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Light Water Reactor Sustainability Program

Post-Service Examination of PWR Baffle Bolts

Part I. Examination and Test Plan

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SUMMARY

In support of extended service and current operations of the US nuclear reactor plants, the Oak Ridge National Laboratory (ORNL), through the Department of Energy (DOE), Light Water Reactor Sustainability (LWRS) Program, is coordinating with Ginna Nuclear Power Plant, The Westinghouse Electric Company, LLC, and ATI Consulting, the selective procurement of baffle bolts that were withdrawn from service in 2011 and currently stored on site at Ginna. The goal of this program is to perform detailed microstructural and mechanical property characterization of baffle former bolts following in-service exposures. This report outlines the selection criteria of the bolts and the techniques to be used in this study. The bolts available are the original alloy 347 steel fasteners used in holding the baffle plates to the baffle former structures within the lower portion of the pressurized water reactor vessel. Of the eleven possible bolts made available for this work, none were identified to have specific damage. The bolts, however, did show varying levels of breakaway torque required in their removal. The bolts available for this study varied in peak fluence (highest dose within the head of the bolt) between 9.9 and 27.8×10^{21} n/cm^2 (E>1MeV). As no evidence for crack initiation was determined for the available bolts from preliminary visual examination, two bolts with the higher fluence values were selected for further post-irradiation examination. The two bolts showed different breakaway torque levels necessary in their removal. The information from these bolts will be integral to the LWRS program initiatives in evaluating end of life microstructure and properties. Furthermore, valuable data will be obtained that can be incorporated into model predictions of long-term irradiation behavior and compared to results obtained in high flux experimental reactor conditions. The two bolts selected for the ORNL study will be shipped to Westinghouse with bolts of interest to their collaborative efforts with the Electric Power Research Institute. Westinghouse will section the ORNL bolts into samples specified in this report and return them to ORNL. Samples will include bend bars for fracture toughness and crack propagation studies along with thin sections from which specimens for bend testing, subscale tensile and microstructural analysis can be obtained. Additional material from the high stress concentration region at the transition between the bolt head and shank will also be preserved to allow for further investigation of possible crack initiation sites.

CONTENTS

FIGURES

TABLES

ACRONYMS

EBSD	Electron Backscatter Diffraction
EPRI	Electric Power Research Institute
FIB	Focused Ion Beam
IMET	Irradiated Materials Examination and Testing
IASCC	Irradiation Assisted Stress Corrosion Cracking
LAMDA	Low Activation Materials Development and Analysis
ORNL	Oak Ridge National Laboratory
PIE	Post Irradiation Examination
PB-2	Point Beach Unit 2
PWR	Pressurized Water Reactor
SEM	Scanning Electron Microscope / Microscopy
TEM	Transmission Electron Microscope / Microscopy

Post-Service Examination of PWR Baffle Bolts

Part I. Examination and Test Plan

1. Introduction

Long-term exposure degradation mechanisms are a concern of nuclear power plants as they exceed 40-years of service life. Degradation for reactor internals may include irradiation assisted stress corrosion cracking (IASCC), which has been a suggested mode of failure for baffle former bolts (from herein called baffle bolts) in pressurized water reactors (PWRs). The first identification of failed bolts was an outcome of the investigation of flow-induced vibration of fuel rods in elements on the core periphery observed in French 900 MW plants in the 1980's. This abnormal condition was created by water flow through gaps between baffle plates as a result of the failure of baffle bolts [1]. These bolts, made from 10 to 30% cold worked 316-grade stainless steel, were determined to have failed through intergranular cracking following destructive examinations of removed bolts. The location of these bolts being in a high irradiation flux position near to the bottom of the fuel core suggests that crack development was related to IASCC as 316 steel is not prone to intragranular stress corrosion cracking in the hydrogenated primary water of the PWR environment.

Baffle bolts secure the baffle plates to the former plates in the lower internal assembly of the reactor vessel. The baffle plates are the vertical internal structure around periphery of the fuel core, with the former plates being the horizontal structural components distributed along the height of the fuel core separating the baffle plates from the core barrel (see Figure 1). For two-loop PWR designs, such as in the R.E. Ginna PWR (from here on, Ginna), there are thirty-six baffle plates and seven former plates [2]. In addition to the baffle bolts, there are bolts between the core barrel and former plates (barrel bolts), and bolts that pin the baffle plate to baffle plate on the ends (edge bolts). For a two-loop Westinghouse design, there can be up to 728 baffle bolts with up to 104 bolts at each height elevation in the lower internal assembly. The bolts can experience varied neutron flux based on the location of the bolt within the assembly as well as a fluence distribution along the length of the bolt.

