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Light Water Reactor Systems

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EXECUTIVE SUMMARY

The UCSB ATR-2 irradiation experiment is designed to generate a new database on a wide variety of irradiated reactor pressure vessel (RPV) steels to fill a critical gap in predicting high fluence embrittlement for extended plant operation up to 80 years. A major focus in this experiment is to characterize the effects of irradiation temperature, neutron flux and fluence, and alloy chemistry on MNSP evolution, and model how these features impact hardening and embrittlement, manifested as shifts in ductile-to-brittle transition temperature. As part of these efforts, disk compact tension (DCT) specimens of three representative materials were included in the ATR-2 experiment to have direct measurement of transition fracture toughness shift and, thus, to have some benchmark data to compare outcoming predictive models with actual fracture toughness shifts. In addition, the attempts would be made to use these fracture toughness data to address the potential effect of high dose irradiation on the shape of the Master Curve for highly embrittled RPV material.

This report provides data for the fracture toughness characterization of Palisades steam generator weld irradiated in ATR-2 experiment. This weld was made with the same procedure as the reactor pressure beltline weld in Palisades Nuclear Power Plant and is well known for having high sensitivity to irradiation due to high level of copper and nickel.

1. INTRODUCTION

The UCSB ATR-2 irradiation experiment is designed to generate a new database on a wide variety of irradiated reactor pressure vessel (RPV) steels to fill a critical gap in predicting high fluence embrittlement for extended plant operation up to 80 years [1-5]. A major focus in this experiment is to characterize the effects of irradiation temperature, neutron flux and fluence, and alloy chemistry on MNSP evolution, and model how these features impact hardening and embrittlement, manifested as shifts in ductile-to-brittle transition temperature. As part of these efforts, disk compact tension (DCT) specimens of three representative materials were included in the ATR-2 experiment to have direct measurement of transition fracture toughness shift and, thus, to have some benchmark data to compare outcoming predictive models with actual fracture toughness shifts. In addition, the attempts were made to use these fracture toughness data to address the potential effect of high dose irradiation on the shape of the Master Curve for highly embrittled RPV material. As part of these efforts, the weld from Palisades Nuclear Power Plant steam generator has been characterized in the irradiated condition. The report is prepared in satisfaction of Milestone M2LW-18OR0402013 – "Complete report on fracture toughness evaluation of highly irradiated reactor pressure vessel alloys, that are part of the ATR-2 irradiation experiment."

2. MATERIAL DESCRIPTION

The weld material in this experiment came from Palisades Nuclear Power Plant (NPP) steam generator and had designation PBR. The Palisades weld, designated weldment "B" from weld heat 34B009, was provided to UCSB and ORNL by Palisades NNP for research purposes. This weld came from steam generator and it was produced with the same weld wire and heat-treated in the same fashion as beltline weld of Palisades NNP. It is well known to high sensitivity to irradiation due to relatively large content of copper and nickel, see Table 1.

Table 1. Chemical composition of PBR weld

	Composition, wt. %									
	Cu	Ni	Mn	Cr	Mo	P	C	S	Si	Fe
PBR	0.2	1.2	1.3	0.04	0.54	0.01	0.11	0.017	0.18	balance

3. FRACTURE TOUGHNESS CHARACTERIZATION

3.1 Testing and analysis procedure

The disk compact tension DCT specimens were selected for this experiment. The diameter of this specimens was 20 mm and it was determined by the diameter of the ATR-2 capsule. The thickness of the specimens was 6.6 mm. The fracture toughness tests were conducted in accordance with the ASTM E 1921 [6] Standard Test Method for Determination of Reference Temperature, T_o, for Ferritic Steels in the Transition Range, with a computer-controlled test and data acquisition system. The specimens were fatigue precracked to a ratio of the crack length to specimen width (a/W) of about 0.5. The unloading compliance method used for measuring the J-integral. All tests were conducted in strain control, with an outboard clip gage having a central flexural beam that was instrumented with four strain gages in a fullbridge configuration. The broken specimens were photographed and digital images were used to measure final crack lengths.

Values of J-integral at cleavage instability, J_c, were converted to their equivalent values in terms of stress intensity K_{Jc} by the following equation [6]:

$$K_{Jc} = \sqrt{J_c \frac{E}{1-\nu^2}} \tag{1}$$

where E is Young's modulus and ν is Poisson's ratio.

