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AGING MANAGEMENT OF PWR INTERNALS COMPONENTS

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ABSTRACT

Light Water Reactors (LWR) in US are normally designed and licensed for a plant life of forty years. However, due to embedded design conservatism and favorable plant operation and maintenance conditions, plants have margins to operate beyond their original designed life.

Many US utilities have set a goal to extend plant life to 60+ years through license renewal applications. To achieve this goal, potential degradation of reactor components associated with aging need to be identified and managed. The Standard Review Plan and Generic Aging Lessons Learned (GALL) report issued by the US Nuclear Regulatory Commission (NRC) outline staff's basis on reviewing and accepting the utility aging management program. The Electric Power Research Institute (EPRI) Material Reliability Program (MRP), funded by the US and a number of international utilities, has been conducting research providing data and results to support aging management and the license renewal goal.

Reactor internals components due to their proximity to the core absorb significant radiation dosage. They consequently experience radiation induced degradation such as irradiation-assisted stress corrosion cracking (IASCC), embrittlement (i.e., reduction of ductility and fracture toughness), creep/stress relaxation (loss of preload), and void swelling (dimensional changes and ductility loss). These degradation mechanisms, depending on the extent of their effects, may impact structural integrity and functionality of internals components.

Keywords: PWR Internals Irradiated Materials Aging Management License Renewal

1. INTRODUCTION

Light Water Reactors (LWR) in US are designed and licensed for a plant life of forty years. However, due to embedded design conservatism and favorable plant operation and maintenance conditions, plants have margins to operate beyond their original designed life.

Many US utilities that operate LWRs have set a goal to operate reactors for 60+ years through license renewal applications. The US Nuclear Regulatory Commission (NRC) has issued a Standard Review Plan on license renewal (NUREG 1800, 2001) and a Generic Aging Lessons Learned (GALL) report (NUREG 1801, 2001). These documents address aging management for license renewal and describe staff’s basis for determining when existing generic programs are adequate to manage aging without change and when existing generic programs should be augmented for license renewal. The EPRI Material Reliability Program (MRP) funded by the US and some international utilities has been conducting researches to provide data and results for aging management and license renewal applications.

Reactor internals components require particular aging management attention. Their proximity to the core result in absorbing significant radiation dosage, which causes radiation induced degradation such as irradiation-assisted stress corrosion cracking (IASCC), embrittlement (i.e., reduction of ductility and fracture toughness), creep/stress relaxation (loss of preload), and void swelling (dimensional changes and ductility loss). Cracking in baffle bolts fabricated from strain-hardened 316 stainless steel was found in operating PWRs (Cauvin et al., 1994; Trenty et al., 1998). Examination of the removed bolts revealed that bolt cracking was of IASCC (Scott et al., 2000). The radiation induced degradation and its associated effects over plant life need to be quantified and their impact on the long-term structural integrity and functionality of reactor internals components need to be assessed.

Under the EPRI MRP, the Reactor Internals Issue Task Group (RI-ITG) has been conducting researches to provide technical basis of radiation damage and effects for aging management of PWR internals to support plant license renewal. This paper highlights selected research results that have been achieved in the following areas:

- Hot Cell Test and Microstructure Examination of PWR Irradiated Materials
- Hot Cell Test and Microstructure Examination of Fast Reactor Irradiated Materials
- Determination of Reactor Internals Operating Conditions
- Evaluation of Irradiation Induced Swelling and Stress Relaxation
- Development of Long-term Aging Management Strategy, Criteria and Process

2. HOT CELL TEST AND MICROSTRUCTURE EXAMINATION OF PWR IRRADIATED MATERIALS

Reactor internals materials samples from operating and decommissioned PWRs were retrieved for testing in the hot cell to fully characterize their mechanical behavior, stress corrosion properties, microstructural details and cracking characteristics. Experimental tasks performed included fractography, tensile test, slow strain rate test, crack growth test, fracture toughness test, and scanning and transmission electron microscopy (SEM and TEM). The retrieved samples included baffle and former bolts from two operating PWRs. They are of strain-hardened type 316 stainless steel from one plant and solution annealed Type 347 stainless steel from the other. The maximum fluence for some of the bolts was estimated around 20 dpa (displacement per atom) at the bolt head. From the decommissioned US PWR, large pieces of baffle and former plate samples (304 SS) were harvested as follows:

- Two approximately 1½”x 13” long sections of a baffle plate, one from the top of the plate and one from the bottom of the top half (mid core region)
- One approximately 8”x5”x5” triangle from the corner of one of the second elevation former plates
- One approximately 8”x8” section of the bottom of one of the 78 inch long core barrel pieces containing a vertical weld seam

Tensile specimens were prepared from the retrieved samples and tested in hot cell to characterize mechanical behavior of PWR irradiated stainless steels. As shown in Fig. 1, the yield strength increases as a function of neutron

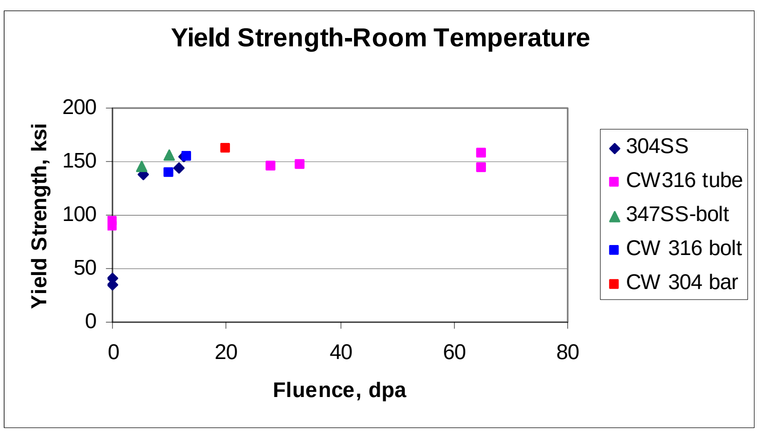

Fig. 1: Yield Strength vs. Fluence of PWR Irradiated Stainless Steels

fluence and approaches saturation around 10 to 15 dpa. Figure 2 shows that the total elongation decreases with increasing neutron fluence. At apparent saturation, the materials remain ductile.

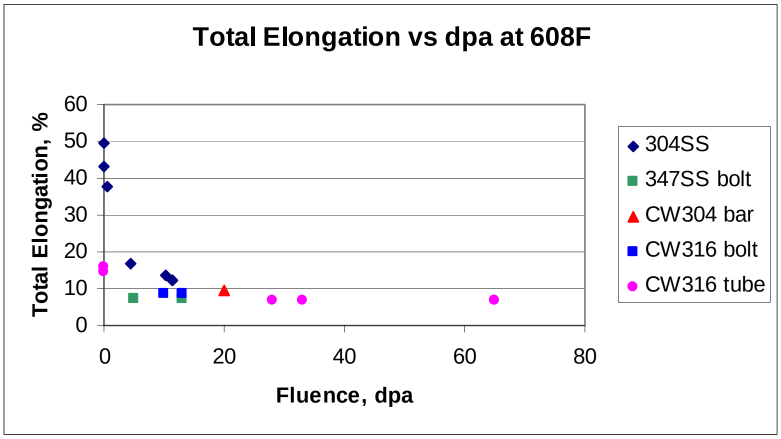

Fig. 2: Total Elongation vs. Fluence of PWR Irradiated Stainless Steels

Slow Strain Rate Test (SSRT) was also performed in simulated PWR water. The SSRT results show similar mechanical behavior characteristics as those from the tensile specimen tests which were conducted in air. Fractography of the failed specimens was performed using a shielded Amray Scanning Electron Microscope (SEM). Figure 3 shows the fracture surface morphology of one of the decommissioned PWR specimens (304 SS) that had very low fluence (~0.08 dpa). The crack initiation sites near the surface are of mixed transgranular and intergranular failure with the remaining cross-sectional area dimpled rupture. Figure 4 shows the SEM examination results of the specimen that experienced comparatively high fluence (17 dpa). For this specimen, the cracking initiation site was of intergranular only and the remaining cross-sectional area dimpled rupture. Transmission Electron Microscopy (TEM) was performed on baffle and former bolts from operating plants (CW 316 SS and SA 347 SS). Voids were observed in the higher temperature regions in the shank but not in the higher fluence heads or bolt locking devices (SA 304 SS). The maximum swelling estimated based on observed voids was ~ 0.03%.