In 1997, the Westinghouse Owners Group created a task force to evaluate the integrity of baffle bolts in U.S. and international PWR's and determined that IASCC was an issue with significant regulatory risk [2]. Baffle bolt cracking can still remain an issue in plants that have not been modified to an up-flow design in which the direction of the water flow between the core barrel and baffle plates has been changed from the traditional downward direction. This results in lower temperatures of the baffle bolts even after gamma heating is accounted for [1]. Plants with the modified flow direction have shown little baffle bolt cracking as compared to the unconverted down flow designs. However, other factors such as improved bolt and replacement bolt designs minimizing stress concentrations between the bolt head and shank could also mitigate crack development.

The original baffle bolts within the Ginna PWR are AISI 347 grad stainless steel machined to ANSI B18.3 standard, consisting of a socket head cap, shank and threads that are rolled and chrome plate (for purposes of preventing galling on initial installation of the bolt). The baffle plates are counter bored, such that the head of installed bolt is flush with the surface. A washer is inserted into the gap between the head and the wall of the counter bore, which is tack welded at both the plate and bolt head. These washers are detached on the removal of the baffle bolts. The bolts to be examined in this study are part of these original bolts. Replacement bolts are now fabricated from SA-193 Class B8M, strain-hardened 316 grade stainless steel with similar dimensions as the original, but without the chrome plating of the threads and include an integrated 304 grade steel locking cup that is welded to the head of the bolt [2,4].

In the late 1990's Framatome developed a program to inspect and quantify the existence of bolt degradation by non-destructive examination (NDE) followed by bolt replacement at the Point Beach Unit 2 (PB-2) and Ginna PWR's [2]. Of the 728 examined bolts from Ginna, 80 were found to be either suspect or defective. Up to 14% of the bolts identified in NDE techniques were found to have fractured bolt heads on removal. Of the 176 type 347-alloy baffle bolts removed from PB-2, inspections confirmed 10 bolts as having developed an in-service crack, with one further bolt as being likely [5]. All cracks were found to initiate around the fillet transition between the underside of the bolt head and the bolt shank, similar to that observed in the earlier French PWR bolts [6]. However, crack directions of the PB-2 bolts were through the cross-section, while those of the French PWR's were more radial and eventually propagating to the center at a 30-45º angle relative to the bolt axis. Room temperature mechanical testing performed on the PB-2 bolts revealed radiation-induced hardening, though ductility remained high (~30% total elongation, ~3% uniform) with ductile transgranular fracture. Possible modes of failure were postulated as IASCC, cyclic stresses causing mechanical fatigue damage due to events such as baffle peening, bolt fabrication irregularities, stress risers within the bolt design, installation irregularities in preload or a combination of the afore mentioned. While IASCC threshold of $5x10^{21}$ n/cm^2 (E>1MeV) was mentioned in the Framatome report on the mechanical properties [5] and cracked bolts were in regions above this threshold, they were not found in conditions representing the highest fluence regions.

The microstructure of a cold worked 316 grade stainless steel baffle bolt harvested from the Belgium Tihange PWR was examined by Edwards and coworkers [7] as a function of temperature and dose according to the position along the length of the bolt from which samples were produced from. Temperature and dose varied from 593 K and 19.5 dpa at the bolt head (1 mm depth), to 616 K and 12.2 dpa at the 25 mm depth point in the shank and to 606 K and 7.5 dpa at 55 mm depth just ahead of the threads of the bolt. Fluences were measured in the bolt independently through measured activity ratio of ^{60}Co (^{59}Co*(n,γ)*, thermal neutron produced) to ^{54}Mn (^{54}Fe*(n,p)* fast neutron produced) by gamma spectroscopy and corrected for decay based on the power history of the reactor. Temperatures were based on previous reactor data with gamma heating variation to account for the higher temperatures away from the bolt head. While all locations examined showed cavities and faulted dislocation loops, the higher temperature locations in the shank showed higher swelling values.

The phenomenon of cavities producing measureable values of swelling has been observed in other austenitic steel components from PWRs despite irradiation temperature and dose in ranges that normally do not contribute to swelling under high dose rate experimental observations [8-10]. The appearance of swelling in reactor-relevant conditions raises concerns over further life extension of components and the prospect for more severe void-induced embrittlement modes of failure.

The effects of higher fluences on core components are an important issue for the LWRS program in predicting structural performance for extended lifetimes. Particularly useful is new information from reactor-harvested materials, given the fact that most studies on IASCC in PWR environments have been conducted on fast reactor material or boiling water reactor irradiated materials [11]. Microstructural changes related to radiation-induced defect development responsible for hardening, solute segregation changes producing corrosion susceptibility, and swelling from void and bubble development are all concerns that can influence sensitivity to IASCC. It is the intention of this work to not only evaluate the microstructural changes occurring in the Ginna bolts with in-service exposure, as compared to non-irradiated 347 steel, but to link microstructural development to the mechanical property changes in the material. This will be achieved through structure property correlations by evaluating possible modes of failure that the material may have sensitivity to. This will be examined through in situ deformation studies as well as microstructural examination of the material specifically in the deformed region. Other effects such as loading changes of the bolt stress state during lifetime as its influenced by swelling and irradiation creep history is currently being addressed by Electric Power Research Institute (EPRI) (MRP-135, MRP-211) and will be addressed for specific reactors. Other aspects of IASCC on reactor internal components such as flux rate and spectrum dependence, swelling, the role of hydrogen, crack initiation and propagation is also being addressed in other projects within the LWRS program.

