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2007-049 BWR Vessel & Internals Project (BWRVIP)

February 21, 2007

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U. S. Nuclear Regulatory Commission
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Rockville, MD 20852

Attention: John Honcharik

Subject: Project No. 704 – BWRVIP-96NP-A: BWR Vessel and Internals Project, Sampling and Analysis Guidelines for Determining the Helium Content of Reactor Internals

Reference: BWRVIP Letter 2007-037 from William A. Eaton (BWRVIP Chairman) to Document Control Desk (NRC), "Project No. 704 – BWRVIP-96-A: BWR Vessel and Internals Project, Sampling and Analysis Guidelines for Determining the Helium Content of Reactor Internals," dated February 8, 2007.

Enclosed are two (2) copies of the non-proprietary report "BWRVIP-96NP-A: BWR Vessel and Internals Project, Sampling and Analysis Guidelines for Determining the Helium Content of Reactor Internals," EPRI Technical Report 1014613NP, December 2006. This report replaces the non-proprietary report previously transmitted to the NRC by the BWRVIP letter referenced above. The enclosed report deletes proprietary wording that appeared in the header of the non-proprietary report previously transmitted to the NRC.

If you have any questions on this subject please contact Randy Stark at EPRI, BWRVIP Program Manager, by telephone at 650.855.2122.

Sincerely,

William A Eaton

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Entergy Operations, Inc.
Chairman, BWR Vessel and Internals Project

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BWRVIP-96NP-A: BWR Vessel and Internals Project

Sampling and Analysis Guidelines for Determining the Helium Content of Reactor Internals

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Technical Report

BWRVIP-96NP-A: BWR Vessel and Internals Project

Sampling and Analysis Guidelines for Determining the Helium Content of Reactor Internals

1014613NP

Final Report, December 2006

EPRI Project Manager
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This report describes research sponsored by the Electric Power Research Institute (EPRI) and its BWRVIP participating members.

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

BWRVIP-96NP-A: BWR Vessel and Internals Project, Sampling and Analysis Guidelines for Determining the Helium Content of Reactor Internals. EPRI, Palo Alto, CA: 2006. 1014613NP.

This report is based on the following previously published report:

BWRVIP-96: BWR Vessel and Internals Project, Sampling and Analysis Guidelines for Determining the Helium Content of Reactor Internals, EPRI, Palo Alto, CA: 2001. 1003019, authored by Framatome ANP, principal investigators M. Michaels and T. Haertel, and Pacific Northwest National Laboratory, principal investigators B. Oliver and L. Greenwood.

REPORT SUMMARY

The Boiling Water Reactor Vessel and Internals Project (B WR VIP), formed in June 1994, is an association of utilities focused primarily on BWR vessel and internals issues. This BWR VIP report provides guidelines for removing a small sample from a reactor to measure helium content (for weldability determinations) as well as other parameters. A previous version of this report was published as BWRVIP-96 (1003019), This report (BWRVIP-96-A) incorporates changes proposed by the BWR VIP in response to U.S. Nuclear Regulatory Commission (NRC) Requests for Additional Information, recommendations in the NRC Safety Evaluation (SE), and other necessary revisions identified since the previous publication of the report. All changes are marked with margin bars. In accordance with an NRC request, the SE is included here as an appendix and the report number includes an "A," indicating the version of the report accepted by the NRC staff.

Background

One of the factors affecting the weldability of stainless steel is the presence of helium in the material. Helium is produced by the transmutation of boron and nickel when they interact with thermal neutrons. If the helium content is low enough, welding can be performed by conventional techniques. However, if helium content is high, welding requires the use of special techniques, or in some cases, cannot be performed. Utilities require a method for determining helium content of in situ materials. One method involves removing a small sample of the subject material from the reactor and performing subsequent spectrographic analysis.

Objective

To develop guidelines describing acceptable methods for sample removal and analysis of helium content.

Approach

As part of a separate effort, a joint BWRVIP/NRC project was conducted to remove samples from three BWRs then analyze the samples for helium content as well as for other parameters. Lessons learned from this effort allowed the project team to develop guidelines for effective removal and analysis of an appropriately chosen sample.

Results

The resulting guidelines include instructions for removing a sample, transmitting the radioactive material to the analysis laboratory, specifying the required analysis, and performing a safety evaluation to demonstrate that sample removal does not create a site that will promote crack initiation. In addition, the guidelines present sufficient details of the sampling performed as part of the BWRVIP/NRC project to give utilities an appreciation for the scope of required activities.

EPRI Perspective

The report presents guidelines that will facilitate utility efforts to measure helium content in a reactor material for purposes of determining weldability. Additional straightforward sample analyses can be performed to determine chemical composition as well as fast and thermal neutron fluence. These latter measurements are useful in benchmarking fluence calculations at the vessel or at other internal locations.

Keywords

BWRVIP
Boiling Water Reactor
Vessel and Internals
Repair
Fluence
Helium
Welding

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RECORD OF REVISIONS

Revision Number	Revisions
BWRVIP-96	Original Report (103019)
BWRVIP-96-A	The report as originally published (1003019) was revised to incorporate changes proposed by the BWRVIP in responses to NRC Requests for Additional Information, recommendations in the NRC Safety Evaluation (SE), and other necessary revisions identified since the last issuance of the report. All changes, except corrections to typographical errors, are marked with margin bars. In accordance with a NRC request, the SE is included here as an appendix and the report number includes an "A" indicating the version of the report accepted by the NRC staff. Non-essential format changes were made to comply with the current EPRI publication guidelines. Appendix D added: NRC Final Safety Evaluation. Details of the revision can be found in Appendix E.

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1
INTRODUCTION

One of the factors that affect the weldability of stainless steels in BWR reactors is the helium content of the metal. Helium is produced when thermal neutrons interact with boron and nickel. The resulting nuclear reactions leave behind a quantity of helium that becomes trapped in the material. When the material is melted during welding, the helium is released and forms small bubbles. If the concentration of helium is above a certain threshold (typically about 0.1 appm), the bubbles will be produced in sufficient quantity to degrade the strength of the material and to cause cracks in the weld heat affected zone. Consequently, if welding is to be considered as a means of repairing an irradiated reactor component, it is important to know the approximate helium content at the weld location.

For some internal reactor components, the helium level can be calculated. This involves estimating the fluence at the location as well as knowledge of the boron and nickel concentrations in the material. While this process is straightforward, the calculated helium concentration may be subject to substantial uncertainty. There is uncertainty in the thermal fluence calculations due to the fact that they have not been accurately benchmarked at many locations of interest (e.g. jet pump riser brace pads or core spray piping). In addition, the boron concentration of the subject material may not be known. Boron was considered a "tramp" element and was not typically recorded as part of the material certification process. Its concentration may vary from less than 1 ppm to over 50 ppm. Consequently, a more direct and accurate method for determining the helium concentration is needed in some cases.

One acceptable method for determining the helium concentration is by direct measurement on a small sample removed from the component of interest. The amount of material required for the analysis is small (<50mg) and the determination of helium content by mass spectroscopy is straightforward and yields accuracies on the order of 1-percent.

Recognizing that utilities may need to perform such sampling and analysis in the future, the BWR Vessel and Internals Project (BWRVIP) and the U.S. Nuclear Regulatory Commission (NRC) conducted a joint project to demonstrate sample removal from a number of BWRs. Samples were removed from the jet pump riser brace pad at four locations in three U.S. BWRs using tooling developed by Framatome – Advanced Nuclear Power (FRA-ANP). The samples were sent to Pacific Northwest National Laboratory (PNNL) where they were analyzed for helium content, initial boron content and accumulated fluence. The results of this project are further described in Reference [1].

The objective of this Sampling and Analysis Guideline is to use the lessons learned from the BWRVIP/NRC project to provide utilities with guidance on performing similar sampling on internal components. Section 2 of the report discusses the basic requirements for removing the sample and obtaining the desired analyses. Section 3 presents an overview of the BWRVIP/NRC jet pump riser brace sampling. It is intended to provide utilities with a better understanding of the scope of the process, a description of typical tooling, sample handling requirements, requisite safety analyses, etc.

The discussion in this report is based primarily on experience using the Framatome tooling and analyses performed by PNNL. However, other methods of sample removal and analysis are also acceptable for determining the helium content of stainless steels for purposes of assessing weldability.

Implementation Requirements

This report is provided for information only. Therefore, the implementation requirements of Nuclear Energy Institute (NEI) 03-08, Guideline for the Management of Materials Issues, are not applicable.

2
SAMPLING/ANALYSIS REQUIREMENTS

This section presents the basic requirements for removing and analyzing a sample including required sample size, sample handling, specification of the required analyses and required safety evaluations.

2.1 Sample Requirements

2.1.1 Sample Size

In order for the laboratory to perform a complete analysis including determination of helium, boron and fluence, a sample of sufficient size must be provided. The sample mass must be at least 20 milligrams. 50 milligrams is preferable. The samples can be in the form of chips, shavings, chunks or strips. The only limitation in form is that the individual chips in each sample need to be at least 1-2 mg. to perform the required analysis. The picture below in figure 2-1 shows the sample collection filter containing the material taken during a test of the sample tooling. The filter is approximately 1" in diameter. This shows the representative size of actual samples for one location. This sample was produced from a ¼" diameter drill bit at a maximum depth of 0.06".

The sample should be removed from a location in close proximity to the potential weld repair location and should consist of the same material on which the weld will be placed. If possible two samples should be removed in order to average out spatial variations in helium content. In selecting locations for sampling, consideration should be give to the shielding effects of nearby obstructions that could cause the fluence (and consequently the helium content) of the sample to differ from that of the material at the weld location.

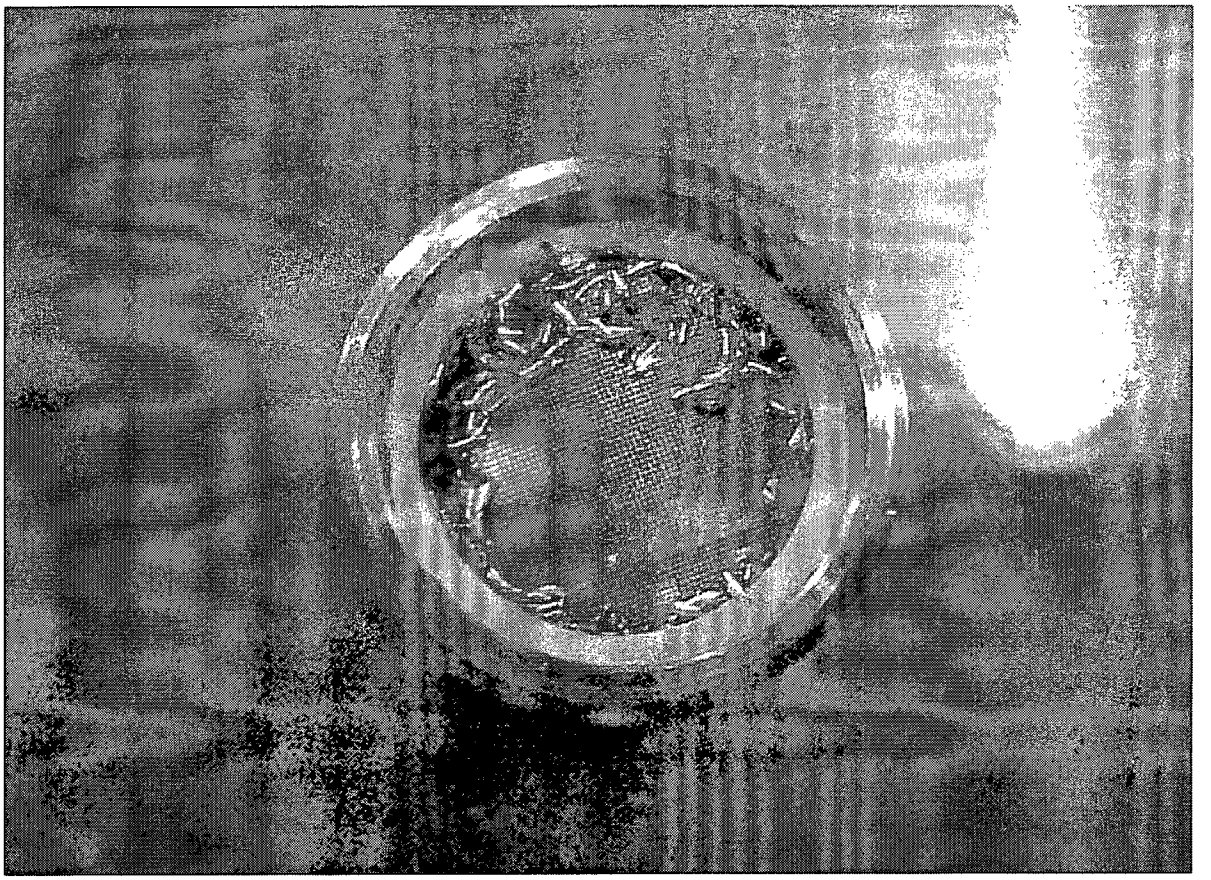

Figure 2-1
The Sample Collection Filter Containing the Material Taken During a Test of the Sample Tooling

2.1.2 Sampling Process

The jet pump riser brace pad samples from the BWRVIP/NRC project were obtained by drilling a small divot (1/4-inch diameter by approximately 0.06-inch deep) in the material of interest and collecting the drill shavings. The tooling used for this process is described in Section 3. The tooling will likely have to be modified (or alternate tooling developed) in order to allow sampling at other plants. Drawings and videos are required for the tooling and mockup designs. After fabrication, the tooling must be tested and qualified. A site procedure to take the samples must be prepared and the tooling should be tested on a full scale mockup using the site procedure.

Using the Framatome tooling, the sample removal process can be performed remotely from either the refuel or auxiliary bridge with the use of long handled poles. Since the work is performed off a bridge, the exposure levels are low. The work over the vessel is typically performed in two 12-hour non-continuous shifts. The actual schedule will depend on what other activities are being performed at that time. Foreign material exclusion (FME) must be employed during all aspects of the sample removal process. The area to be sampled should first be video

inspected to determine if the area requires cleaning to remove loose contamination, and to spot any obstructions that may hinder the sample removal process. Brushing or hydrolazing can be used for this cleaning. The sampling area should be video inspected after the sampling to determine if the sample area is the same configuration as that which was qualified. The details of the tooling, qualification, and the process are described in Section 3.1.

The equipment for sample removal included the following:

- Sample removal tooling
- Underwater video camera suitable for installation and pre/post inspection
- Hydrolazing wands and hoses (Plants supplied the hydrolazing pump), if the sampling location requires to be cleaned
- Support equipment (handling poles, VCR, monitors, videotapes, etc.)
- Sample shipping container(s)

2.1.3 Sample Handling/Shipping

Typically samples of this type are shipped in a lead lined steel container (AKA "pig"). The size of the pig is determined by the dose rate of the samples. The requirements for shipping LSA material are contained in 10CFR49. If the dose rates are low (as was the case in the BWRVIP/NRC project), the samples can be shipped as a limited quantity shipment through an express mail service like Fed-Ex. The requirement (10CFR49 Part 173) for shipping limited quantity is that the outside of the container is reading less than 0.5 mR. If the samples have a higher dose rate, a "pig" will have to be used. The shipper must also verify that the samples are within the limits of the analysis laboratory's license.

2.1.4 Definition of Desired Analysis

The samples may be analyzed for helium content, boron content, fluence (fast and/or thermal) and general chemical composition. Helium content is the primary measurement required for determining weldability. Boron may be useful if, in the future, it is contemplated that analytical methods may be used for determining helium content of similar heats of materials. Measured fluence is useful as a benchmark for future analyses and also provides direct information that can be used to "calibrate" reactor physics codes.

The following language may be used to specify the helium, boron and fluence analyses to the laboratory:

> Measure the helium and boron contents in each sample using mass spectrometric methods of small specimens taken from the drill fillings provided. The specimens shall be chosen, either using optical methods, or by gamma counting, to be representative of the bulk of the material. Specimens with surface irregularities, or with evidence of oxidation, should not be used for the analyses. Helium determinations will be conducted by gas mass spectrometry. Results will be reported as atomic concentrations (appb ^{4}He) relative to the total number of atoms in the sample to an accuracy of approximately 2%. Boron measurements will be conducted by determining the increase in helium content from the ^{10}B (n, α) ^{7}Li reaction after

supplemental irradiation of a sub-set of the samples in a thermalized neutron environment. Determination of the thermal neutron fluence will be made via the ^{6}Li (n,α) ^{3}H reaction using well-characterized Al-^{6}Li alloy included in the irradiation assembly, or by using other accepted thermal neutron dosimeter materials. Natural Boron (NB) measurements will be reported as weight concentrations (wt.ppm NB) to an accuracy of approximately 5%.

Characterize the neutron spectra at the location of each sample using retrospective dosimetry methods. The measurements will include compositional analysis by X-ray fluorescence, and radiometric counting to determine the activities of ^{51}Cr, ^{58}Co, ^{60}Co, ^{54}Mn, ^{59}Fe, and ^{63}Ni, ^{55}Fe, and ^{93m}Nb, as appropriate. Cobalt impurity concentrations in the samples will be determined from gamma analysis of the samples irradiated as part of the boron determinations. The radiometric and composition data will be used to determine the fast and thermal neutron fluence exposures. If the compositional analysis indicates sufficient ^{93}Nb, the activity of ^{93}mNb will also be reported and used in the fluence determinations. Thermal and fast (> 0.1 MeV and > 1. MeV) neutron fluences will be reported as neutrons/cm2 to an accuracy of approximately 10% or better.

Assuming that prior contractual arrangements have been made, results of the helium analysis and an estimate of the initial (i.e., pre-irradiation) boron content of the samples can be available within approximately one week of receipt of samples by the lab. Complete analysis for fluence and chemical composition may require as much as one to two months.

2.2 Required Safety Analysis

A technical justification is required to support development of a safety evaluation for the sampling activity. The technical justification should address testing, qualification and use of the equipment as well as relevant structural and material issues associated with the as-left condition of the sampling location(s).

The removal of the sample will result in a stress concentration that was not considered in the original design calculations. The following discussion shows how the effect of this stress concentration on the fatigue life of the sampling location can be addressed so that the rules of ASME Section III are not violated.

The existing stress report and thermal cycle drawing must be reviewed. Not all the stresses required to perform the fatigue analysis may be contained in the stress report. Supplementary analysis may be required to calculate other required stresses for the fatigue analysis. The significant transient loads considered in the original calculations are:

- Design Hydrotest
- Sudden Heatup and Sudden Cooldown
- Normal Heatup and Normal Cooldown

The new peak stresses and fatigue life must be calculated at the location of the sample removal for the transients that contribute to the usage factor. This is accomplished in the following steps:

- Calculate stress concentration factor (SCF) caused by sample removal

- Apply SCF to range of primary + secondary + peak stresses
- Calculate revised usage factor

A detailed discussion of the structural analyses performed for one plant in the BWRVIP/NRC program is contained in Section 3.2.1.

A material evaluation of the divot in the sampling location must be performed. The cold worked surface layer due to machining must be evaluated. The crevice condition that will exist in the machined divot must also be evaluated. The detailed description of the material evaluation is contained in Section 3.2.2.

The plant should consider if there is a need for an independent reviewer for the safety analysis.

2.3 Plant Information Required

The following plant specific information is typically required for the design of the tooling and mock-ups, and the engineering technical justification for the sampling process. Additional information may be required:

- Stress report (including stress/load changes due to power uprate)
- Reactor thermal cycle drawing
- General RPV and internals arrangement drawings
- Detail drawings of the location(s) to be sampled
- Configuration and materials of construction of the location(s) to be sampled
- Video footage (IVVI and general area footage) of the location(s) to be sampled
- Drawings which may identify potential obstructions
- Drawings of other components in the region of the sampling location(s)

2.4 Overview of Helium Analysis Procedure

The following is a summary description of the technique used by PNNL for helium analysis.

2.4.1 Definitions

Materials received for helium analysis are prepared as described below (or as specially requested), with portions of the sample being cut into "specimens" for analysis. Except as noted, the sampling includes two independent helium analyses of a pair of specimens cut by the sampling lab from adjacent locations in a single solid sample. This sampling includes cutting, cleaning, and also etching the pieces, if requested, to remove material that could have enhanced or depleted helium due to recoil (the distance an α particle travels from when it is generated until it comes to rest) or other effects. Helium recoil can be a significant surface effect for samples which were irradiated in a helium environment (i.e., helium knock-on), or for samples where the produced helium has high recoil energy (i.e., accelerator environments). Maximum recoil

distances for fusion reactor environments are typically a few tenths of a mil (i.e., 0.0001 to 0.0002 in.). For the type of sampling discussed in the present report, where the material has been collected from the bulk material beneath the component surface (e.g., by drilling), then the effects of recoil and surface contamination are not a factor, and therefore no etching is required.

2.4.2 Sample Preparation

Upon arrival at the sampling lab, the sample packages are identified and compared with the customer's correspondence. Each sample is then examined to check for adhesions or other surface irregularities, which might affect the helium analysis results. As discussed above, specimens should be selected t o be representative of the bulk of the material. Following examination, two smaller specimens are then taken from each sample for duplicate helium analysis. Each specimen is ultrasonically cleaned in alcohol and air dried. The specimens are weighed with a 1 σ uncertainty of less than 1% of the mass value (generally to $\pm$ 0.002 mg), using a mass balance with calibration traceable to the U. S. National Institute of Standards and Technology (NIST).

2.4.3 Helium Analysis Procedure

Determination of the helium content in a sample is performed by vaporizing the sample in a resistance-heated graphite or tungsten-wire crucible in a vacuum furnace connected to a mass spectrometer. The absolute amount of ^{4}He released is then measured with respect to a known quantity of added ^{3}He "spike." Each helium spike is obtained by expanding and partitioning a known quantity of gas through a succession of calibrated volumes. The mass spectrometer is calibrated for mass sensitivity during each series of runs by analyzing known mixtures of ^{3}He and ^{4}He.

2.4.4 Accuracy

The amount of helium that can be measured with an absolute accuracy of ~ 1 % ranges from ~10^{11} to 10^{17} atoms. Lower or higher helium levels can be measured with some potential reduction in accuracy, but the lower limit is usually set by the background helium released by the vaporization process. The uncertainty in this background ranges from ~5 x 10^8 to ~5 x 10^9 atoms of ^{4}He, depending on which furnace type is used to vaporize the sample.

For comparison, a 1 mg sample of steel with 0.1 appm helium contains approximately 10^{12} atoms of helium.

2.4.5 Retaining Unused Sample Material

Usually, additional sample material remains after the specimens are taken. This material is normally held for a period of 6 months in case additional analyses are requested. This also depends on the lab performing the analysis.

3
BWRVIP/NRC JET PUMP RISER BRACE PAD SAMPLING

As discussed previously, the BWRVIP and NRC have removed samples from the jet pump riser brace pad (Figures 3-1 and 3-2) at three reactors and have had the samples analyzed for helium, initial boron content, and fast and thermal fluence. An overview of that sampling and analysis is presented in this section to provide utilities with an appreciation for the scope of the process, a description of typical tooling, sample handling requirements, requisite safety analyses, personnel and time requirements, etc.

