Characterization of Decommissioned PWR Vessel Internals Material Samples

Material Certification, Fluence, and Temperature (Nonproprietary Version)

This report describes research sponsored by EPRI and the U.S. Department of Energy (Award No. DE-FC07-00NE22796).

Export U.S. Control

Technical Report

Characterization of Decommissioned PWR Vessel Internals Material Samples

Material Certification, Fluence, and Temperature (Nonproprietary Version)

1009799

Topical Report, September 2004

Cosponsor
U.S. Department of Energy,
Nuclear Energy Plant Optimization (NEPO)
Office of Nuclear Energy, Science and Technology
1000 Independence Avenue, S.W.
Washington, D.C. 20585-1290

EPRI Project Manager
H.T. Tang

EPRI • 3412 Hillview Avenue, Palo Alto, California 94304 • PO Box 10412, Palo Alto, California 94303 • USA
800.313.3774 • 650.855.2121 • askepri@epri.com • www.epri.com

DISCLAIMER OF WARRANTIES AND LIMITATION OF LIABILITIES

ORGANIZATION(S) THAT PREPARED THIS DOCUMENT

Westinghouse Electric Company LLC

ORDERING INFORMATION

Requests for copies of this report should be directed to EPRI Orders and Conferences, 1355 Willow Way, Suite 278, Concord, CA 94520, (800) 313-3774, press 2 or internally x5379, (925) 609-9169, (925) 609-1310 (fax).

CITATIONS

This report was prepared by

Westinghouse Electric Company LLC
Science and Technology Department
1310 Beulah Road
Pittsburgh, PA 15235

Principal Investigators
M. Krug
R. Shogan

Westinghouse Electric Company LLC
Nuclear Services Business Unit
4350 Northern Pike
Monroeville, PA 15146-2094

Principal Investigators
A. Fero
M. Snyder

This report describes research sponsored by EPRI and the U.S. Department of Energy (Award No. DE-FC07-00NE22796. Task FY02 3-27).

The report is a corporate document that should be cited in the literature in the following manner:

Materials Reliability Program: Characterization of Decommissioned PWR Vessel Internals Material Samples: Material Certification, Fluence, and Temperature (Nonproprietary Version), EPRI, Palo Alto, CA, and U.S. Department of Energy, Washington, DC: 2004. 1009799.

PRODUCT DESCRIPTION

Pressurized water reactor (PWR) cores operate under extreme environmental conditions due to coolant chemistry, operating temperature, and neutron exposure. Extending PWR life requires detailed knowledge of the changes in mechanical and corrosion properties of structural austenitic stainless steel components (internals) adjacent to the fuel that are inherent to such conditions. Materials samples of baffle plate, former plate, and core barrel from a decommissioned PWR were studied to determine the effects of reactor service on the materials' mechanical and corrosion properties.

Results & Findings

Retrieval of materials certification documentation verified that all material chemistries conformed to the American Society for Testing and Materials (ASTM) specifications for Type 304 stainless steel. All plate material (former, baffle, and core barrel) was confirmed to have been in the annealed condition [Yield Stress $\sigma_Y \approx$ 35 ksi (240 MPa), Tensile Strength $\approx$ 80 ksi (550 MPa)] on installation. The two bolts retrieved may have been in either of two conditions: annealed or with 6-8% cold work. Hardness of the harvested materials upon Westinghouse receipt was approximately 80 HRB (150 HV) for nearly unirradiated material and increased to a saturation value of approximately 35 HRC (350 HV) for neutron exposures of 5 dpa (displacement per atom) and above. Radiation exposures ranged from several hundredths of a dpa (top of core baffles) to about 22 dpa (core midplane baffles). Average material temperatures were in the range of 560-620°F (293-327°C) during the first seven cycles and 535-595°F (279-313°C) for cycles 8-11 due to a reduction of core inlet temperature at that time.

Challenges & Objective(s)

Sections of the reactor's baffle, former, barrel, and baffle-former bolts, all manufactured from Type 304 stainless steel, were recovered. By calculating radiation and temperature history, changes in mechanical and corrosion properties can be correlated against influences that allow generic comparison with other plants. The magnitude of these changes (to be presented in future reports) that evolve with radiation and thermal environments will become part of a data set to support continued plant operation beyond its designed life.

Applications, Values & Use

The radiation and thermal histories in this report will be used in subsequent studies as variables to correlate with changes in strength, fracture toughness, and corrosion resistance. The methods used in these calculations can be applied for similar future investigations in other plants where samples become available or for evaluating conditions in an operating unit.

EPRI Perspective

This report contains basic material characterization information of the as-installed samples of reactor internals material harvested from a decommissioned PWR. In addition, the methodology and calculation of the thermal and radiation history of the plant are reported. Results provide quantitative information of operating variables that will be needed to understand and interpret hot cell test results from future studies. Two reports document the results. A non-proprietary version, the current one, provides a summary. The related proprietary version, EPRI technical report 1008202, contains detailed results and data.

This program represents the first time that significant amounts of PWR-irradiated internals materials are available to allow study of the effects of reactor service on the materials' mechanical and corrosion properties. Related EPRI reports include *Materials Reliability Program: Determination of Operating Parameters of Extracted Bolts - MRP-52* (1003076, October 2001), *Materials Reliability Program: Hot Cell Testing of Baffle /Former Bolts Removed From Two Lead Plants - MRP-51* (1003069, November 2001), and *Materials Reliability Program: Characterizations of Type 316 Cold Worked Stainless Steel Highly Irradiated Under PWR Operating Conditions - MRP 73* (1003525, August 2002).

Approach

Baseline chemical and mechanical property data were established by retrieving the original materials certification documentation for the reactor internals. The radiation exposure calculations used a three-dimensional model of roughly one-eighth of the core. Radiation exposure values were converted into units of stainless steel dpa at three state-points per cycle: beginning of cycle (BOC), middle of cycle (MOC), and end of cycle (EOC) for each of the plant's eleven cycles for a total of 33 state-points. Temperatures of program materials were estimated by three 3-dimensional finite element models (former and midplane baffles, top of core baffles, and midplane barrel) generated with ANSYS code. In addition to reactor coolant temperature data, this model used results from the radiation exposure calculation to incorporate gamma heating effects.

Keywords

Pressurized water reactor
Radiation effects
Radiation history
Thermal history
Reactor internals
Stainless steel

ABSTRACT

Pressurized water reactor (PWR) cores operate under extreme environmental conditions due to coolant chemistry, operating temperature, and neutron exposure. Extending the life of PWRs requires detailed knowledge of the changes in mechanical and corrosion properties of the structural austenitic stainless steel components (internals) adjacent to the fuel that are inherent to such conditions. This program represents the first time that significant amounts of PWR internals have been available to allow the effects of reactor service on the materials' mechanical and corrosion properties to be determined. This report contains basic material characterization information of the as-installed samples of reactor internals material which were harvested from a decommissioned PWR. In addition, the methodology and results of the calculation of the thermal and radiation history of the plant are also reported.

Retrieval of materials certification documentation verified that all material chemistries conformed to ASTM specifications for Type 304 stainless steel. All plate material (former, baffle, and core barrel) was confirmed to have been in the annealed condition [$\sigma_Y \approx 35$ ksi (240 MPa), T.S. (Tensile Strength) ≈ 80 ksi (550 MPa)] upon installation. The two program bolts may have been in either of two conditions: annealed or with 6-8% cold work. Hardness of the program materials upon Westinghouse receipt was approximately 80 HRB (150 HV) for nearly unirradiated material, and increased to a saturation value of approximately 35 HRC (350 HV) for neutron exposures of 5 dpa and above. Radiation exposures for the program materials were calculated to range from several hundredths of a dpa (top of core baffles) to about 22 dpa (core midplane baffles). The core temperature in general was approximately 25°F (14°C) cooler for cycles 8-11 than for the first seven cycles due to a reduction of core inlet temperature at that time. Program materials were calculated to reach average temperatures in the range of 560-620°F (293-327°C) during the first seven cycles, and 535-595°F (279-313°C) during the remaining four cycles.

CONTENTS

LIST OF FIGURES

LIST OF TABLES

1
OBJECTIVES

The program to test and characterize reactor internals material samples harvested from a decommissioned PWR consists of several parts. The objectives of the part of the program that is documented in this report can be summarized as follows:

- To establish baseline unirradiated materials property data from materials certification sheets
- To perform visual inspection of received materials
- To perform dimensional measurement on received materials to facilitate specimen location selection
- To create a three-dimensional time-dependant model of the reactor core and to calculate the irradiation histories of the program materials
- To create a three-dimensional time-dependant model of the reactor core and to calculate the thermal histories of the program materials

2 INTRODUCTION

Austenitic stainless steels were selected for the internals structures surrounding the core in the PWR design because of their relative strength, ductility and resistance to corrosion in the PWR water environment. More recently studies have shown that the neutron radiation environment affects this corrosion resistance[1,2]. A few actual failures of internals bolting as well as of similar alloys used for control rod sheathing have been reported[3,4]. These failures have led to studies of irradiation effects on stainless steels, particularly for bolting applications. The alloys studied were mostly Types 316 and 347 since they were available after bolt or bottom mounted instrument tube replacement. Few studies included Type 304 at high fluence levels since it is used for larger structures not easily removed. Irradiation effects in Type 304 are of particular interest, however, since most of the reactor internals are constructed of this alloy and BWR experience shows that it is susceptible to irradiation assisted stress corrosion cracking (IASCC).

In addition Type 304 stainless steel is the austenitic alloy most susceptible to irradiation induced void swelling from breeder reactor experience. While little evidence of significant swelling has been found in other PWR austenitic stainless steel alloys irradiated to high fluence, Type 304 material has generally not been available for study.

When sections of stainless steel from the internals region became available during the decommissioning of a PWR reactor, the EPRI Materials Reliability Program (MRP) Reactor Internals Issue Task Group (RI-ITG) was able to obtain material for detailed characterization. Studies planned for these materials include:

- Thermal and irradiation history calculations
- Tensile properties
- Fracture toughness measurements

---

[1] Conermann, J., Shogan, R., International IASCC Advisory Committee, Phase 2 Final Report, Westinghouse Electric Co. LLC, January, 2004.