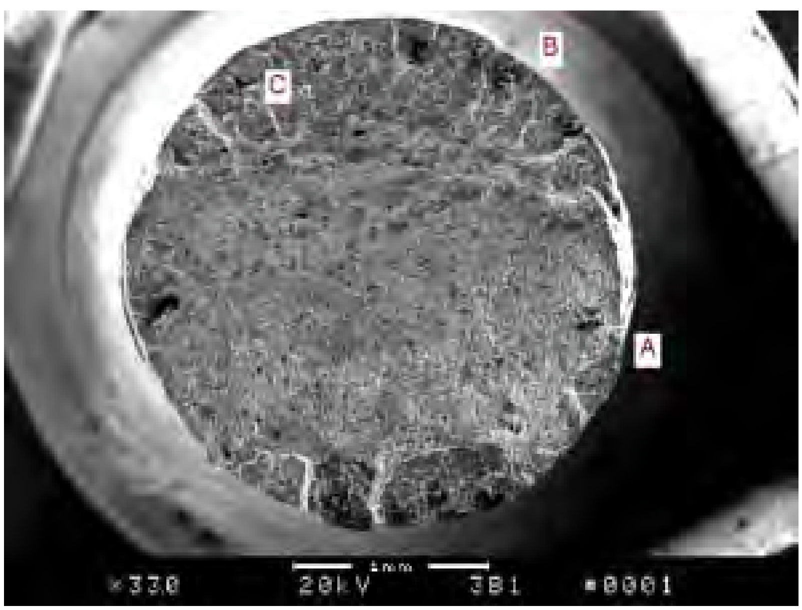

Fig. 3: Fracture Surface of Low Fluence 304 Specimen (~0.08 dap)

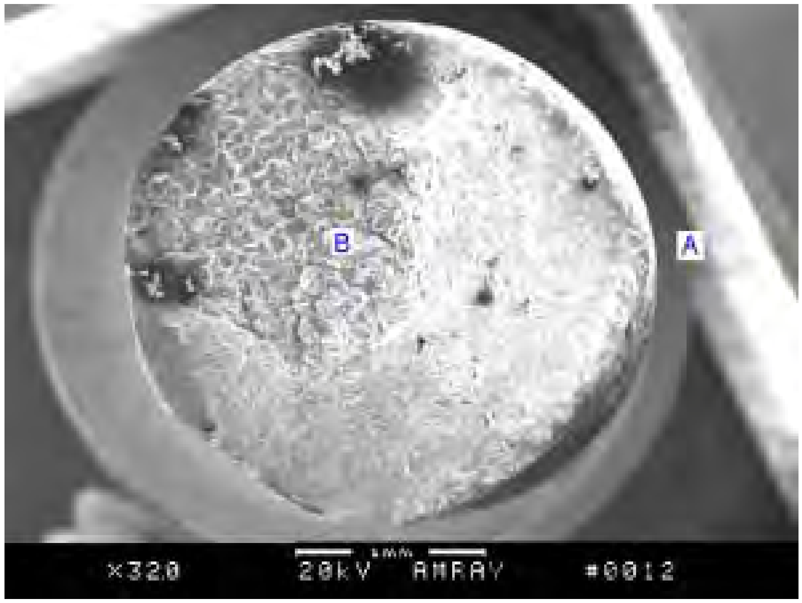

Fig. 4: Fracture Surface of Higher Fluence 304 SS Specimen (~17 dpa)

Fracture toughness test was also conducted on actual bolt configurations (not a standard specimen) and CT specimens of decommissioned 304 SS materials. Results show toughness reduction due to radiation. Additional TEM and fracture toughness tests are being performed. Also being performed are crack growth tests which would shed further insights into corrosion behavior of PWR irradiated materials.