3.1 Sampling Hardware Description

FRA-ANP developed the sample removal tooling and provided the necessary personnel and equipment to support the removal of samples.

3.1.1 Sampling Tooling

The tooling was designed for removal of a sample from the pad on the reactor vessel to which the jet pump riser brace is welded. It was desired that the tooling be useable at a number of different plants with minor modification. During the tooling development, a number of BWR plants provided drawings and footage from in-vessel visual inspection (IVVI) for the riser brace and pad configurations.

The equipment for sample removal includes the following:

- Sample removal tooling
- Underwater video camera suitable for installation and pre/post inspection
- Hydrolazing wands and hoses (Plants supplied the hydrolazing pump)
- Support equipment (handling poles, VCR, monitors, videotapes, etc.)
- Sample shipping container(s)

The sampling tool (see Figure 3-3) is a remotely operated drill fixture that can be lowered into position using handling poles from the refuel or auxiliary bridge. The fixture is secured to the riser brace leaf via a rotating hydraulic clamp arrangement. The pneumatic drill motor carries the sample removal bit. A hydraulic feed cylinder is used to feed the drill motor to and from the pad. Plant de-mineralized water was used in all hydraulic cylinders so as not to affect plant water chemistry in the unlikely event of a leak.

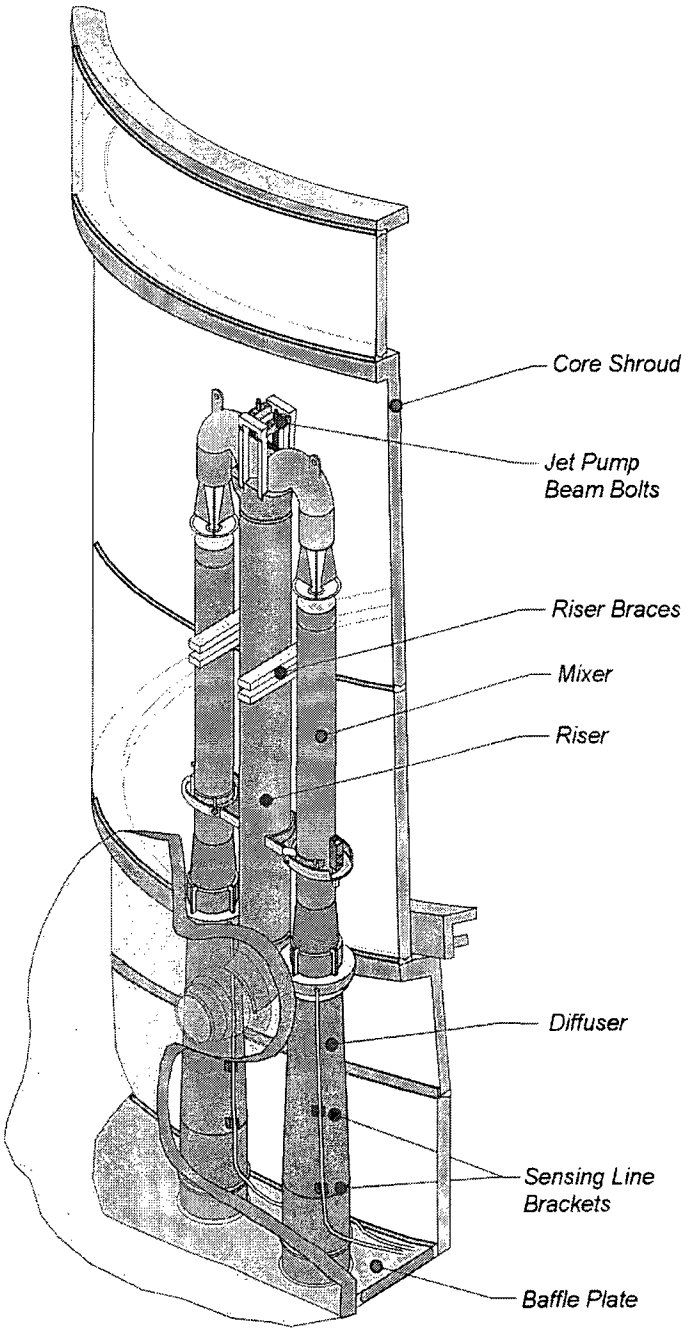

Figure 3-1
Typical BWR Jet Pump Configuration

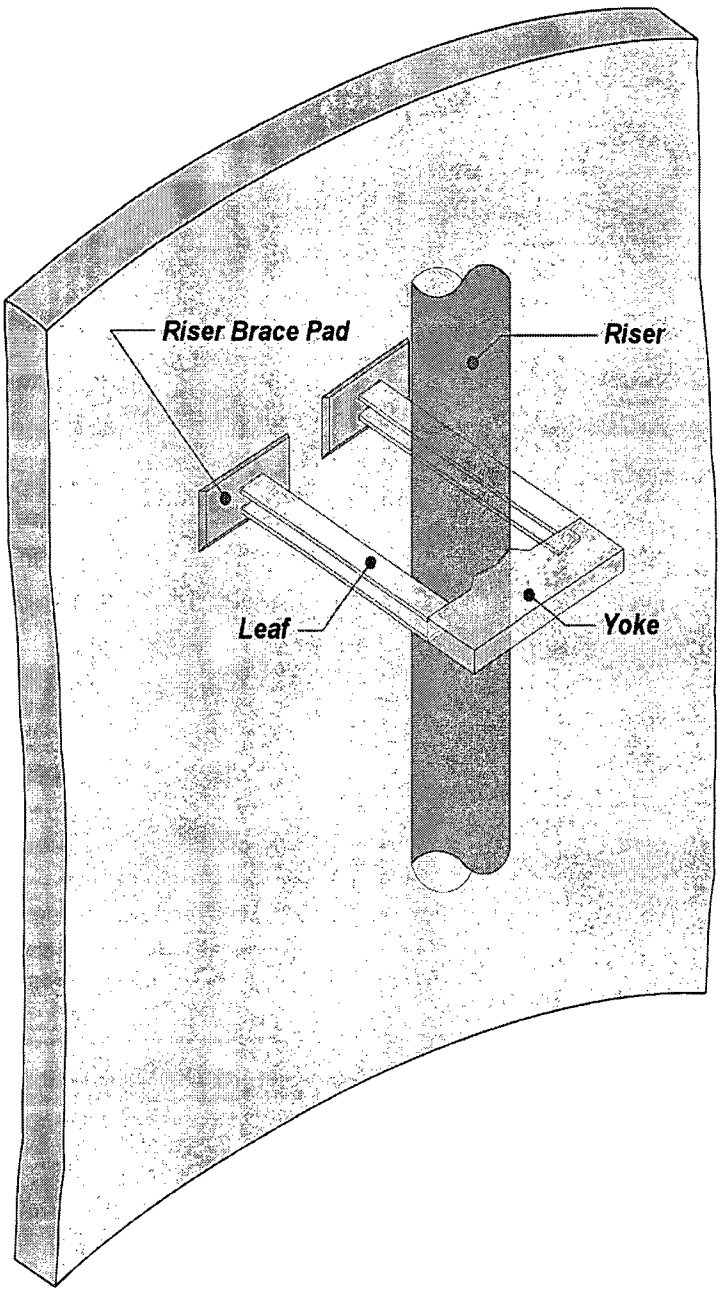

Figure 3-2
Typical Riser Brace/Pad Configuration

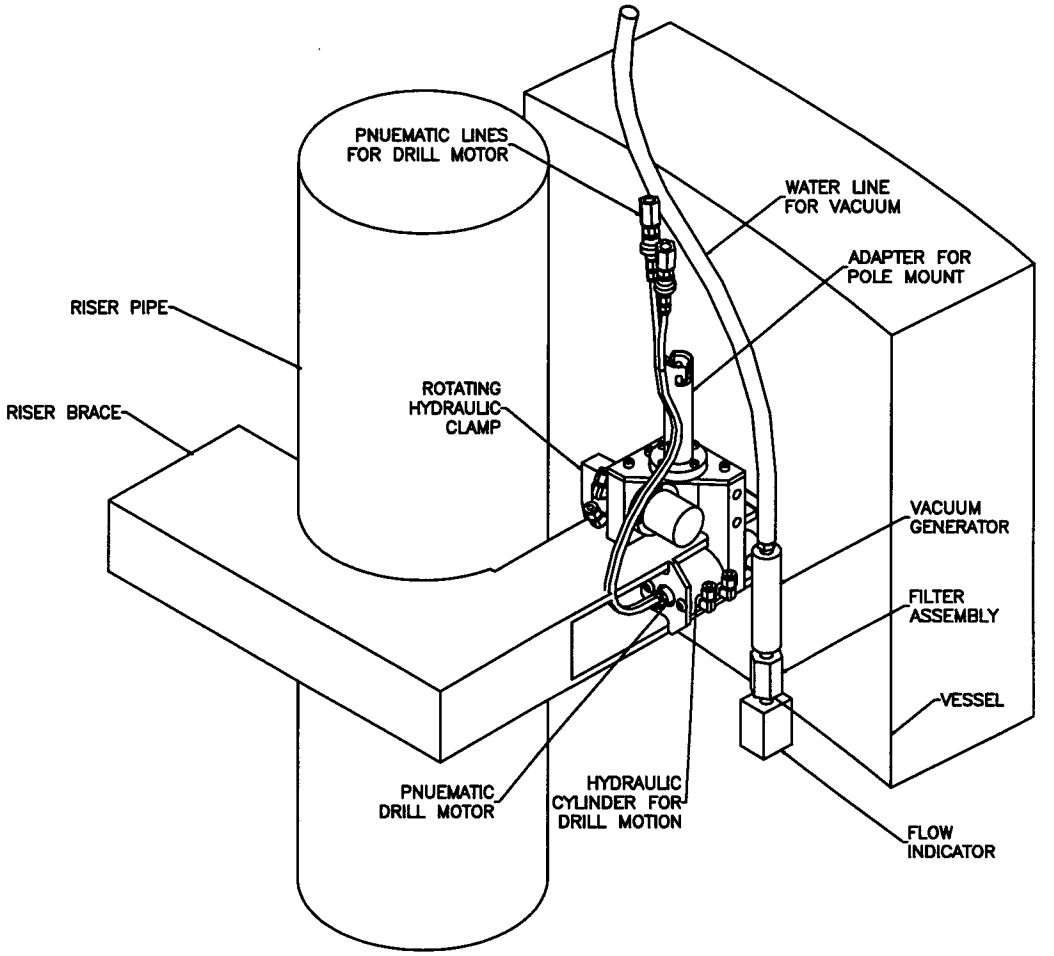

Figure 3-3
Jet Pump Riser Brace Pad Sampling Tool

A sheath protects the vessel internals (and bit) during tool installation. It also acts as a hard stop to control the depth of cut and provides containment of the sample. Sharp tool bits are used for each sample removal to minimize work hardening of the material. During the sampling process, a vacuum generator (a simple water powered venturi) is used to create a suction in the annular region between the bit and the sheath. The small chips created are pulled through the sheath into a filter cartridge so as not to create a foreign material exclusion (FME) concern. The following methods are used to determine that the vacuum generator is operating properly. The flow indicator is monitored before and during the cut to verify that proper suction is maintained. The operators ensure that the wheel turns at a comparable speed to that identified during qualification testing. If not, the cut is not started (or the cut will be stopped if previously started). There is also an air bubbler near the inlet of the debris sheath. The bubbler provides a redundant method for verifying proper suction. By visually verifying that the bubbles enter the sheath, it demonstrates that the debris is being sucked into the debris sheath during the cutting operation. Also to determine that the vacuum generator is receiving the proper flow, a pressure gage is installed on the water supply line to the tool. This pressure is monitored just before and during the cut to ensure that the pressure is greater than that used during qualification.

3.1.2 Testing and Qualification

Bench testing of the sample removal tooling was performed on mockups with representative jet pump riser brace (JPRB) configurations. In addition to bench testing, the test plan included demonstrating that the tooling can be remotely delivered using a full height mockup (Figure 3-4). Also demonstrated during testing was the ability to adequately contain the sample in the filter cartridge, eliminating the samples as an FME concern.

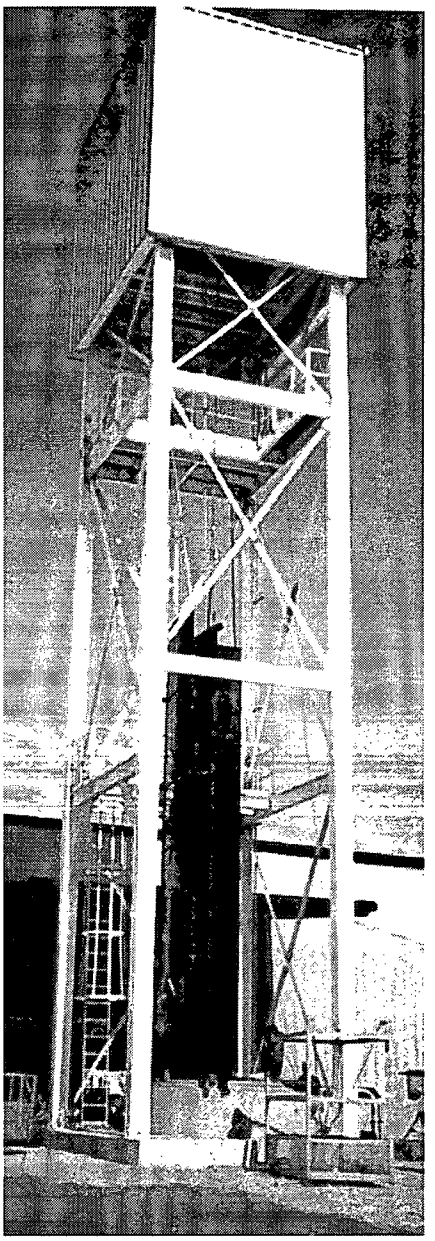

Figure 3-4
Full Height Mockup Tower

The sampling process was performed in accordance with a qualified procedure. Weld pad mockups were manufactured using a weld buildup on low alloy material. Both Alloy 182 (ENiCrFe-3) and Type 308L/309L weld pads were constructed with representative surface finishes. The weld pads were deposited in the flat position using qualified welding procedures. The Alloy 182 material was deposited using manual shielded metal arc welding technique. The 308L/309L material was deposited using a manual gas tungsten arc welding technique.

The weld pad mockup materials used for the qualification were in an un-irradiated condition.

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The qualification plan required that the sampling process leave no crevices (see Section 3.2.2) on the weld pad mockups. After sampling, both PT (color contrast-solvent removable method) and visual inspections (EVT-1) were performed on the divots in the weld pad mockups. The acceptance criteria allow no indications of a crevice being generated as a result of the sampling process. Photographs and video of the qualification samples were taken so that they may be compared to the as-left condition of the riser brace pads on site.

3.1.3 Sample Removal Process

The sampling process involved removal of a small amount of material from the surface of the JPRB weld pad and resulted in a small "divot" being left. The configuration of the "divot" is shown in Figure 3-5. The sampling process was performed in accordance with a qualified procedure. Prior to sample removal, the pad may be either brushed or hydrolazed (cleaned with high-pressure water) at the option of the utility and was visually inspected (EVT-1) in the region where the sampling is performed.

A new filter was placed in the tool. The sampling tool was lowered into position using handling poles from either the refuel or auxiliary bridge. The fixture was secured to the riser brace leaf via a rotating hydraulic clamp arrangement.

Once the sample has been taken, the fixture was removed from the vessel and the filter cartridge was detached, placed in a suitable shipping container and shipped to a lab for processing. Approximately 100 to 200 mg of material was removed from each location sampled. A final visual inspection (EVT-1) of the sampled region was performed to ensure that the as-left condition is comparable to the qualification samples.

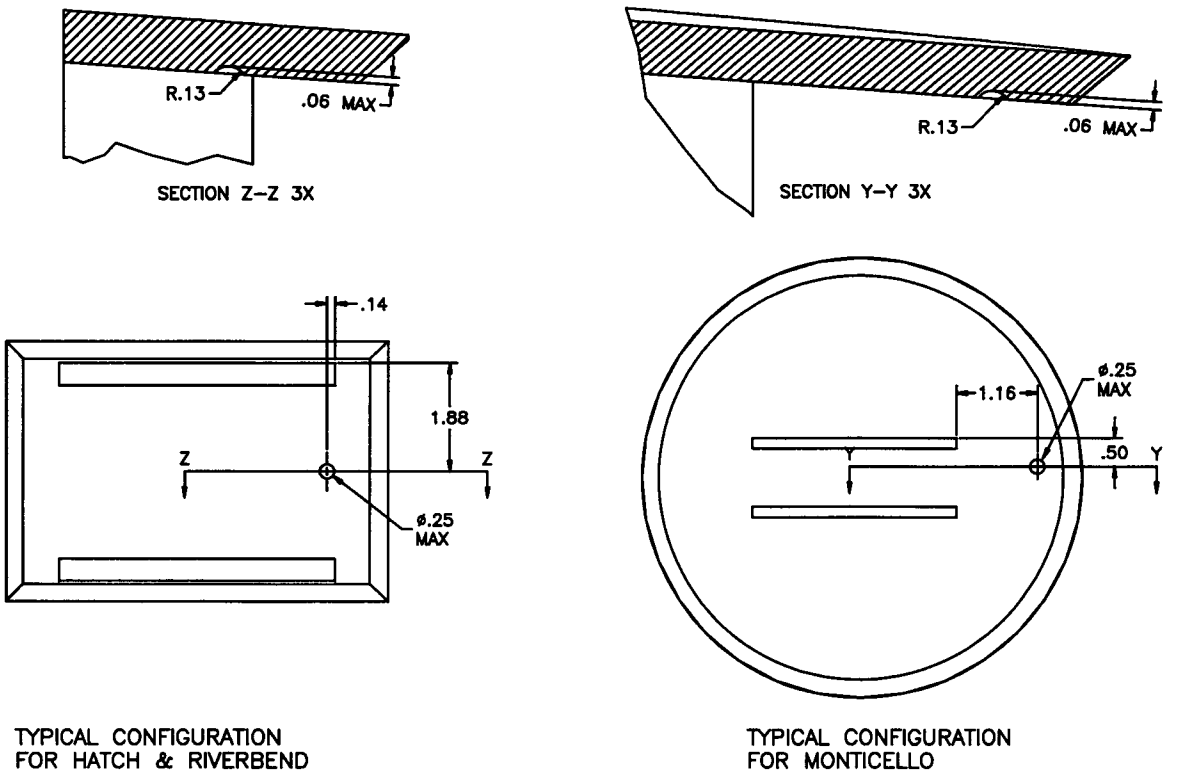

Figure 3-5
Proposed As-Left Riser Brace Pad Configuration

3.1.4 Sample Site Support

The utility supplied the following general site support:

- Office facilities for 4 people with phone, fax, modem and photocopier available
- Mobilization of equipment to and from the refuel floor
- Health Physics
- QA support (as required by site)
- Qualified bridge operators (as required by site)

The utility supplied the following Site supports for the refuel floor:

- Equipment laydown/setup area (Approx 15' x 15')
- Clean/dry plant air (10 scfm @ 80-100 psi) (Air which has been filtered and passed through a moisture separator.)
- Demin water supply 40 psi (8-10 gal/min)
- Demin water for hydraulics (2-5 gallons)
- 110V, 20A, 60 Hz power (on bridge and in setup area)
- Site supplied consumables (duct tape, Loctite, etc.)
- Sample shipment

3.1.5 Sampling Personnel

A single four-person crew consisting of one Task Lead, two Reactor Services Technicians and one Tooling Engineer were supplied to support the sampling activities. With this crew, the sampling activities can be performed over a two-day period working two 12-hour non-continuous shifts. In addition, one 12-hour shift was required before and after the sampling effort for equipment setup and demobilization. If the Plant has two work platforms available, it is possible that the sampling activities could be performed in parallel with other in-vessel activities.

3.1.6 Sampling Locations

Six JPRB pad locations were selected for each of the three plants in the BWRVIP/NRC project. Three locations had high fluence levels and three locations had low fluence levels. Two locations were for contingency in case a sample could not be obtained from the primary locations.

3.1.7 Sample Shipment

The samples taken from each of the three plants had dose rates of <2mr. Since the dose rates were so low, it was possible to ship the samples to PNNL as a limited quantity shipment by Fed-Ex.

3.1.8 Lessons Learned

In performing the sampling at the three plants, there were several lessons learned that were incorporated into the project. Some factors that were incorporated into tooling modifications include: a secondary means was needed to ensure that the vacuum was operating properly; a flow meter was added to the water supply side of the vacuum and a bubbler was added in front of the sampling tube to ensure that the sample chips would be sucked into the sampling tube.

After the first plant was sampled, the need for hydrolazing was revisited. There was very little difference in the surface condition before and after hydrolazing. Therefore, it was not performed at the two other plants. There was no difference in the dose levels of the samples with and without hydrolazing.

Foreign material exclusion (FME) is always a concern when operating tooling inside the reactor vessel. Tooling needs to be designed to ensure that parts do not become a FME concern. An example of this is that a quartz window of a camera light had become separated from the housing causing the reflector light assembly inside the housing to also become separated and lost in the Reactor Vessel. The thoughts are that the epoxy, which seals the lens to the housing, had degraded after time and had failed. The housings are now being visual inspected for signs of degradation. Tape is also being placed over the outer edges of the lens to secure the lens to the housing to prevent a lost parts situation even if the lens were to fail.

A concern was raised regarding cross contamination of the samples. To a certain degree, the location of the sampled chips can be determined from the measured fluence level. However, in order to avoid resorting to this means of sample identification, the tooling should be checked to ensure that chips are not left in the tool from the previous sampling. As a minimum, a visual inspection should be performed; a more positive verification can be performed by flushing the lines.

3.2 Safety Analysis

A safety analysis was conducted to ensure that the sampling would not leave the surface in a degraded condition. The safety analysis consisted of a structural analysis and a material evaluation of the sample divot in the JPRB weld pad. The safety analysis was reviewed by an independent reviewer, Structural Integrity Associates (SIA). While the analyses performed for the three plants differed somewhat, details of one safety analysis are presented below to indicate the types of evaluations performed.

3.2.1 Structural Analysis

Removal of the sample from the weld pad will result in a stress concentration that was not considered in the original design calculations. The following discussion demonstrates that the effect of this stress concentration on the fatigue life of the reactor vessel is small and that the rules of ASME Section III are not violated.

In the following discussion, the stresses due to the loads on the shell from the riser braces are not considered. From the "Stress Analysis Brackets" section of the reactor vessel stress report, the stresses are on the order of 1 ksi, which is negligible and will have no appreciable effect on the calculated cumulative usage factor (CUF).

The first step in the analysis is to calculate the stresses at the location of the sample. From the "Top Head and Cylindrical Shell" section of the reactor vessel stress report, it was determined that the range of local membrane plus secondary bending stress at the inside of the vessel at the jet pump riser brace pad location is 22.6 ksi. The report indicated that thermal stresses are not included. Therefore, this stress is due solely to pressure. The report also stated that fatigue requirements were met by demonstrating that the rules of paragraph N-415.1 (Exemption from Fatigue) were satisfied.