[2] Material Reliability Program: Hot Cell Testing of Baffle/Former Bolts Removed From Two Lead PWR Plants (MRP-51), EPRI, Palo Alto, CA:2001.1003069

[3] Matsuoka, T., et al., Performance of PWR RCCA Rodlets Regarding Cladding Tube Cracking Caused by Absorber Swelling, Proceedings of ICON 5, 5th International Conference on Nuclear Engineering, May 26-30, 1997, Nice, France.

[4] Sipush, P. J., et al., Lifetime of PWR Silver-Indium-Cadmium Control Rods, EPRI-NP-4512.

- Slow strain rate tensile (SSRT) testing to measure IASCC susceptibility
- Corrosion crack initiation
- Corrosion crack growth
- Microstructural studies to correlate microstructural changes with mechanical and corrosion property changes
- Swelling measurements

The program materials were taken from the baffle, former, barrel and baffle-former bolts of the decommissioned reactor. All of these components were constructed of Type 304 stainless steel. In this report background information about the materials and sections obtained for study and their calculated thermal and fluence histories are reported.

3
MATERIAL CHARACTERIZATION

3.1 Introduction

Sections removed from the internals of a decommissioned pressurized water reactor were characterized prior to sectioning and machining into mechanical and corrosion test specimens. The characterization included photography, dimensioning, surface flatness profiling, and hardness measurements.

In addition, the materials certification sheets for the four materials were retrieved. This section summarizes the relevant materials properties from these certifications which supply unirradiated baseline data for determining the extent of irradiation induced property changes in the materials.

3.2 Material Description

3.2.1 Visual Exam

The components that were sampled for specimen material were the core barrel, the former, and the baffle plates. For the latter, material was taken from both the core midplane and the top of core positions. For the other materials only the near core midplane locations were retrieved. The reactor locations of the actual baffle and former samples are shown in Figure 3-1, and the location of the barrel sample is shown in Figure 3-2.

Photographs of the as-received baffle samples are shown in Figures 3-3 through 3-6. The four baffle samples were arbitrarily numbered from 1 through 4. Based on a visual examination of the edge conditions (machined, water jet cut, etc.), wear marks, and relative radiation levels from the ends of the samples, the orientation of the baffle samples in the reactor was determined. This orientation is shown in Figure 3-7. Photographs of the as-received former and barrel samples are shown in Figures 3-8 and 3-9, respectively.

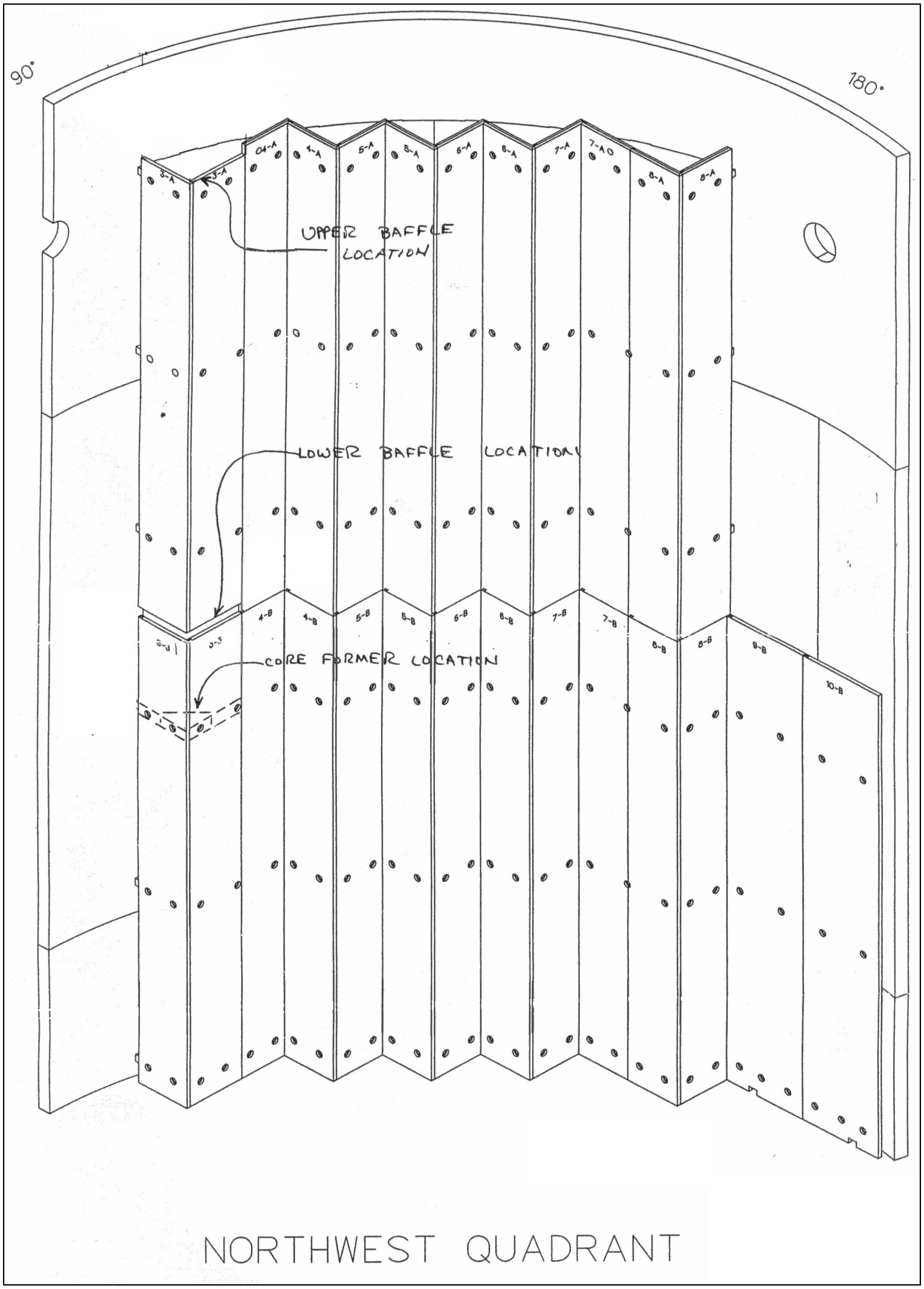
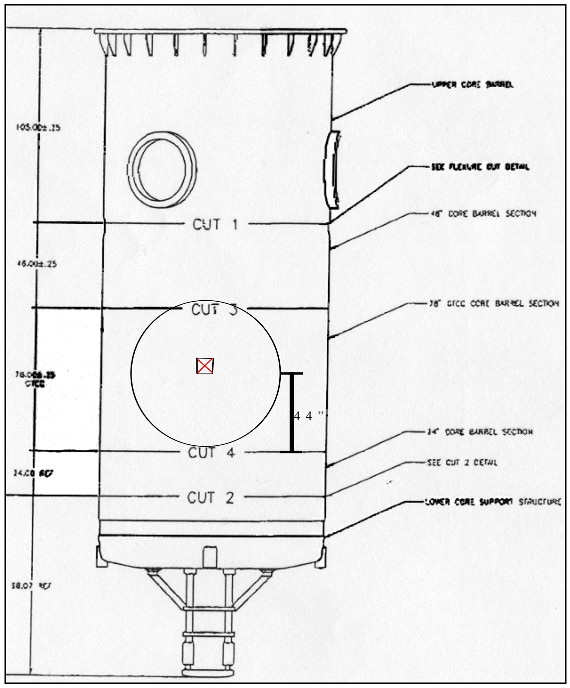
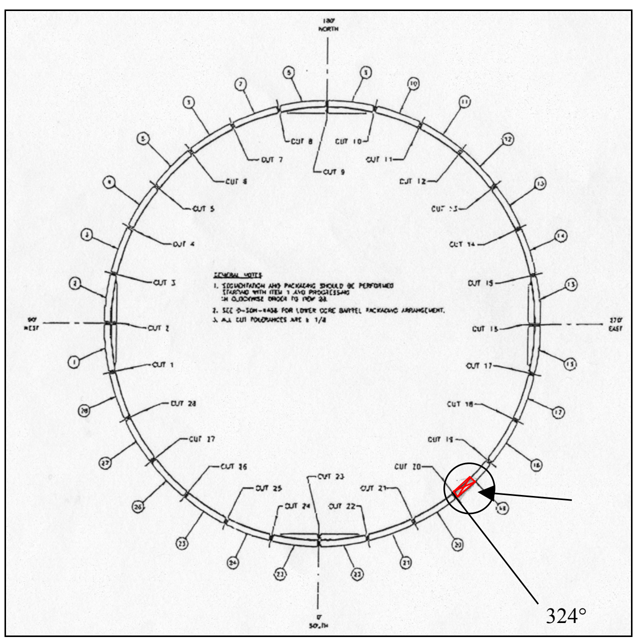


Figure 3-1
Core Locations of the 4 Removed Baffle Samples and the Removed Former Sample



(a) A side view of the barrel showing the height of the sample in the core.



(b) A top view of the barrel showing the diametral location of the sample.