3. HOT CELL TEST AND MICROSTRUCTURE EXAMINATION OF FAST REACTOR IRRADIATED MATERIALS

Availability of PWR irradiated materials is limited, particularly highly irradiated ones that could be studied to simulate extended life conditions. An accelerated approach was thus taken to irradiate materials in the Bor-60 fast reactor in Russia. This test reactor can yield fluence up to 20 dpa per year, 10 to 20 times faster than normal commercial PWRs.

Table 1 shows the initial batch of US PWR commercial materials irradiated in Bor-60. These samples are of 347, 316CW, 316SA, 304SA and 308. All samples were for tensile specimens only with the highest radiation goal of 80 dpa.

Table 1: First Batch US Materials Irradiated in Bor-60

Material - No. of heats (Code) Supplier	Fluence 20dpa	Fluence 40dpa	Fluence 60dpa	Fluence 80dpa
Type 347 - 1 heat (EC) (W)	4	2		3
Type 316CW - 2 heats (EA & EB) (W)	5	2		5
Type 316SA - 1 heat (ED) (CE)		3	4	
Type 304SA - 2 heats (FD & EH) (FTI & CE)	4	4	4	3
Type 308 - 1 heat (FE) (FTI)	5			4

Table 2 lists the second batch of US PWR core internals material samples irradiated in Bor-60. These samples are of 308 Tig/Mig weld, 304 HAZ, 304SA, 304CW, 316 and as-received and thermally aged cast austenitic stainless steel (CASS). Irradiated specimens were expanded to cover tensile test, slow strain rate test, crack initiation test (O-ring), fracture toughness test (CT) and microstructure characterization (TEM).

Table 2: Second Batch US Materials Irradiated in Bor-60

Materials	Specimen type	5 dpa	10 dpa	20 dpa	40 dpa
308 TIG/MIG weld and 304 HAZ	Tensile O-ring CT 3mm disc	12 2 1 16	9 2 2 16	15 4 2 16	4 0 0 16
304 SA and CW	Tensile O-ring CT 3mm disc	6 1 0 8	12 2 1 16	14 3 2 16	5 2 2 16
316 two heats	Tensile O-ring CT 3mm disc	0 0 0 0	12 3 0 16	7 3 0 16	5 1 0 16
Cast austenitic - as-received & two levels of thermal aging	Tensile O-ring CT 3mm disc	19 5 6 24	13 3 6 24	0 0 0 0	0 0 0 0

Selected tensile test results of the Bor-60 irradiated materials are shown in Figs. 5 and 6. They are similar to those obtained from the PWR irradiated materials with yield strength (YS) and ultimate tensile strength (UTS) saturated around 10 dpa. Figures 7 and 8 show uniform elongation and total elongation. For 304 and 316, saturation was observed around 20 and 30 dpa, respectively.