A K_{Jc} datum was considered invalid if this value exceeded the $K_{Jc(limit)}$ requirement of the ASTM Standard E 1921 [6]:

$$K_{Jc(limit)} = \sqrt{\frac{b_o \sigma_{YS}}{30} \cdot \frac{E}{1-\nu^2}} \tag{2}$$

where b_o is the remaining ligament and σ_{YS} was the yield strength of the material at the test temperature. All invalid data were censored and substituted by the $K_{Jc\,(limit)}$ values for calculation of the transition fracture toughness temperature, T_o. After that, all K_{Jc} data (valid and substituted) were converted to 1T equivalence, $K_{Jc(1T)}$, using the size adjustment procedure of ASTM Standard E1921 [6]:

$$K_{Jc(1T)} = 20 + \left[K_{Jc(x)} - 20\right] \cdot \left(\frac{B_x}{B_{1T}}\right)^{1/4} \tag{3}$$

where $K_{Jc(x)}$ = measured K_{Jc} value,
B_x = gross thickness of test specimen,
B_{1T} = gross thickness of 1T C(T) specimen.

The reference fracture toughness transition temperature, T_o, was determined using the multi-temperature equation from E1921 [6]:

$$\sum_{i=1}^{N} \delta_i \frac{\exp\left[0.019\left(T_i - T_o\right)\right]}{11 + 77\exp\left[0.019\left(T_i - T_o\right)\right]} - \sum_{i=1}^{N} \frac{\left(K_{Jc(i)} - 20\right)^4 \exp\left[0.019\left(T_i - T_o\right)\right]}{\left\{11 + 77\exp\left[0.019\left(T_i - T_o\right)\right]\right\}^5} = 0 \tag{4}$$

where δ_i= 1.0 if the datum is valid or zero if datum is invalid,
T_i= test temperature corresponding to $K_{Jc(i)}$.

3.2 PALISADES WELD FRACTURE TOUGHNESS RESULTS

The reference fracture toughness temperature, T_o, of Palisades weld in the unirradiatied condition was -102°C as previously reported in [7]. The irradiated fracture toughness data are presented in Table 2. All specimens cleaved with two specimens exhibiting ductile tearing prior to cleavage. That prevented to fully address the issue of effect of high embrittlement on the shape of the fracture toughness transition region. This work will be continued with two other alloys in this irradiation experiment, CM17 and LP. The results of these two welds will be published in next fiscal year. The analysis of the data yield T_o = 108°C. Thus, the shift of the transition temperature as result of irradiation is 210°C. Figure 1 presents unirradiated and irradiated fracture toughness data for Palisades weld and its master curves in both unirradiated and irradiated conditions.

Table 2. Fracture toughness data of PBR weld

Specimen ID	Temperature, °C	Measured K_{Jc}, MPa√m	1T-adjusted K_{Jc}, MPa√m	Δa, mm
PBR-7	75	85.6	66.84	
PBR-9	90	93.6	72.55	
PBR-15	75	77	60.69	
PBR-16	88	59.6	48.27	
PBR-17	23	44.6	37.56	
PBR-18	80	99.6	76.83	
PBR-20	83	80.0	62.84	
PBR-27	50	85.7	66.91	
PBR-10	94	147.8	111.24	0.30
PBR-29	90	120.4	91.68	0.97