Figure 3-2
Location of the Barrel Sample

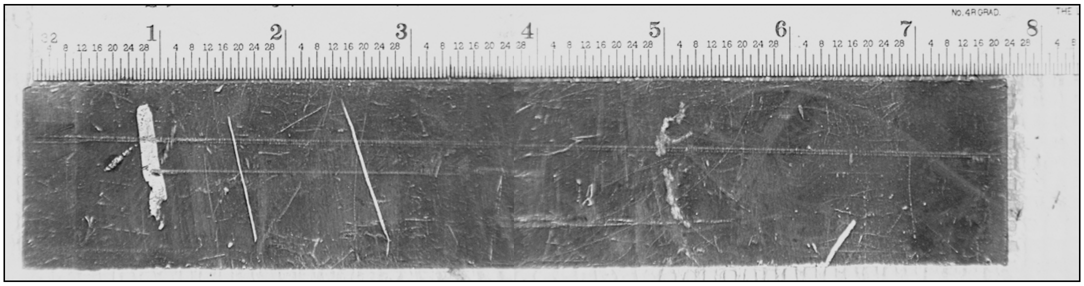


Figure 3-3
Baffle Section 1

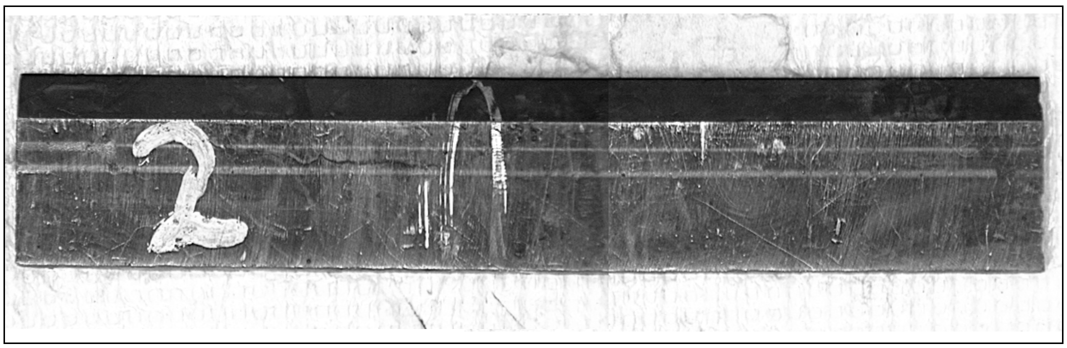


Figure 3-4
Baffle Section 2

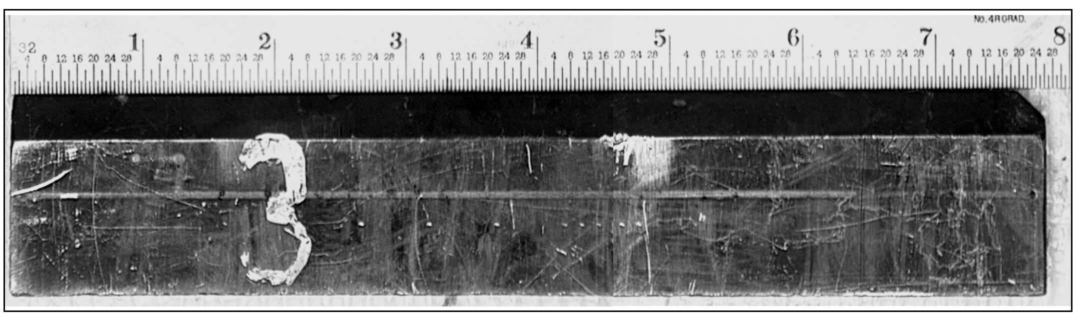


Figure 3-5
Baffle Section 3

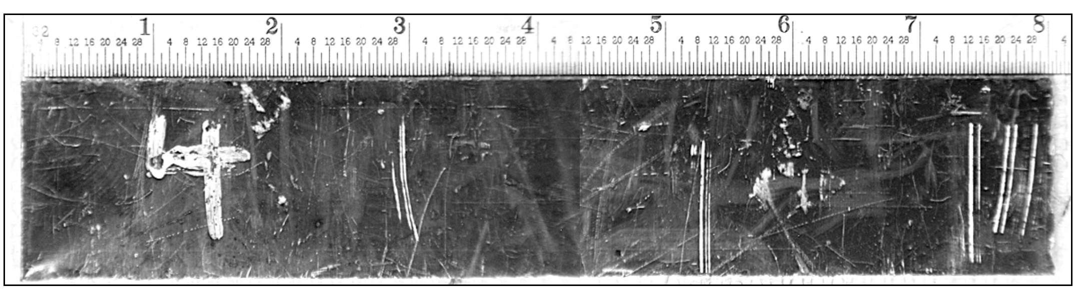


Figure 3-6
Baffle Section 4

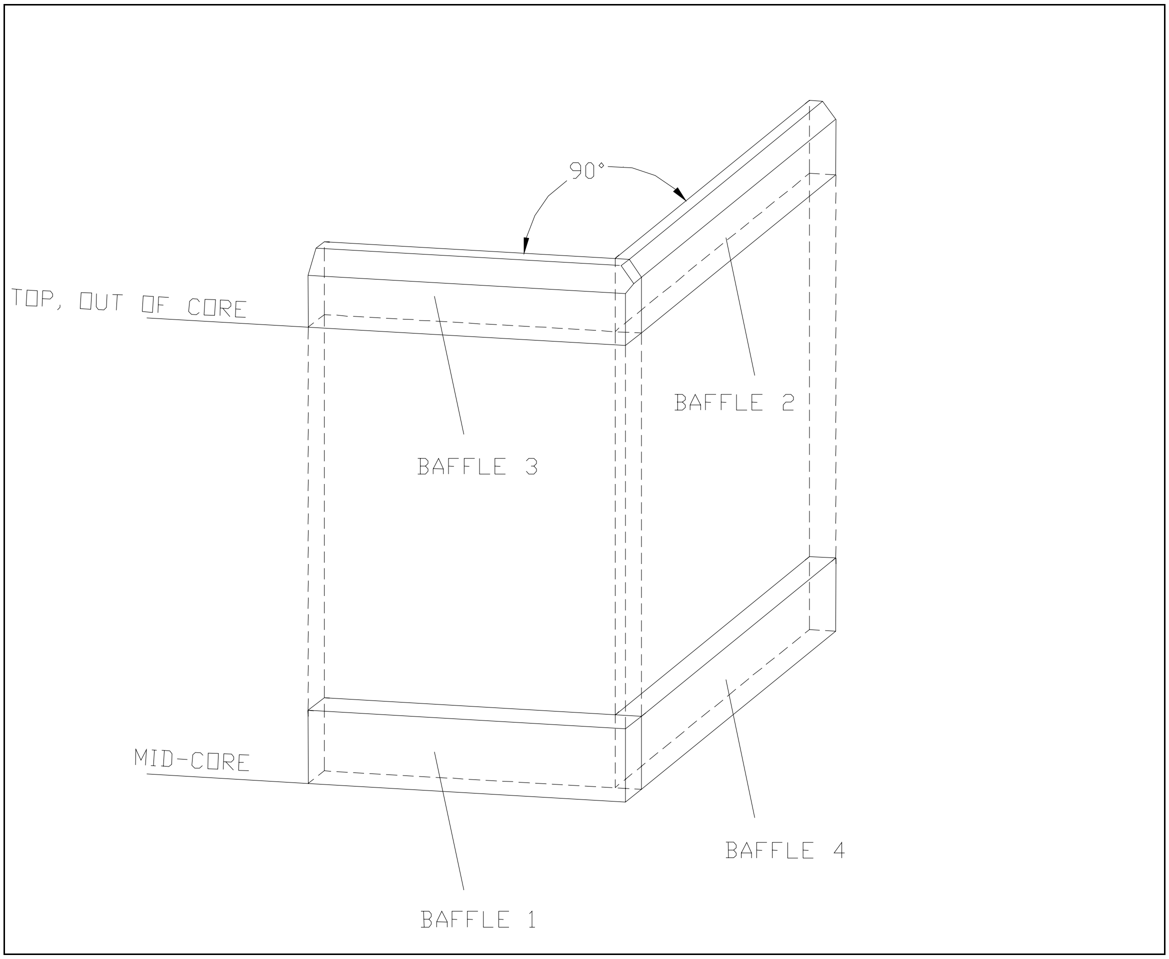


Figure 3-7
Truncated Schematic Showing Relative Baffle Orientation

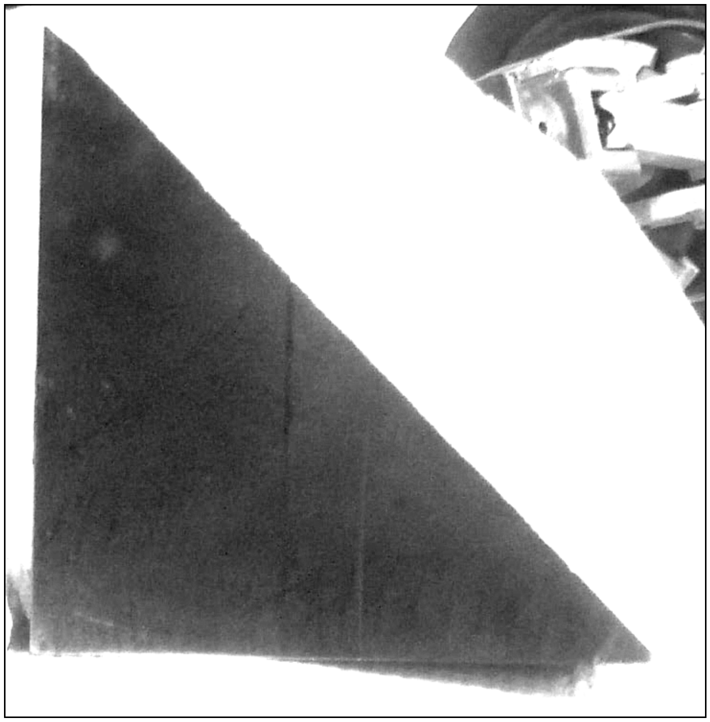


Figure 3-8
Sample Taken from Former #3

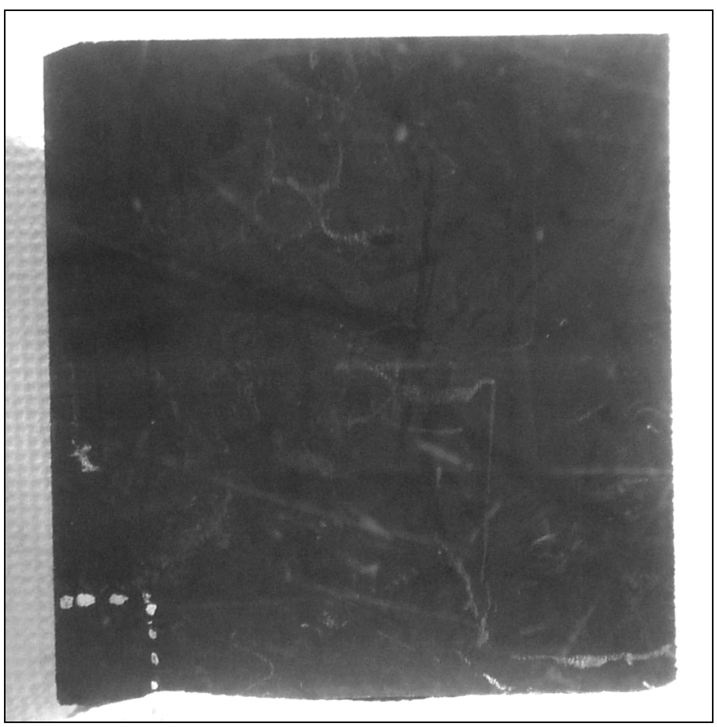
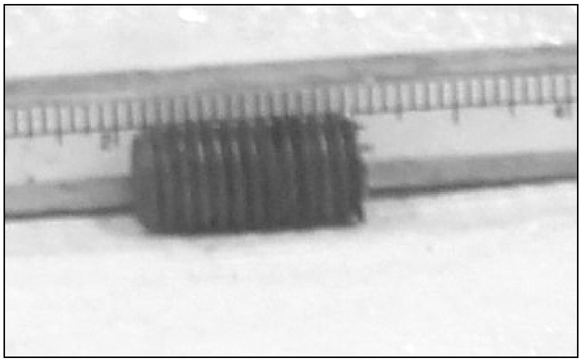
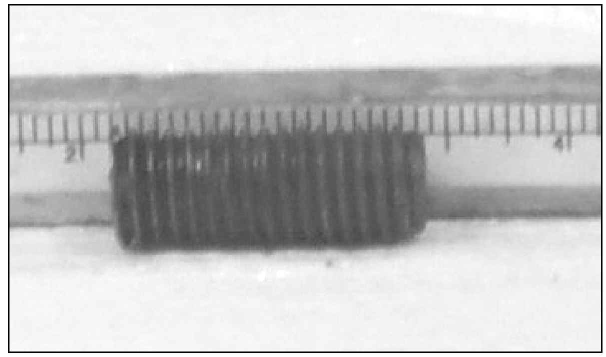


