Duke Power.
A Duke Energy Company

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January 22, 2003

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington D.C. 20555

Subject: Duke Energy Corporation
Catawba Nuclear Station Units 1 & 2
Docket Nos. 50 -413, 414
McGuire Nuclear Station Units 1 & 2
Docket Nos. 50 -369, 370
Oconee Nuclear Stations Units 1, 2 & 3
Docket Nos. 50-269, 270, 287

Request for Additional Information (RAI) Associated With the 60 Day Response to NRC Bulletin 2002-01:
Reactor Pressure Vessel Head Degradation
Reactor Coolant Pressure Boundary Integrity

This letter and the associated attached Enclosures provide the Duke Energy Corporation (Duke) response to the RAI associated with the 60 day response to NRC Bulletin 2002-01 for the Catawba, McGuire and Oconee Nuclear Stations. The RAI responses are provided in Enclosures I, II and III for Catawba, McGuire and Oconee, respectively.

The enclosed responses represent Duke's current programs. Due to the current industry issues with degradation of Alloy 600 and other pressure boundary materials, Duke is conducting a review of the effectiveness of the methods used at its nuclear plants to detect borated water leakage. This review is focusing on the adequacy of the programs to detect degradation due to leakage onto the reactor coolant pressure boundary and on assuring a strong interface between programs that detect leakage and those that correct leakage or degradation. In addition, Duke is developing a comprehensive Alloy 600 program to manage nickel based wrought and weld pressure boundary materials.

Duke has not made any specific regulatory commitments in response to this RAI.

If you have questions or need additional information, please contact Gregory S. Kent at (704)373-6032.

Very truly yours,

H B Barron

H. B. Barron, Jr.
Senior Vice President
Nuclear Operations

ENCLOSURES

A095

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H. B. Barron, Jr., affirms that he is the person who subscribed his name to the foregoing statement, and that all the matters and facts set forth herein are true and correct to the best of his knowledge.

HB Barron

H. B. Barron, Jr., Senior Vice President

Subscribed and sworn to me: Jan 22, 2003
Date

Mary P. Nelms
Notary Public

My Commission Expires: JAN 22, 2006
Date

SEAL

Enclosure I
Catawba Nuclear Station
Response to NRC Bulletin 2002-01 RAI

RAI 1

Provide detailed information on, and the technical basis for, the inspection techniques, scope, extent of coverage, and frequency of inspections, personnel qualifications, and degree of insulation removal for examination of Alloy 600 pressure boundary material and dissimilar metal Alloy 82/182 welds and connections in the reactor coolant pressure boundary (RCPB). Include specific discussion of inspection of locations where reactor coolant leaks have the potential to come in contact with and degrade the subject material (e.g., reactor pressure vessel (RPV) bottom head).

Catawba Nuclear Station relies on multiple methods to determine the condition of Alloy 600 pressure boundary material, including base metal and Alloy 82/182 welds that connect to the reactor coolant pressure boundary. These methods consist of code examinations to detect degradation, various walkdowns for the purpose of detecting borated water leakage, and specific inspections to address recent industry issues. The following paragraphs discuss these primary methods.

ASME Code In-service Inspection

Catawba Nuclear Station conducts In-service Inspections (ISI) in accordance with the applicable ASME Section XI code. As part of a comprehensive ISI program, those connections with the RCPB that contain Alloy 600/82/182 and are greater than 1" in diameter for Class 1 receive a surface examination once per ten-year interval. Class 1 piping equal to or greater than 4" in diameter receives a surface and volumetric examination once per ten-year interval.

Other connections containing Alloy 600/82/182 that the code does not require to be inspected volumetrically are inspected by area surveillance conducted as part of the ASME Section XI code required system leakage test that occurs at operating pressure and temperature during plant startup. All Class 1 piping is within the inspection scope, including the reactor vessel upper head, the incore instrument nozzles on the lower reactor vessel head, and other areas subject to recent maintenance. A VT-2 examination method in accordance with ASME Section XI is used during the system leakage test, which does not require that insulation be removed during inspection.

Catawba Nuclear Station, as part of its developing Alloy 600 Program, has compiled a list of locations in the primary coolant system that contain Alloy 600 or Alloy 690, either as base metal or weld metal. These locations are listed in spreadsheets attached to these RAI's. The spreadsheets detail the inspection techniques utilized for each location and the extent of coverage achieved during the last examination. The frequencies of the exams are dictated by ASME code requirements. The inspection techniques dictate whether insulation removal was necessary to provide access to the bare metal surface. Inspections for the Catawba Nuclear Station ISI plan are conducted per applicable procedures under the Duke Power Quality Assurance program. The program and applicable procedures govern the qualification requirements of the inspectors. Prior to May 2000, personnel were qualified and certified in accordance with ASNT SNT-TC-1A 1984 Edition. After May 2000, ANSI/ASNT CP-189, 1991 Edition has been used.

Reactor Vessel Closure Head Bare Metal Inspection

The Catawba Nuclear Station Unit 1 reactor vessel closure head was inspected in April 2002. The Catawba Unit 2 reactor vessel closure head is scheduled to be inspected at the refueling outage beginning in March, 2003. The technical basis for the reactor vessel closure head bare metal inspection is the information provided in EPRI report 1006269 *Visual Examination for Leakage of PWR Reactor Head Penetrations on Top of RPV Head* indicating that degradation (i.e., through wall cracking) of the CRD nozzle to reactor head weld will exhibit itself as boric acid deposits on the surface of the head at the nozzle penetration. Visual inspection is an effective method of determining whether weld cracking, if present, has progressed to the leakage stage. If discovered, evidence of leakage is further evaluated by the appropriate means, which could include NDE.

