$\frac{a_{c_irradiated}}{a_{c_unirradiated}} = \left(\frac{K_{Ic_irradiated}}{K_{Ic_unirradiated}}\right)^2$$

Reduction in critical crack size would be a concern if the critical crack size is less than what could be detected by In-Service Inspection or if the critical crack size would violate the leak before break concept of vessel design.

Therefore, EDF-TRA-ATR-1823 adequately justifies that the ATR reactor vessel fracture toughness and projected neutron fluence are conservative and supports aging assessment to 2014. It should be noted that based on more recent neutron fluence data, the projected fluence could be extended beyond 2014.

EXTRAPOLATION BEYOND 2014

In the ATR Vessel Lifetime And Primary Coolant System IGSCC Assessment (Durney, 1987) it was determined that:

> "The relationship between the critical flaw sizes versus K1 in Figures 14(a), (b), and (c) for upset conditions and in Figures 15(b) and (d) for emergency conditions are used in conjunction with KI_c =114 ksi•in$^{1/2}$ (JI_c = 464 in .•lb /in^2 in Table 14) to calculate the critical flaw sizes for the weld metal . This value was selected because it is the lowest of the results in Table 14 and it is assumed that a fluence of 3 .13 x 10^{18} n/cm^2 (E > 1 MeV) will not reduce the fracture toughness."

Further RV specimen irradiation analysis found the fracture toughness relationship is as shown in Figure One. To determine an approximation of the safe operating life for the RV, the fluence to reach the fracture toughness limit of 464 in-lb/in^2 was determined (). Refer to Figure One below for the remainder of this discussion. The line through the data (upper red line) is a least squares fit and the parallel line below (blue line) the fit is the lower 95% confidence interval. The tic mark on the fluence axis at 1.27 x 10^{18} (all fluences in units of n/cm^2) represents the fluence projected to be accumulated at the 2004 CIC, and the tic mark at 4.75 x 10^{18} represents

the fluence estimated to be accumulated in an additional fifty years operating (the year 2054) at 175 MW at 90% efficiency. Extrapolating out to a fluence of 1.5×10^{19} (which is still above the fracture toughness limit) would be to the year 2171. This extrapolation is not intended to be used as a technical basis for the ATR safety analysis report review or approval but for a general understanding of the robustness or fragility of the ATR RV.

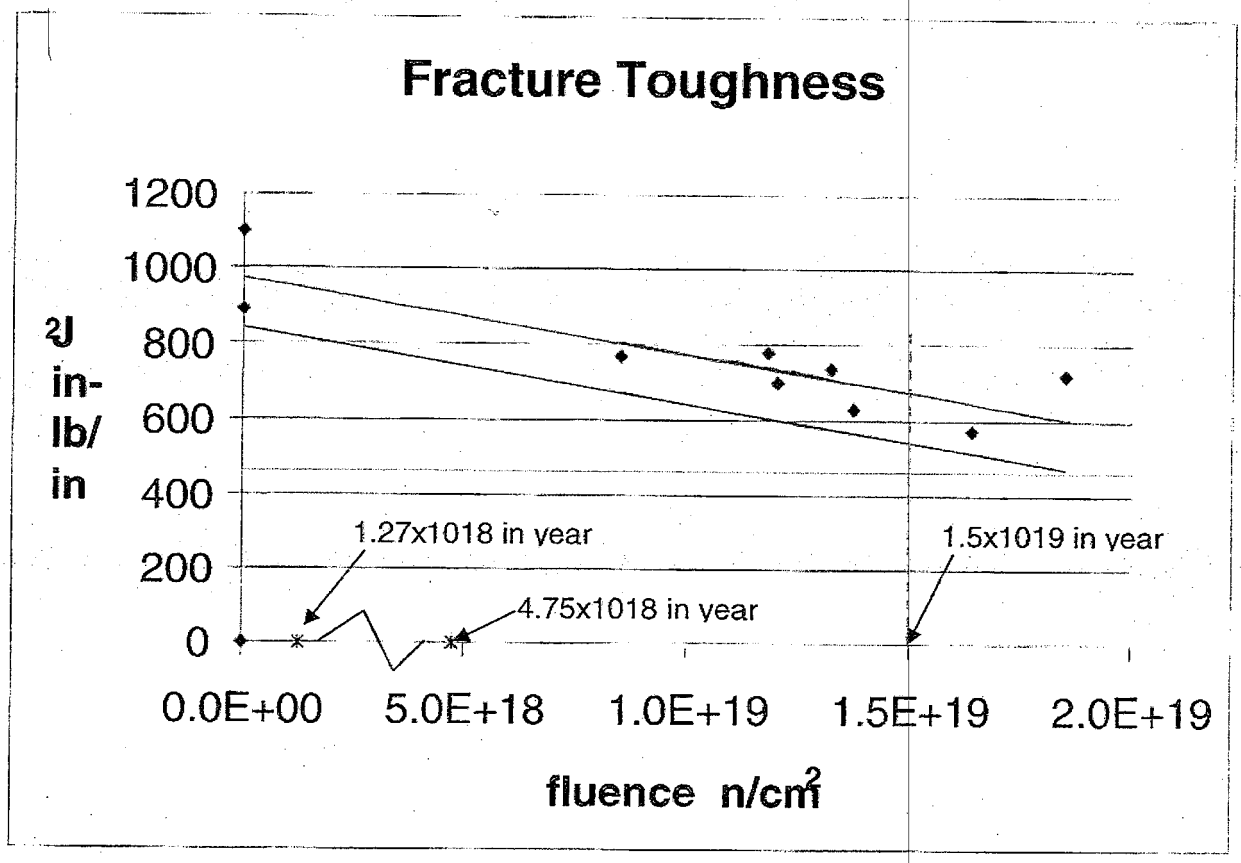
Fracture Toughness
1200
1000
800
600
400
200
0
2J in-lb/in
1.27x1018 in year
4.75x1018 in year
1.5x1019 in year
0.0E+00
5.0E+18
1.0E+19
1.5E+19
2.0E+19
fluence n/cm2