Figure 3-9
Core Barrel Section Showing Area Containing Weld Material

The former sample contained the thread section of two baffle-former bolts. These were center-drilled and removed with an easy-out. The bolt segments turned easily out of the former. Photographs of the bolt segments are shown in Figure 3-10.



(a)



(b)

Figure 3-10
Two Bolt Segments Joining the Former to the Baffle Plates Were Removed from the Former Sample

No visible indication of weld metal could be seen on the barrel sample. A small magnet was scanned over the sample and one corner was found to be magnetic (weld metal contains some ferrite phase which is magnetic). The scan was repeated with a (eddy current) ferrite probe. This scan also verified that only a small sliver of weld might exist on one corner. It is not expected that any specimens could be taken from the small available volume.

3.2.2 Surface Profiles

A thorough examination of the high probability swelling areas was performed for visible indications of swelling in the materials. No fit-up edge mushrooming or wear, nor any visual evidence of swelling could be observed.

Since swelling might be present, if at all, as small localized internal volumes at maximum irradiation temperature locations, surface flatness profiles were measured across high probability swelling areas on samples taken from the core midplane region to look for swelling-induced surface bulges. No evidence of surface bulges was found in any of the materials tested.

3.3 Material Certification Data

Baseline properties of the program materials were obtained from the original material certification sheets retrieved from the quality control records. The records indicate that there were single heats of material used for the core barrel, baffle plates and baffle-former bolts in this plant. However, there were four heats of Type 304 stainless steel used to manufacture the formers. It is not known if chemical analyses performed in future program tasks will be able to determine which heat was used for the program former section because of the similarity of the four heats. The chemistries of all materials, shown in Table 3-1, meet the requirements of

Type 304 stainless steel, with some with slight material-to-material variations in the trace element composition.

Table 3-1
Nominal Compositions of Program Materials (wt %)

Source	C	Mn	P	S	Si	Cr	Ni	Co	Mo	Cu
Core Barrel	0.061	1.22	0.016	0.01	0.52	18.54	9.17	0.07	—	—
Baffle Plate	0.050	1.09	0.014	0.007	0.62	18.12	9.14	0.08	—	—
Former[1]	0.057 0.050 0.058 0.045	1.61 1.56 1.73 1.67	0.020 0.028 0.023 0.023	0.011 0.011 0.012 0.013	0.69 0.79 0.66 0.46	18.32 18.05 18.49 18.32	9.3 9.36 9.35 9.27	0.04 0.04 0.04 0.04	0.22 0.23 0.34 0.46	0.25 0.29 0.28 0.34
Baffle/Former Bolt	0.050	1.65	0.025	0.024	0.55	18.64	9.24	0.12	0.4	0.34
ASTM A240 Type 304	0.08 max	2.2 max	0.045 max	0.03 max	1.0 max	17–19	8–10	—	—	—
Note: 1. four heats of material										

Table 3-2 shows the mechanical properties of the program materials from the material certification records. The room temperature mechanical properties of the plate materials (the baffle, former and barrel) showed yield strengths ranging from 35 to 40.8 ksi indicating the fully annealed condition. The plant baffle–former bolts appear to have all been made from one heat of steel, but the reported mechanical properties (Table 3-2) on the final bar material used for machining the bolts indicates two distinct sets of properties. The properties shown by one test, 30.9 ksi yield strength and 70.7% elongation, are typical of annealed material. The second set showed a 76.4 ksi yield strength and 45% elongation which is typical of material with 6–8% cold work. It is not possible at the present to determine into which category either of the two bolts available to this program fit.

Table 3-2
Typical Mechanical Properties of Program Materials, Room Temperature

Source	YS 0.2% ksi (MPa)	UTS ksi (MPa)	% Elong.	% R.A.	BHN
Core Barrel	34.5 (238) 35.5 (245)	78.5 (541) 80.5 (555)	60 63	69 71	137 156
Baffle Plate	36.6 (252) 39.0 (269) 39.4 (272)	83.8 (578) 83.0 (572) 83.8 (578)	60 60 62	62 61 61	156 149 149
Former[1]	43.6 (301) 45.0 (310)	84.9 (585) 85.0 (586)	65 64	68 67	170
	35.0 (241) 35.4 (244)	81.4 (561) 83.0 (572)	60 60	70 70	148
	42.9 (296) 43.2 (298)	84.3 (581) 84.4 (582)	60 59	—	154
	40.8 (281) 41.0 (283)	80.6 (556) 81.0 (558)	60 62	—	156
Baffle/Former Bolt	30.9 (213) 76.4 (527)	84.3 (581) 100.3 (692)	70.7 45.0	78.5 72.0	215 180

Note:
1. Four heats, two tests each

3.4 Hardness Measurements

Exposure to high levels of neutron irradiation results in an increase in strength and hardness.

A Rams Rockford Model 30R hardness tester was used on the Rockwell B and C scales to determine the hardness of selected specimen blanks machined from the program materials. The top-of-core baffle samples serve as a source of low-fluence data for Type 304 steel, and remaining samples exhibit significant gradients in fluence. The program 304SS ranges from neutron exposures of several hundredths of a dpa to roughly twenty dpa. Neutron exposures at each hardness testing location were calculated by a method to be explained in Section 4.

Nearly unirradiated material was ~80 on the Rockwell B scale. Hardness increased with increasing neutron exposure until roughly 5 dpa, where it saturated at ~35 on the Rockwell C scale.

4
RADIATION EXPOSURE CALCULATION

4.1 Introduction

This section describes the calculations performed to determine the radiation environment that each of the irradiated stainless steel samples removed from the reactor internals of a decommissioned PWR experienced. The samples included core baffle plates (sections near the core midplane and near the top of the core), an inner corner section of a core former plate near the core midplane, and a section of the core barrel near the core midplane. The calculated radiation environment included gamma ray heat generation rates and fast neutron exposure parameters such as fast (E > 1.0 MeV and E > 0.1 MeV) neutron flux and fluence and stainless steel atomic displacements per atom (dpa). This data will be used to correlate observed changes in material properties among the various samples and to calculate the operating temperatures of the samples. The neutron exposure of each program test specimen was also calculated and will be reported with specimen test results in subsequent reports.

4.2 Method Discussion

The method of analysis consists of running three dimensional discrete ordinates radiation transport calculations using the TORT™ code [1] to determine the fast neutron fluxes and gamma ray heat generation rates in the baffle plates, former plates, and core barrel. The calculations are run in X-Y-Z and R-Θ-Z geometry for the outer portion of slightly more than an octant of the core. Calculations were run at three statepoints, beginning of cycle (BOC), middle of cycle (MOC), and end of cycle (EOC), for each fuel cycle.

4.3 Input–General

The input for the X-Y-Z and R-Θ-Z TORT calculations consists of the following:

- Basic geometry of fuel assemblies (assembly pitch, length of fuel rod end plugs, and top and bottom nozzle heights), core baffle plates, former plates, core barrel, thermal shield, and upper core plate.
- Assembly-wise core power distribution data including the initial ^{235}U enrichment, BOC and EOC burnups and relative assembly powers, cycle burnup, BOC, MOC, and EOC axial core power distributions, initial boron concentrations, fuel rod end plug lengths, top and bottom nozzle heights, and assembly pitch were obtained from the referenced Nuclear Design Report (NDR) WCAPs. Representative relative pin-by-pin power distributions for the peripheral fuel assemblies are used.

- Core inlet and outlet temperatures, reactor coolant system pressure, and the reactor thermal power.
- Water temperatures in the baffle-former-barrel bypass region.

4.4 Results of Analysis

There were a total of 99 TORT runs made to describe the radiation environment for the baffle-former-core barrel regions of the Decommissioned PWR reactor internals from which material was harvested. Each of the three TORT geometries were run for 11 fuel cycles and three statepoints (BOC/MOC/EOC) for each cycle.

Results are provided for fast (E > 1.0 MeV and E > 0.1 MeV) neutron flux and fluence and for stainless steel atomic displacements per atom (dpa). In addition gamma ray heat generation rates were calculated for input to the ANSYS finite element code to determine the operating temperatures of the samples. The thermal analysis of the Decommissioned PWR reactor internals is described in Section 5.

4.4.1 Macroscopic Sample Results

Results are provided to describe the radiation environment for the macroscopic samples that were harvested from the Decommissioned PWR reactor internals.

Table 4-1 provides the fast (E > 1.0 MeV) neutron fluence and stainless steel dpa data for the two core baffle plate samples harvested near the core midplane. Data are tabulated at several locations on the core side surface, at the middle of the plate, and at the back side surface of each of the samples.

Table 4-2 provides the fast (E > 1.0 MeV) neutron fluence and stainless steel dpa data for the former plate and core barrel samples harvested near the core midplane. Data are tabulated at several locations on the bottom surface, at the middle of the plate, and at the top surface of the former plate and at the core side surface, at the middle of the sample, and at the back side surface of the core barrel sample.