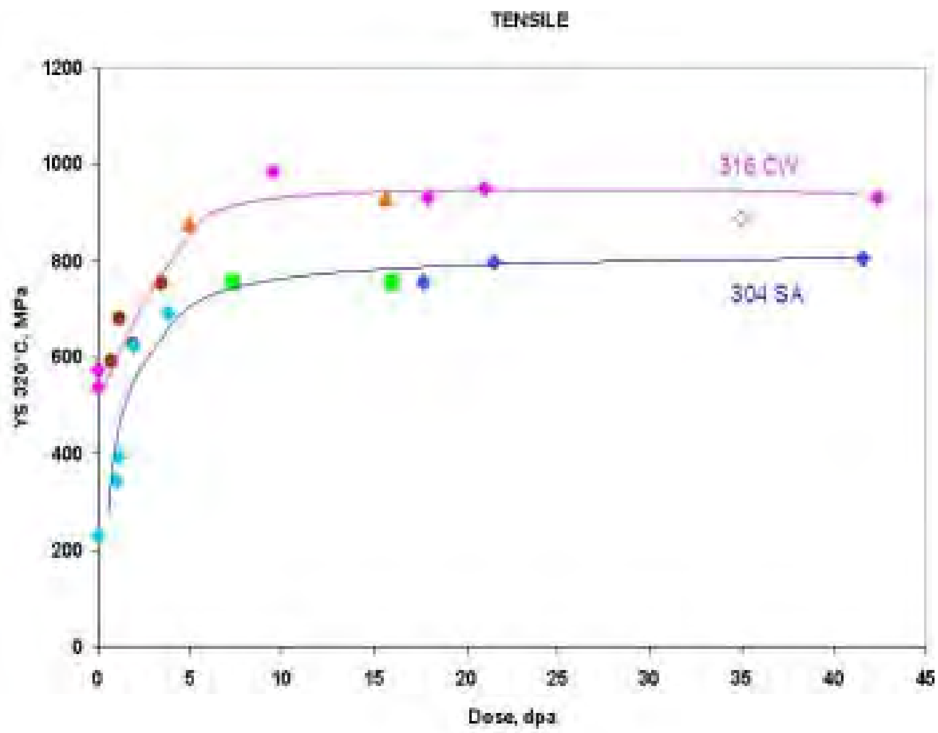

Fig. 5: Yield Strength of Bor-60 Irradiated Materials

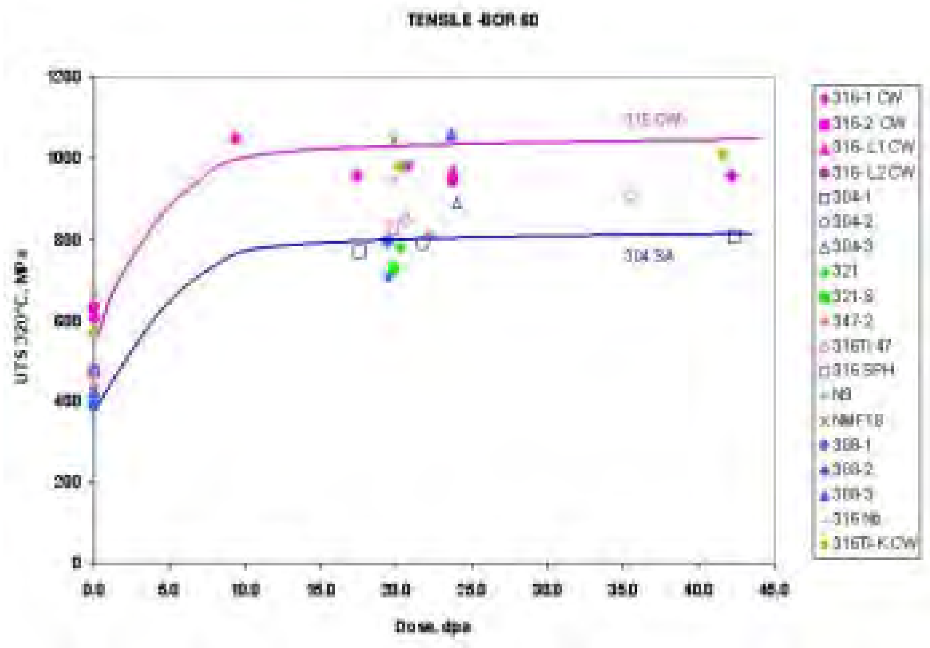

Fig. 6: Ultimate Tensile Strength of Bor-60 Irradiated Materials

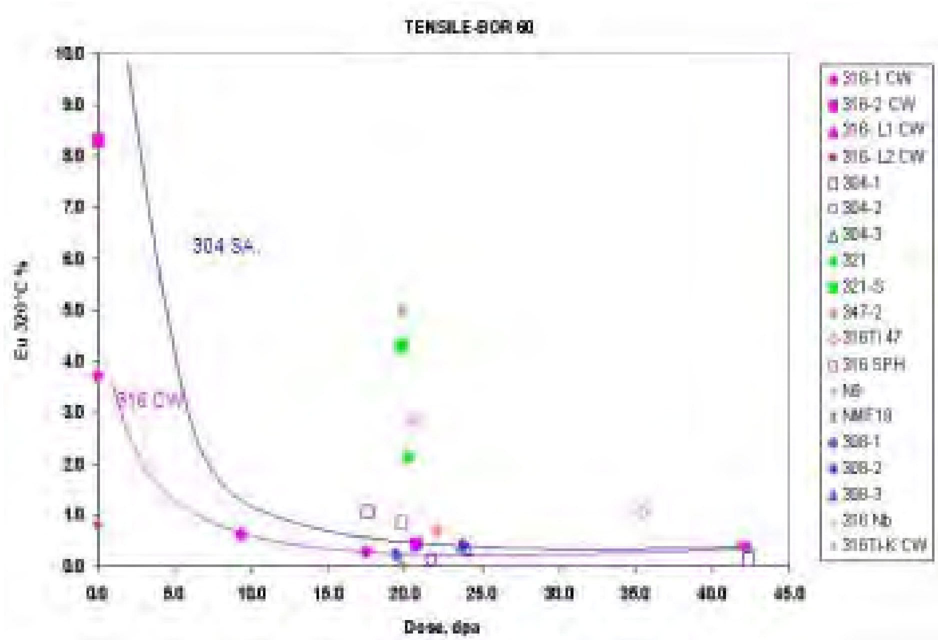

Fig. 7: Uniform Elongation of Bor-60 Irradiated Materials

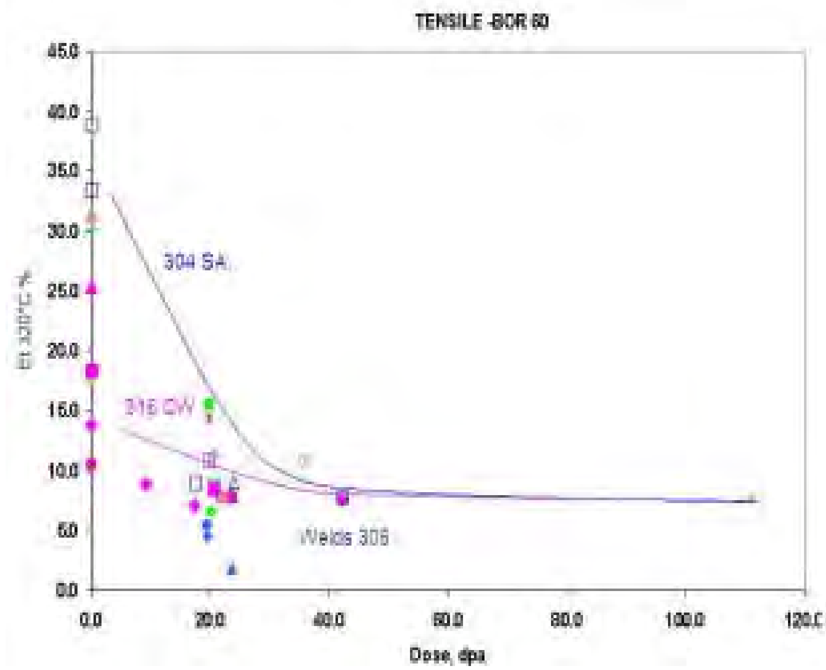

Fig. 8: Total Elongation of Bor-60 Irradiated Materials

Since Bor-60 is a fast reactor and its operating conditions (e.g., neutron flux spectrum, temperature, and coolant) are different from those of PWRs, more data including tensile test, corrosion test, fracture toughness test, TEM, etc. will be collected to provide correlation and benchmark between the two.

4. DETERMINATION OF REACTOR INTERNALS OPERATING CONDITIONS

Analyses were performed based on operating histories to characterize radiation and temperature conditions of PWR irradiated specimens. This was needed for interpreting test results. A three-dimensional discrete ordinate neutron transport code, TORT, was used to calculate fluence and gamma ray heat generation rates. The latter was then input into a finite element code (ANSYS) to calculate the core temperature histories and distribution. Figure 9 is an example of temperature contours within a given bolt.