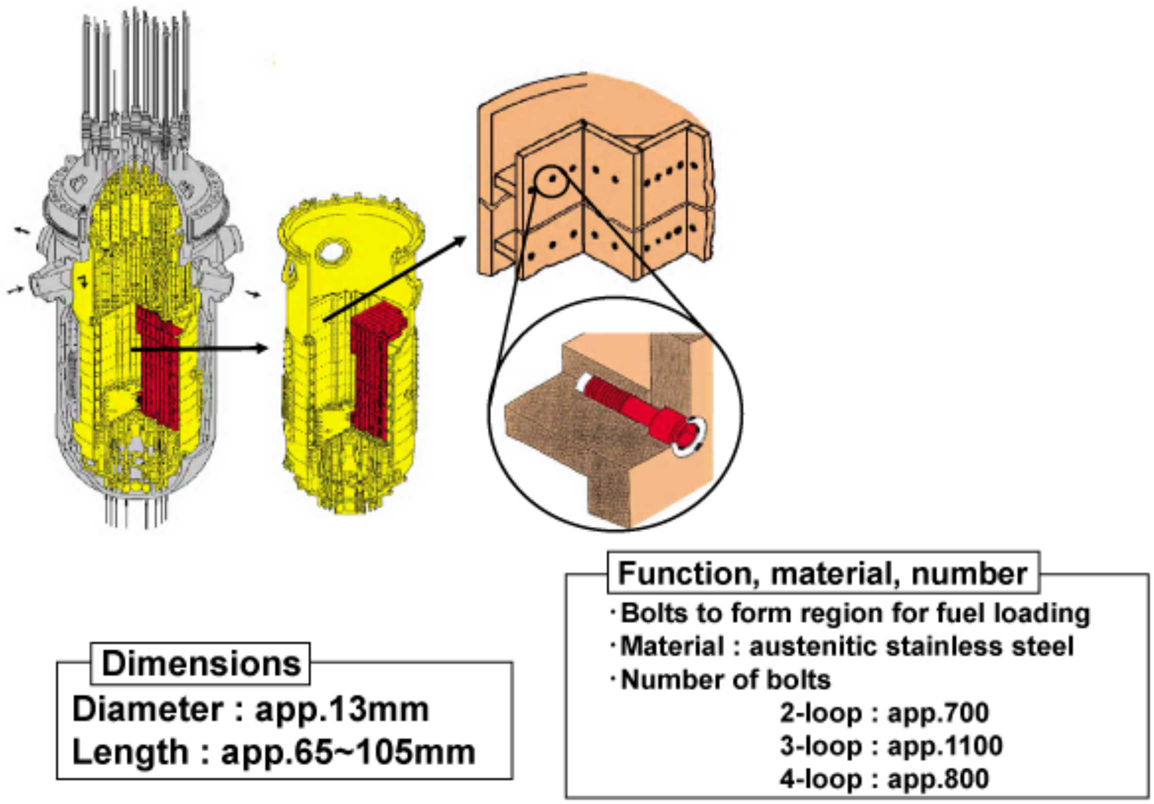

Figure 1. Schematic diagram of the reactor vessel lower internal assembly showing the wall facing the fuel assemblies consisting of the baffle and former plates secured by bolts, from [3].

2. Selection of Baffle Bolts

The baffle bolts were harvested during the spring 2011 outage and are in storage at the spent fuel pool at the Ginna site. A listing of the available bolts is provided in Table 1. The Ginna ID number for the bolts follows a 4 digit code with the first number being the quadrant location in the reactor, then the associated baffle plate number, the column location of the bolt associated with the particular baffle plate number, and finally the former location where the bolt originates [2,12]. A further description and illustration of the bolt locations is shown in Figure 2. The distribution in fluence is listed for locations within the head, mid-shank and mid-thread of the bolts. This distribution of dose is graphically presented in Figure 3. Fluences within the bolt head are generally 20 to 25% higher than the mid shank and double that of the mid-thread values, with the exception of bolts 17 and 26. These two have a flatter dose distribution across the bolt length. The estimated temperature distribution in the bolts has been requested.