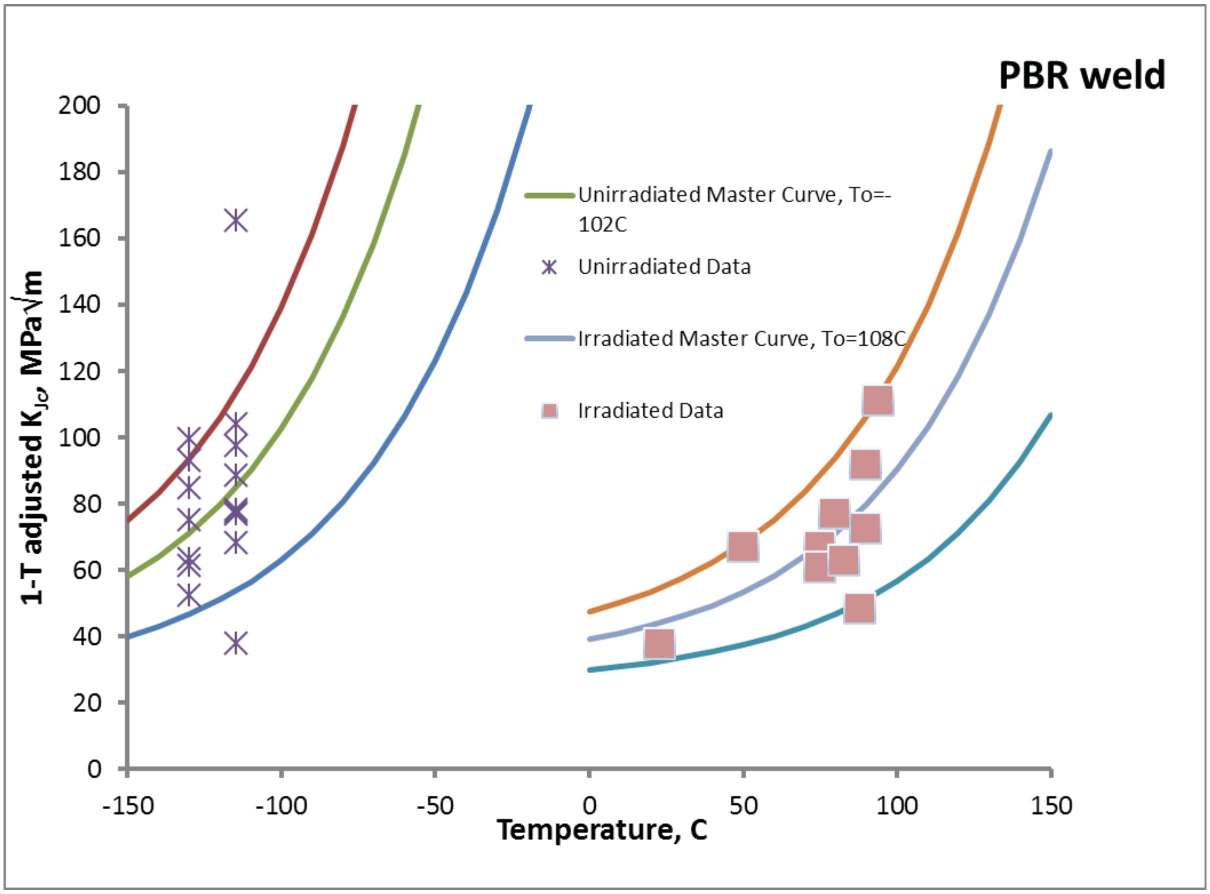

Figure 1. Fracture toughness data and the master curve of PBR weld in the unirradiated and irradiated conditions

SUMMARY

Fracture toughness characterization of Palisades weld, PBR, has been performed using DCT specimens. These specimens were irradiated in ATR-2 experiment. Analysis of the data showed very high level of embrittlement of this material. Shift of T_o is 210^{o}C.

REFERENCES

1. Nanstad, R. K., G. R. Odette, and T. Yamamoto, "Progress Report on Disassembly of UCSB ATR-2 Capsule and Revision to Post-Irradiation Plan," ORNL/TM-2014/525, Oak Ridge National Laboratory, September 2014.
2. Nanstad, R. K. and G. R. Odette, "Reactor Pressure Vessel Task of Light Water Reactor Sustainability Program: Milestone Report on Materials and Machining of Specimens for the ATR-2 Experiment," ORNL/LTR-2011/413, Oak Ridge National Laboratory, January 2011.
3. Nanstad, R. K., "Reactor Pressure Vessel Task of Light Water Reactor Sustainability Program: Assessment of High Value Surveillance Materials," ORNL/LTR-2011/172, Oak Ridge National Laboratory, June 2011.
4. Nanstad, R. K., G. R. Odette, T. Yamamoto and M. A. Sokolov, "Post-irradiation Examination Plan for ORNL and University of California Santa Barbara Assessment of UCSB ATR-2 Irradiation Experiment," ORNL/TM-2013/598, Oak Ridge National Laboratory, December 2013.
5. R. K. Nanstad, G. R. Odette, N. Almirall, J. P. Robertson, W. L. Server, T. Yamamoto, and P. Wells, "Effects of ATR-2 Irradiation to High Fluence on Nine RPV Surveillance Materials", ORNL/TM-2017/172, Oak Ridge National Laboratory, December 2016
6. Standard Test Method for Determination of Reference Temperature, To, for Ferritic Steels in the Transition Range, Designation E 1921, Annual Book of ASTM Standards, Vol. 03.01.
7. M.A. Sokolov, X. Chen, R.K. Nanstad, G.R. Odette, T. Yamamoto, P. Wells, "Fracture Toughness Characterization of Reactor Pressure Alloys from the ATR-2 Experiment", ORNL/TM-2017/358, Oak Ridge National Laboratory, July 2017.