Other plant information indicated that the location of interest experiences little fatigue and is exempt from detailed fatigue analysis. Therefore, the fatigue usage at this location (before sample removal) is low, which is consistent with the discussions above.

To complete the stress analysis, a through-wall gradient stress must be determined and the stresses due to the material interface (base metal/cladding) must be accounted for. The relevant parameters are:

Material	A533 Gr B Cl 1
Thickness	5.00"
Inside radius	103.19"

Since the through-wall gradient stress for heatup and cooldown (HU and CD) was not calculated in the reactor vessel stress report, this was determined using information from the vessel stress report for a plant with similar characteristics. It is reasonable to use the results from the analysis of the similar plant in this assessment since the shells are practically identical for the purpose of calculating the through-wall gradient and the heatup/cooldown rates are the same (100°F/hr).

For a linear temperature gradient through a shell, the thermal bending stress is

$$E\alpha(\Delta T)/(2)(1-\nu) \qquad \textbf{Equation 3-1}$$

Where: E = Modulus of elasticity

α = Coefficient of thermal expansion

ΔT = Temperature gradient across the wall

ν = Poisson's ratio = 0.3

It should be noted that the factor 2 in the denominator does not appear in Equation 3-1 when the ΔT used is from mid-wall to the inside surface.

From the reactor vessel stress report, the temperature differences between the inside surface and mid-wall for normal HU and CD are –27.5°F and +16.5°F, respectively. For the subject reactor vessel base material listed above at 550°F, E = 26.7E6 psi and α = 7.77E-6/°F. Using ΔT = [(16.5) - (-27.5)] = 44.0°F in Equation 3-1 (with the factor of 2 removed in the denominator since the ΔT used is from mid-wall to the inside surface), the stress range due to normal HU and CD is 13.1 ksi.

Finally, the effect of the difference in thermal expansion between base metal and cladding must be included. The cladding/base metal interface stress is due to differences in the coefficient of thermal expansion only, as the interface is at the same temperature. The equation for this interface stress is

$$E(\Delta\alpha)(\Delta T)/(1-\nu) \qquad \textbf{Equation 3-2}$$

Where: E = Modulus of elasticity

$\Delta\alpha$ = Difference in material coefficients of thermal expansion

ΔT = Material temperature – stress free temperature

ν = Poisson's ratio = 0.3

For 304 stainless steel (used for conservatism versus a nickel-based weld metal) at 550°F, E = 25.6E6 psi and α = 9.45E-6/°F, the corresponding difference between the cladding and low alloy base metal coefficients of thermal expansion is $\Delta\alpha$ = [(9.45E-6)-(7.77E-6)] = 1.68E-6/°F. With a stress free temperature of 70°F, ΔT = (550-70) = 480°F. Therefore, using Equation 3-2 with E = 25.6E6 psi, the interface stress is 29.5 ksi.

Next, the stress concentration factor needs to be determined. This will be based on Reference [4], Figure 6-1 (c). Using R = 0.13", h = 0.10", D/d = 1, a stress concentration factor of 2.8 is obtained. It should be noted that using h = 0.10 adds additional conservatism for slight variations in the depth of the sample (nominally 0.06" max) due to the weld pad surface variations that may be expected in the field.

To enter the ASME Section III design fatigue curve (1965 Edition), the alternating, i.e., (half-range) stress intensity is formed by multiplying the product of stress range (22.6 + 13.1 + 29.5 = 65.2 ksi) and stress concentration factor (2.8) by one-half. This yields a value of 91 ksi. From ASME Section III, Figure N-415 (B), the allowable number of cycles is 1500.

The discussion in the Monitoring and Equipment Cycles report indicates that different parts of the vessel were analyzed for different numbers of heatup/cooldown cycles. A value of 300 cycles envelopes the range of values identified in the Monitoring and Equipment Cycles report, including those required for power re-rate. Of course, some of these cycles have already been used but we will use the value of 300 cycles for conservatism. The cumulative usage factor (CUF) is then CUF=300/1500 = 0.2. This is well below the allowable value of 1.0 and, as such, the formation of a fatigue crack is not expected.

3.2.2 Material Evaluation

The BWR JPRB weld pad is fabricated from Type 308L/309L weld metal. Stainless steel (e.g., Type 308L) weld metal has been found to be very resistant to intergranular stress corrosion crack (IGSCC) initiation in a BWR environment and has been used to provide a protective cladding on the recirculation piping inside surface [5]. However, these materials may be susceptible to IGSCC should a heavily cold worked surface or a metallurgical crevice be present [6]. A metallurgical crevice is defined as a lack of fusion, a visible crack, or smeared metal.

3.2.2.1 Cold Work

Numerous evaluations of IGSCC in stainless steel materials in BWRs [7 and 8] have shown that cold work increases the susceptibility to cracking of plate and piping stainless steel materials (e.g., Type 304/304L). However, the duplex microstructure (i.e., austenite and ferrite) of these weld metals makes them significantly more resistant to IGSCC [5].

If cold work were to introduce an initiating defect or crack on the surface of the Type 308L/309L stainless steel weld metal, crack growth would be very limited or non-existent due to the combination of low carbon and significant ferrite in the weld metal. Laboratory studies and examination of field components have provided evidence that austenitic stainless steel weld metals are essentially immune to IGSCC when they contain ferrite levels typical of those present in 308L/309L material [9].

There are many locations throughout the system where Type 308L/309L weld metal has been ground during manufacturing, thereby introducing cold work to the surface, and there has been no evidence to date of IGSCC. Therefore, the small amount of cold work resulting from the sampling process, which would be significantly less than that introduced in other BWR components during original construction, is considered acceptable.

3.2.2.2 Crevices

Type 308L/309L stainless steel weld material is susceptible to IGSCC in a crevice situation [8, 10, and 11]. Therefore, a metallurgical crevice cannot be left in place. In the absence of crevices, there is little experience or information that suggests cracks would initiate in these pad materials. The process qualification (see Section 3.1.2) will show that no crevices (PT exam and visual inspection consistent with BWRVIP requirements, i.e. EVT-1) have been created during the sample removal process. Therefore, IGSCC initiation, resulting from the introduction of a crevice during the sample removal process, is not an issue.

3.2.2.3 Metallurgical Examination

Metallurgical examinations were performed on the jet pump riser brace weld pad qualification samples. The samples were examined for evidence of laps, tears, or crevices, and none were found. The samples were also examined for excessive cold work, and none was found.

Two samples used by FTI in the performance qualification of the jet pump riser brace sampling tool were analyzed. The samples were identified as RBW-001 and RBW-002. A sample which encompassed a machining divot was removed from each sample, see Figures 3-6 and 3-7. The sample from RBW-001 was identified as 1-2 and the sample removed from RBW-002 was identified as 2-3.

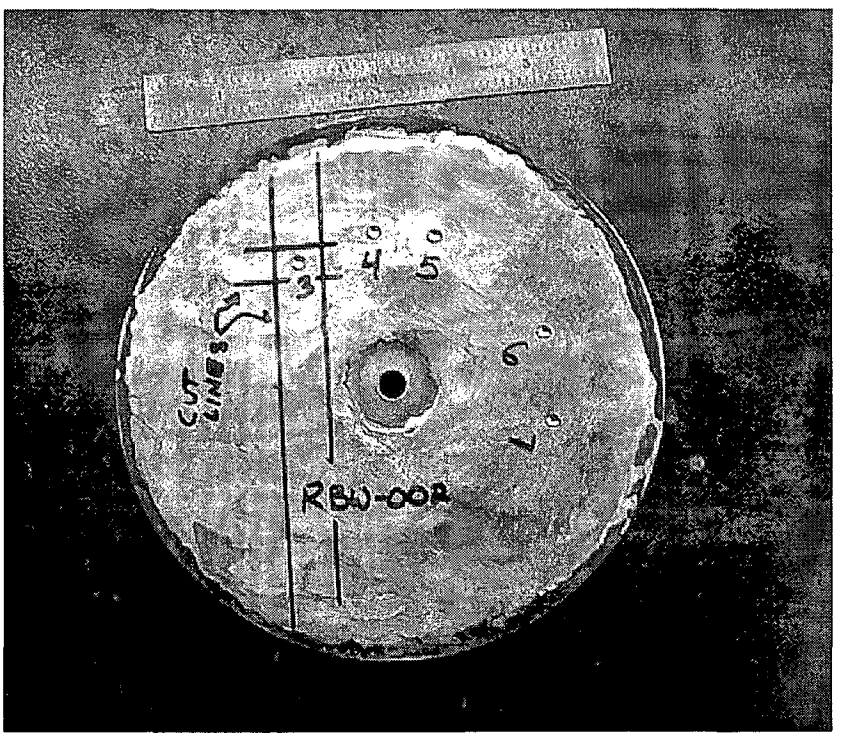

Figure 3-6
RBW-002 Cut Line Layout for Sample 2-3

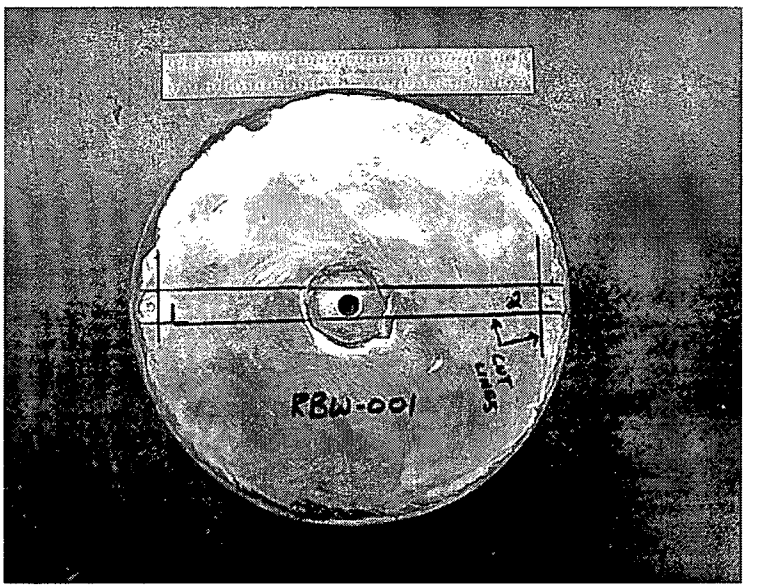

Figure 3-7
RBW-001 Cut Line Layout for Sample 1-2

Each sample was cross-sectioned through the center line of the divot, mounted in phenolic resin and polished according to accepted metallurgical practices. The cross sections were examined in the un-etched and etched conditions at 50X and 100X. A typical etched cross section of sample 2-3 is shown in Figures 3-8 and 3-9.

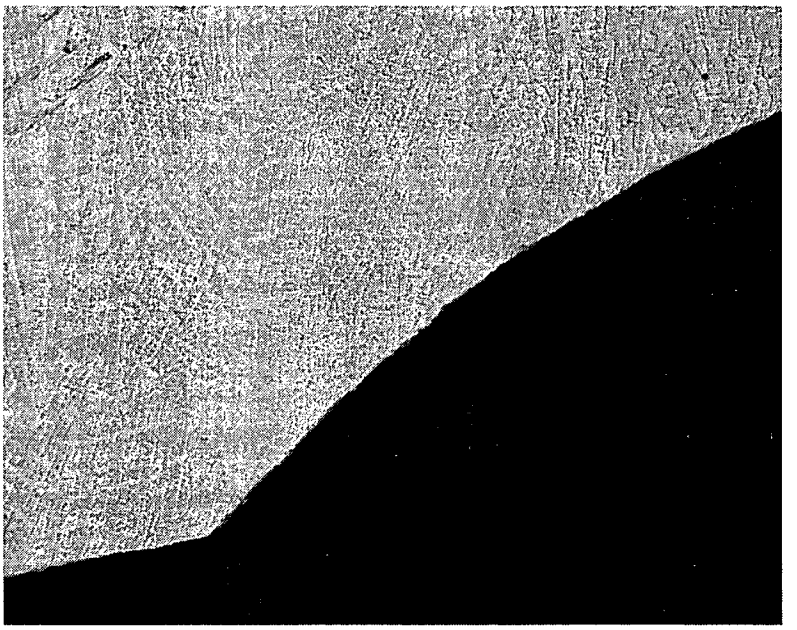

Figure 3-8
Cross Section of Sample 2-3 (100X)

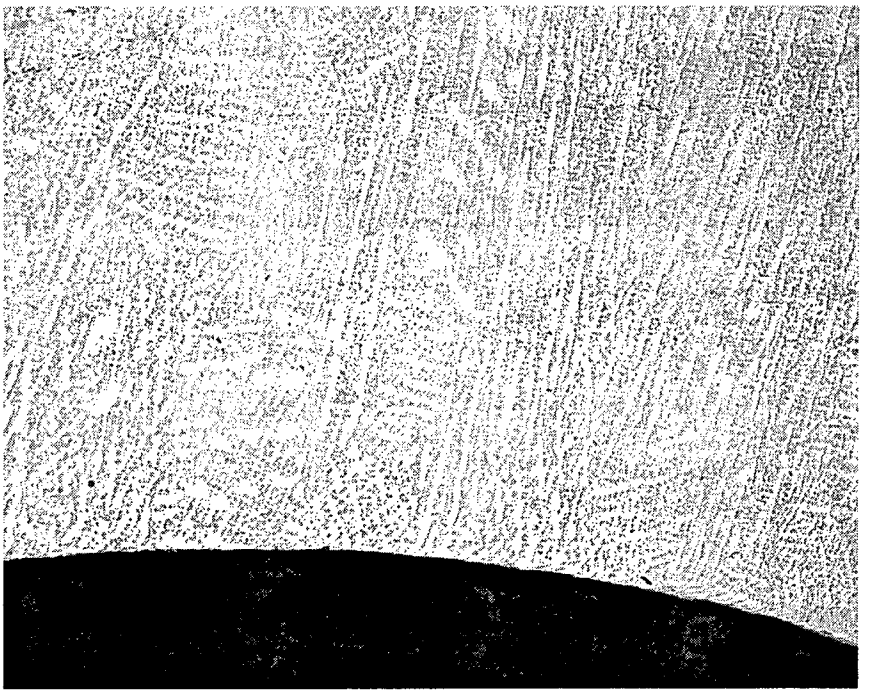

Figure 3-9
Cross Section of Sample 2-3 (100X)

The examination revealed a dendritic grain structure with small nonmetallic inclusions. This structure is typical of stainless steel weld metal deposits. Both cross sections had a very distinct edge with no evidence of laps, tears or crevices. There was very little deformation of the dendritic structure indicating minimal cold work had occurred during the sample removal process.

A hardness test was performed on the surface of the weld metal build up on sample RBW-002. This sample was adjacent to divot 7, see Figure 3-6, and had a Rockwell hardness of 16.4C (~ 96Rb). A hardness test was performed at the bottom of divot 7, see Figure 3-6, and had a Rockwell hardness of 15.4C (~ 95Rb). This indicates that the sample removal process does not induce excessive cold work in to the weld metal.

Based upon the above examinations and tests performed on samples RWB-001 and RWB-002, it would appear that the machining process proposed for removal of samples from the jet pump riser pads is acceptable for use. The machining process did not create laps, tears or crevices and did not introduce excessive cold work into the weld metal pad. Based upon the examinations and tests, it would appear that the divots in the weld pad resulting from the removal of samples should not produce an initiation site for stress corrosion cracking.

3.3 Sample Analysis Results

All samples removed in the BWRVIP/NRC program were analyzed for helium content, initial boron content and fast and thermal fluence. Detailed results for all three plants are presented in Appendices A, B and C to show the types of results that can be expected from the analysis laboratory.

No information from the plant operator is generally required for the helium and boron determinations. Correcting the measured as-received boron contents back to their original conditions prior to irradiation requires an estimate of the plant thermal fluence exposures. Usually these corrections are small (<1%), and therefore large uncertainties in these estimates can be tolerated.

Additional information is usually required for determining the plant fluences. As a minimum, this includes the end of irradiation date and a reactor power time history. With this information, fluences can be determined using a generic BWR neutron spectrum. If more accurate spectra are available, this will reduce the overall uncertainties in the fluence determinations to some extent. For one plant (Plant 1, Appendix A), additional information on the fuel pin power distribution was also required to improve the uncertainties in the measured fluence values.

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A
SAMPLE MEASUREMENT RESULTS – PLANT 1

Presented below are the results of the helium, boron and fluence analyses for one of the three plants that were sampled as part of the BWRVIP/NRC project. The results are presented essentially as received from the laboratory. The results were originally reported by B. M. Oliver and L. R. Greenwood of PNNL.

A.1 Helium and Boron Analyses

A.1.1 Summary

Results of helium and boron analyses on four jet pump riser brace (JPRB) weld pad samples provided by FRA-ANP are reported.

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Estimated uncertainty in the boron concentrations ranged from ~3 % to ~6 %. Uncertainties greater than about 2% are attributed to heterogeneities in the boron contents in the material for the milligram and sub-milligram sized samples analyzed. Overall, however, the range of heterogeneity is very similar to that observed in previous sets of reactor steel. The range of boron concentration values, however, is higher than can likely be explained by measurement uncertainty or boron heterogeneity, and suggests real variations in the boron contents between the four JPRB samples.

A.1.2 Analysis Samples

Four sets of stainless steel samples were received from FRA-ANP on February 1, 2000 for helium, boron, and activation analysis. All of the samples had been taken remotely from jet pump riser weld pads from a commercial reactor between January 15 and January 18, 2000, and were individually contained in stainless steel filter assemblies. Each of the samples was in the form of multiple small machine chips.

Sample 1 was intermingled with rust and organic matter from the flow water supply. This material had essentially no activity, and was separated out prior to subsequent sample preparation.

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A.1.3 Sample Preparation

Samples for helium and boron analyses were taken from a subset of the as-received material. For the initial helium analyses, single discrete metal pieces were selected. The pieces were chosen with the aid of a low-power stereo microscope to be as representative of the each sample set as possible. Pieces that were potentially oxidized, or that had other features not representative of the lot as a whole, were not used for the analyses.

For the boron determinations, multiple chips were selected as described above and wrapped in aluminum foil for neutron exposure at the University of Missouri Research Reactor (MURR). Following the MURR exposure, single specimens were selected from the set for additional helium analysis to determine the boron content. Details of the neutron exposure are given in Section A.1.4.

Prior to analysis, each specimen was cleaned in acetone, air-dried, and then weighed using a microbalance with calibration traceable to the National Institute for Standards and Technology (NIST). Specimen masses ranged from ~0.2 to ~2 mg each. Mass uncertainty is conservatively estimated to be ±0.002 mg.

A.1.4 Neutron Exposure

Determination of the boron content in the samples was made by first irradiating a selected subset of the sample material in a thermalized neutron field in the MURR and then measuring the increase in the sample helium content. In a thermalized neutron field, helium generation will be predominately through (n,α) reactions with ^{6}Li or ^{10}B, or with Ni through the two-stage reaction ^{58}Ni(n,γ)^{59}Ni(n,α). Lithium is assumed to be an improbable impurity in steels due to its relatively high volatility. If the materials have already been exposed in a neutron environment, as was the case here, corrections are also applied to determine the original level of boron in the material.

The boron characterization samples and the dosimetry samples were packaged together inside an aluminum rabbit assembly originally supplied by Brookhaven National Laboratory (BNL). The samples were contained inside a small aluminum capsule, wrapped with aluminum foil, and held in the approximate axial center of the rabbit. The aluminum rabbit, capsule, and foil were fabricated from Type 1100 aluminum.

Figure A-1 shows a diagram of the sample loading arrangement inside the rabbit. The rabbit was backfilled with argon, and sealed prior to shipment to MURR. Table A-1 lists the individual samples in the irradiation capsule. The aluminum rabbit was irradiated at MURR in a reflector location for 24 hours. Details of the irradiation are given in Table A-2.

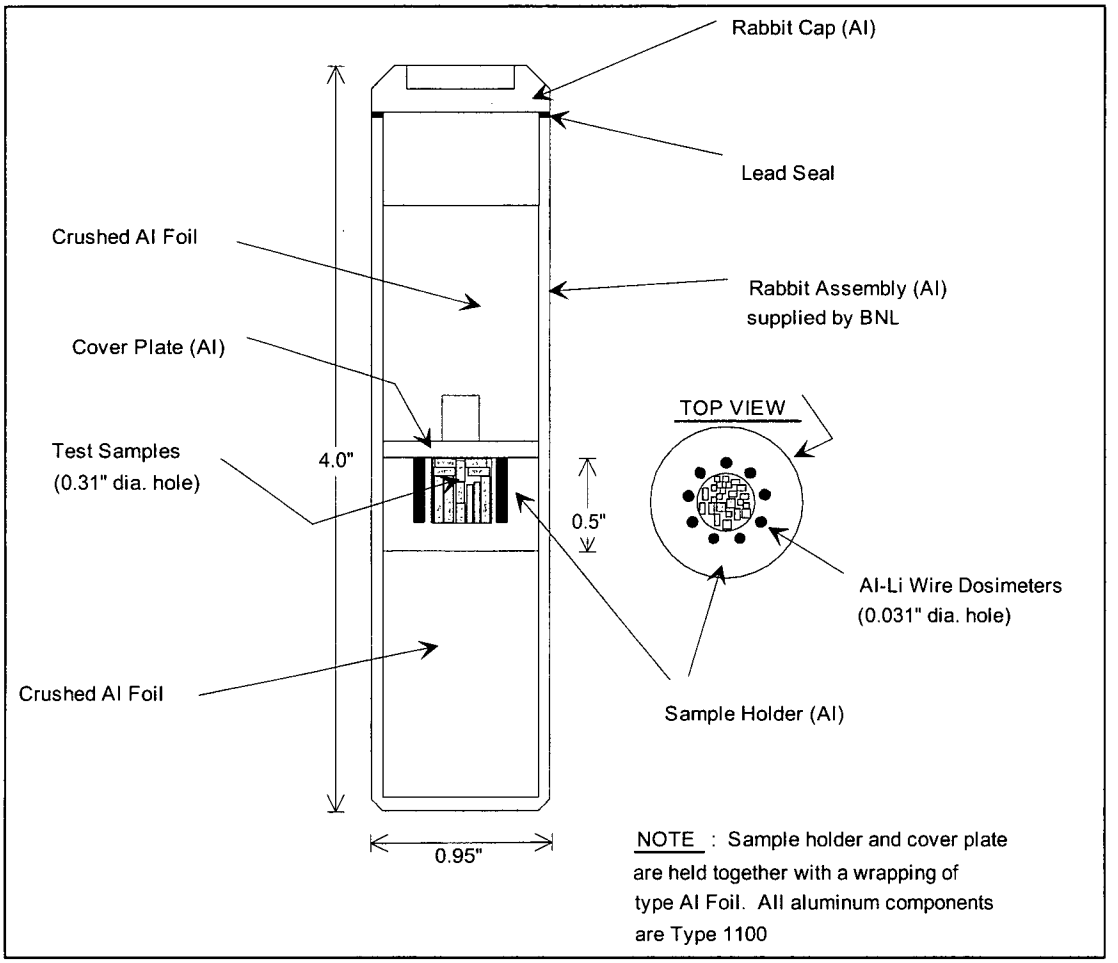

Figure A-1
MURR Irradiation Assembly

Table A-1
Reactor Steel and Dosimetry Samples Irradiated in MURR

Sample Name	Material	No.[a]	Mass (mg)	Sample Description
SS-1	Stainless steel	1	3.2	Multiple tool chips
SS-3	Stainless steel	1	3.6	Multiple tool chips
SS-4	Stainless steel	1	10.4	Multiple tool chips
SS-5	Stainless steel	1	11.5	Multiple tool chips
AlLi-1 to -4	Al-0.7 wt.%^{6}Li alloy	4	12.2	Bare wire, 0.5 mm dia. x ~6 mm long
V-1	Vanadium (Lot 7)	1	29.7	Bare wire, 0.9 mm dia. x ~6 mm long

[a]Number of specimens of sample included in irradiation capsule. The steel samples consisted of multiple chips wrapped in aluminum foil.