Figure One

CONCLUSION

Therefore, EDF-TRA-ATR-1823 conservatively justifies that the ATR reactor vessel fracture toughness and projected neutron fluence are conservative and supports aging assessment to 2014. It should be noted that based on more recent neutron fluence data, the projected fluence could be extended beyond 2014. Performance of a realistic assessment by extrapolating the available data shows that the ATR Reactor Vessel should be resistant to neutron embrittlement and corrosion for well into the future.

REFERENCES

1. BBWI 2002, ATR Vessel Specimen Irradiation Program, EDF TRA-ATR-1843
2. BEA 2005, Facility Certification Report, ATR CRD-15E (contains the white paper, - E. Carlson, Summary of Inservice Inspections Planned and Performed on ATR Primary Systems)
3. BEA 2007a, ATR Upgraded Final Safety Analysis Report, SAR-153
4. BEA 2007b, ATR inservice Inspection Plan Fourth Interval, PLN-859
5. Durney 1987, ATR Vessel Lifetime And Primary Coolant System IGSCC Assessment, PG-T-87-009
6. Dwight 1986, Neutron Embrittlement of the ATR Reactor Vessel, JEDw-25-86
7. R.I. Stephens, et. al., *Metal Fatigue in Engineering*, 2nd ed., John Wiley and Sons, New York, 2001.
8. Rice, J.R., "A Path Independent Integral and the Approximate Analysis of Strain Concentration by Notches and Cracks", *Journal of Applied Mechanics*, Vol 35, 1968, pp 379-386

APPENDIX ONE
FRACTURE TOUGHNESS

The difference between K and J values in fracture toughness is that K is based on stress intensity at the crack root and has units of ksi√in, where J is based on strain energy has units of in-lb/in^2 (sometimes lb/in). The equivalencies are:

$$J = \frac{K^2}{E}$$ for plane stress

$$J = \frac{K^2}{E}\left(1-\nu^2\right)$$ for plain strain [8]

Assuming Young's modulus (E) is 30,000 ksi and a Poisson's ratio (υ) of 0.3, the fracture toughness value of K=135 ksi√in would be an equivalent of J=553 in-lb/in^2. It should be noted that this correlation is only valid under linear elastic conditions, and ASTM E1737 does not allow the J value results according to that test standard to be used to obtain the K values.

The value of 464 in-lb/in^2 was referenced by PG-T-87-009, which appears to be the lowest value listed in a referenced table from J.D. Landes and D.E. McCabe, Toughness of Austenitic Stainless Steel Pipe Welds, EPRI NP-4768, October 1986.

The value of 464 in-lb/in^2 seems conservative since all our tests resulted in much higher Js.

To clarify fracture toughness notations, the commonly seen subscripts are: Roman numerals used to describe the mode shown in Figure 2, and c stands for critical values.

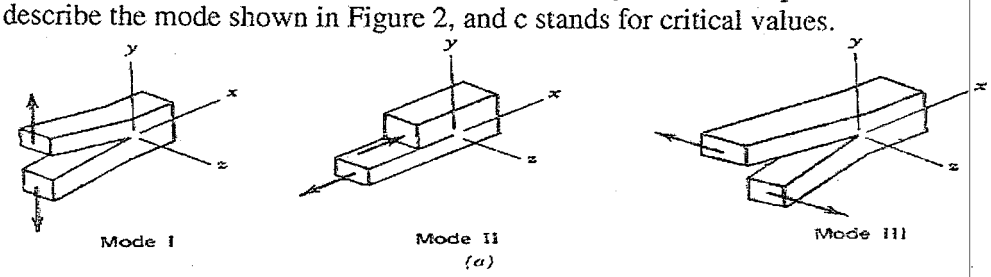

Figure 2: Three Basic Modes of Loading [7]