Further information [12] available on the bolt extraction on the original alloy 347 bolts and on the operational experience in bolt replacement is detailed in Table 2. The bolts available in this study were not part of the examination work performed in the 1990's by Framatome and reported to EPRI in reference 2.

Table 1. Available 347 grade stainless steel bolts in the spent fuel pool at Ginna, available for retrieval and examination.

Bolt #	Ginna ID	Fluence (x10 ^{22} n/cm^2) E>1 MeV Head	Mid-Shank	Mid-Thread
22	4412	2.78	2.27	1.46
21	3933	2.21	1.78	1.09
24	4424	2.20	1.77	1.09
23	4416	1.91	1.56	1.00
17	3814	1.84	1.56	1.11
20	3925	1.62	1.30	7.90
25	4435	1.62	1.30	0.79
26	3326	1.34	1.14	0.84
5	1262	1.00	0.80	0.49
15	3212	1.00	0.80	0.49
16	3215	0.99	0.80	0.49

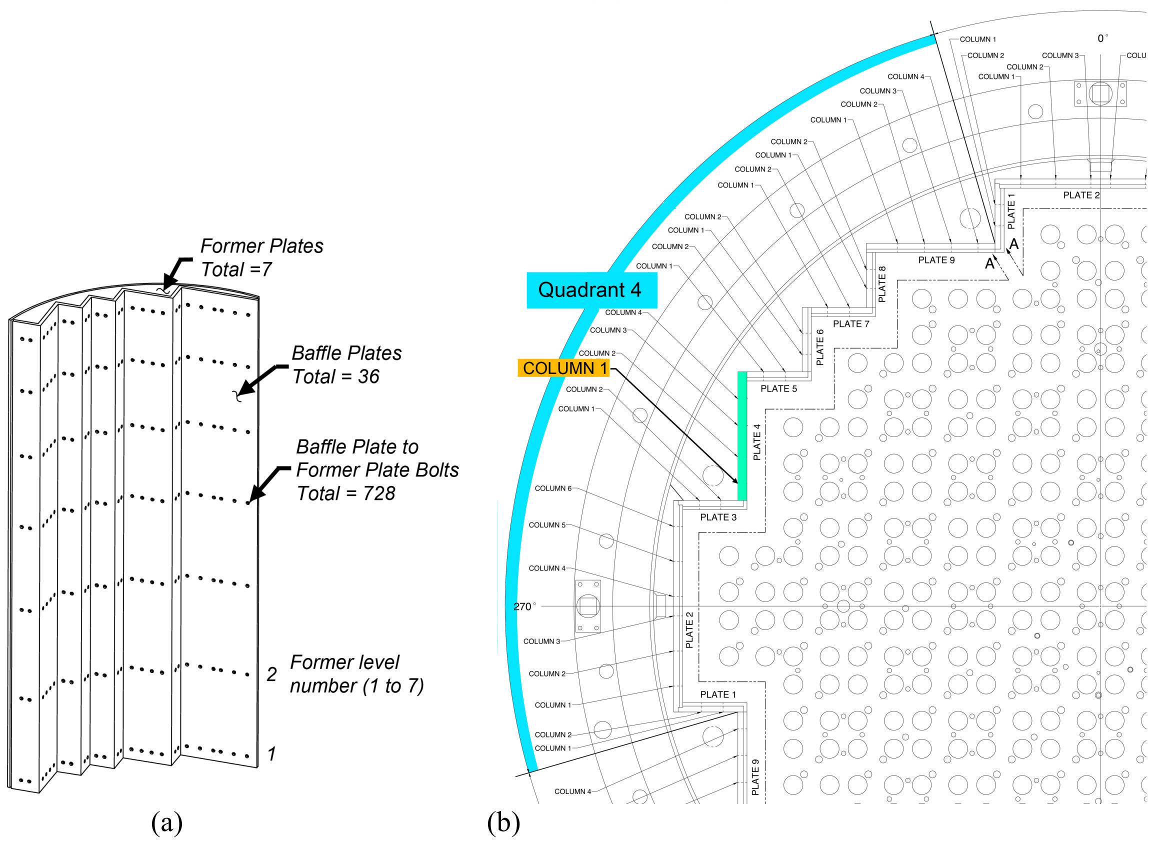

Figure 2. Diagrams of the baffle plate and bolt locations within the PWR. (a) Vertical section illustrating the various components and quantities. (b) Top view of one of four quadrants in a PWR. Diagrams help illustrate the location of position of the 4412 (Ginna ID) bolt of interest to this work. Illustrations adapted from references 2 and 12.