Table A-2
Summary of MURR Irradiation Parameters

Parameter	Value
Irradiation Start Date	March 3, 2000
Irradiation Location	H-1 (reflector position)
Irradiation Temperature (°C)	46 (nominal)
Irradiation Time (hours)	24
Calculated Thermal Neutron Fluence (n/cm^2)[a]	5.71×10^{18}
Estimated Fast Neutron Fluence (n/cm^2)[b]	2×10^{17}
^{10}B Burnup (%)	2.2

[a]Thermal neutron fluence determined from the measured helium generation in the Al-Li alloy samples (see text).

[b]Estimated fast neutron fluence (>1 MeV) in H-1 location.

Multiple samples of Al-Li alloy wire [12] were included in the irradiation capsule in order to characterize the thermal neutron fluence. This alloy was originally fabricated by the Central Bureau for Nuclear Measurements (CBNM) in Belgium specifically for neutron dosimetry applications. The stated CBNM composition of the Al-Li alloy is 0.705±0.025 wt.% Li, with a lithium isotopic composition of 95.82±0.08 at.% ^{6}Li. Independent analysis at Argonne National Laboratory showed a lithium content of 0.730±0.004 wt.%, and an isotopic composition of 95.67±0.04 at.% ^{6}Li; essentially in agreement with the CBNM values but with lower uncertainties. For analysis purposes, a composition of Al-0.73±0.01 wt.% Li, with a ^{6}Li content of 95.7±0.1 at.% is assumed.

Also included as a control material for the boron determinations was a sample of PNNL's Lot 7 vanadium material. This material is used for the fabrication of Helium Accumulation Fluence Monitor (HAFM) capsules and radiometric monitors, and has been characterized for boron content in a variety of irradiation environments yielding a current average value of 4.5 ± 0.2 wt. ppm.

A.1.5 Helium Analysis Procedure

Helium analyses on all specimens were conducted by isotope-dilution gas mass spectrometry following vaporization in a resistance-heated tungsten-wire or graphite crucible in one of the mass spectrometer system's high-temperature vacuum furnaces. [13]. Duplicate helium analyses are routinely performed to give an indication of the analysis reproducibility and to give an indication of the helium homogeneity in each sample. This was important for the present work, where heterogeneity in the boron impurity was a concern, particularly for Samples -1 and -3, where limited sample material was available. To help clarify the heterogeneity question, triplicate analyses were conducted on selected samples as indicated in Table A-3.

Table A-3
Helium Concentrations in Al-Li Alloy Dosimetry Samples

Specimen	Material	Mass[a] (mg)	Measured ^{4}He (10^{15} atoms)	Helium Concentration (appm)	
				Measured[b]	Average[c]
AlLi-1-A	Al-Li Alloy	0.983	3.505	5136	
AlLi-2-A	"	0.870	3.035	5025	5048 ± 79
AlLi-3-A	"	1.013	3.504	4983	

[a]Mass of analyzed specimen. Mass uncertainty is ±0.002 mg.

[b]Helium concentration in atomic parts per million (10^{-6} atom fraction) with respect to the total number of ^{6}Li atoms in the specimen.

[c]Mean and standard deviation (1σ) of analyses.

The absolute amount of ^{4}He released was measured relative to a known quantity of added ^{3}He "spike." The ^{3}He spikes were obtained by expanding and partitioning known quantities of gas through a succession of calibrated volumes. [14]. The mass spectrometer was calibrated for mass sensitivity during each series of runs by analyzing known mixtures of ^{3}He and ^{4}He, as specified in PNNL Procedure RPG-MSL-1000.

A.1.6 Helium Analysis Results

The results of the helium measurements are given in Table A-3 and A-4. Table A-3 gives the results of the helium analyses on the Al-Li alloy dosimetry material. Table A-4 gives the helium concentrations measured in the characterization samples both before and after neutron exposure at MURR.

Helium concentrations were measured in three of the four Al-Li alloy wires included in the irradiation. These concentrations are given in Table A-4, and are listed as total atoms of helium released, and as helium concentrations in atomic parts per million (10^{-6} atom fraction) relative to the total number of ^{6}Li atoms in each specimen. Conversion from total helium to helium concentration was based on a calculated number of ^{6}Li atoms per gram of 0.06942×10^{22}.

Helium concentrations in the Al-Li alloy were used to determine the thermal neutron exposure in MURR. From Table A-3, the mean helium concentration (in terms of atom fraction) is 0.005048 ± 0.000079. Correcting for burnup of the ^{6}Li isotope and for neutron self-shielding yields a value of 0.005370 ± 0.000084, which represents the ^{6}Li reaction probability $\sigma\varphi t$. Burnup corrections were made using the standard exponential formulation. Self-shielding corrections were based on transport theory approximations from Case et al. [15], and were 6.1%. Uncertainty in the self-shielding correction is conservatively estimated to be ~25% of the correction factor. Assuming a 2200 m/s ^{6}Li(n,α)^{3}H neutron cross section of 941 barns, yields an effective thermal neutron fluence of 5.71×10^{18} n/cm^2. It should be noted, that this value is in excellent agreement with an independent determination of 5.75×10^{18} n/cm^2 obtained from the gamma analysis data used to determine the plant fluences.

Table A-4
Measured Helium Concentrations

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Helium concentrations measured in the samples, both before and after neutron exposure at MURR, are given in Table A-4. The results are listed as total atoms of helium released and as helium concentrations in atomic parts per billion (10^{-9} atom fraction). Residual (pre-exposure) helium concentration in the PNNL vanadium has been determined previously to be <0.03 appb. Conversion from total helium to helium concentration was based on the following calculated values:

308L SS	1.09×10^{22} atoms/gram
Vanadium	1.182×10^{22} atoms/gram

The value for the 308L SS material is a nominal value for Type 308L stainless steel. It should be noted, however, that these numbers, and the helium concentrations obtained using it, are not very sensitive to small changes in material composition.

Absolute uncertainty in the measured helium concentrations is estimated to be ~2% (1σ). This uncertainty results from the cumulative uncertainties in the sample mass, the isotope ratio measurement, and the spike size.

A.1.7 Boron Determinations

Calculated boron contents in the samples are given in Table A-5. Column 6 gives the calculated helium contribution from the nickel component in the as-received material, accounting for burnup of ^{58}Ni and in growth of ^{59}Ni from the material's prior reactor exposure. Cross sections used for the nickel two-stage reaction were those determined by Greenwood et. al. [16]. The nickel contributions in Table A-5 are essentially negligible at ~0.1 appb. A conservative error estimate of 10% is assigned to this value. Helium contributions from threshold neutron reactions (greater than ~2 MeV) were calculated to be <1 %, and were neglected.

The Column 7 values in Table A-5 represent the calculated ^{10}B content in the as-received samples *prior* to the MURR exposure. These values were calculated from the "net" increase in the helium content attributable to the boron, obtained by subtracting the values in Columns 4 and 6 from those in Column 5, and assuming a 2200 m/s cross-section for the ^{10}B(n,α) reaction of 3838 barns. Burnup of ^{10}B during the MURR irradiation, which amounted to 2.2 %, was accounted for in the calculations.

Columns 8 and 9 gives an estimate of the original (unirradiated) natural boron content in each of the samples, and the estimate overall uncertainty. These values were determined from the calculated as-received ^{10}B contents in Column 7, corrected for burnup of ^{10}B during the previous plant exposure, and assuming a ^{10}B isotopic content of 19.9 atom %. Calculated burnup levels for the previous plant exposure, based on the thermal neutron fluences in Column 2 calculated at PNNL from the radiometric analyses, ranged from ~0.3 % to ~0.7 %.

Table A-5
Calculated Boron Contents in Steel Samples

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A.1.8 Uncertainty in Boron Determinations

Error sources in the determination of the boron contents are given in Table A-6. The largest error source is the variability in the measured pre- and post-irradiation helium concentrations. Variability in the measured helium concentrations is attributed largely to heterogeneity in the boron contents for the milligram and sub-milligram sample masses analyzed. Total uncertainty estimates, determined from the quadrature sum of the various uncertainty components, are given in the last column of Table A-6.

Table A-6
Error Sources for Boron Determinations

Parameter	Estimated Error in Parameter (1σ)	Resultant Error in Boron Content (%)[a]
Measured		
He concentrations[b]	0.5 % - 8 %	1 – 7
Neutron fluence (MURR irradiation)	2.2%	2.2
Nickel content (wt. %)	0.5 wt. %	<0.1
Calculated		
He generation from nickel component	10%	<0.1
Prior thermal fluence (n/cm^2)	10%	<0.1

[a]Standard error range in calculated boron contents resulting from errors in the corresponding parameter.

[b]Standard deviations in replicate analyses (pre- and post-exposure combined).

Uncertainty in the nickel content of the stainless steel material is given as ±0.5 wt. % based on the x-ray fluorescence analyses performed as part of the radiometric analyses. Uncertainty

in the calculated plant thermal neutron fluence values was arbitrarily assigned as ±10 %. For the present measurements, however, uncertainty in the plant thermal fluence has a negligible effect on the final calculated boron contents.

A.1.9 Discussion of Results

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A.2 Retrospective Neutron Dosimetry

A.2.1 Summary

Four scraping samples were obtained from the jet pump riser brace pads at positions 1, 3, 4, and 5 around the pressure vessel of the reactor and delivered to PNNL for analysis. Each sample consisted of multiple, small stainless steel turnings. Sample 1 showed some contamination from residual material in the sampler. This residual material had little contamination and did not appear to affect our analyses. Selected subsamples from each group of turnings were initially gamma counted to determine the residual activation products. Subsamples were also analyzed for the beta-emitters ^{55}Fe and ^{63}Ni. The chemical composition of each steel sample was measured by x-ray fluorescence. All of the measured activities were then converted to saturated activation rates by taking into consideration the reactor power history. Thermal and fast neutron fluences were finally determined at each position using spectral-averaged cross sections determined for a typical BWR pressure vessel spectrum. The neutron fluences were first determined using the total reactor power history. However, this analysis showed that shorter-lived reactions predicted lower neutron fluences than the longer-lived reactions. This effect appears to be readily explained by changes to the reactor core configuration that were made to reduce neutron leakage, thereby reducing the neutron fluxes at the locations that were sampled. The analyses and results are detailed below.

A.2.2 Gamma Energy Analysis

Selected subsamples were weighed and then gamma counted in duplicate using high efficiency intrinsic germanium detectors following procedure PNL-ALO-450. The detectors are calibrated relative to NIST and other accepted standards, and control counts are performed daily to ensure continuing energy and efficiency calibrations. All four of the samples showed the presence of ^{51}Cr, ^{54}Mn, ^{57}Co, ^{58}Co, ^{59}Fe, ^{60}Co, and ^{65}Zn. The measured activities and total propagated uncertainties in microCuries/milligram are listed in Table A-7. All of the activities were corrected for decay to the date of the reactor shutdown at the end of cycle 19 on January 6, 2000. The samples were removed from the reactor between January 15 and 18, 2000; however, it is assumed that they did not see any significant neutron exposure following the reactor shutdown.

The ^{51}Cr, ^{59}Fe, ^{60}Co, and ^{65}Zn activities are due to thermal neutron activation of Cr, Fe, and Co, Cu, and Zn impurities in the samples. The other activities are produced by the fast neutron reactions $^{54}Fe(n,p)^{54}Mn$, $^{58}Ni(n,p)^{58}Co$, and $^{58}Ni(n,2n+np)^{57}Co$. Multiple counts on independent subsamples were averaged to determine the values in Table A-8. The uncertainties include both the counting statistics and scatter in the duplicate counts. The ^{60}Co value for position 1 showed considerable scatter, possibly due to the contamination of this sample by the residual material in the sampling mechanism. ^{60}Co is also produced by two other reactions, namely $^{60}Ni(n,p)^{60}Co$ and from the decay of ^{59}Fe to ^{59}Co. However, calculations indicate that both of these reactions make a negligible (< 0.7%) contribution to the ^{60} Co that is produced from the ^{59}Co impurity in the steel.

A.2.3 ^{55}Fe Measurements

Subsamples measuring about 1.5 mg were dissolved in heated HCl and brought to a known volume in distilled water. Aliquots were taken and spiked with 2-mg Fe carrier to be used as a yield monitor. The iron fraction was purified using extraction chromatography following procedure PNL-ALO-435. The sample matrix was converted to 8M HNO_3 and loaded onto a TRU Resin column. Iron was retained on the column, impurities were eluted with 8M HNO_3. The iron was then eluted with 2M HNO_3. An $Fe(OH)_3$ precipitate was then formed with stirring by adding concentrated NH_4OH. The $Fe(OH)_3$ precipitate was loaded onto a Tuffryn® filter, dried, and covered with 6-micron mylar film. The filters were then counted using thin-window germanium detectors (Low Energy Photon Spectrometers or LEPS). Prior tests have demonstrated that x-ray fluorescence is not significant with this mounting scheme. The LEPS detectors were calibrated using ^{55}Fe NIST standards prepared in identical geometries. The sample activities were corrected for decay and x-ray self-absorption was calculated to be negligible. After counting, the $Fe(OH)_3$ was dissolved in a known volume of 3M HNO_3, and an aliquot analyzed by Inductively Coupled Plasma Atomic Emission Spectroscopy (ICP-AES) to determine Fe concentration and thus radiochemical yield. The radiochemical yields averaged 95%. Sample duplicates showed good repeatability and the blank spike and matrix spike averaged 95% yield-corrected recovery. The ^{55}Fe activities and uncertainties are listed in Table A-7. The ^{55}Fe activity is produced by the $^{54}Fe(n,\gamma)$ thermal neutron reaction.

Table A-7
Activity Measurements in µCI/MG

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A.2.4 ^{63}Ni Measurements

Subsamples measuring about 1.5 mg were dissolved in HCl and brought to a known volume in distilled water. Aliquots were taken and spiked with 2-mg Ni carrier to be used as a yield monitor. After a matrix adjustment, the nickel fraction was purified using an on-column nickel dimethyglyoxime precipitation reaction following procedure PNL-ALO-495. Nickel was retained on the column; impurities were eluted with a basic ammonium citrate solution. The nickel was then eluted with 3M HNO_3. The strip solution was carefully evaporated to near dryness, brought to a known volume with 0.1M HCl, and split for radiochemical recovery determination by ICP-AES and ^{63}Ni analysis by liquid scintillation counting following procedure PNL-ALO-474. The chemical yields averaged 99%. A minor correction for ^{60}Co tailing into the ^{63}Ni region was made. Calibrations were performed using ^{63}Ni standards obtained from NIST. Sample duplicates showed good repeatability and the blank spike and matrix spike averaged 101% yield-corrected recovery. The decay-corrected ^{63}Ni activities are listed in Table A-7. The ^{63}Ni activity is produced by the ^{62}Ni(n,γ) thermal neutron reaction.

A.2.5 Calculation of Saturated Activation Rates

Table A-7 lists the measured activities and uncertainties corrected to the end of irradiation time for each sample and reaction. Nuclear decay data were taken from the Table of Radioactive Isotopes, E. Browne, R. Firestone, and V. Shirley, Wiley, 1986. The activities were converted to saturated activation rates by correcting for the decay during irradiation, atomic weight, elemental abundance, isotopic abundance, gamma self-absorption, neutron self-absorption, and nuclear burnup. These factors are shown in Table A-8 and are discussed below.

Table A-8
Correction Factors and Cross Sections for Each Reaction

Reaction	At.Wt	Iso.Abn.	Gabs	History	Cross Sections, Barns	
Thermal					Thermal	**Epi.Cor.**
$^{54}Fe(n,\gamma)^{55}Fe$	55.847	0.058	0.998	1.135	2.25	2.30
$^{62}Ni(n,\gamma)^{63}Ni$	58.69	0.0363	N/A	0.1803	14.5	14.8
$^{58}Fe(n,\gamma)^{59}Fe$	55.847	0.0028	0.993	1.282	1.28	1.35
$^{50}Cr(n,\gamma)^{51}Cr$	51.996	0.0435	0.987	1.261	15.9	16.2
$^{59}Co(n,\gamma)^{60}Co$	58.933	1.0	0.993	1.065	37.2	40.4
Fast					**>.1 MeV**	**>1 MeV**
$^{54}Fe(n,p)^{54}Mn$	55.847	0.059	0.990	1.202	0.1058	0.1846
$^{58}Ni(n,p)^{58}Co$	58.69	0.6808	0.990	1.284	0.1337	0.2333

At.Wt. = atomic weight
Iso. Abn. = isotopic abundance of target (also need elemental abundance in alloy)
Gabs = calculated gamma self-absorption in wires
History = reactor power history correction for decay
Thermal = cross section and fluence derived with 2200 m/s cross section
Epi. Cor. = thermal neutron cross section with epithermal corrections assuming a 53% epithermal flux

The decay during irradiation was determined using the BCF computer code that integrates the decay of each isotope for each period of reactor operation and downtime. Reactor time history as provided by the client for the reactor, is shown in Figure A-2.

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The accuracy of the reactor power history correction factors depend on the halflife of each isotope relative to the details of the irradiation histories. The longer-lived isotopes, such as ^{63}Ni truly integrate over the entire irradiation history. However, shorter-lived isotopes, such as ^{59}Fe (45 day) or ^{58}Co (71 day) are only dependent on the last year or so of reactor operation and the decay corrections are thus more uncertain since they depend critically on the fine details of the irradiation history. For this reason, neutron fluence results are more accurate for the longer-lived isotopes. In an attempt to improve the reliability of the shorter-lived isotopes, fine details of the power history from cycle 19 were used in the calculations. For the other cycles, the average power level was used for each cycle, as shown in Figure A-2.

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Figure A-2
Power History. Cycle 19 is Expanded to Show Fine Detail

Estimated gamma self-absorption corrections varied from 1-2% for the direct counting of the turnings. As mentioned above, there were no significant x-ray absorption or fluorescence effects for ^{55}Fe. Neutron self-absorption corrections were not performed since we do not have knowledge of the geometry of the individual turnings during reactor operation. It is assumed that the samples were removed from the surface of the pressure vessel. In this case, the neutron self-absorption corrections would be negligible for stainless steel. Due to the relatively low neutron fluence levels, nuclear burnup and transmutation corrections for both target and product isotopes were negligible.

Examination of the neutron fluences derived later from the data in Table A-9 showed a trend with the halflife of the reaction product. This effect was most likely due to efforts by the utility to reduce the neutron leakage in order to improve plant efficiency. The net effect of these changes was that the neutron flux at a specified position on the JPRB's might not depend linearly on the reactor thermal power. Corrections to account for these changes are discussed below.

A.2.6 X-Ray Fluorescence Measurements

The steel samples have the nominal composition of Type-308 stainless steel. In order to determine the exact composition of each sample more accurately, subsamples were analyzed by energy dispersive x-ray fluorescence (EDXRF) techniques at KLM Analytical in Richland, WA. In this procedure, samples are bombarded with an electron beam and the x-rays are measured with a high-resolution detector. Since the samples were radioactive, x-ray spectra were also taken with the beam off and this background was subtracted from each EDXRF x-ray spectrum. NIST 304 stainless steel standards were analyzed at the same time to calibrate and verify the performance of the equipment and interelement corrections. The results of the x-ray measurements are listed in Table A-9.

Table A-9
EDXPF Analyses of the JPRB Steel Samples

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The samples were also analyzed by wavelength dispersive x-ray fluorescence to determine the level of the Co impurity. The results were inconclusive for samples 1 and 3 due to the small amount of available material in these samples. However, good results were obtained for samples 4 and 5. We also calculated the Co impurity concentration from the boron measurement in the Missouri University Research Reactor, as described in the accompanying helium and boron report. The Co concentration can be directly calculated from a gamma energy analysis of the

irradiated samples using a neutron activation technique, including subtraction of the initial ^{60}Co activity. These results are also shown on Table A-9. The Co concentrations for positions 4 and 5 are in excellent agreement with the WDXRF measurements so that we have confidence in these values for positions 1 and 3 where the WDXRF values have high uncertainties.

A.2.7 Neutron Fluence Evaluations

Saturated activities are equal to the integral over neutron energy of the neutron activation cross section times the neutron flux spectrum. In principle, the neutron flux spectra can be determined using neutronics calculations; however, such information was not provided. An estimate of the thermal neutron fluence can be obtained by dividing the saturated activities by the 2200 m/s thermal neutron cross section and multiplying by the total irradiation times using cross section data obtained from the National Nuclear Data Center at Brookhaven National Laboratory (values are listed in Table A-8). However, a better estimate of the thermal neutron fluence can be obtained by correcting for the presence of epithermal neutrons. A simple way to do this is to set each saturated activity equal to the sum of the thermal flux times the thermal cross section plus an epithermal flux times the resonance integral. This correction is also listed in Table A-8 and is shown as a corrected thermal neutron cross section. More accurate estimates of the thermal and epithermal fluences require a calculation of the neutron flux spectrum at each position rather than using a generic BWR neutron spectrum. For the present analyses, a neutron energy spectrum obtained previously from General Electric for a surveillance capsule position for a GE BWR reactor was used.

The neutron fluences from the calculated saturated activities and cross sections are listed in Table A-10. As was discussed above, the values from the various thermal neutron reactions show a trend with halflife. Due to the short halflifes of ^{59}Fe and ^{51}Cr, results from these reactions are not as reliable as those from the other reactions and the thermal fluences were determined by averaging fluences from the longer-lived reactions. The standard deviations on the three thermal neutron measurements range from 8 to 17%.