Table 4-3 provides the fast (E > 1.0 MeV) neutron fluence and stainless steel dpa data for the two core baffle plate samples harvested near the top of the core. Data are tabulated at several locations on the core side surface, at the middle of the plate, and at the back side surface of each of the samples.

Table 4-1
Macroscopic Baffle Plate Sample Data from Decommissioned PWR Core Midplane XYZ TORT

Macroscopic Sample Data from Decomissioned PWR Core Midplane XYZ TORT
Fast (E > 1.0 MeV) Neutron Fluence (n/cm^2) and Stainless Steel 304 dpa

Baffle Plate - Left

Core Side Surface			Middle of Plate			Back Side Surface		
Point	**Phi**	**dpa**	**Point**	**Phi**	**dpa**	**Point**	**Phi**	**dpa**
1	2.9E+21	4.2	**1**	2.5E+21	3.7	**1**	2.2E+21	3.2
2	1.6E+22	23.2	**2**	1.5E+22	22.2	**2**	1.5E+22	21.4
3	2.9E+21	4.2	**3**	2.5E+21	3.7	**3**	2.2E+21	3.2
4	1.6E+22	23.2	**4**	1.5E+22	22.2	**4**	1.5E+22	21.4

Looking at Sample from Core Side of Sample

Macroscopic Sample Data from Decomissioned PWR Core Midplane XYZ TORT
Fast (E > 1.0 MeV) Neutron Fluence (n/cm^2) and Stainless Steel 304 dpa

Baffle Plate - Right

Core Side Surface			Middle of Plate			Back Side Surface		
Point	**Phi**	**dpa**	**Point**	**Phi**	**dpa**	**Point**	**Phi**	**dpa**
1	1.5E+22	21.4	**1**	1.3E+22	19.2	**1**	1.2E+22	17.3
2	8.2E+21	11.9	**2**	6.9E+21	10.1	**2**	5.7E+21	8.4
3	1.5E+22	21.4	**3**	1.3E+22	19.2	**3**	1.2E+22	17.3
4	8.2E+21	11.9	**4**	6.9E+21	10.1	**4**	5.7E+21	8.4

Looking at Sample from Core Side of Sample

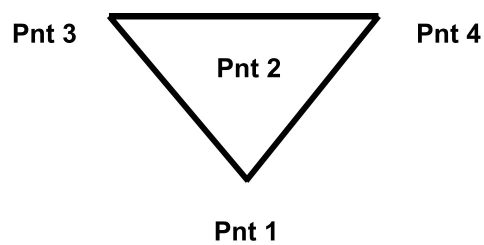
Table 4-2
Macroscopic Former Plate and Core Barrel Sample Data from Decommissioned PWR Core Midplane XYZ and RTZ TORTs

Macroscopic Sample Data from Decomissioned PWR Core Midplane XYZ TORT
Fast (E > 1.0 MeV) Neutron Fluence (n/cm^2) and Stainless Steel 304 dpa

Former Plate 3 - Inner Corner Sample

Bottom Surface			Middle of Plate			Top Surface		
Point	**Phi**	**dpa**	**Point**	**Phi**	**dpa**	**Point**	**Phi**	**dpa**
1	1.2E+22	18.1	**1**	1.2E+22	18.4	**1**	1.2E+22	18.2
2	3.5E+21	5.3	**2**	3.6E+21	5.5	**2**	3.5E+21	5.3
3	4.2E+21	6.3	**3**	4.4E+21	6.4	**3**	4.2E+21	6.3
4	7.5E+21	11.1	**4**	7.7E+21	11.4	**4**	7.5E+21	11.1

Looking Down at Sample from Above



Macroscopic Sample Data from Decomissioned PWR Core Midplane RTZ TORT
Fast (E > 1.0 MeV) Neutron Fluence (n/cm^2) and Stainless Steel 304 dpa

Core Barrel Sample

Core Side Surface			Middle of Sample			Back Side Surface		
Point	**Phi**	**dpa**	**Point**	**Phi**	**dpa**	**Point**	**Phi**	**dpa**
1	3.2E+20	0.462	**1**	2.5E+20	0.366	**1**	1.6E+20	0.240
2	3.6E+20	0.525	**2**	2.9E+20	0.415	**2**	1.8E+20	0.271
3	3.1E+20	0.451	**3**	2.5E+20	0.357	**3**	1.6E+20	0.234
4	3.6E+20	0.515	**4**	2.8E+20	0.406	**4**	1.8E+20	0.265

Looking at Sample from Core Side of Sample

Table 4-3
Macroscopic Baffle Plate Sample Data from Decommissioned PWR Top of Core XYZ TORT

Macroscopic Sample Data from Decomissioned PWR Top of Core XYZ TORT
Fast (E > 1.0 MeV) Neutron Fluence (n/cm^2) and Stainless Steel 304 dpa

Baffle Plate - Left

Core Side Surface			Middle of Plate			Back Side Surface		
Point	**Phi**	**dpa**	**Point**	**Phi**	**dpa**	**Point**	**Phi**	**dpa**
1	2.5E+19	0.037	**1**	2.4E+19	0.036	**1**	2.3E+19	0.035
2	6.9E+19	0.100	**2**	6.9E+19	0.100	**2**	6.8E+19	0.099
3	1.7E+19	0.025	**3**	1.7E+19	0.025	**3**	1.6E+19	0.025
4	4.1E+19	0.060	**4**	4.1E+19	0.060	**4**	4.1E+19	0.060

Looking at Sample from Core Side of Sample

Macroscopic Sample Data from Decomissioned PWR Top of Core XYZ TORT
Fast (E > 1.0 MeV) Neutron Fluence (n/cm^2) and Stainless Steel 304 dpa

Baffle Plate - Right

Core Side Surface			Middle of Plate			Back Side Surface		
Point	**Phi**	**dpa**	**Point**	**Phi**	**dpa**	**Point**	**Phi**	**dpa**
1	6.8E+19	0.099	**1**	6.6E+19	0.096	**1**	6.3E+19	0.092
2	4.9E+19	0.071	**2**	4.7E+19	0.068	**2**	4.4E+19	0.064
3	4.1E+19	0.060	**3**	4.1E+19	0.060	**3**	4.0E+19	0.058
4	3.0E+19	0.045	**4**	3.0E+19	0.044	**4**	2.9E+19	0.043

Looking at Sample from Core Side of Sample

4.5 References

1. Oak Ridge National Laboratory RSICC Computer Code Collection CCC-650, "DOORS 3.1 One-, Two- and Three-Dimensional Discrete Ordinates Neutron/Photon Transport Code System," August 1996.

5
THERMAL ANALYSIS

5.1 Introduction

The purpose of this section is to present the results of the thermal calculations performed, utilizing the internal heat generation rate data from Section 4, to determine the normal operating temperatures of the reactor internal components.

5.2 Method of Analysis

Temperatures were calculated using three-dimensional finite element models of each sample removed from the decommissioned PWR. In order to establish appropriate thermal boundary conditions, the models also included portions of the internals structure (core barrel, formers and baffles) adjacent to each sample. Three separate finite element models are used in the analyses: the former sample and the mid-baffle samples (Baffle samples #1 and #4), the top-baffle samples (Baffle samples #2 and #3), and the core barrel sample. The finite element models were constructed using the ANSYS computer code.

5.3 Results

As presented in Section 4, the internal heat generation rates vary from cycle to cycle as well as within each cycle. As a result, the component temperatures also vary within and from cycle to cycle. Three-heat generation rate state points {i.e., beginning of cycle (BOC), middle of cycle (MOC), and end of cycle (EOC)} for each cycle were utilized in these calculations. However, plant operating data (e.g. plant overpower events, reactor trips) and the resulting impact on variations in component temperatures have not been included in this assessment.

5.3.1 Time-Histories of Temperatures at Test Specimen Locations

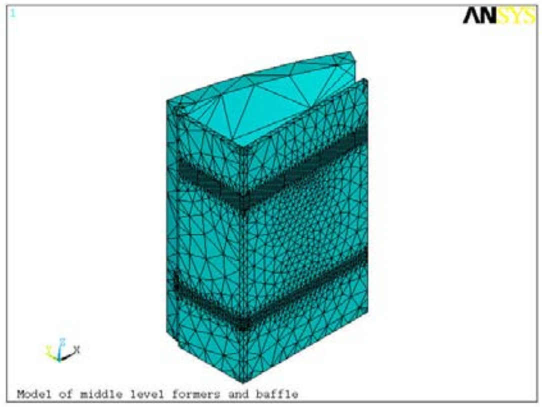
Based upon time measured in equivalent full power seconds (EFPS), temperature-time histories at each expected test sample location were calculated. The final number and location of each type of mechanical or corrosion test may differ somewhat from the current provisional data. Temperatures for final specimen locations will be provided with test results in subsequent reports.

All time histories show that the temperatures in cycles 1-7 are approximately 25°F higher in cycles 1-7 compared to cycles 8-11. This is due to a reduction of 25°F in the core inlet temperature during cycles 8-11 compared to cycles 1-7.

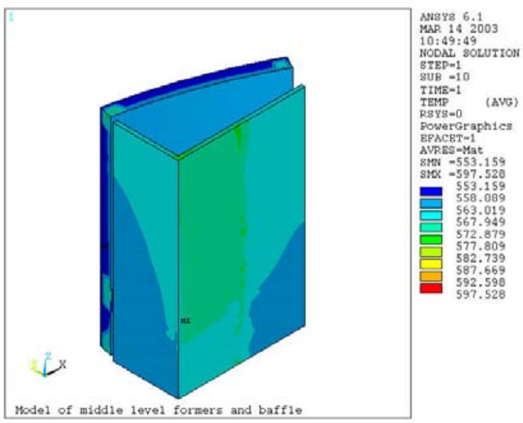
5.3.2 Temperature Contours and Profiles

Figures 5-1 through 5-7 present views of contours and temperature profiles from the thermal analysis. In these figures, 4 different statepoints were chosen for plotting: Cycle 1, BOC; Cycle 6, EOC; Cycle 8, BOC; and Cycle 11, EOC. These represent approximately the coolest and hottest statepoints from cycles 1-7 and 8-11.