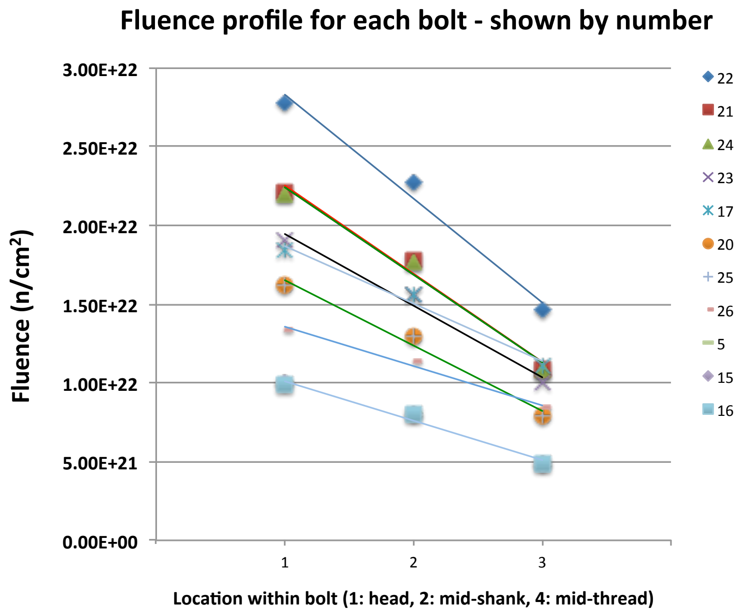

Figure 3. Graphical representation of the fluence distribution within the bolts available at Ginna.

Table 2. Available information on the bolt unloading and replacement performed during the spring 2011 outage at the Ginna nuclear power plant.

Bolt #	Ginna ID	Removal: Breakaway torque (ft-lbs)	Install: Final Torque (ft-lbs)	Comments
22	4412	20	55	No issues on removal or installation.
21	3933	150	55	No issues on removal or installation.
24	4424	20	55	No issues on removal or installation.
23	4416	5	55	No issues on removal or installation.
17	3814	0	55	No issues on removal or installation.
20	3925	2	55	No issues on removal or installation.
25	4435	25	55	No issues on removal or installation.
26	3326	20	55	Bolt removed without issue. Replacement bolt locking cup not flush against baffle plate. Broken head on backing off replacement bolt. After several attempts at performing a square head burn, a contingency nail burn was performed.
5	1262	132	55	No issue on removal. Two attempts at seating replacement bolt.
15	3212	110	75	No issues on removal. Installation required 75 ft-lbs torque to seat bolt.
16	3215	95	75	No issues on removal. Installation required 75 ft-lbs torque to seat bolt.

As no identification of cracking or potential damage was determined for the bolts listed, the selection of two bolts for the ORNL study was based on a couple of factors: the selection of the highest fluence condition and one that will provide a matching fluence and bolts which showed any difference in known behavior (i.e. extraction torque required). Therefore, bolt numbers 22 and 23 were selected. The highest fluence bolt (number 22) will provide the maximum level of degradation possible, based on radiation-induced or radiation-enhanced damage, through which changes in microstructure and mechanical properties can be measured and compared to experimental test reactor data. This will allow a better assessment of the dose rate effects on the microstructural development through comparison of the material representative of actual conditions as opposed to high flux experimental data generated using test reactors. Bolt 23 offers a relatively close matching of the fluences of the mid-shank to the mid-thread of bolt 22, and the fluence in the head of bolt 23 to that of the mid-shank of bolt 22. This will allow for a comparison of the bolt structure/properties as they may be influenced by differences in irradiation temperature or stress distribution. Furthermore, the torque required in removing bolt 23 from the baffle plate was among the lowest of the bolts available. Characterization of the microstructure may provide insight into variances for torque removal values between the bolts listed in Table 2.

The bolts are to be retrieved from the spent fuel storage pool at the Ginna nuclear plant during the summer of 2015 (timeline being negotiated) by a team from Westinghouse. The bolts specific to the interest of ORNL will be shipped to Westinghouse, combined with those of interest to Westinghouse. The Westinghouse work is in conjunction with EPRI and separate from the LWRS Program. The ORNL bolts will be inspected at the Westinghouse hot cells for any visible cracks or abnormalities. Afterwards, the two ORNL bolts will be sectioned according to specifications shown in the next section of this report and shipped back to ORNL for further examination and testing.

3. Bolt Sectioning Plan

Bolts 22 and 23 will be sectioned into several test pieces designed for mechanical and microstructural evaluation. A schematic of the cutting plan is shown in Figure 4, with a sample numbering scheme in Figure 5. From the mid-shank and thread positions, bend bars will be fabricated. As a fluence distribution will be present within the bend bars, the high-dose ends of each bar will be marked with identifier to track extracted sample orientation relative to the original bolt orientation. Adjacent to these bars, additional slices for microstructural characterization and possible bend or subscale tensile bar fabrication will be extracted. The thinner, ~0.5 mm thick slices, will be utilized for microstructural examinations. Ideally, pending dose and handling limits, these thin sections are to be used in preparing 3 mm diameter transmission electron microscopy (TEM) discs thinned to electron transparency by electropolishing rather than focused ion beam milling (FIB). This will provide information on the defect evolution within the microstructure without damage introduced into the materials by focused ion beam milling. If further samples are needed for TEM investigations, FIB specimens can be prepared from the outer rim portion of these 3 mm discs, therefore optimizing utilization of available samples.