The gamma counting data can also be used to estimate the fast neutron fluence. In order to accurately determine the fast fluence above 1 MeV, or any other energy threshold, it is necessary to know the energy dependence of the neutron flux spectrum. These spectral-averaged activation cross sections were calculated from the GE BWR neutron spectrum mentioned above and they are listed in Table A-8. The standard deviations in the averages of the two fast neutron reactions range from 13 to 30%. However, absolute uncertainties may be somewhat larger due to the uncertainty in the choice of the spectral-averaged cross sections. The fast fluences derived from ^{54}Mn (312 d) are consistently higher than those from ^{58}Co (71 d) due to the flux non-linearity problem with the reactor power history, as discussed above. The difference between the ^{54}Mn and ^{58}Co derived fast fluences also depends on the spectral-averaged cross sections. As was the case for the thermal and epithermal fluences discussed above, more accurate determinations would require detailed neutron flux spectra calculations for each of the four JPRB positions.
The fluence ratios from Fe/Ni are very consistent at about 1.21 for positions 1, 4, and 5. However, the Fe/Ni ratio at position 3 is 1.54. This was checked by triplicate gamma counts of separate turnings from sample 3 and all three results agreed within ±1%. This may be due to a difference in the neutron energy spectrum at this location. Table A-10 lists the fast neutron fluences for thresholds of 0.1 and 1.0 MeV at each of the four positions. These values were calculated by averaging the Fe and Ni results.

Table A-10
Neutron Fluences

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A.2.8 Edge Fuel Bundle Power History Corrections

As was discussed above, the various thermal neutron fluence estimates from different reactions appear to show a trend with the reaction product halflife, indicative of a potential problem with the reactor power history corrections for each reaction. Additional information was received from the plant operator indicating that changes had been made to the reactor fuel configuration in order to reduce the neutron leakage, thereby lowering the flux on the JPRB positions. The average edge power peaking factor for the reactor ran at about 60% of the core average for the first 11 fuel cycles, dropping to about 40% for the last 7 fuel cycles. These factors were used to determine a core edge power history, as shown in Figure A-4. This modified reactor power history was then used to determine revised saturation factors for each of the reactions. The resultant neutron fluence values are shown in Table A-11. The revised saturated activity values are given in Table A-12. As can be seen, the previous trend with reaction product halflife is

virtually eliminated and all of the thermal reactions give consistent results with much lower standard deviations for the average fluences than shown previously in Table A-10. It is important to note that the thermal neutron fluences derived from the longest-lived reactions are not affected very much by these changes and that the new average thermal fluences are in excellent agreement with the longest-lived values.

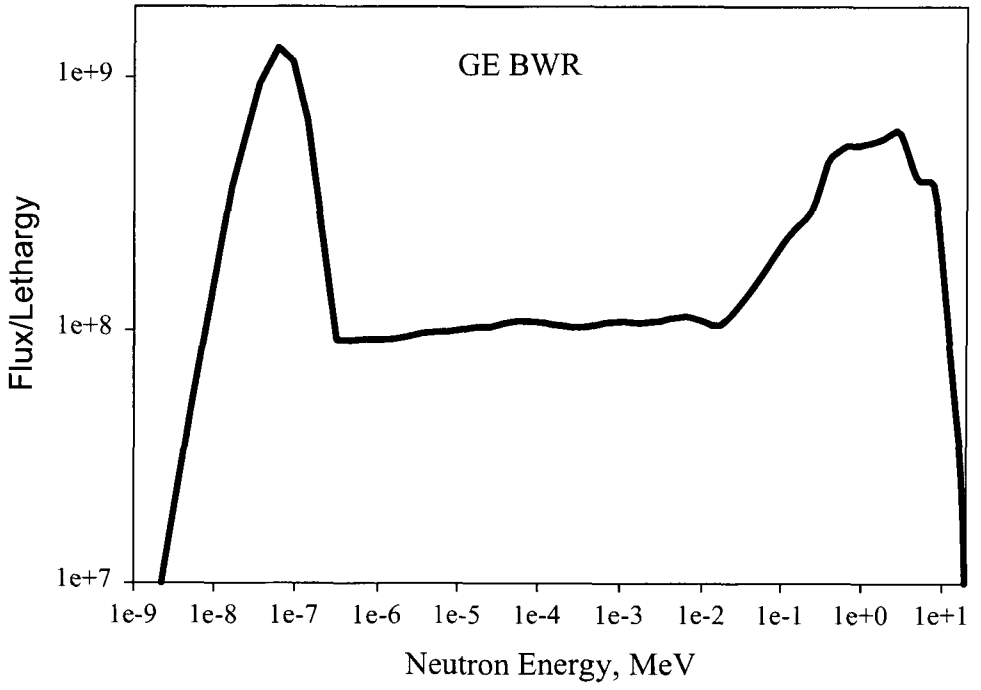

Figure A-3
Neutron Flux Spectrum for a Typical GE BWR Pressure Vessel Location

The fast neutron fluences are derived from two reactions that are both relatively short-lived (312 day and 71 day). Hence, using the revised core edge power history significantly changes both reaction rates. A comparison of Tables A-10 and A-11 shows that the fast neutron fluences are about 30% higher using the core edge power history. This difference is to be expected since the total reactor power history generally shows increased power as a function of time whereas the core edge power history shows a distinct drop around cycle 12 due to the efforts to reduce neutron leakage. Hence, the reaction correction factors for shorter halflives are about 30% higher for the core edge power history, as was seen with the thermal neutron reactions.

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Figure A-4
Reactor Power History using Edge Fuel Bundle Peaking Factors

Position 3 appears to show more scatter in the neutron fluences than the other positions for reasons that are not understood. Multiple gamma counts from separate subsamples showed excellent agreement, confirming that the differences are real and not due to sample inhomogeniety or contamination.

A.2.9 Conclusions

The thermal and fast neutron fluences have been determined with uncertainties ranging from 6 to 29% at each of the four JPRB positions. The data clearly show the effects of the decrease in the core edge neutron fluxes caused by the changes in the core configuration to reduce the neutron leakage and the fluences in Table A-11 are thus recommended. The absolute accuracy of the fast neutron fluences might be significantly improved if neutron spectral calculations were performed at each of the four JPRB positions, taking into account the core configurations over the lifetime of the reactor. Such calculations would provide more reliable estimates of the spectral-averaged cross sections for the Fe and Ni reactions, hopefully improving the agreement between these independent measurements of the fast neutron fluences. However, this would require a considerable additional effort that might only provide a marginal improvement in the fluence determinations. The thermal neutron fluences from the different reactions are in good agreement and the absolute accuracy should be close to the standard deviations quoted in Table A-11.

Table A-11
Revised Neutron Fluences Using Core Edge Power History

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Table A-12
Saturated Activation Rates in Atom/Atom-Second (Corrected for Core Edge Peak Bundle Factors)

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B
SAMPLE MEASUREMENT RESULTS–PLANT 2

B.1 Helium and Boron Analyses of Plant 2 Weld Pad Samples

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B.1.1 Summary

Results of helium and boron analyses on four jet pump riser brace (JPRB) weld pad samples from Plant 2 are reported.

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The observed variability in the boron content of the four samples is larger than can be explained by the estimated measurement uncertainties, suggesting that the variability between the various riser pad locations is real. This could perhaps be due to some mix in the samples of rider pad and associated weld material as a result of the mechanical sampling operation.

B.1.2 Analysis Samples

Four sets of Inconel 182 weld pad samples were received on October 19, 2000 for helium, boron, and activation analysis. All of the samples had been taken remotely from jet pump riser weld pads from a commercial reactor (Plant 2) in early October 2000, and were individually contained in stainless steel filter assemblies. Each of the samples were in the form of multiple small machine scrapings.

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B.1.3 Sample Preparation

Following receipt, each of the filter assemblies was disassembled in a radioactive materials fume hood. Removal of the scraping material was accomplished by flushing the filter element and internal sections of the filter with alcohol into a gravity fed filter paper/funnel assembly. After air drying, the contents of the filter paper were collected into a small glass vial.

Samples for helium and boron analyses were then taken from a subset of the collected material. For the initial helium analyses, single discrete metal pieces were selected. The pieces were chosen with the aid of a low-power stereo microscope to be as representative of the each sample set as possible. Pieces that had features not representative of the lot as a whole were not used for the analyses.

For the boron determinations, a single large scraping piece was used when available. Otherwise, multiple scrapings were selected as described above. Each selected piece (or pieces) was wrapped in aluminum foil for neutron exposure at the McMaster University Nuclear Reactor (MNR). Following the MNR exposure, single specimens were either cut or selected from the set for additional helium analysis to determine the boron content. Details of the neutron exposure are given in Section D.

Prior to analysis, each specimen was cleaned in acetone, air-dried, and then weighed using a microbalance with calibration traceable to the National Institute for Standards and Technology (NIST). Specimen masses ranged from ~0.5 to ~3 mg each. Mass uncertainty is conservatively estimated to be ±0.002 mg.

B.1.4 Neutron Exposure

Determination of the boron content in the samples was made by first irradiating a selected subset of each sample material in a thermalized neutron field in the MNR and then measuring the increase in the samples helium content. In a thermalized neutron field, helium generation is predominately through (n,α) reactions with ^{6}Li or ^{10}B, or with Ni through the two-stage reaction ^{58}Ni(n,γ)^{59}Ni(n,α). Lithium is assumed to be an improbable impurity in steels due to its relatively high volatility. If the materials have already been exposed in a neutron environment, as was the case here, corrections are also applied to determine the original level of boron in the material.

The boron characterization samples and the dosimetry samples were packaged together inside an aluminum can provided by McMaster University. The samples were contained inside a small aluminum capsule, wrapped with aluminum foil, and held in the approximate axial center of the can. The aluminum capsule and foil were fabricated from Type 1100 aluminum.

Figure B-1 shows a diagram of the sample loading arrangement inside the can. Table B-1 lists the individual samples in the irradiation capsule. The aluminum can was irradiated at the MNR for a period of approximately 27 hours. Details of the irradiation are given in Table B-2.

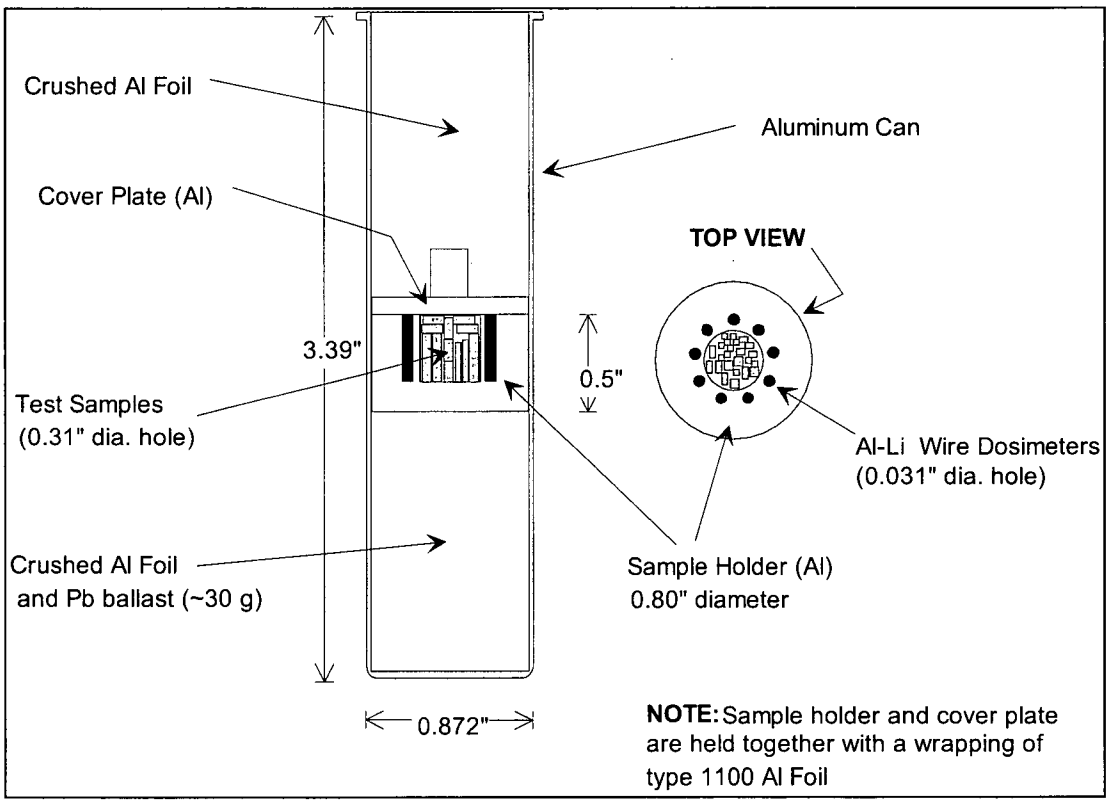

Figure B-1
MNR Irradiation Assembly

Table B-1
Weld Pad and Dosimetry Samples Irradiated in the MNR

Sample Name	Material	No.[a]	Mass (mg)	Sample Description
H-3	Inconel 182	1	8.6	Machine to-ol scraping(s)
H-5	Inconel 182	1	18.1	Machine tool scraping(s)
H-13	Inconel 182	1	8.0	Machine tool scraping(s)
H-15	Inconel 182	1	15.6	Machine tool scraping(s)
AlLi-1 to -4	Al-0.7 wt.%^{6}Li alloy	4	12.2	Bare wire, 0.5 mm dia. x ~6 mm long
V-1	Vanadium (Lot 7)	1	26.4	Bare wire, 0.9 mm dia. x ~6 mm long
Ni-1	Nickel (Lot 4)	1	27.1	Bare wire, 1.0 mm dia. x ~4 mm long
Fe-1	Iron (Lot 11A)	1	24.7	Bare wire, 0.8 mm dia. x ~6 mm long

[a]Number of specimens of sample included in irradiation capsule. The weld pad samples consisted of single or multiple scrapings wrapped in aluminum foil (see text).

Table B-2
Summary of MNR Irradiation

Parameter	Value
Irradiation Start Date	November 8, 2000
Irradiation Location	5C
Irradiation Temperature (°C)	33 (nominal)
Irradiation Time (hours)	27.3
Thermal Neutron Fluence (n/cm^2)[a]	4.10×10^{18}
Fast Neutron Fluence (n/cm^2)[b]	8×10^{17}
^{10}B Burnup (%)	1.6

[a]Thermal neutron fluence determined from the measured helium generation in the Al-Li alloy samples (see text).

[b]Fast neutron fluence (>1 MeV) in 5C location from activation data.

Multiple samples of Al-Li alloy wire [1] were included in the irradiation capsule in order to characterize the thermal neutron fluence. This alloy was originally fabricated by the Central Bureau for Nuclear Measurements (CBNM) in Belgium specifically for neutron dosimetry applications. The stated CBNM composition of the Al-Li alloy is 0.705±0.025 wt.% Li, with a lithium isotopic composition of 95.82±0.08 at.% ^{6}Li. Independent analysis at Argonne National Laboratory showed a lithium content of 0.730±0.004 wt.%, and an isotopic composition of 95.67±0.04 at.% ^{6}Li; essentially in agreement with the CBNM values but with lower uncertainties. For analysis purposes, a composition of Al-0.73±0.01 wt.% Li, with a ^{6}Li content of 95.7±0.1 at.% is assumed.

Samples of iron and nickel were also included to characterize the fast neutron contribution to the helium generation in the samples, as the fast neutron component in the MNR was expected to be somewhat higher than that in the University of Missouri Research Reactor (MURR) that was used for Plant 1. MURR was not available for the present work, and will likely not be available for Plant 3 due to changing programmatic commitments at MURR. Boron contents in the iron and nickel have previously been determined to be ~0.005 and ~0.004 wt. ppm respectively, and are thus negligible.

Also included as a control material for the boron determinations was a sample of PNNL's Lot 7 vanadium material. This material is used for the fabrication of Helium Accumulation Fluence Monitor (HAFM) capsules and radiometric monitors, and has been characterized for boron content in a variety of irradiation environments yielding a current average value of 4.5 ± 0.2 wt. ppm.

B.1.5 Helium Analysis Procedure

Helium analyses on all specimens were conducted by isotope-dilution gas mass spectrometry following vaporization in a resistance-heated tungsten-wire or graphite crucible in one of the mass spectrometer system's high-temperature vacuum furnaces [2]. The absolute amount of ^{4}He released was measured relative to a known quantity of added ^{3}He "spike." The ^{3}He spikes were obtained by expanding and partitioning known quantities of gas through a succession of calibrated volumes [3]. The mass spectrometer was calibrated for mass sensitivity during each series of runs by analyzing known mixtures of ^{3}He and ^{4}He, as specified in PNNL Procedure RPG-MSL-1000.

B.1.6 Helium Analysis Results

The results of the helium measurements are given in Table B-3 and B-4, and are listed as total atoms released and as atom concentrations. Table B-3 gives the results of the helium analyses on the Al-Li alloy, Fe, and Ni dosimetry materials. Helium concentrations for the Al-Li alloy are relative to the total number of ^{6}Li atoms in each specimen. For all other materials, the concentrations are relative to the total number of atoms in the samples.

Table B-3
Helium Concentrations in Dosimetry Samples

Specimen	Material	Mass[a] (mg)	Measured ^{4}He (10^{15} atoms)	Helium Concentration (appm)[b]
AlLi-1-A	Al-Li Alloy	0.722	1.809	3609
AlLi-2-A	"	0.689	1.744	3646
AlLi-3-A	"	0.748	1.889	3638
Fe-1	Pure iron	1.984	0.0000058	0.00027
Ni-1	Pure nickel	3.147	0.000206	0.00638

[a]Mass of analyzed specimen. Mass uncertainty is ±0.002 mg.

[b]Helium concentration in atomic parts per million (10^{-6} atom fraction) with respect to the total number atoms in the specimen. For the Al-Li, the concentration is relative to the number of ^{6}Li atoms.

Table B-4 gives the helium concentrations measured in the characterization samples both before and after neutron exposure at the MNR. Residual (pre-exposure) helium concentration in the PNNL vanadium, iron and nickel has been determined previously to be <0.03 appb.

In Tables B-3 and B-4, conversion from atoms to concentration was based on the following calculated values:

Al-Li alloy.....0.06942 x 10^{22} atoms ^{6}Li/gram
Iron.....1.078 x 10^{22} atoms/gram
Nickel.....1.026 x 10^{22} atoms/gram
Inconel 182.....1.05 x 10^{22} atoms/gram
Vanadium.....1.182 x 10^{22} atoms/gram

The value for the Inconel 182 material is based on an average composition determined from the x-ray fluorescence analyses (XRF) analyses conducted for the separate radiometric analyses (see Attachment B). It should be noted that these conversion factors, and the helium concentrations obtained using it, are not very sensitive to small changes in material composition.

Helium concentrations were measured in three of the four Al-Li alloy wires. From Table B-3, the mean helium concentration (in terms of atom fraction) is 0.003631 ± 0.000019. Correcting for burnup of the ^{6}Li isotope and for neutron self-shielding yields a value of 0.003860 ± 0.000020, which represents the ^{6}Li reaction probability $\sigma\varphi t$. Burnup corrections were made using the standard exponential formulation. Self-shielding corrections were based on transport theory approximations from Case et al.[4], and were 6.1%. Uncertainty in the self-shielding correction is conservatively estimated to be ~25% of the correction factor. Assuming a 2200 m/s ^{6}Li(n,α)^{3}H neutron cross section of 941 barns, yields an effective thermal neutron fluence of 4.10 x 10^{18} n/cm^{2}.

Table B-4
Measured Helium Concentrations in Weld Pad Samples

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Absolute uncertainty in the measured helium concentrations is estimated to be ~1 % (1σ). This uncertainty results from the cumulative uncertainties in the sample mass, the isotope ratio measurement, and the spike size.

B.1.7 Boron Determinations

Calculated boron contents in the samples are given in Table B-5. Column 6 gives the measured helium contribution from the nickel component in the as-received material. As mentioned earlier, the nickel content in the Inconel 182 material was determined from the separate XRF analyses. Corrections for burnup of ^{58}Ni and in growth of ^{59}Ni from the material's prior reactor exposure are negligible.

The nickel contributions in Table B-5 are 4.6 appb, or about 4% of the total incremental helium generation. This value was calculated from the measured helium generation in the pure nickel material included in the MNR irradiation, adjusted for the measured nickel content in the Inconel 182 material. Most of this helium is from the threshold neutron (>1 MeV) reaction ^{58}Ni(n,He). Helium contributions from other threshold neutron reactions were calculated to be <1 %, and were neglected. This is confirmed by the very low observed helium generation in the iron of ~0.3 appb.

Table B-5
Calculated Boron Contents in Steel Samples

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The Column 7 values in Table B-5 are the calculated ^{10}B contents in the as-received samples *prior* to the MNR exposure. These values were calculated from the net increase in the helium content attributable to the boron, obtained by subtracting the values in Columns 4 and 6 from those in Column 5, and assuming a 2200 m/s cross-section for the ^{10}B(n,α) reaction of 3838 barns. Burnup of ^{10}B during the MNR irradiation, which amounted to 1.6 %, was accounted for in the calculations.

Columns 8 and 9 gives an estimate of the original (unirradiated) natural boron content in each of the samples, and the estimate overall uncertainty. These values were determined from the calculated as-received ^{10}B contents in Column 7, corrected for burnup of ^{10}B during the previous plant exposure, and assuming a ^{10}B isotopic content of 19.9 atom %. Calculated burnup levels for the plant exposure, based on the thermal neutron fluences in Column 2 calculated at PNNL from the radiometric analyses, ranged from ~0.2 % to ~0.4 %.

B.1.8 Uncertainty in Boron Determinations

Error sources in the determination of the boron contents are given in Table B-6. The largest error source is the variability in the measured pre- and post-irradiation helium concentrations. Total uncertainty estimates, determined from the quadrature sum of the various uncertainty components, are given in the last column of Table B-5.

Table B-6
Error Sources for Boron Determinations

Parameter	Estimated Error in Parameter (1σ)	Resultant Error in Boron Content (%)[a]
Measured		
He concentrations[b]	1 % - 3 %	1 – 2
Neutron fluence (MNR irradiation)	1.5%	1.5
Nickel content (wt. %)	1.0 wt. %	<0.1
He generation from nickel component	1%	<0.1
Calculated		
Prior thermal fluence (n/cm^2)	10%	0.1

[a]Standard error range in calculated boron contents resulting from errors in the corresponding parameter.

[b]Standard deviations in replicate analyses (pre- and post-exposure combined).

Uncertainty in the nickel content of the Inconel material is given as ±1.0 wt. % based on the XRF analyses. Uncertainty in the calculated plant thermal neutron fluence values were arbitrarily assigned as ±10 %. Uncertainty in the nickel contribution to the helium generation, and in the plant thermal fluence, have a negligible effect on the final calculated boron contents.

B.1.9 Discussion of Results

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B.2 References

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2. H. Farrar and B. M. Oliver, "A Mass Spectrometer System to Determine Very Low Levels of Helium in Small Solid and Liquid Samples," J. Vac. Sci. Technol. A, **4**, 1740 (1986).

3. B. M. Oliver, J. G. Bradley, and H. Farrar, "Helium Concentration in the Earth's Lower Atmosphere," Geochim. et Cosmochim. Acta **48**, 1759 (1984).
4. K. M. Case, F. de Hofmann and G. Placzek, "Introduction to the Theory of Neutron Diffusion", Los Alamos Scientific Laboratory, Los Alamos, New Mexico, June 1953.