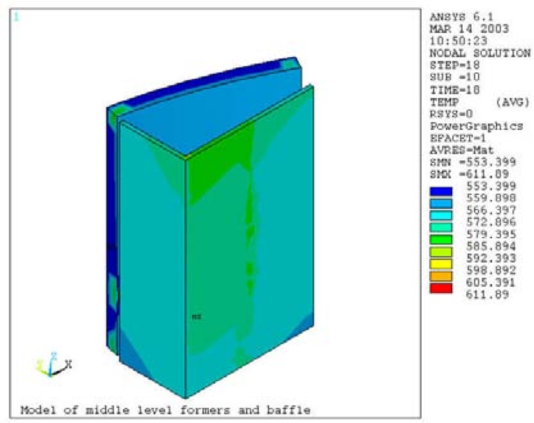
(a) Finite Element Mesh



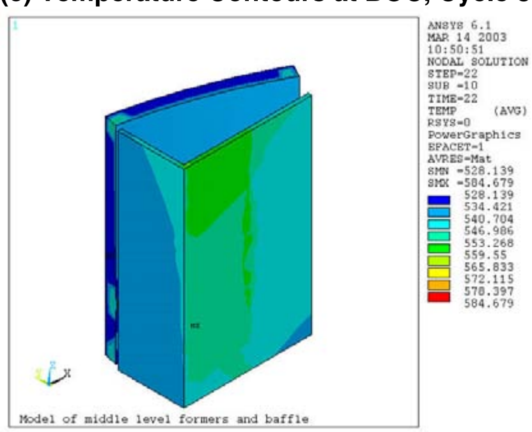
(b) Temperature Contours at BOC, Cycle 1



(c) Temperature Contours at EOC, Cycle 6



(c) Temperature Contours at BOC, Cycle 8



(d) Temperature Contours at EOC, Cycle 11

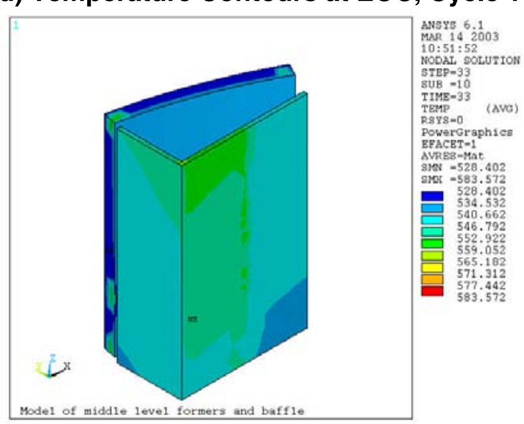


Figure 5-1
Model of Middle-Baffle and Former Samples, Finite Element Mesh and Temperature Contours Viewed from Core Side

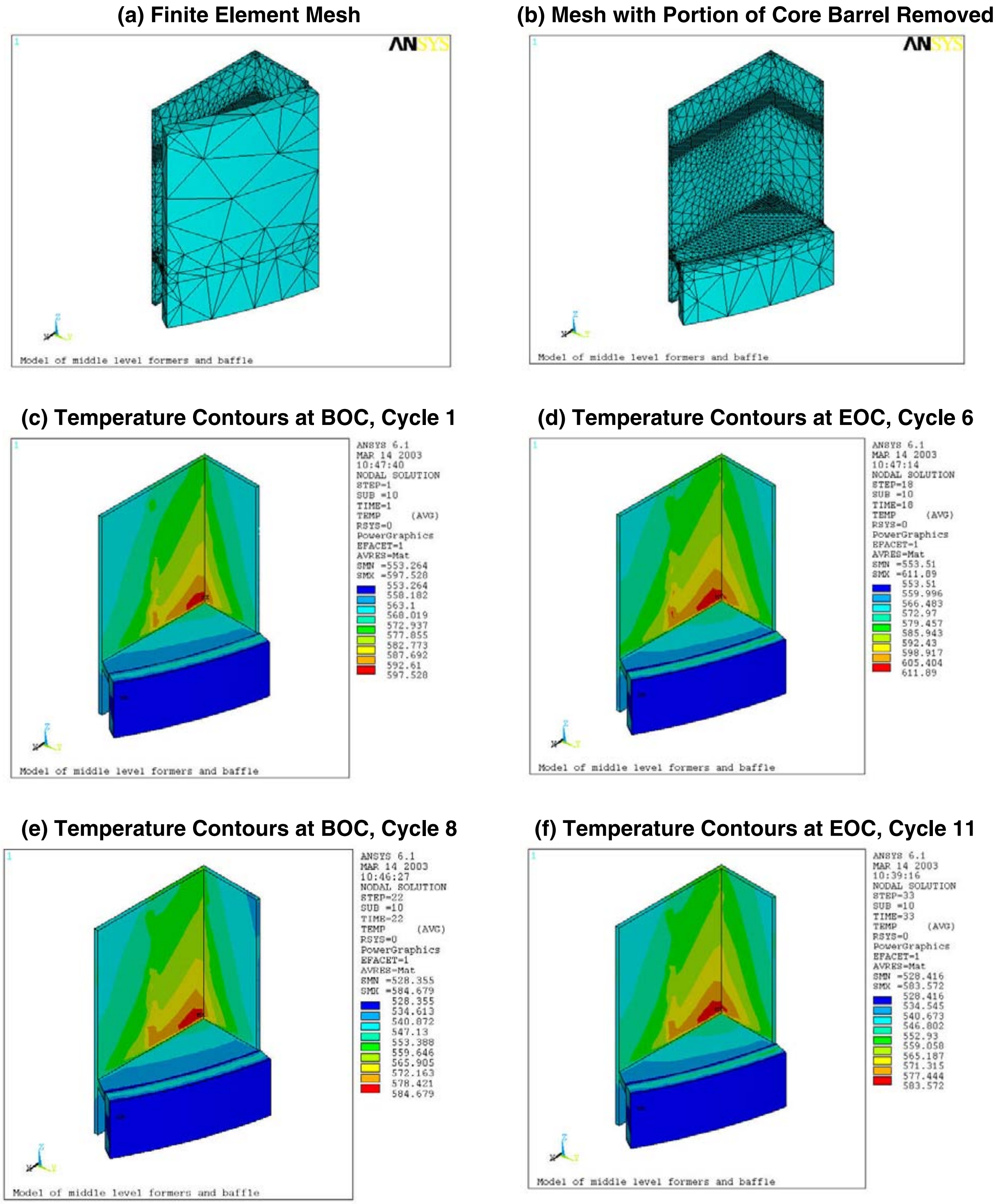
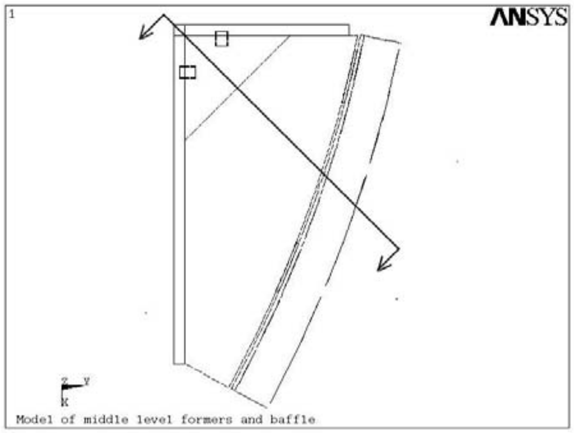
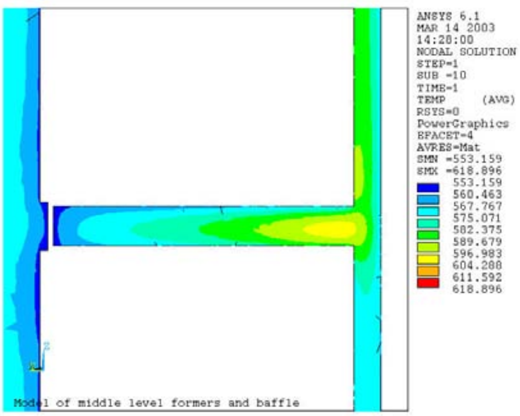


Figure 5-2
Model of Middle-Baffle and Former Samples, Finite Element Mesh and Temperature Contours Viewed from Downcomer Side

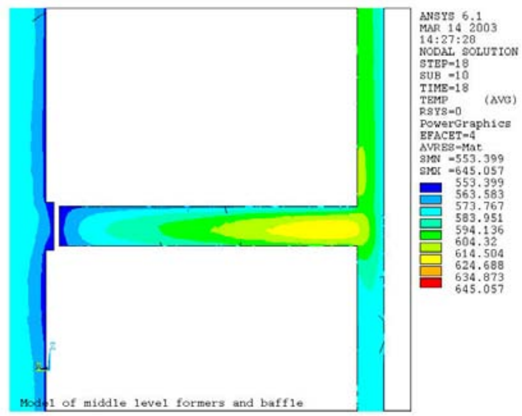
(a) Orientation of View of Cross-Section



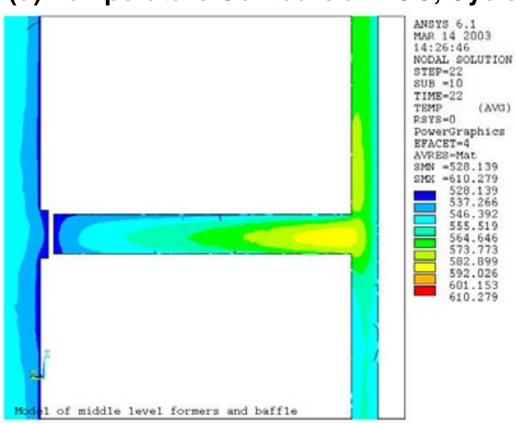
(b) Temperature Contours at BOC, Cycle 1



(c) Temperature Contours at EOC, Cycle 6



(d) Temperature Contours at BOC, Cycle 8



(e) Temperature Contours at EOC, Cycle 11

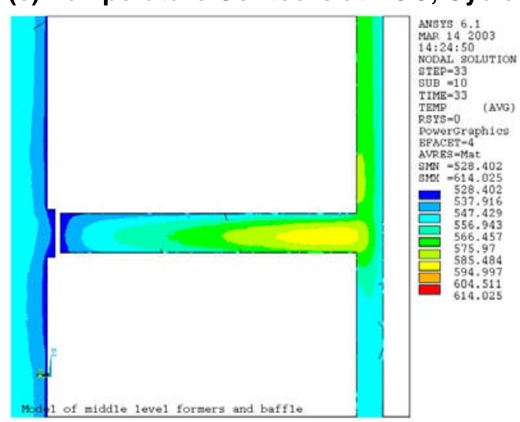


Figure 5-3
Model of Former Samples, Temperature Contours Viewed on Vertical Cross-Section Oriented at 45 Degrees