In addition to the materials sections removed at the mid-shank and thread locations, material will also be saved from the head of the bolt. To preserve the region consisting of the high stress concentration area that contains the fillet transition between the bolt head and shank. From the bolt head, sliced sections can be removed from which further microhardness, nano-indentation and electron microscopy could potentially be performed, pending sample dose levels. Furthermore, the material containing the high stress region will be preserved for further investigation and documentation in the ORNL hot cells should these bolts reveal visible damage during the Westinghouse inspections.

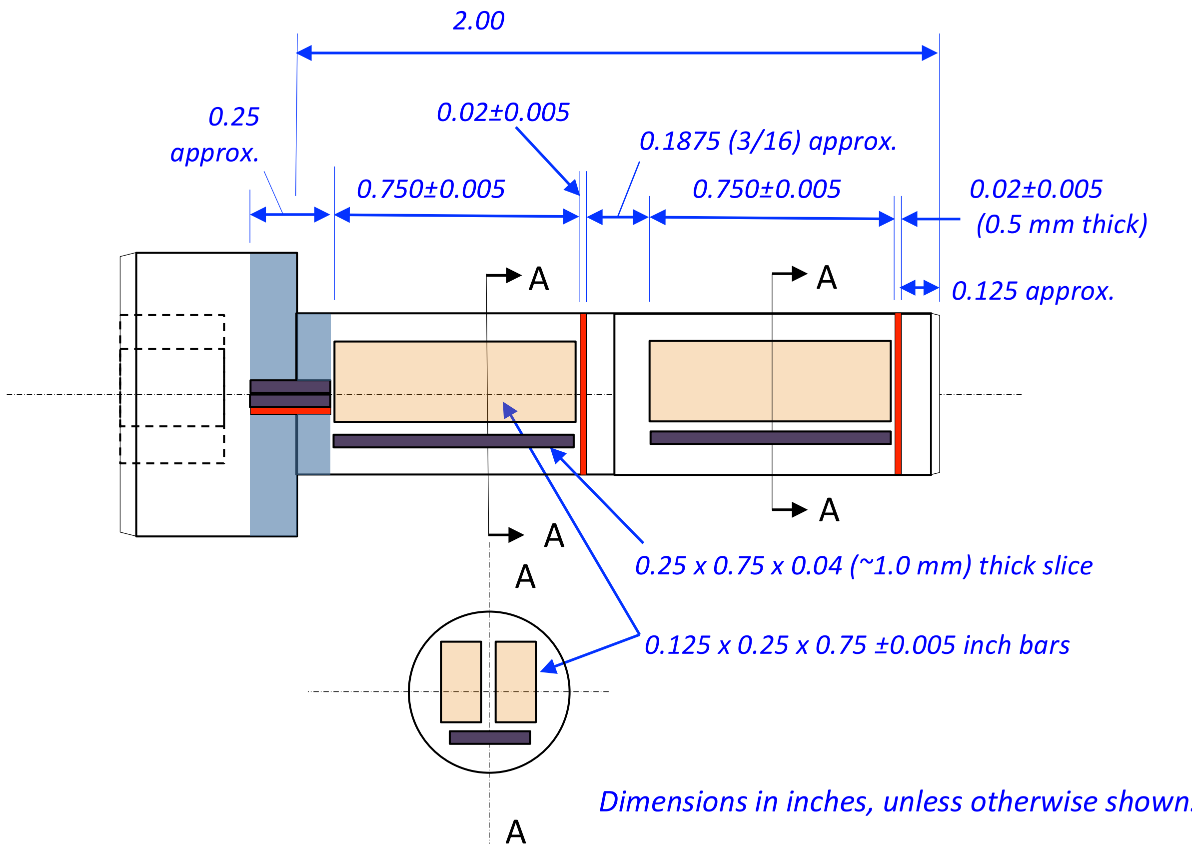

Figure 4. Simplified cutting diagrams for the baffle bolts showing the color-coded sample types to be sectioned from the bolts at the head, mid-shank and thread regions (shaded diagonal line region) for mechanical and microstructural examinations.