B.3 Retrospective Neutron Dosimetry for the Plant 2 Reactor

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B.3.1 Summary

Four scraping samples were obtained from the jet pump riser brace pads at positions 3, 5, 13, and 15 around the pressure vessel of the Plant 2 reactor and delivered to PNNL for analysis. Each sample consisted of multiple, small inconel turnings. Selected subsamples from each group of turnings were initially gamma counted to determine the residual activation products. Radiochemical separations were subsequently performed to measure the ^{55}Fe, ^{63}Ni, and ^{93m}Nb activities. The chemical composition of each inconel sample was measured by x-ray fluorescence. All of the measured activities were then converted to saturated activation rates by taking into consideration the reactor power history. Thermal and fast neutron fluences were then determined at each position using spectral-averaged cross sections calculated for a typical BWR pressure vessel spectrum. Details of the analyses and results are given below.

B.3.2 Gamma Energy Analyses

Selected subsamples were weighed and then gamma counted in duplicate using high efficiency intrinsic germanium detectors following procedure PNL-ALO-450. The detectors are calibrated relative to NIST and other accepted standards, and control counts are performed daily to ensure continuing energy and efficiency calibrations. All four of the samples showed the presence of ^{51}Cr, ^{54}Mn, ^{57}Co, ^{58}Co, ^{59}Fe, ^{60}Co, and ^{65}Zn. The measured activities and total propagated uncertainties in microCuries/milligram are listed in Table B-7. All of the activities were corrected for decay to the date of the reactor shutdown at the end of cycle 19 on September 30, 2000.

The ^{51}Cr, ^{59}Fe, ^{60}Co, and ^{65}Zn activities are due to thermal neutron activation of Cr, Fe, Co, Cu, and Zn impurities in the samples, respectively. The other activities are produced by the fast neutron reactions ^{54}Fe(n,p)^{54}Mn, ^{58}Ni(n,p)^{58}Co, and ^{58}Ni(n,2n+np)^{57}Co. Multiple counts on independent subsamples were averaged to determine the values in Table B-7. The uncertainties include both the counting statistics and scatter in the duplicate counts. ^{60}Co is also produced by two other reactions, namely ^{60}Ni(n,p)^{60}Co and from the decay of ^{59}Fe to ^{59}Co. However, calculations indicate that both of these reactions make a negligible (< 0.7%) contribution to the ^{60}Co that is produced from the ^{59}Co impurity in the steel.

B.3.3 ^{55}Fe and ^{63}Ni Measurements

An accurately weighed amount of each sample (about a milligram) was dissolved in HCl + HNO_3, then taken to exactly 10.0 mL in 6M HCl. Subsamples of this solution were used for ^{55}Fe and ^{63}Ni analysis.

Each subsample was aliquoted into a glass vial with 2 mg of stable iron carrier and 2 mg of stable nickel carrier. The solution was evaporated dry, then dissolved in concentrated HCl. The solution was passed through a strong base anion exchanger (AG MP-1, 50-100 mesh). Iron, cobalt, and many other elements load onto the column while nickel passes through. After the sample was loaded onto the column, the vial was rinsed with concentrated HCl and the rinses were loaded onto the column. Finally, the column was rinsed with clean concentrated HCl.

The column load and wash was saved for nickel analysis and set aside. The column effluent has the nickel, chromium, and part of the manganese, but no iron or cobalt. The column was washed with 6M HCl to strip cobalt. The cobalt fraction was discarded.

Iron was stripped from the column by passing 6M HCl, 0.1M NH_4I through the column. The iodide ion reduces the iron and strips it from the column. The iron fraction was wet-ashed to eliminate the ammonium salts, then dissolved in exactly 10.0 mL of 0.5M HCl. To mount the samples for x-ray counting for ^{55}Fe, 5.00 mL of the iron solution was measured into a plastic centrifuge tube. NH_4OH was added until $Fe(OH)_3$ precipitated. The $Fe(OH)_3$ was sucked down onto a glass fiber filter. The filter was dried under a heat lamp, then mounted on a cardboard back and covered with thin Mylar. The mounts were counted on a Low Energy Photon Spectrometer (LEPS) detector for the 6 keV ^{55}Fe x-ray. The chemical recovery of the iron was measured by ICP-AES analysis of part of the iron product solution. The x-ray counting results for ^{55}Fe were corrected for chemical recovery of the iron.

The nickel solution (previously set aside for nickel analysis) was evaporated dry, then dissolved in dilute HCl. The solution was adjusted to pH 10 with NH_4OH, and then passed through an EiChrom nickel column. The column is made of dimethylglyoxime adsorbed onto an inert plastic powder. Nickel holds up in the column, but most interferences pass through. (The most serious interference is cobalt, which was previously removed by the anion exchange column.) Nickel was then stripped off the column, free of any interference, by adding dilute HNO_3 to the column. The nickel product solution was evaporated dry, wet-ashed to eliminate organics, then dissolved in exactly 10.0 mL of dilute HCl. An aliquot of the solution was mixed with liquid scintillation cocktail for beta counting for ^{63}Ni. The liquid scintillation beta energy spectra showed a peak that matched the spectrum of a ^{63}Ni standard that was used to determine the efficiency. No other beta activities were observed in the beta spectra. The chemical recovery of the nickel was measured by ICP-AES on part of the nickel product.

B.3.4 ^{93m}Nb Measurements

A few milligrams of each sample were dissolved in HF + HCl + HNO_3 in a Teflon beaker. A blank and two reagent-only spikes were prepared with 250 µg of stable Nb, but without Inconel. Sample 15 was prepared in duplicate, and one of the duplicates was spiked with ^{93m}Nb. The sample solutions were taken nearly dry with concentrated HF (to expel most of the HCl and HNO_3), then dissolved in several mL of 10M HF. The sample solutions were then passed through a strong-base anion exchanger (Bio-Rad AG 1-X8, 100-200 mesh, chloride form, converted to

fluoride form just before use). Niobium loads onto the resin, but chromium, manganese, iron, cobalt, nickel, and zinc pass through. The anion exchange columns were washed with clean 10M HF to eliminate other radionuclides, then the niobium was stripped from the columns with a solution of 14% NH_4F, 4% HF. The niobium fractions were warmed and taken nearly dry with HNO_3 to eliminate the ammonium salts, then dissolved in a small volume of concentrated HF.

Each separated niobium fraction was then dried onto a disc of filter paper and mounted in a cardboard holder. The x-ray emission from ^{93m}Nb was then measured using a LEPS (low energy photon spectrometer) detector. The absolute emission rates were determined by direct comparison with a NIST ^{93m}Nb standard. The radiochemical yield, determined by measuring the chemical recovery of two reagent-only spikes as well as a sample spike, averaged 95 ±5%. During the mounting process for the sample from position 15, the Nb fraction ran off the filter paper onto the cardboard mount. Hence, the activity measurement has a slightly higher uncertainty for this sample.

B.3.5 Calculation of Saturated Activation Rates

Table B-7 lists the measured activities and uncertainties corrected to the end of irradiation time for each sample and reaction. Nuclear decay data were taken from the Table of Radioactive Isotopes, E. Browne, R. Firestone, and V. Shirley, Wiley, 1986. The activities were converted to saturated activation rates by correcting for the decay during irradiation, atomic weight, elemental abundance, isotopic abundance, gamma self-absorption, neutron self-absorption, and nuclear burnup. These factors are shown in Table B-8 and are discussed below. Saturated activation rates are shown in Table B-9.

Table B-7
Activity Measurements (μCi/mg) with 1σ Uncertainties for the Plant 2 Reactor

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Table B-8
Correction Factors and Cross Sections for Each Reaction

Reaction	At.Wt	Iso.Abn.	Gabs	History	Cross Sections, barns	
Thermal					**Thermal**	**Epi.Cor.**
$^{54}Fe(n,\gamma)^{55}Fe$	55.847	0.058	0.998	1.290	2.25	1.0233
$^{62}Ni(n,\gamma)^{63}Ni$	58.69	0.0363	N/A	0.1672	14.5	1.0199
$^{58}Fe(n,\gamma)^{59}Fe$	55.847	0.0028	0.993	1.531	1.28	1.0580
$^{50}Cr(n,\gamma)^{51}Cr$	51.996	0.0435	0.987	1.536	15.9	1.0214
$^{59}Co(n,\gamma)^{60}Co$	58.933	1.0	0.993	1.158	37.2	1.0869
Fast					**>.1 MeV**	**>1 MeV**
$^{93}Nb(n,n')^{93m}Nb$	92.906	1.0	1.0	0.722	0.1240	0.2110
$^{54}Fe(n,p)^{54}Mn$	55.847	0.059	0.990	1.410	0.1058	0.1846
$^{58}Ni(n,p)^{58}Co$	58.69	0.6808	0.990	1.521	0.1337	0.2333

At.Wt. = atomic weight
Iso. Abn. = isotopic abundance of target (also need elemental abundance in alloy)
Gabs = calculated gamma self-absorption in wires
History = reactor power history correction for decay
Thermal = cross section and fluence derived with 2200 m/s cross section
Epi. Cor. = correction to thermal fluence assuming a 53% epithermal flux

Table B-9
Saturated Activation Rates (Atom/Atom-Second) with 1σ Uncertainties. Values are Normalized to a Power Level of 2436 MWt

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The decay during irradiation was determined using the BCF computer code that integrates the decay of each isotope for each period of reactor operation and downtime. Reactor time history as provided by the client for the Plant 1 reactor, is shown in Figure B-2.

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The accuracy of the reactor power history correction factors depend on the halflife of each isotope relative to the details of the irradiation histories. The longer-lived isotopes, such as ^{63}Ni (100 y) and ^{93m}Nb (16.1 y) truly integrate over the entire irradiation history. However, shorter-lived isotopes, such as ^{59}Fe (45 day) or ^{58}Co (71 day) are only dependent on the last year or so of reactor operation and the decay corrections are thus more uncertain since they depend critically on the fine details of the irradiation history. For this reason, neutron fluence results are more accurate for the longer-lived isotopes. In an attempt to improve the reliability of the shorter-lived isotopes, fine details of the power history from cycle 19 were used in the calculations. For the other cycles, the average power level was used for each cycle, as shown in Figure B-2.

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Figure B-2
Power History for the Plant 2 Reactor. Cycle 19 is Expanded to Show Fine Detail. Earlier Cycles only Show the Average Power Level the Entire Cycle

Estimated gamma self-absorption corrections varied from 1-2% for the direct counting of the turnings. As mentioned above, there were no significant x-ray absorption or fluorescence effects for ^{55}Fe. Neutron self-absorption corrections were not performed since we do not have knowledge of the geometry of the individual turnings during reactor operation. It is assumed that the samples were removed from the surface of the jet pump riser brace pad welds. In this case, the neutron self-absorption corrections would be negligible for Inconel. Due to the relatively low neutron fluence levels, nuclear burnup and transmutation corrections for both target and product isotopes were negligible.

B.3.6 X-Ray Fluorescence Measurements

The samples have the nominal composition of Inconel 182, which is "weld butter" for welding inconel. In order to determine the exact composition of each sample more accurately, subsamples were analyzed by energy dispersive x-ray fluorescence (EDXRF) techniques at KLM Analytical in Richland, WA. In this procedure, samples are bombarded with an electron beam and the x-rays are measured with a high-resolution detector. Since the samples were radioactive, x-ray spectra were also taken with the beam off and this background was subtracted from each EDXRF x-ray spectrum. NIST Standard Reference Materials (SRM) were analyzed at the same time to calibrate and verify the performance of the equipment and interelement corrections. The results of the x-ray measurements are listed in Table B-10.

Table B-10
EDXRF Analyses of the JPRB Inconel 182 Samples

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The Co impurities in each sample were determined by neutron activation analysis following the irradiation at the McMaster Reactor that was performed to determine the boron content (see Attachment A). The neutron fluence in the brief irradiation at McMaster was much higher (thermal fluence about 4.E+18 n/cm2) than the integrated thermal fluence at the positions of the jet pump riser brace pads in Plant 2 (see Table B-10). The Co impurity was determined by taking the difference between the ^{60}Co activity measured before and after the McMaster irradiation. The thermal neutron fluence was determined from the activation of ^{59}Fe and ^{51}Cr, which gave excellent agreement (within 5%).

B.3.7 Inteferring Nuclear Reactions from Inconel

Previous retrospective dosimetry for Plant 1 was performed with stainless steel samples which had a composition of about 67% Fe, 11% Ni, and 1.5-2.0% Mn. However, in the present case for Plant 2 the samples are Inconel with a composition of about 69% Ni, 7% Fe, and 7% Mn. This change in composition has a deleterious effect on four of the nuclear reaction products that we normally use for the neutron fluence calculations. Calculations were performed for various reactions using the BWR neutron spectrum obtained from EPRI. The results are discussed below for the four interfering reactions.

The main reaction that we rely on to determine the fast neutron fluence is ^{54}Fe(n,p)^{54}Mn. In stainless steel samples, such as those obtained in Plant 1, the Fe composition is about 67% and Mn is only 1.5%. The contribution to the production of ^{54}Mn from Mn in the steel is thus less than 1%. However, for the Inconel 182 specimens in the present case, the Fe and Mn content is nearly the same at about 7% and we calculate that the Mn will contribute about equally with the Fe to the production of ^{54}Mn. Unfortunately, the ^{55}Mn(n,2n) reaction has a high threshold of about 10 MeV and the neutron spectral calculations are not very accurate above this energy. It would be necessary to perform a spectral calculation at the JPRB positions to obtain high confidence in these results. Looking at the measured data for the ^{54}Mn and ^{58}Co reactions, we calculate that the Mn contributes about 42% of the ^{54}Mn, in reasonable agreement with the calculations based on the EPRI BWR neutron spectrum. The uncertainties are high for this correction so that the ^{54}Fe(n,p)^{54}Mn data are less reliable (about 20% total uncertainty) than the other fast neutron reactions.

Another serious interference due to Inconel is for the ^{54}Fe(n,γ)^{55}Fe reaction. We calculate that the ^{55}Fe production from the ^{58}Ni(n,α)^{55}Fe reaction is about 23% higher than from the ^{54}Fe(n,γ)^{55}Fe reaction. Due to the high uncertainties of this calculation, we have dropped this reaction from the final thermal fluence evaluation.

A much smaller effect occurs for the ^{59}Co(n,γ)^{60}Co reaction, where we calculate that the ^{60}Ni(n,p)^{60}Co reaction contributes about 12.5% of the ^{60}Co activity. Similarly, the ^{62}Ni(n,α)^{59}Fe reaction produces about 3.4% of the ^{59}Fe activity normally attributed to the ^{58}Fe(n,γ)^{59}Fe reaction. In both of these cases, the reaction rates were corrected for these interfering reactions.

B.3.8 Neutron Fluence Evaluations

The corrected saturated activities listed in Table B-9 are quoted in product atom per target atom per second. These values are equal to the integral over neutron energy of the neutron activation cross section times the neutron flux spectrum. In principle, the neutron flux spectra can be determined using neutronics calculations; however, such information was not provided. An estimate of the thermal neutron fluence can be obtained by dividing the saturated activities by the 2200 m/s thermal neutron cross section and multiplying by the total FPD values, as is done in Table B-11, using cross section data obtained from the National Nuclear Data Center at Brookhaven National Laboratory (values are listed in Table B-8). A better estimate of the thermal neutron fluence can be obtained by correcting for the presence of epithermal neutrons. A simple way to do this is to set each saturated activity equal to the sum of the thermal flux times the thermal cross section plus an epithermal flux times the resonance integral. This correction is also listed in Table B-8. More accurate estimates of the thermal and epithermal fluences require a calculation of the neutron flux spectrum at each position rather than using a generic BWR neutron spectrum. For the present analyses, a neutron energy spectrum obtained from EPRI for the pressure vessel of a BWR reactor was used. This spectrum also agrees well with a spectrum previously obtained from General Electric, as shown in Figure B-3.

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Figure B-3
Neutron Flux Spectrum for a Typical BWR Pressure Vessel Location as Provided by General Electric (GE) and the Electric Power Research Institute (EPRI)

The final neutron fluences are listed in Table B-11. Four of the reaction rates have been corrected for interfering reactions as noted in the previous sections and the footnotes to the tables. As can be seen, the values from the various thermal neutron reactions are in reasonable agreement. In order to accurately determine the fast fluence above 1 MeV, or any other energy threshold, it is necessary to know the energy dependence of the neutron flux spectrum. These values were calculated from the GE BWR neutron spectrum mentioned above and they are listed in Table B-7. The absolute uncertainties may be somewhat larger due to the uncertainty in the choice of these spectral-averaged cross sections.

As mentioned previously, the uncertainties are relatively high for the corrections to the $^{54}Fe(n,p)^{54}Mn$ reaction and the $^{58}Ni(n,p)^{58}Co$ reaction product is relatively short-lived at 71 days. The $^{93}Nb(n,n')^{93m}Nb$ reaction, which has a halflife of 16.1 years and no problem with interferences from other impurities in the samples, is thus judged to be the best indicator of the fast neutron fluence. Comparing these three reactions at each reactor position shows that the results are in remarkably good agreement, considering the problems discussed above.

Table B-11
Neutron Fluences for the Plant 1 Reactor Neutrons/cm^2 x 10^{17}

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B.3.9 Conclusions

The thermal and fast neutron fluences have been determined with uncertainties of 10% at each of the four JPRB positions for Plant 2. The absolute accuracy of the fast neutron fluences might be significantly improved if neutron spectral calculations were performed at each of the four JPRB positions, taking into account the core configurations over the lifetime of the reactor. Such calculations would provide more reliable estimates of the spectral-averaged cross sections for the Nb, Fe, and Ni reactions. However, this would require a considerable additional effort that might only provide a marginal improvement in the fluence determinations.

For future retrospective dosimetry with Inconel samples, we would recommend not measuring the $^{54}Fe(n,\gamma)^{55}Fe$ reaction due to the strongly interfering reaction.

C
SAMPLE MEASUREMENT RESULTS–PLANT 3

C.1 Helium and Boron Analyses of Plant 3 Weld Pad Samples

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C.1.1 Summary

Results of helium and boron analyses on four jet pump riser brace (JPRB) weld pad samples from Plant 3 are reported.

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C.1.2 Analysis Samples

Four sets of 308L stainless steel weld pad samples were received on May 15, 2001 for helium, boron, and activation analysis. All of the samples had been taken remotely from jet pump riser weld pads from a commercial reactor (Plant 3) in April 2001, and were individually contained in stainless steel filter assemblies. Each of the samples were in the form of multiple small drill turnings.

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C.1.3 Sample Preparation

Following receipt, each of the filter assemblies was disassembled in a radioactive materials fume hood. Removal of the sample material was accomplished by flushing the filter element and internal sections of the filter with alcohol into a gravity fed filter paper/funnel assembly. After air drying, the contents of each filter paper were collected into a small glass vial. Following the collection, each sample was divided into three groups. The first group, which consisted of a single large drill turning, was to be used for the gamma counting; the second group, which consisted of multiple smaller drill turnings, was to be used for the supplemental neutron irradiation (discussed in more detail below); and the third group, consisted of all remaining sample material.

Initially, gamma counts were conducted on the first two groups of material. The results of these initial counts, however, showed some variability in three of the four samples, DA-3, DA-11, and DA-13. As a result, counts were made on additional subsections of each sample from all three groups. As these additional counts continued to show heterogeneity in the three samples, it was decided to group the sample material into as nearly homogeneous groups as possible, and to use only sub-samples from these groups for the subsequent helium, boron, and gamma analyses. The rationale for this grouping is discussed in Attachment B.

Samples for helium and boron analyses were taken from the selected groups discussed above. For the initial helium analyses, single discrete metal pieces were selected. The pieces were chosen with the aid of a low-power stereo microscope to be as representative of the each sample set as possible. Pieces that had features not representative of the lot as a whole were not used for the analyses.

For the boron determinations, a single large scraping piece was used when available. Otherwise, multiple scrapings were selected as described above. Each selected piece (or pieces) was wrapped in aluminum foil for neutron exposure at the McMaster University Nuclear Reactor (MNR). Two groups of material were irradiated for Sample DA-11. The first, DA-11-A, was the original group No. 2 material collected prior to the initial gamma counting. This group showed an activity ~25% higher than the other two groups, and thus was not analyzed further. The second sample, DA-11-B, was a specimen cut from the original group No. 1 material. The "-A" sample was included only for future analysis purposes.

Following the MNR exposure, single specimens were either cut or selected from the set for additional helium analysis to determine the boron content. Prior to analysis, each specimen was cleaned in acetone, air-dried, and then weighed using a microbalance with calibration traceable to the National Institute for Standards and Technology (NIST). Specimen masses ranged from ~0.5 to ~3 mg each. Mass uncertainty is conservatively estimated to be ±0.002 mg. Details of the neutron exposure are given below in Section D.

C.1.4 Neutron Exposure

Determination of the boron content in the samples was made by first irradiating a selected subset of each sample material in a thermalized neutron field in the MNR and then measuring the increase in the samples helium content. In a thermalized neutron field, helium generation is predominately through (n,α) reactions with ^{6}Li or ^{10}B, or with Ni through the two-stage reaction $^{58}Ni(n,\gamma)^{59}Ni(n,\alpha)$. Lithium is assumed to be an improbable impurity in steels due to its relatively high volatility. If the materials have already been exposed in a neutron environment, as was the case here, corrections are also applied to determine the original level of boron in the material.

The boron characterization samples and the dosimetry samples were packaged together inside an aluminum can provided by McMaster University. The samples were contained inside a small aluminum capsule, wrapped with aluminum foil, and held in the approximate axial center of the can. The aluminum capsule and foil were fabricated from Type 1100 aluminum.

Figure C-1 shows a diagram of the sample loading arrangement inside the can. Table C-1 lists the individual samples in the irradiation capsule. The aluminum can was irradiated at the MNR for a period of 27 hours. Details of the irradiation are given in Table C-2.