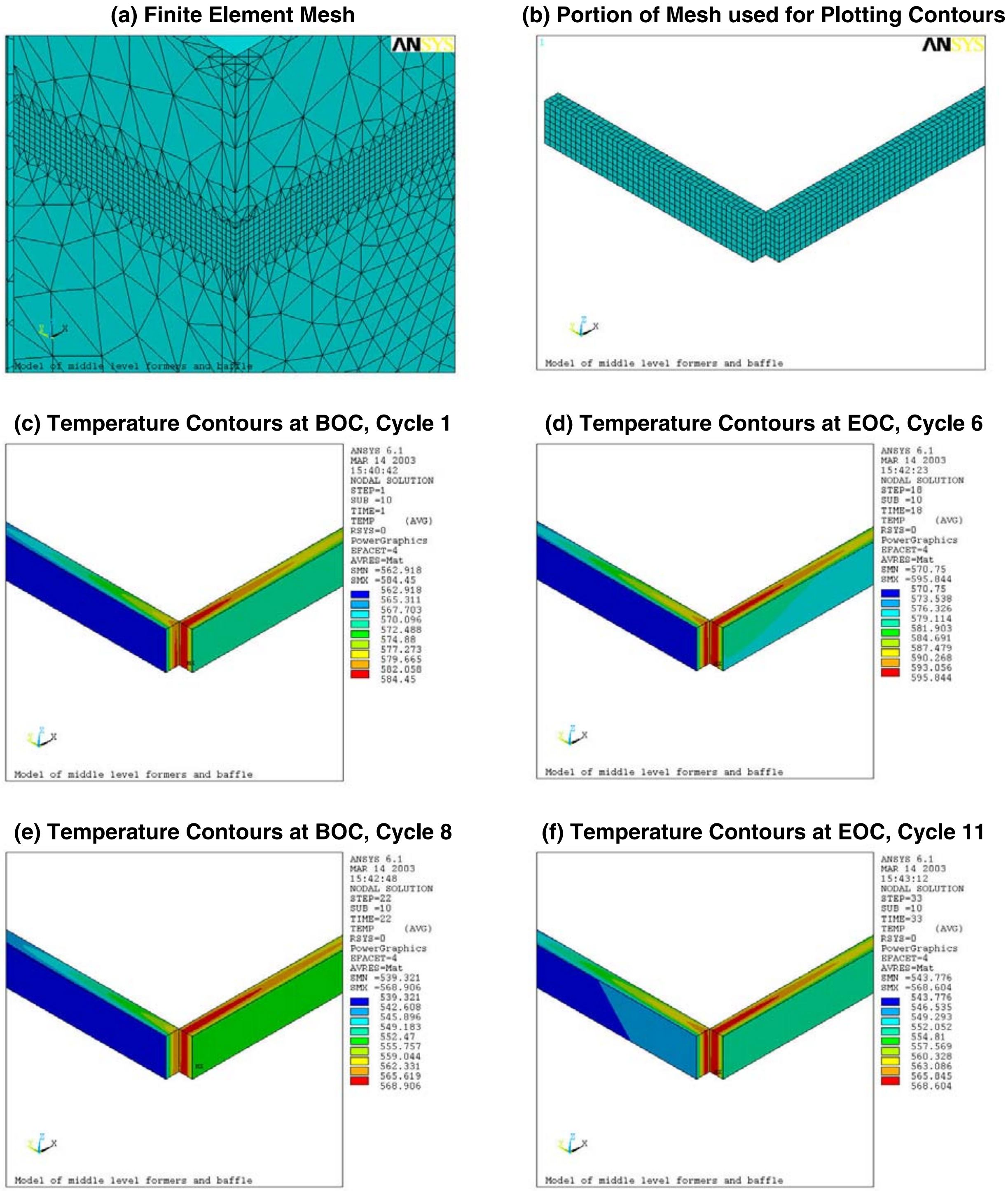
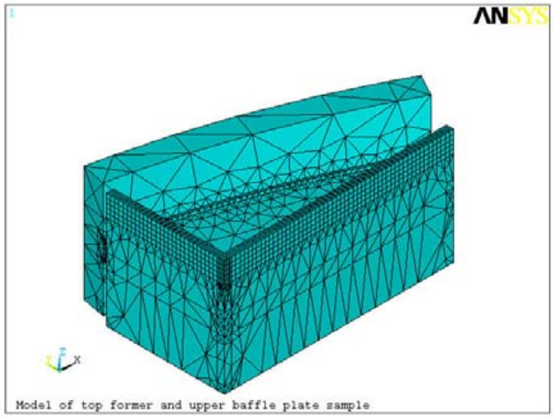
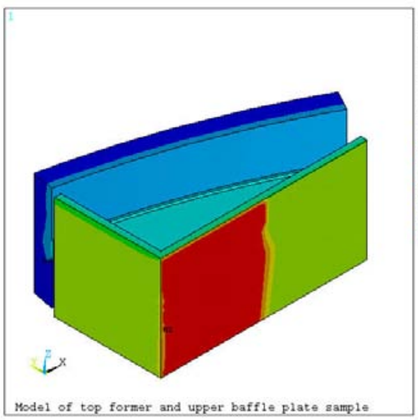


Figure 5-4
Model of Middle-Baffle Samples, Finite Element Mesh and Temperature Contours Viewed from Core Side, (Enlarged from View of Figure 5-1)

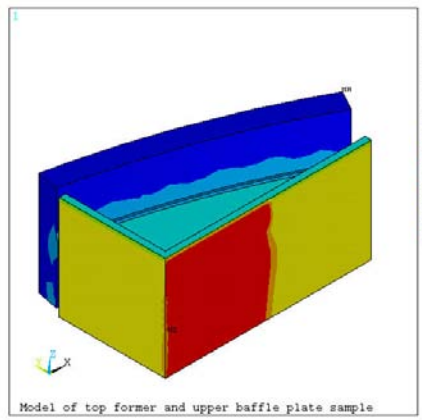
(a) Finite Element Mesh



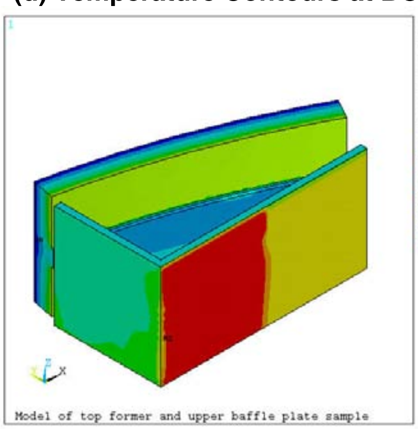
(b) Temperature Contours at BOC, Cycle 1



(c) Temperature Contours at EOC, Cycle 6



(d) Temperature Contours at BOC, Cycle 8



(e) Temperature Contours at EOC, Cycle 11

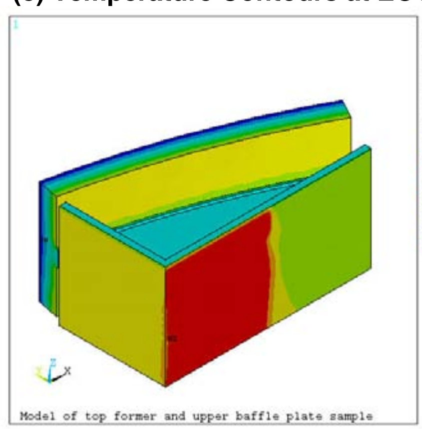


Figure 5-5
Model of Top-Baffle Samples, Finite Element Mesh and Temperature Contours Viewed from Core Side

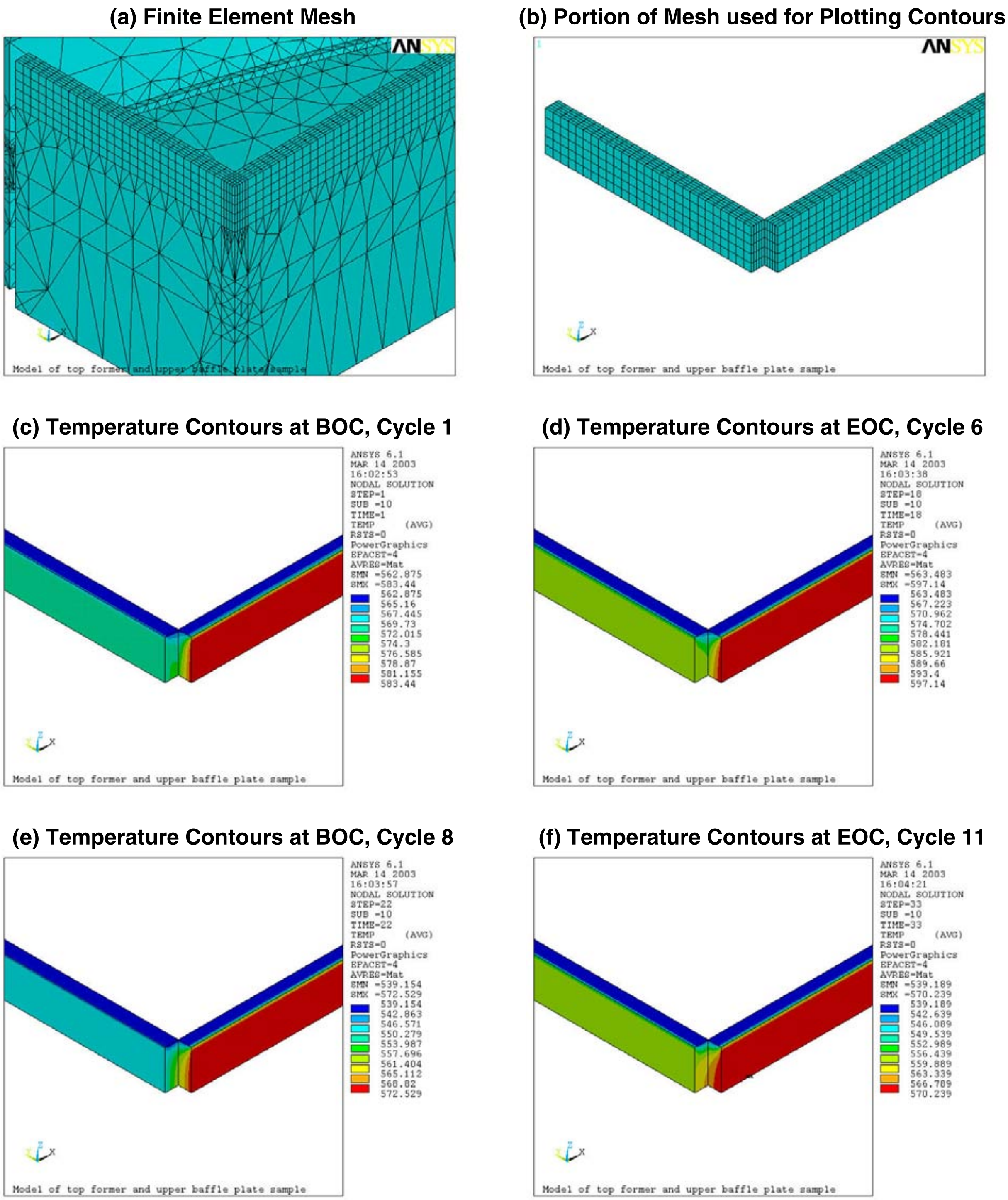


Figure 5-6
Model of Top-Baffle Samples, Finite Element Mesh and Temperature Contours Viewed from Core Side. (Enlargement of Figure 5-5)