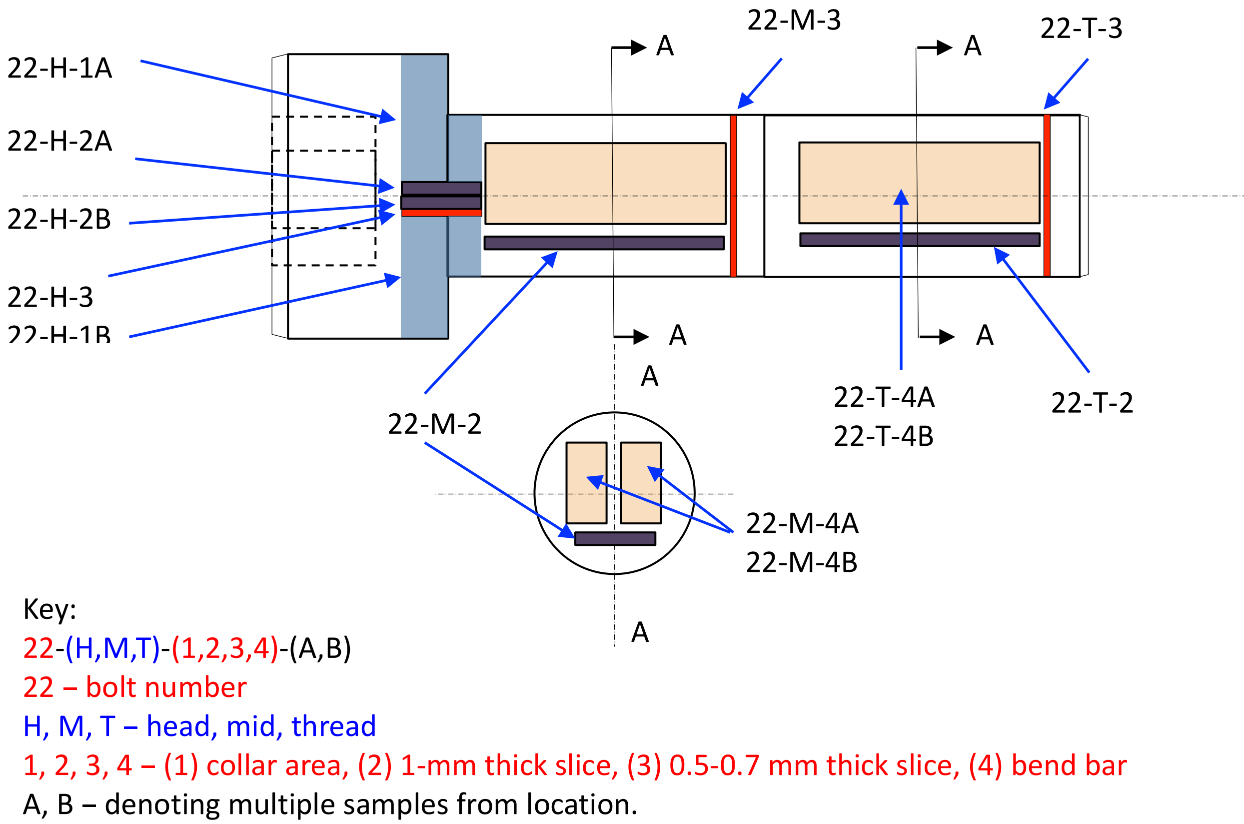

Figure 5. Sample identification diagram for the bend bar and section slices taken from the baffle bolts under investigation.

4. Materials Test Plan

Upon arrival at the ONRL hot cells, the specimens will undergo inspection and further preparation. This includes the evaluation of the 22-H-1A, 22-H-1B, 23-H-1A and 23-H-1B series of specimens that contain the high stress region between the bolt head and bolt shank that has been the location of observed cracks in previous baffle bolt examinations. In addition to these samples, slices that are 1 mm and 0.5 mm thick slices will also be available for further in-depth examination.

The 22-X-2 and 23-X-2 series (with X being either H=head, M=mid-shank or T=thread) of samples, which compose of 1 mm thick slices will be transferred to the Low Activation Materials Development and Analysis (LAMDA) lab, dose permitting. Additional cutting may be considered prior to the transfer if dose levels approach or exceed the LAMDA limit (100 mR/1h at 1ft).

For each of the 1 mm thick samples, density measurements will be performed to estimate the swelling level (if any) as a function of sample position along the bolt length. The results will be compared with TEM data on the presence, density, and average size of voids. Also, magnetic phase amount will be measured as a function of dose and irradiation temperature to provide information on phase stability of the material under irradiation. After these tests, samples will be further prepared into a combination of bend test bars or sub-sized tensile bars for evaluation of mechanical properties.

Mechanical tests (bend or tensile) will be conducted at room temperature to evaluate material strength and deformation hardening behavior. Specimens deformed at different strain levels, will be prepared for microstructure evaluation via scanning electron microscopy (SEM) using an electron backscatter diffraction (EBSD) detector. The goal is to analyze the structure evolution, deformation localization degree, twinning, and phase transformations (if any) during the straining. The most interesting areas will be prepared via FIB milling for detailed TEM analysis.