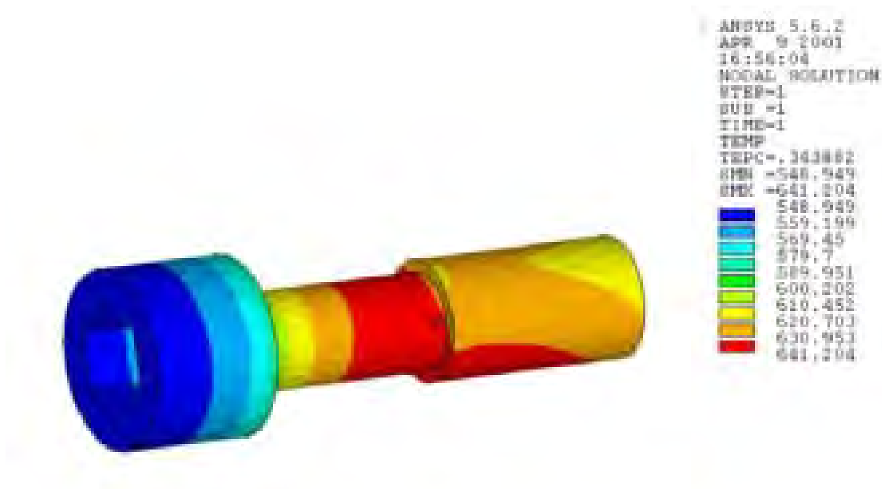

Fig. 9: Temperature Contours within a Given Bolt

Figures 10 and 11 respectively sshow finite element mesh and temperature contours of a sector of the decommissioned PWR baffle and former assembly.