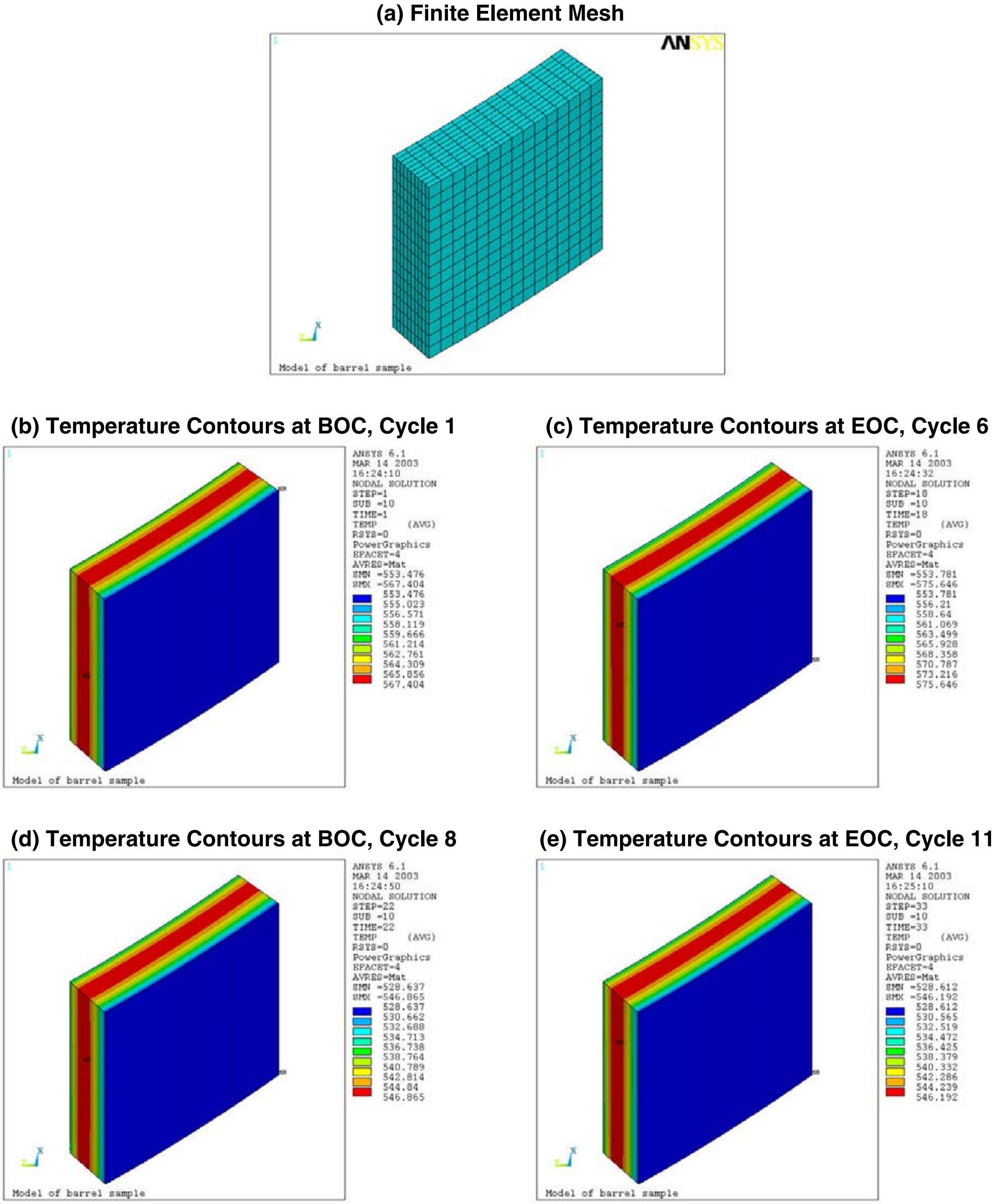


Figure 5-7
Model of Barrel Sample, Finite Element Mesh and Temperature Contours Viewed from Downcomer Side

6 SUMMARY

Quantification of the evolution of mechanical and corrosion properties of structural austenitic stainless steel PWR internals is necessary to ensure the safe operation of plants seeking life extension. This program represents the first time significant amounts of such material has been available for analysis. The results of this program will provide data free of the uncertainties implicit in fast-reactor exposure and irradiation in environments other than the PWR core. These data will be used to interpret the results of future testing of mechanical and corrosion properties of the program material. For example, the slow strain rate tensile (SSRT) results and the specimens' neutron exposure data will be combined to measure irradiation assisted stress corrosion cracking (IASCC) susceptibility as it varies with fluence. Also, the data presented in this report will be used for specimen location selection. For example, using a predictive swelling model such as Foster-Flinn, microstructural (TEM) specimen location will be selected with the appropriate combination of neutron fluence and irradiation temperature to optimize conditions for swelling.

Internals material was harvested from both the top-of–core and mid-core locations in order to obtain a broad spectrum of neutron exposure levels. Baseline heat-specific data were established via retrieval of the materials certification sheets. These sheets listed basic mechanical properties and chemical composition for the heats of steel used to manufacture the baffle plates, former, and barrel from which the program specimens were machined. The radiation exposure calculation utilized a three-dimensional model of roughly one-eighth of the core. Radiation exposure values were converted into units of stainless steel dpa at three state-points per cycle (BOC, MOC, EOC) for each of the plant's eleven cycles for a total of 33 state-points. The temperatures of the program materials were estimated by three 3-dimensional finite element models (former and mid-plane baffles, top of core baffles, and mid-plane barrel) generated with ANSYS code. In addition to reactor coolant temperature data, this model utilized the results from the radiation exposure calculation to incorporate gamma heating effects. Temperatures were calculated for all of the planned sample locations at each of the 33 state-points.

Retrieval of materials certification documentation verified that all material chemistries conformed to ASTM specifications for Type 304 stainless steel. All plate material (former, baffle, and core barrel) was confirmed to have been in the annealed condition [$\sigma_Y \approx$ 35 ksi (240 MPa), T.S. $\approx$ 80 ksi (550 MPa)] upon installation. The two program bolts may have been in either of two conditions: annealed or with 6-8% cold work. Hardness of the program materials upon Westinghouse receipt was approximately 80 HRB (150 HV) for nearly unirradiated material, and increased to a saturation value of approximately 35 HRC (350 HV) for neutron exposures of 5 dpa and above. Calculated radiation exposures for the program materials ranged from several hundredths of a dpa (top of core baffles) to about 22 dpa (core mid-plane baffles). The estimated core temperature in general was approximately 25°F (14°C) cooler for cycles 8-11 than for the first seven cycles due to a reduction of core inlet temperature at that time. Program materials were calculated to reach average temperatures in the range of 560-620°F (293-327°C) during the first seven cycles, and 535-595°F (279-313°C) during the remaining four cycles.

Program:
Nuclear Power

About EPRI

EPRI creates science and technology solutions for the global energy and energy services industry. U.S. electric utilities established the Electric Power Research Institute in 1973 as a nonprofit research consortium for the benefit of utility members, their customers, and society. Now known simply as EPRI, the company provides a wide range of innovative products and services to more than 1000 energy-related organizations in 40 countries. EPRI's multidisciplinary team of scientists and engineers draws on a worldwide network of technical and business expertise to help solve today's toughest energy and environmental problems.

EPRI. Electrify the World

WARNING: This Document contains information classified under U.S. Export control regulations as restricted from export outside the United States. You are under an obligation to ensure that you have a legal right to obtain access to this information and to ensure that you obtain an export license prior to any re-export of this information. Special restrictions apply to access by anyone that is not a United States citizen or a Permanent United States resident. For further information regarding your obligations, please see the information contained below in the section entitled "Export Control Restrictions."

Export Control Restrictions

Access to and use of EPRI Intellectual Property is granted with the specific understanding and requirement that responsibility for ensuring full compliance with all applicable U.S. and foreign export laws and regulations is being undertaken by you and your company. This includes an obligation to ensure that any individual receiving access hereunder who is not a U.S. citizen or permanent U.S. resident is permitted access under applicable U.S. and foreign export laws and regulations. In the event you are uncertain whether you or your company may lawfully obtain access to this EPRI Intellectual Property, you acknowledge that it is your obligation to consult with your company's legal counsel to determine whether this access is lawful. Although EPRI may make available on a case by case basis an informal assessment of the applicable U.S. export classification for specific EPRI Intellectual Property, you and your company acknowledge that this assessment is solely for informational purposes and not for reliance purposes. You and your company acknowledge that it is still the obligation of you and your company to make your own assessment of the applicable U.S. export classification and ensure compliance accordingly. You and your company understand and acknowledge your obligations to make a prompt report to EPRI and the appropriate authorities regarding any access to or use of EPRI Intellectual Property hereunder that may be in violation of applicable U.S. or foreign export laws or regulations.

Printed on recycled paper in the United States of America

1009799

EPRI • 3412 Hillview Avenue, Palo Alto, California 94304 • PO Box 10412, Palo Alto, California 94303 • USA
800.313.3774 • 650.855.2121 • askepri@epri.com • www.epri.com