In-situ straining experiments will be considered depending on the availability of special equipment (deformation stage for in-SEM testing) considered for this specific project, aiming the analysis of strain localization and interaction of dislocation channels with grain boundaries. As expected, the in-situ experiments will allow for direct measurements of acting stress and strain at channel – grain boundary intersection points.

In addition to the above mentioned samples, the ~0.5 mm thick materials sections taken from the bolt (22-X-3 and 23-X-3, series of samples) will also be transferred to LAMDA, pending dose limits. These samples will undergo microhardness testing (following light mechanical polishing) then preparation into TEM samples through electropolishing. Where dose limits of the samples exceed LAMDA limits, further section of the materials will be performed at the hot cells prior to transfer to LAMDA. This may require that all TEM samples be produced through the FIB milling technique (which can introduce defects from the ion milling). Experimental work will be performed on materials sliced from the bolt head. Nano-indentation studies as well as FIB preparation of TEM specimens from these high stress locations will provide useful information on the response of the material to reactor conditions under stress. These two test and sample preparation techniques allow for more specific site selection of the regions of interest being studied. Furthermore, these sections can be prepared metallographically for SEM-EBSD techniques to evaluated strain levels within the material grains within the stressed location. If dose levels exceed LAMDA limits, then smaller sized samples can be prepared at the hot cell for transfer to LAMDA. Techniques such as transmission EBSD could also be performed, utilizing the micro-size samples produced by the FIB.

Samples prepared for TEM will be examined for the changes in microstructural features and solute segregation as a function of position along the length of the bolt. Features to be examined include the radiation-induced defect distribution and type (dislocation loop and cavity), radiation induced precipitate development and changes to the solute segregation profiles at grain boundaries. Furthermore, atom probe tomography may be utilized to further evaluate solute cluster formation not easily visible through TEM examination.

Post-radiation annealing will be considered depending on overall progress and available funds. The recovery of material property by annealing is an important topic with very limited available literature for 347 steel. Structure evolution (change in loop density and size, evolution of void population and radiation induced precipitations) may be of interest.

The 22-X-4A/B and 23-X-4A/B (with X being either H=head, M=mid-shank or T=thread) series of bend bar preforms will be further machined at the ORNL hot cells to include a mid-span notch. It is anticipated that one specimen from each bolt will be tested according to ASTM Standard Test Method for Measurement of Fracture Toughness, E1820, and other will be tested according to ASTM Standard Test Method for Measurement of Fatigue Crack Growth Rate, E647. These testing will allow the evaluation of fracture toughness and fatigue crack growth rates for this material at various fluences. These results will be compared to basic properties in the unirradiated condition.

Depending on the fracture test results, the bend bars will be sectioned into subsamples and transferred to LAMDA for detailed analysis of fracture mechanisms in highly irradiated steel.

5. Summary

A total of eleven baffle former bolts removed from service in the Ginna PWR during the 2011 spring outage, are available for examination. Two of the alloy 347 bolts were selected for inspection and analysis through the LWRS program at ORNL. The two bolts selected were higher in fluence of the materials available, with one exhibiting the highest fluence of the group. None of the bolts available were identified as having cracks or other structural anomalies during removal from the core. However, the two bolts did show a difference in the level of torque required during their removal from the baffle plates. Bolt number 22 has the higher overall fluence, with bolt 23 being less. The bolts show a fluence distribution along their length with the head of the bolt having the highest exposure. The head of the bolt 23 is similar in fluence to that of mid-shank of bolt 22. Similarly, the mid-shank of bolt 23 has a fluence level similar to that of the mid-thread region of 22. In addition to the fluence distribution along the length of the bolt a temperature profile also exists, with temperatures in the mid-shank being higher due to gamma heating than the ends of the bolt. While temperature profiles remain to be determined, the materials response to slight differences in irradiation temperature and possible stress levels for similar fluence is a very useful comparison.

Future work includes the shipping of the bolts to Westinghouse for inspection and sectioning into specified samples that will be evaluated for mechanical and microstructural properties later at ORNL. Mechanical testing will include fracture toughness testing, fatigue crack growth rate, bend testing and if feasible, miniature tensile tests. Microstructural examination will include defect analysis and radiation induced solute segregation and/or precipitate evaluation. High stress regions associated with the transition between the head and shank of the bolt will be evaluated through SEM-EBSD. In situ techniques of examining the deformation of the material under evaluation by the SEM and EBSD analysis are being considered.

The results obtained from the study of materials exposed to actual commercial PWR environment are invaluable. The data obtained will provide further information to model predictions of material behavior and highlight differences observed between actual PWR components to materials irradiated through high flux experimental reactors.

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