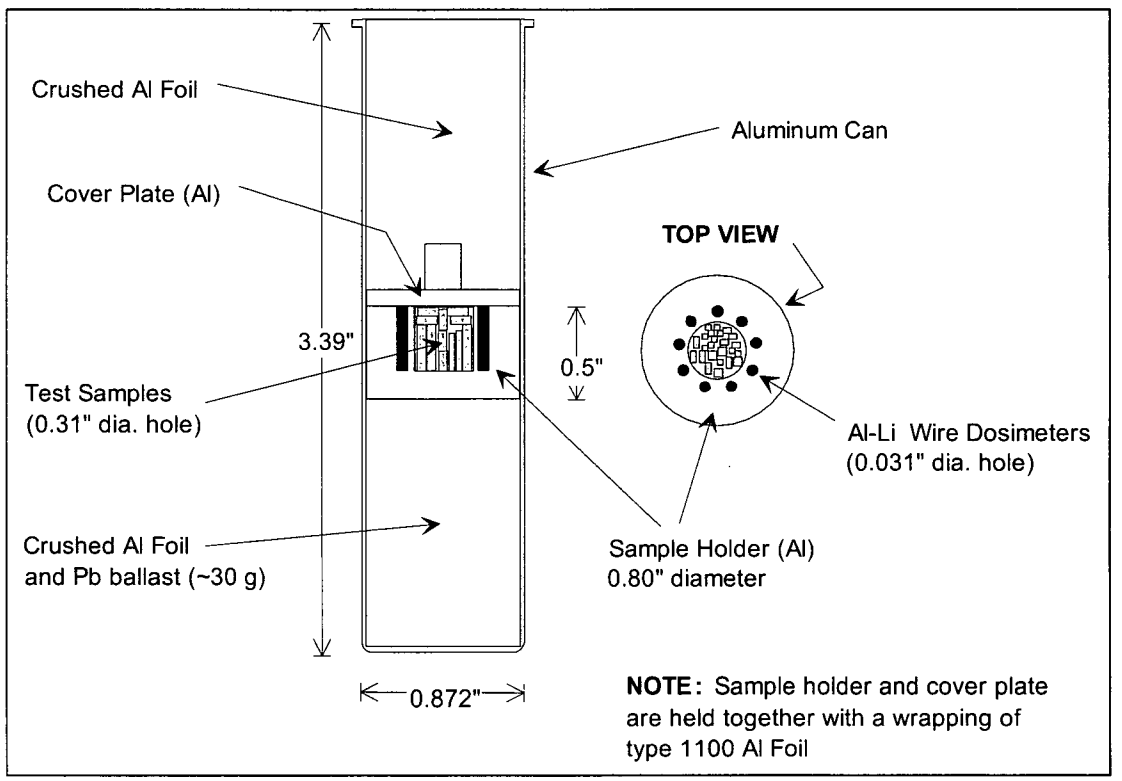

Figure C-1
MNR Irradiation Assembly

Table C-1
Weld Pad and Dosimetry Samples Irradiated in the MNR

Sample Name	Material	No.[a]	Mass (mg)	Sample Description
DA-3	SS	1	5.6	Al wrapped drill turnings
DA-5	SS	1	12.9	Al wrapped drill turnings
DA-11-A	SS	1	16.0	Al wrapped drill turnings
DA-11-B	SS	1	9.1	Al wrapped drill turnings
DA-13	SS	1	10.6	Al wrapped drill turnings
DA-AlLi-1 to -3	Al-0.7 wt.%^{6}Li	3	8.3	Bare wire, 0.5 mm dia. x ~6 mm long
DA-V-1	Vanadium (Lot 7)	1	24.9	Bare wire, 0.9 mm dia. x ~6 mm long
DA-Ni-1	Nickel (Lot 4)	1	25.5	Bare wire, 1.0 mm dia. x ~4 mm long
DA-Fe-1	Iron (Lot 11A)	1	18.6	Bare wire, 0.8 mm dia. x ~5 mm long

[a]Number of specimens of sample included in irradiation capsule. The weld pad samples consisted of single or multiple scrapings wrapped in aluminum foil (see text).

Table C-2
Summary of MNR Irradiation

Parameter	Value
Irradiation Start Date	June 19, 2001
Irradiation Location	5C
Irradiation Temperature (°C)	33 (nominal)
Irradiation Time (hours)	27.0
Thermal Neutron Fluence (n/cm^2)[a]	3.96 x 10^{18}
Fast Neutron Fluence (n/cm^2)[b]	6.7 x 10^{17} (>1 MeV)
^{10}B Burnup (%)	1.5

[a]Thermal neutron fluence determined from the measured helium generation in the Al-Li alloy samples (see text).

[b]Fast neutron fluence (>1 MeV) in 5C location from activation data.

Multiple samples of Al-Li alloy wire [1] were included in the irradiation capsule in order to characterize the thermal neutron fluence. This alloy was originally fabricated by the Central Bureau for Nuclear Measurements (CBNM) in Belgium specifically for neutron dosimetry applications. The stated CBNM composition of the Al-Li alloy is 0.705±0.025 wt.% Li, with a lithium isotopic composition of 95.82±0.08 at.% ^{6}Li. Independent analysis at Argonne National Laboratory showed a lithium content of 0.730±0.004 wt.%, and an isotopic composition of 5.67±0.04 at.% ^{6}Li; essentially in agreement with the CBNM values but with lower uncertainties. For analysis purposes, a composition of Al-0.73±0.01 wt.% Li, with a ^{6}Li content of 95.7±0.1 at.% is assumed.

Samples of iron and nickel were also included to characterize the fast neutron contribution to the helium generation in the samples. Boron contents in the iron and nickel have previously been determined to be ~0.005 and ~0.004 wt. ppm respectively, and are thus negligible.

Also included as a control material for the boron determinations was a sample of PNNL's Lot 7 vanadium material. This material is used for the fabrication of Helium Accumulation Fluence Monitor (HAFM) capsules and radiometric monitors, and has been extensively characterized for boron content in a variety of irradiation environments yielding a current average value of 4.5 ± 0.2 wt. ppm.

Following irradiation, the samples were analyzed for helium content to enable calculation of the boron content. The Plant 3 samples (i.e., drill turnings) were also gamma counted, and the results compared with data obtained prior to the irradiation, to determine the cobalt content in the steel, and to provide a separate independent determination of the thermal fluence in the McMaster reactor.

C.1.5 Helium Analysis Procedure

Helium analyses on all specimens were conducted by isotope-dilution gas mass spectrometry following vaporization in a resistance-heated tungsten-wire or graphite crucible in a high-temperature vacuum furnace [2]. The absolute amount of ^{4}He released was measured relative to a known quantity of added ^{3}He "spike." The ^{3}He spikes were obtained by expanding and partitioning known quantities of gas through a succession of calibrated volumes [3]. The mass spectrometer was calibrated for mass sensitivity during each series of runs by analyzing known mixtures of ^{3}He and ^{4}He, as specified in PNNL Procedure RPG-MSL-1000.

C.1.6 Helium Analysis Results

The results of the helium measurements are given in Tables C-3 and C-4, and are listed as total atoms released and as atom concentrations. Table C-3 gives the results of the helium analyses on the Al-Li alloy, Fe, and Ni dosimetry materials included in the MNR irradiation. Helium concentrations for the Al-Li alloy are relative to the total number of ^{6}Li atoms in each specimen. For all other materials, the concentrations are relative to the total number of atoms in the sample.

Table C-3
Helium Concentrations in MNR Dosimetry Samples

Specimen	Material	Mass[a] (mg)	Measured ^{4}He (10^{15} atoms)	Helium Concentration (appm)[b]
AlLi-1	Al-Li Alloy	0.487	1.198	3544
AlLi-2	"	0.971	2.348	3483
AlLi-3	"	1.048	2.534	3483
Fe-1	Pure iron	2.245	0.000029	0.0012
Ni-1	Pure nickel	4.039	0.00025	0.0060

[a]Mass of analyzed specimen. Mass uncertainty is ±0.002 mg.

[b]Helium concentration in atomic parts per million (10^{-6} atom fraction) with respect to the total number atoms in the specimen. For the Al-Li, the concentration is relative to the number of ^{6}Li atoms.

Table C-4
Measured Helium Concentrations in Weld Pad Samples

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Table C-4 gives the helium concentrations measured in the weld-pad samples both before and after exposure in the MNR. Residual (pre-exposure) helium concentration in the PNNL vanadium, iron and nickel has been determined previously to be <0.03 appb. In Tables C-3 and C-4, conversion from atoms to concentration was based on the following calculated values:

Al-Li alloy.....0.06942 x 10^{22} atoms ^{6}Li/gram
Iron.....1.078 x 10^{22} atoms/gram
Nickel.....1.026 x 10^{22} atoms/gram
308L SS.....1.09 x 10^{22} atoms/gram
Vanadium.....1.182 x 10^{22} atoms/gram

It should be noted that these conversion factors, and the helium concentrations obtained using it, are not very sensitive to small changes in material composition.

Helium concentrations measured in the three Al-Li alloy wires are given in Table C-3. The mean helium concentration (in terms of atom fraction) is 0.003503 ± 0.000035. Correcting for burnup of the ^{6}Li isotope and for neutron self-shielding yields a value of 0.003724 ± 0.000037, which represents the ^{6}Li reaction probability $\sigma\varphi t$. Burnup corrections were made using the standard exponential formulation. Self-shielding corrections were based on transport theory approximations from Case et al.[4], and were 6.1%. Uncertainty in the self-shielding correction is conservatively estimated to be ~25% of the correction factor. Assuming a 2200 m/s $^{6}Li(n,\alpha)^{3}H$ neutron cross section of 941 barns, yields an effective thermal neutron fluence of 3.96 x 10^{18} n/cm^2. This compares well with a value of 4.06 x 10^{18} n/cm^2 obtained from the gamma analysis of the MNR-irradiated material discussed earlier.

Absolute uncertainty in the measured helium concentrations is estimated to be ~1% (1σ). This uncertainty results from the cumulative uncertainties in the sample mass, the isotope ratio measurement, and the spike size.

C.1.7 Boron Determinations

Calculated boron contents in the samples are given in Table C-5. Column 6 gives the measured helium contribution from the nickel component in the as-received material. Nickel content in the 308L steel was determined from separate XRF analyses conducted as part of the gamma analysis effort (see Attachment B). Corrections for burnup of ^{58}Ni and in growth of ^{59}Ni from the material's prior reactor exposure are negligible.

The nickel contributions in Table C-5 are 0.7 appb, or ~1% of the total incremental helium generation. This value was calculated from the measured helium generation in the pure nickel material included in the MNR irradiation, adjusted for the measured nickel content in the steel material. Most of this helium is from the threshold neutron (>1 MeV) reaction $^{58}Ni(n,He)$. Helium contributions from other threshold neutron reactions were calculated to be <1%, and were neglected.

Table C-5
Calculated Boron Contents in Steel Samples

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The Column 7 values in Table C-5 are the calculated ^{10}B contents in the as-received samples *prior* to the MNR exposure. These values were calculated from the net increase in the helium content attributable to the boron, obtained by subtracting the values in Columns 4 and 6 from those in Column 5, and assuming a 2200 m/s cross-section for the $^{10}B(n,\alpha)$ reaction of 3838 barns. Burnup of ^{10}B during the MNR irradiation, which amounted to 1.6%, was accounted for in the calculations.

Columns 8 and 9 gives an estimate of the original (unirradiated) natural boron content in each of the samples, and the estimated overall uncertainty. These values were determined from the calculated as-received ^{10}B contents in Column 7, corrected for burnup of ^{10}B during the previous plant exposure, and assuming a ^{10}B isotopic content of 19.9 atom%. Calculated burnup levels for the plant exposure, based on the thermal neutron fluences calculated at PNNL from the radiometric analyses in Column 2, ranged from 0.4% to 0.7%.

C.1.8 Uncertainty in Boron Determinations

Error sources in the determination of the boron contents are given in Table C-6. Generally, the largest error source is the variability in the measured pre- and post-irradiation helium concentrations. Total uncertainty estimates, determined from the quadrature sum of the various uncertainty components, are given in the last column of Table C-5.

Table C-6
Error Sources for Boron Determinations

Parameter	Estimated Error in Parameter (1σ)	Resultant Error in Boron Content (%)[a]
Measured		
He concentrations[b]	1% - 3%	1 - 3
Neutron fluence (MNR irradiation)	1.5%	1.5
Nickel content (wt.%)	1.0 wt.%	<0.1
He generation from nickel component	1%	<0.1
Calculated		
Prior thermal fluence (n/cm^2)	10%	<0.1

[a]Standard error range in calculated boron contents resulting from errors in the corresponding parameter.

[b]Standard deviations in replicate analyses (pre- and post-exposure combined).

Uncertainty in the nickel content of the steel is given as ±1.0 wt.%, based on the XRF analyses. Uncertainty in the calculated plant thermal neutron fluence values were arbitrarily assigned as ±10%. Uncertainty in the nickel contribution to the helium generation, and in the plant thermal fluence, have a negligible effect on the final calculated boron contents.

C.1.9 Discussion of Results

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C.2 References

1. J. Van Audenhove, Central Bureau for Nuclear Measurements, Geel, Belgium, CBNM Lot SP 3335, July 1979.
2. H. Farrar and B. M. Oliver, "A Mass Spectrometer System to Determine Very Low Levels of Helium in Small Solid and Liquid Samples," J. Vac. Sci. Technol. A, **4**, 1740 (1986).
3. B. M. Oliver, J. G. Bradley, and H. Farrar, "Helium Concentration in the Earth's Lower Atmosphere," Geochim. et Cosmochim. Acta **48**, 1759 (1984).
4. K. M. Case, F. de Hofmann and G. Placzek, "Introduction to the Theory of Neutron Diffusion", Los Alamos Scientific Laboratory, Los Alamos, New Mexico, June 1953.

C.3 Retrospective Neutron Dosimetry for the Plant 3 Reactor

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C.3.1 Summary

Samples were obtained by Framatome from four jet pump riser brace pads near the pressure vessel of the Plant 3 reactor and delivered to PNNL for analysis. The jet pumps and locations were 3 at 68°, 5 at 104°, 11 at 248°, and 13 at 284°. Each sample consisted of multiple, small stainless steel turnings drilled out of the jet pump riser brace pads and collected on a filter by a water suction system. Sample 3 showed some contamination from a broken drill bit that was noted during the sampling. Selected subsamples from each group of turnings were initially gamma counted to determine the residual activation products. Subsamples were also analyzed for the beta-emitter ^{63}Ni. The chemical composition of each steel sample was measured by x-ray fluorescence. All of the measured activities were then converted to saturated activation rates by taking into consideration the reactor power history. Thermal and fast neutron fluences were determined at each position using spectral-averaged cross sections determined for a typical BWR pressure vessel spectrum. The neutron fluences were then calculated using the total reactor power history. The analyses and results are discussed in detail below.

C.3.2 Gamma Energy Analyses

Selected subsamples were weighed and then gamma counted using high efficiency intrinsic germanium detectors following Procedure PNL-ALO-450. The detectors are calibrated relative to NIST and other accepted standards, and control counts are performed daily to ensure continuing energy and efficiency calibrations. All four of the samples showed the presence of ^{51}Cr, ^{54}Mn, ^{57}Co, ^{58}Co, ^{59}Fe, ^{60}Co, and ^{65}Zn. The measured activities and total propagated uncertainties in microCuries/milligram are listed in Table C-7. All of the activities were corrected for decay to the date of the reactor shutdown at the end of cycle 17 on April 12, 2001.

Table C-7
Activity Measurements (μCi/mg) with 1σ Standard Deviations for the Plant 3 Reactor. Values are Corrected to April 12, 2001 and Normalized to a Power Level of 1593 MWt

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The ^{51}Cr, ^{59}Fe, ^{60}Co, and ^{65}Zn activities are due to thermal neutron activation of the major constituents Cr and Fe, and the Co, Cu, and Zn impurities in the samples. The other activities are produced by the fast neutron reactions $^{54}Fe(n,p)^{54}Mn$, $^{58}Ni(n,p)^{58}Co$, and $^{58}Ni(n,2n+np)^{57}Co$. The ^{60}Co value for position 1 showed considerable scatter, possibly due to the contamination of this sample by residual material in the sampling mechanism (see discussion below). ^{60}Co is also produced by two other reactions, namely $^{60}Ni(n,p)^{60}Co$ and from the decay of ^{59}Fe to ^{59}Co. However, calculations indicate that both of these reactions make a negligible (< 0.7%) contribution to the ^{60}Co that is produced from the ^{59}Co impurity in the steel.

Each sample consisted of multiple drill turnings collected from each jet pump location. Several subsamples were gamma counted at each location to check for homogeneity of the results. At location 3, the sub sample counting results showed only fair agreement, apparently due to the presence of samples from the broken drill bit. One small piece of material was found to have a totally different signature of activation products suggesting that it did not come from the current sampling at location 3. One rather large turning was assumed to be the best representation of the location 3 activities since it was known to be uncontaminated with the drill bit pieces. A second sub sample consisting of several pieces gave only slightly higher results. The values in Table C-7 are the average of these two subsamples. Most of the other subsamples showed lower activities, presumably due to contamination with the drill bit or other materials (such as the one piece that we found that did not have the correct activation signature).

At position 5, three separate sub samples were counted and the results agreed within counting statistics. The values in Table C-7 represent the average of these three measurements.

At positions 11 and 13, three subsamples were initially gamma counted. At position 11, two of the subsamples agreed within counting statistics; however, the third sub sample gave higher results by about 30%. The same behavior was also seen at position 13 except that one of the sub samples was lower by about 30%. In both cases, two of the original subsamples were further divided and gamma counted. (The third original sub sample consisted of one large turning.) However, these results were identical to the undivided sub samples indicating that the results could not be easily explained by contamination with other materials. Furthermore, in both cases the ratios of all the activation products were very nearly constant ruling out contamination with a different type of activated material. Due to the very small area of the jet pumps that were sampled at each location, neutron flux and spectral gradients would be negligible and we would expect good agreement for all the turnings that were collected. It is possible that the samples were contaminated or mixed during the sampling process. After each sample is taken, the entire sampling apparatus is removed from the water and the drill bit and sampling tube are visually inspected. No abnormalities were reported during the sampling other than the broken drill bit for sample 3. However, it is possible that a turning from one sample remained in the sampling tube or the internal section of the venturi tube and was then sucked into the next sample. Private conversations with staff from Framatome view this type of sample mix-up as unlikely; however, this appears to be the only plausible scenario for explaining the variances in the gamma counting data.

For both jet pumps 11 and 13, the two sub samples that agreed within counting statistics were assumed to represent the most likely samples collected from these locations. These results were averaged and reported in Table C-7 since this appears to be the most likely data from each location. These sub samples were also used for the He and B measurements discussed in Attachment A.

C.3.3 ^{63}Ni Measurements

Subsamples measuring about 1-2 mg were dissolved in HCl and brought to a known volume in distilled water. Aliquots were taken and spiked with a 2-mg Ni carrier to be used as a yield monitor. After a matrix adjustment, the nickel fraction was purified using an on-column nickel dimethyglyoxime precipitation reaction following procedure PNL-ALO-495. Nickel was retained on the column; impurities were eluted with a basic ammonium citrate solution. The nickel was then eluted with 3M HNO_3. The strip solution was carefully evaporated to near dryness, brought to a known volume with 0.1M HCl, and split for radiochemical recovery determination by ICP-AES and ^{63}Ni analysis by liquid scintillation counting following procedure PNL-ALO-474. The chemical yields averaged 98%. A minor (<7%) correction for ^{60}Co tailing into the ^{63}Ni region was made using a ^{60}Co standard to determine the beta fraction in the ^{63}Ni energy window. Calibrations were performed using ^{63}Ni standards obtained from NIST. Sample duplicates showed good repeatability and the blank spike and matrix spike averaged 103% yield-corrected recovery. The decay-corrected ^{63}Ni activities are listed in Table C-7. The ^{63}Ni activity is produced by the ^{62}Ni(n,γ) thermal neutron reaction. ^{63}Ni is an excellent thermal neutron monitor since the long half-life of 100 years makes it relatively independent of the reactor power history.

C.3.4 Calculation of Saturated Activation Rates

Table C-7 lists the measured activities and uncertainties corrected to the end of irradiation time for each sample and reaction. Nuclear decay data were taken from the Table of Radioactive Isotopes, E. Browne, R. Firestone, and V. Shirley, Wiley, 1986. The activities were converted to saturated activation rates by correcting for the decay during irradiation, atomic weight, elemental abundance, isotopic abundance, gamma self-absorption, neutron self-absorption, and nuclear burnup. These factors are shown in Table C-8 and are discussed below. The resultant saturated activation rates are shown in Table C-9.

The decay during irradiation was determined using the BCF computer code that integrates the decay of each isotope for each period of reactor operation and downtime. Reactor time history as provided by the client for the Plant 3 reactor, is shown in Figure C-2.

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The accuracy of the reactor power history correction factors depend on the halflife of each isotope relative to the details of the irradiation histories. The longer-lived isotopes, such as ^{63}Ni

truly integrate over the entire irradiation history. However, shorter-lived isotopes, such as ^{59}Fe (45 day) or ^{58}Co (71 day) are only dependent on the last year or so of reactor operation and the decay corrections are thus more uncertain since they depend critically on the recent details of the irradiation history. Comparison of neutron fluences derived later from the activation data shows that different thermal and fast neutron reactions give excellent agreement verifying the accuracy of the power history corrections.

Table C-8
Correction Factors and Cross Sections for Each Reaction

Reaction	At.Wt	Iso.Abn.	Gabs	History	Cross Sections, Barns	
Thermal					**Thermal**	**Epi.Cor.**
$^{62}Ni(n,\gamma)^{63}Ni$	58.69	0.0363	N/A	0.1722	14.5	15.3
$^{58}Fe(n,\gamma)^{59}Fe$	55.847	0.0028	0.993	1.446	1.28	1.48
$^{50}Cr(n,\gamma)^{51}Cr$	51.996	0.0435	0.987	1.446	15.9	16.8
$^{59}Co(n,\gamma)^{60}Co$	58.933	1.0	0.993	1.156	37.2	46.1
Fast					**>.1 MeV**	**>1 MeV**
$^{54}Fe(n,p)^{54}Mn$	55.847	0.059	0.990	1.365	0.1058	0.1846
$^{58}Ni(n,p)^{58}Co$	58.69	0.6808	0.990	1.443	0.1337	0.2333

At.Wt. = atomic weight
Iso. Abn. = isotopic abundance of target (also need elemental abundance in alloy)
Gabs = calculated gamma self-absorption in wires
History = reactor power history correction for decay
Thermal = 2200 m/s thermal neutron cross section
Epi. Cor. = thermal neutron cross sections with epithermal corrections, using best fit to the thermal reaction data

Table C-9
Saturated Activation Rates (Atom/Atom-Second) with 1σ Standard Deviations Values are Normalized to a Power Level of 1593 MWt

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Figure C-2
Power History for the Plant 3 Reactor

Estimated gamma self-absorption corrections varied from 1-2% for the direct counting of the turnings. Neutron self-absorption corrections were not performed since we do not have knowledge of the geometry of the individual turnings during reactor operation. Since the samples were removed from the surface (<0.060") of the pressure vessel, the neutron self-absorption corrections would be negligible for stainless steel. Due to the relatively low neutron fluence levels, nuclear burnup and transmutation corrections for both target and product isotopes are negligible.

C.3.5 X-Ray Fluorescence Measurements

The steel samples have the nominal composition of Type-308 stainless steel. In order to determine the exact composition of each sample more accurately, subsamples were analyzed by energy dispersive x-ray fluorescence (EDXRF) techniques at KLM Analytical in Richland, WA. In this procedure, samples are bombarded with an electron beam and the x-rays are measured with a high-resolution detector. Since the samples were radioactive, x-ray spectra were also taken with the beam off and this background was subtracted from each EDXRF x-ray spectrum. NIST 304L stainless steel standards were analyzed at the same time to calibrate and verify the performance of the equipment and interelement corrections. The results of the x-ray measurements are listed in Table C-10.

Due to the very low concentration of Co in these samples, the Co content was determined by neutron activation analysis using the irradiation of the samples at McMaster University, which was done to measure the boron content. The ^{60}Co activity was measured before and after this new irradiation although the new irradiation greatly increased the ^{60}Co content. Thermal neutron standards included with the irradiation were used to determine the neutron fluence so that the Co content could then be calculated. All of the samples gave an average Co content of 813±40 wppm.