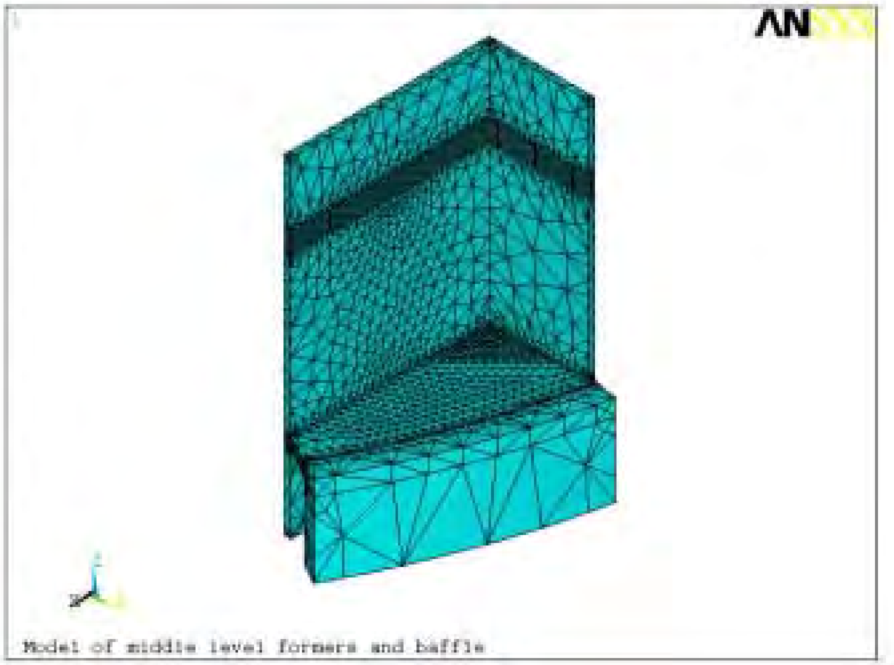

Fig. 10: Finite Element Mesh of Baffle and Former Samples

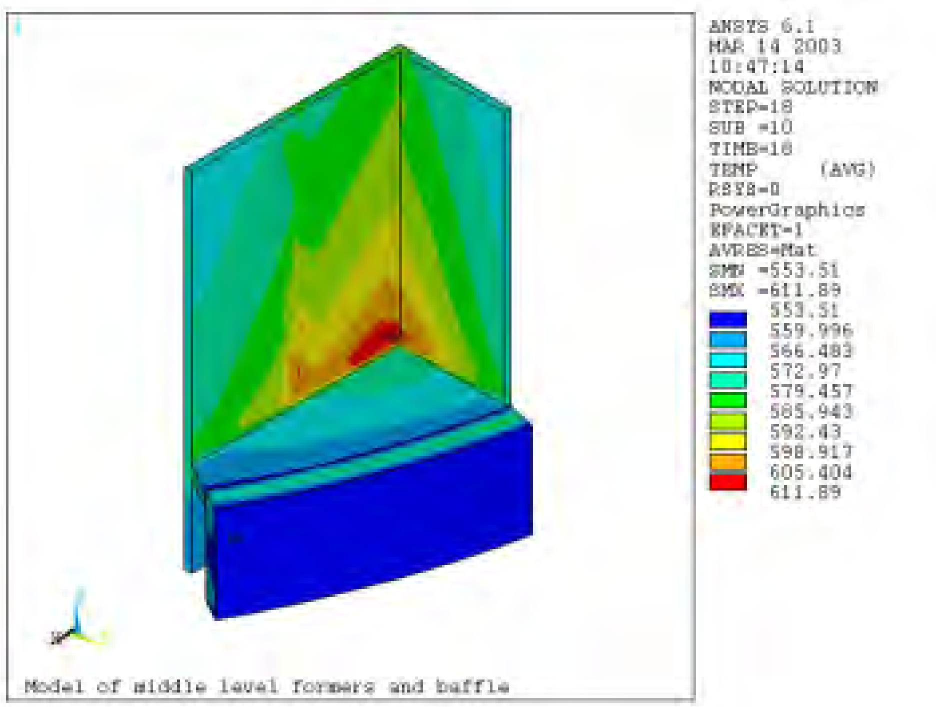

Fig. 11: Temperature Contours of Baffle and Former Assembly

5. EVALUATION OF IRRADIATION INDUCED SWELLING AND STRESS RELAXATION

Irradiation-induced void swelling and irradiation-enhanced stress relaxation (creep) are two degradation mechanisms, depending on their evolution over time, might impact long-term performance of core internals components.

Void formation is a mechanism in which radiation-induced vacancies accumulate in metal to form microscopic voids. A large number of voids can lead to volumetric increase (dimensional change), termed void swelling, in the affected component. This volumetric increase could then result in unaccounted for loads at connection points (for example, at bolted or welded joints of structural members). If void swelling is sufficiently large, it can further reduce material's toughness.

Irradiation-enhanced stress relaxation (or creep in a displacement controlled situation) is caused by neutron irradiation facilitated microscopic displacement of stressed metal much like that resulting from exposure to elevated temperatures. Irradiation-enhanced stress relaxation can result in bolt losing its preload. While this is undesirable, it has a beneficial effect by mitigating loads resulting from void swelling.

Currently available void swelling data are primarily from fast and fusion reactor research having non-typical PWR conditions. To date, there is no evidence that PWR spectrum induced void swelling would be equivalent or even close to that of fast reactors. The fast reactor conditions (the level of high temperature and high fluence) under which large swelling was possible are in general do not occur in PWR internals since the region facing the core (high fluence) is well cooled by the reactor coolant while the regions away from the core where higher temperatures are possible is of moderated fluence. High flux and high temperature locations in PWR internals are limited and highly localized. Consequently, void swelling is not likely to have negative effect on long-term performance of reactor internals components. TEM of the PWR 304 SS materials which have fluence up to about 20 dpa is being performed to provide benchmark data needed. Also, controlled experiment in test reactor is being initiated for investigating fundamental mechanism of PWR spectrum induced void swelling.

6. DEVELOPMENT OF LONG-TERM AGING MANAGEMENT STRATEGY, CRITERIA AND PROCESS

A strategic plan for aging management of PWR is being developed. This strategy will use 1) available knowledge of internals design, materials of construction, and material properties; 2) operating experience and age-related degradation mechanism knowledge that provides susceptibility criteria for known aging mechanisms; and 3) supplementary plant surveillance, testing, monitoring, and examinations, as appropriate for the susceptible internals components. Continued evaluation of the aging degradation effects will be incorporated as the results become available, and data from research test programs will be examined for improved understanding of the aging and degradation mechanisms. The ultimate goal of the strategy is to produce a set of inspection and functionality assessment guidelines for aging management of PWR internals.

7. REFERENCES

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