Table C-10
EDXRF Analyses of the JPRB Steel Samples for Plant 3

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C.3.6 Neutron Fluence Evaluations

The corrected saturated activities listed in Table C-9 are quoted in product atom per target atom per second. These values are equal to the integral over neutron energy of the neutron activation cross section times the neutron flux spectrum. In principle, the neutron flux spectra can be determined using neutronics calculations; however, such information was not provided. An estimate of the thermal neutron fluence can be obtained by dividing the saturated activities by the 2200 m/s thermal neutron cross section and multiplying by the total irradiation times using cross section data obtained from the National Nuclear Data Center at Brookhaven National Laboratory (values are listed in Table C-8). A better estimate of the thermal neutron fluences can be obtained by correcting for the presence of epithermal neutrons, as is done for the values in Table C-11. A simple way to do this is to set each saturated activity equal to the sum of the thermal flux times the thermal cross section plus an epithermal flux times the resonance integral. This correction is also listed in Table C-8 using a ratio of the epithermal to thermal flux of 0.12. For the previous reports on Plant 1 and Plant 2, these epithermal corrections were determined using a generic GE BWR spectrum, as shown in Figure C-3, with an epithermal ratio of 0.044. However, in the present case, this ratio was varied to obtain the best fit to the data. It should be noted that using higher epithermal fluence lowers the apparent thermal fluence relative to the other plants we have studied. (A similar adjustment might improve the fits to the thermal data for these plants as well, although the differences are on the order of 10%.) More accurate estimates of the thermal, epithermal, and fast neutron fluences require a calculation of the neutron flux spectrum at each position rather than using a generic BWR neutron spectrum.

The gamma counting data was also used to estimate the fast neutron fluence. In order to accurately determine the fast fluence above 1 MeV, or any other energy threshold, it is necessary to know the energy dependence of the neutron flux spectrum. These spectral-averaged activation cross sections were calculated from the GE BWR neutron spectrum shown in Figure C-8 and the values are listed in Table C-8. The standard deviations in the averages of the two fast neutron reactions are less than 11%, indicating very good agreement. However, absolute uncertainties may be somewhat larger due to the uncertainty in the calculation of the spectral-averaged cross sections. Table C-11 lists the neutron fluences and estimated uncertainties above 0.1 and 1.0 MeV at each position obtained by averaging the Fe and Ni results.

C.3.7 Conclusions

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Table C-11
Neutron Fluences with 1σ Uncertainties for the Plant 3 Reactor (Neutrons/cm^2 x 10^{18})

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Figure C-3
Neutron Flux Spectra for a Typical BWR Pressure Vessel Location, as Provided by General Electric (GE) and the Electric Power Research Institute (EPRI)

D
NRC FINAL SAFETY EVALUATION

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

BWRVIP 2005-079A

February 25, 2005

Bill Eaton, BWRVIP Chairman
Entergy Operations, Inc.
Echelon One
1340 Echelon Parkway
Jackson, MS 39213-8202

SUBJECT: SAFETY EVALUATION OF PROPRIETARY EPRI REPORT, "BWR VESSEL AND INTERNALS PROJECT, SAMPLING AND ANALYSIS GUIDELINES FOR DETERMINING THE HELIUM CONTENT OF REACTOR INTERNALS (BWRVIP-96)" (TAC NO. MB3947)

Dear Mr. Eaton:

The NRC staff has completed its review of the Electric Power Research Institute (EPRI) proprietary report, "BWR Vessel and Internals Project, Sampling and Analysis Guidelines for Determining the Helium Content of Reactor Internals (BWRVIP-96)," dated November 2001. This report was submitted by letter dated November 29, 2001, for NRC staff review and approval. The BWRVIP also submitted the non-proprietary version of this report by letter dated February 10, 2005. The BWRVIP-96 report provides guidelines for removing small samples from irradiated reactor internals to measure the helium content for weldability determinations. The NRC staff has reviewed your submittal and finds that the report presents an acceptable technical justification of its proposed methodology. The staff's safety evaluation is attached.

Please contact Meena Khanna of my staff at 301-415-2150 if you have any further questions regarding this subject.

Sincerely,

William H. Bateman, Chief
Materials and Chemical Engineering Branch
Division of Engineering
Office of Nuclear Reactor Regulation

Enclosure: As stated

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U.S. NUCLEAR REGULATORY COMMISSION

OFFICE OF NUCLEAR REACTOR REGULATION

SAFETY EVALUATION OF BWR VESSEL AND INTERNALS PROJECT,

"SAMPLING AND ANALYSIS GUIDELINES FOR DETERMINING

THE HELIUM CONTENT OF REACTOR INTERNALS (BWRVIP-96)"

1.0 INTRODUCTION

1.1 Background

By letter dated November 29, 2001, the Boiling Water Reactor Vessel and Internals Project (BWRVIP) submitted the Electric Power Research Institute (EPRI) Proprietary Report TR-1003019, "Sampling and Analysis Guidelines for Determining the Helium Content of Reactor Internals (BWRVIP-96)," for staff review. It was supplemented by letter dated June 7, 2004, in response to the staff's request for additional information (RAI) dated January 8, 2003. In addition, by letter dated February 10, 2005, the BWRVIP submitted the non-proprietary report, TR-1003019NP, BWRVIP-96NP: "BWR Vessel and Internals Project, Sampling and Analysis Guidelines for Determining the Helium Content of Reactor Internals." The BWRVIP-96 report provides the guidelines that describe the acceptable methods for sample removal from irradiated reactor internals and analysis of helium content for weldability determinations.

1.2 Purpose

The staff reviewed the BWRVIP-96 report to determine whether its proposed guidance will provide an acceptable method for sample removal from irradiated reactor internals and analysis of the helium content for weldability determinations by direct measurement. The review assessed the proposed guidelines regarding sample removal, sample transmission, analysis methodology, and safety evaluation.

1.3 Organization of this Report

Because the BWRVIP report is proprietary, this SE was written to not repeat proprietary information contained in the report. The staff does not discuss in detail the provisions of the guidelines or the parts of the guidelines it finds acceptable. A brief summary of the contents of the BWRVIP-96 report is provided in Section 2 of this SE, with the evaluation presented in Section 3. The conclusions are summarized in Section 4. The presentation of the evaluation is structured according to the organization of the BWRVIP-96 report.

ENCLOSURE

2.0 SUMMARY OF BWRVIP-96 REPORT

The BWRVIP-96 report addresses the following topics in the following order:

- Introduction - The BWRVIP-96 report states that the one of the factors that affect the weldability of stainless steels in BWR reactors is the helium content of the metal. If the concentration of helium is above a certain threshold, helium bubbles will be produced in sufficient quantity to degrade the strength of the material and to cause cracks in the weld heat affected zone (HAZ). Therefore, the report states that if welding is to be considered as a means of repairing an irradiated reactor component, it is important to know the approximate helium content at the weld location.

  For some reactor internal components, the helium concentration can be calculated. This involves estimating the neutron fluence at the location as well as knowledge of the boron and nickel concentrations in the material. This process is straightforward, however, the calculated helium concentration may be subject to substantial uncertainty. There is uncertainty in the thermal neutron fluence calculations due to the fact that they have not been accurately benchmarked at many locations of interest, e.g., jet pump riser brace pads or core spray piping. In addition, the boron concentration of the subject material may not be known. Therefore, a more direct and accurate method for determining the helium concentration is needed in some cases. One acceptable method for determining the helium concentration is by direct measurement on a small sample removed from the component of interest.

  Recognizing that utilities may need to perform such sampling and analysis in the future, the BWR Vessel and Internals Project (BWRVIP) and the NRC conducted a joint project to demonstrate sample removal from a number of BWRs. Samples were removed from the jet pump riser brace pad at four locations in three US BWRs using tooling developed by Framatome-Advanced Nuclear Power (FRA-ANP). The samples were sent to Pacific Northwest National Laboratory (PNNL) and were analyzed for helium content, initial boron content and neutron fluence.

  The objective of the sampling and analysis guideline is to use the lessons learned from the BWRVIP/NRC project to provide utilities with guidance on performing similar sampling of internal components. The discussion in this report is based primarily on experience using the Framatome tooling and analyses performed by PNNL. However, other methods of sample removal and analysis are also acceptable for determining the helium content of stainless steels and high nickel alloys for purposes of assessing weldability.

- Sampling/Analysis Requirement - Section 2 of the report discusses the basic requirements for removing and analyzing a sample. This includes a discussion of the required sample size, sampling process, and sample handling/shipping. The report also provides a description of the required safety analysis. It states that a technical justification is required to support the development of a safety evaluation by the licensee for the sampling activity. The technical justification is to address testing, qualification, the use of the equipment, as well as relevant structural and material issues associated with the as-left condition of the sampling location(s).

In addition, the report lists the plant-specific information that is typically required for the design of the tooling and mock-ups and the engineering technical justification for the sampling process.

An overview of the helium analysis procedure, which is the technique used by PNNL, is included in the report.

- BWRVIP/NRC Jet Pump Riser Brace Sampling - In Section 3 of the report, the BWRVIP discusses that samples were removed from the jet pump riser brace pad at three reactors and analyzed for helium, initial boron content, and fast and thermal neutron fluence. An overview of that sampling and analysis is provided in the report. This section of the report provides utilities with a better understanding of the scope of the process. In addition, a description of the typical tooling, testing and qualification of the sampling process, sample removal process, sample site support, sampling personnel and locations, sample handling requirements, requisite safety analyses, and sample analysis results are provided in the report.

  Section 3 of the report also discusses that a safety analysis was conducted to ensure that the sampling would not leave the surface in a degraded condition. The safety analysis consisted of a structural analysis, a material evaluation, and a metallurgical examination of the jet pump riser brace weld pad qualification samples. The report indicates that the structural analysis demonstrated that the rules of ASME Section III are not violated. With respect to the material evaluation, the report indicated that the small amount of cold work resulting from the sampling process is considered acceptable because it is significantly less than that introduced in other BWR components during original construction. In addition, the report stated that the process qualification will show that no crevices have been created during the sample removal process; therefore, IGSCC initiation, resulting from the introduction of a crevice during the sample removal process, is not an issue.

  Metallurgical examinations were performed on the jet pump riser brace weld pad qualification samples. Based upon the examinations and tests performed, it appeared that the machining process proposed for removal of samples from the jet pump riser pads is acceptable for use because the machining process did not create laps, tears or crevices and did not introduce excessive cold work into the weld metal pad. Based on the results of the examinations, it appeared that the "divots" in the weld pad, resulting from the removal of samples, should not produce an initiation site for stress corrosion cracking.

  The report provides details of the safety analysis that was conducted to indicate the types of evaluations performed. Also, the report discusses the sample analysis results.

- Example of Analytical Results - In Appendix A of the report, the results of the helium, boron, and neutron fluence analyses for one of the three plants that were sampled as part of the BWRVIP/NRC project are provided. Based on these results, the BWRVIP provides recommendations in the report.

The thermal and fast neutron fluences were determined with uncertainties ranging from 6% to 29% at each of the four jet pump riser brace positions. The data clearly display the effects of the decrease in the core edge neutron fluxes caused by the changes in the core configuration to reduce the neutron leakage; therefore, the BWRVIP provided recommended fluences in Table A-11 of the report. The BWRVIP concluded that the thermal neutron fluences calculated from the different transmutation/decay reactions are in good agreement and the absolute accuracy should be close to the standard deviations listed in Table A-11 of the report.

3.0 STAFF EVALUATION

3.1 Sampling Size/Analysis Requirements

The staff finds that the BWRVIP adequately addressed the basic requirements for removing and analyzing a sample. The staff determined that the BWRVIP-96 report includes comprehensive information on the preferable sample size and sample extraction locations because the sampling process is performed in accordance with a qualified procedure and multiple high and low fluence locations are utilized, which will provide a good representation of sampling locations.

The staff finds that the report adequately provides the basis and the rationale for a required safety analysis which ensures that a sample removal would not lead to a stress concentration that would adversely impact the fatigue life of the reactor vessel. This refers to the potential for a fatigue crack to grow from the riser brace pad into the reactor pressure vessel that could effect the pressure boundary. In addition, the rules of ASME Section III are not violated because the BWRVIP's structural analysis showed that the resulting stress concentration yields an allowable number of cycles of 1500, per Figure N-415 (B) of ASME Section III, which will lead to a cumulative usage factor well below the allowable value of 1, and therefore, fatigue crack formation is not expected. In addition, the material evaluation from the metallurgical examination, as stated in the BWRVIP-96 report, concluded that cold work and crevices created from the sample removal were acceptable and would not impact compliance with the fatigue analysis requirements of ASME Section III.

The staff concludes that the sampling size/analysis requirements provided in the BWRVIP-96 report are comprehensive and, where applicable, adequately meet the fatigue analysis requirements of ASME Section III, and are, therefore, acceptable.

3.2 BWRVIP/NRC Jet Pump Riser Brace Sampling

The staff evaluated the testing and qualification plan for the sampling process and noted that the process was performed in accordance with a qualified procedure, and the weld pad mockups were constructed with representative surface finishes. Also, the staff agreed that the machining characteristic of the un-irradiated mockup materials are representative of the field conditions. Therefore, the staff found that the BWRVIP adequately addressed the testing and qualification plan for the sampling process.

The staff found the sample removal process acceptable because it was performed in accordance with a qualified procedure. In addition, the staff finds that the guidelines regarding

sampling hardware, sites, personnel, procedures, and lessons learned are clearly described in the report.

As stated in Section 3 of the report, the removal of the sample from the weld pad will result in a stress concentration that was not considered in the original design calculations. However, with respect to the structural analysis that was conducted, the staff found that the BWRVIP adequately demonstrated that the effects of stress concentration on the fatigue life of the reactor vessel is small and that the fatigue analysis requirements of ASME Section III will still be met, and the as-left condition would, therefore, be acceptable.

The staff noted that the BWRVIP adequately addressed the material evaluation of the BWR jet pump riser brace weld pad. The staff agreed that the small amount of cold work resulting from the sampling process, which would be significantly less than that introduced in other BWR components during original construction, is considered acceptable. In addition, the BWRVIP indicated that the process qualification shows that no crevices have been created during the sample removal process; therefore, the staff agrees that IGSCC initiation, resulting from the introduction of a crevice during the sample removal process, is not an issue.

Examinations and tests were performed to adequately demonstrate that stress corrosion cracking should not be a concern for the removal of samples from the jet pump riser pads; therefore, the staff agreed that the "divots" in the weld pad resulting from the removal of samples should not produce an initiation site for stress corrosion cracking.

Based on the above evaluation, the staff finds that the BWRVIP-96 report provides comprehensive guidelines regarding the BWRVIP/NRC jet pump riser brace sampling.

3.3 Example of Analytical Results

Appendix A of the report provided details of the sampling results for one of the three plants. The staff reviewed Appendix A and found that the discussions on the retrospective neutron dosimetry to be reasonable and adequate because established procedures had been used and the thermal neutron fluences from different transmutation/decay reactions are in good agreement. However, in order to complete the review on the helium and neutron dosimetry analysis, the staff required additional information. Accordingly, by letter dated January 8, 2003, the staff requested additional information. The first RAI (RAI 96-1) related to the fact that Section A.1.9 of the BWRVIP-96 report states that the reproducibility of the helium content measurements between the replicate analysis averaged 4%. Section A.1.9 also states that reproducibility in the analysis system for these types of samples is 0.5%. The review of Table A-4 on which these statements are based indicates that the spread of the reproducibility is between about 0.5% and 8% for the pre-exposure samples and between 2.4% and 8% for the post-exposure samples. The staff requested clarification on whether this spread, in reproducibility, is a reflection of the helium homogeneity issue discussed in Section A.1.5 of the report. The staff requested that the BWRVIP explain the wider spread in the reproducibility in the pre-exposure samples compared to the post-exposure samples and whether similar spreads in reproducibility have been observed in samples analyzed for the other two plants as part of the BWRVIP/NRC project. The BWRVIP was also requested to provide a copy of Reference 1 (as stated in the report), memo from Vaughn Wagoner to all BWRVIP Committee

Members, dated October 4, 2001, "Transmittal of Summary Reports on Jet Pump Riser Brace Pad Sample Analysis."

By letter dated June 7, 2004, the BWRVIP responded to RAI 96-1, by stating that the NRC correctly noted that the spread of reproducibility of the helium measurements for individual samples range from approximately 0.5% to 8%. The helium content of samples is proportional to the initial boron content. Variations in the boron measurements for samples taken from the same location ranged from approximately 3% to 6%. Variations higher than 2% are attributed to inhomogeneity in the boron content in the samples and may reflect real variations in the boron content across the riser brace pad. The BWRVIP further stated that the inhomogeneity in the boron concentration gives rise to similar inhomogeneity in the helium concentration. Therefore, the BWRVIP concluded that the spread of the helium reproducibility is attributed to a combination of measurement uncertainty and boron inhomogeneity.

The BWRVIP stated that the variability observed in these samples is very small in terms of their intended use. The helium measurements will be used in making determinations regarding the weldability of reactor internal components. It was noted that BWRVIP-97, "Guidelines for Performing Weld Repairs to Irradiated BWR Internals," has established a factor of safety of about 10 on the helium concentration threshold for determining weldability. The BWRVIP concluded that uncertainties or inhomogeneity of a few percent are inconsequential.

The BWRVIP also provided a copy of the memo requested in RAI-96-1, with the related PNNL reports describing the analysis results from all three plants. The BWRVIP stated that reproducibility of the helium measurements from the other two plants was generally better than that for the plant reported in the BWRVIP-96 report and ranged from approximately 0.1% - 3%.

The staff reviewed the BWRVIP's response and the reports that were attached and agrees that the spread of the helium concentration reproducibility is attributable to a combination of measurement uncertainty and boron inhomogeneity. The staff agrees that uncertainties or inhomogeneity of a few percent in the helium concentration are inconsequential since the BWRVIP-97 report has established a factor of safety of about 10 on the helium threshold for determining weldability. The staff notes that according to Table 6 of the two PNNL reports, regarding Plants 2 and 3, the estimated error for the measured helium concentration ranged from 1% to 3%. This is consistent with the BWRVIP's statement that reproducibility of the helium measurements from the other two plants was generally better than that for the plant reported in the BWRVIP-96 report.

RAI 96-2 related to the fact that since every atom of ^{10}B will eventually transmute to ^{4}He and one weight-percent natural B can produce one atomic percent He (see 3rd paragraph in Section 2.1 of the BWRVIP-97 report), the weldability would vary considerably with the variations in the boron content in the riser brace pad regions of the plants. Sections A.1.1 and A.1.9 of the BWRVIP-96 report state that the range of the boron concentration values is higher than can be explained by the measurement uncertainty or boron inhomogeneity, suggesting real variations in the boron contents between the four jet pump riser brace samples which all came from the same plant. The staff requested clarification on the source of these real variations, the variations in boron contents that has been observed in samples from the other two plants, and the source of these variations.

In the letter dated June 7, 2004, the BWRVIP responded to RAI 96-2 by stating that similar to their response to RAI 96-1, the variations in the boron content are attributed to a combination of measurement uncertainty (~2%) coupled with real inhomogeneity of the boron content of the riser brace pad. The BWRVIP further stated that boron variability in the samples from the other two plants ranged from 2 to 4 percent. The BWRVIP concluded that this degree of variability is not significant in the context of determining weldability.

The staff reviewed the BWRVIP's response and the reports that were attached. The staff found the response adequate because the BWRVIP clarified that the spread of the boron measurement is attributed to a combination of measurement uncertainty and boron inhomogeneity and that uncertainties or inhomogeneity of a few percent for all three plants are inconsequential in the context of determining weldability. Since the requests for additional information regarding the helium and boron analysis in Appendix A of the BWRVIP-96 report (as documented in RAIs 96-1 and 96-2) have been adequately addressed by the applicant's responses in its June 7, 2004 letter, the staff finds the discussion on the helium and boron analysis in Appendix A, as supplemented by the applicant's response to RAIs 96-1 and 96-2, to be acceptable.

Based on its review of the report, the additional material provided by the BWRVIP, and the responses to the staff's RAIs, the staff finds the guidelines provided by this subject report acceptable.

4.0 CONCLUSION

The staff has reviewed the BWRVIP-96 report and the supplemental information that the BWRVIP has provided in its response to the staff's RAIs. On the basis of the staff's evaluation above, the staff finds that the guidelines provided in the BWRVIP-96 report for removing a small sample from a reactor to measure the helium content to be acceptable. The staff requests that the BWRVIP incorporate the information that was provided in its response to the staff's RAIs into the -A version of the BWRVIP-96 report.

5.0 REFERENCES

1. EPRI TR-1003019, "BWRVIP-96, Sampling and Analysis Guidelines for Determining the Helium Content of reactor Internals," dated November 2001.

2. Vaughn Wagoner/Tom Mulford to All BWRVIP Committee Members, "Transmittal of Summary Reports on Jet Jump Brace Pad Sample Analyses," dated October 4, 2001, with attached reports.

3. NRC Request for Additional Information, "Proprietary Request for Additional Information, Review of BWR Vessel and Internals Project Reports, BWRVIP-96, -97, -99, and -100," dated January 8, 2004.

4. BWRVIP Response to NRC Request for Additional Information, "Response to NRC Request for Additional Information on BWRVIP-96," dated June 7, 2004.

E
RECORD OF REVISIONS

BWRVIP-96-A	Information from the following documents was used in preparing the changes included in this revision of the report: 1. *BWRVIP-96: BWR Vessel and Internals Project, Sampling and Analysis Guidelines for Determining the Helium Content of Reactor Internals.* EPRI, Palo Alto, CA: 2001. 1003019. 2. Letter from Meena Khanna (NRC) to Carl Terry (BWRVIP Chairman), "Proprietary Request for Additional Information - Review of BWR Vessel and Internals Project Reports BWRVIP-96, -97, -99 and -100 (TAC NOS. MB3947, MB3948, MB3951 and MB3946" dated 1/08/03 (BWRVIP Correspondence File Number 2003-022). 3. Letter from William Eaton (BWRVIP Chairman) to Meena Khanna (NRC), "Project 704 - BWRVIP Response to NRC Request for Additional Information on BWRVIP-96" dated 6/7/04 (BWRVIP Correspondence File Number 2004-214). 4. Letter from William Bateman (NRC) to Bill Eaton (BWRVIP Chairman), "Safety Evaluation of Proprietary EPRI Report, "BWR Vessel And Internals Project, Sampling and Analysis Guidelines for Determining the Helium Content Of Reactor Internals (BWRVIP-96)" (TAC NO. MB3947)" dated 2/25/05 (BWRVIP Correspondence File Number 2005-149). Details of the revisions can be found in Table E-1.

Table E-1
Revision Details

Required Revision	Source of Requirement for Revision	Description of Revision Implementation
Revise report to include information provided to NRC in response to NRC RAI.	NRC Final SE (2005-149)	Discussion added in Section 3.3 ("Sample Analysis Results"); Results of sample analysis for plants 2 and 3 added as Appendices B and C.
Add NEI 03-08 Requirements		Requirements added to Section 1
End of Revisions		

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