Visual inspection of the reactor head and CRD nozzles is performed per the controlling procedure during reactor shutdown conditions. Access to the areas of interest is provided by removing all head insulation and lifting the CRDM vent shroud. These openings provide line of sight access to the areas of interest, allowing direct visual inspection.

The entire outer surface of the reactor head is inspected, including all CRD nozzles and their penetrations through the head. Extent of coverage is 100 percent of the head and nozzle penetrations.

Reactor head inspections are currently performed by personnel qualified to perform VT-2 Visual examinations in accordance with the ASME Code. The inspectors are accompanied by experienced plant engineering personnel.

All insulation that would prevent access to the outer surface of the head and to the nozzle penetrations is removed.

Catawba Nuclear Station is working on developing future visual and non-visual inspection plans for the reactor vessel heads in accordance with NRC Bulletin 2002-02.

Other Walkdowns

In parallel with unit shutdown, a containment walkdown is performed. The purpose is to systematically inspect inside containment for identification of boric acid leaks. The procedure controlling this walkdown identifies specific primary system piping/vessel locations that are considered highly susceptible to leakage. This walkdown is performed at normal operating pressure and temperature, during which insulation is not removed. The scope of this walkdown inspection includes primary system piping and components that are in accessible areas. Personnel performing this walkdown are operations, maintenance, and radiation protection technicians, with engineering support to assess specific system leakage and any boric acid corrosion issues. Any identified system leakage and boric acid corrosion issues are addressed by the corrective action program.

RAI 2

Provide the technical basis for determining whether or not insulation is removed to examine all locations where conditions exist that could cause high concentrations of boric acid on pressure boundary surfaces or locations that are susceptible to primary water stress corrosion cracking (Alloy 600 base metal and dissimilar metal Alloy 82/182 welds). Identify the type of insulation for each component examined, as well as any limitations to removal of insulation. Also include in your response actions involving removal of insulation required by your procedures to identify the source of leakage when relevant conditions (e.g., rust stains, boric acid stains, or boric acid deposits) are found.

At Catawba Nuclear Station, insulation is removed from the components or piping whenever access to the bare metal surface is required. Access may be required to perform NDE, inspect for leakage or a leak path, or determine extent of condition of a potentially degraded component. Catawba Nuclear Station has removed insulation on like equipment to determine extent of condition whenever PWSCC of Alloy 600 base metal or Alloy 82/182 weld metal has been identified.

The type of insulation for the primary coolant system components at Catawba Nuclear Station is provided in the attached spreadsheet. The insulation is removed to investigate relevant conditions indicative of leakage. The investigative procedure is activated through the Fluid Leak Management program and requires an inspection of each leak source as well as target components of leakage. The procedure requires an inspection of each component for signs of corrosion and pitting. Specific instruction for insulation removal is not given in the procedure. However, the insulation would usually have to be removed to satisfy the intent of the procedure. If boron can be seen coming from openings in the insulation, the insulation will be removed to identify the source of the leak and to determine if the component under the insulation has been damaged.

There are no locations on the primary coolant system at Catawba Nuclear Station where insulation cannot be removed when an inspection is required. Although accessible, access to certain limited locations may not be practical for inspection due to time, difficulty and potential extensive dose associated with removal of such insulation.

RAI 3

Describe the technical basis for the extent and frequency of walkdowns and the method for evaluating the potential for leakage in inaccessible areas. In addition, describe the degree of inaccessibility, and identify any leakage detection systems that are being used to detect potential leakage from components in inaccessible areas.

During power operation, access to primary system components in lower containment is restricted. This includes access to the reactor vessel upper head, reactor coolant loops, reactor coolant pumps, pressurizer, and steam generators. When system leakage is suspected and inspection is warranted, limited containment access is possible into the containment pipe chase, cold leg accumulator rooms, and upper pressurizer cavity. Containment can be made more accessible for inspection when system leakage is suspected by reducing power level. Such inspections are not routinely performed due to the elevated radiation levels.

With the reactor at power, the primary means of detecting primary system leakage is the reactor coolant system water inventory balance calculation. In addition, monitoring of the containment floor and equipment sump, the containment ventilation unit condensate drain tank, and containment atmosphere radiation monitors, in accordance with technical specifications provides an additional indication of system leakage during the times when the equipment is inaccessible.

Routine walkdown inspections in lower containment are performed at normal operating temperature and pressure with the reactor shutdown. Reduced radiation levels allow access to most areas inside containment that are not accessible with the reactor at power. These walkdowns include the general locations of most Alloy 600 pressure boundary materials and dissimilar metal Alloy 82/182 welds that are susceptible to primary water stress corrosion cracking. Access to the reactor vessel shell located behind the biological shield wall is very limited. However, there is no Alloy 600/82/182 material in this region. Leakage is identified by the presence of boric acid crystals and deposits. Mirror insulation is not routinely removed for these inspections, unless the presence of boric acid deposits makes investigation of potential leakage necessary.

RAI 4

Describe the evaluations that would be conducted upon discovery of leakage from mechanical joints (e.g., bolted connections) to demonstrate that continued operation with the observed leakage is acceptable. Also describe the acceptance criteria that was established to make such a determination. Provide the technical basis used to establish the acceptance criteria. In addition,

a. if observed leakage is determined to be acceptable for continued operation, describe what inspection/monitoring actions are taken to trend/evaluate changes in leakage, or

b. if observed leakage is not determined to be acceptable, describe what corrective actions are taken to address the leakage.

Upon discovery of leakage of borated water from a mechanical joint, the typical action would be to repair the leak and assess the leak path for corrosion damage. These activities are conducted through the Fluid Leak Management Program. Should continued operation be necessary in the presence of leakage resulting in corrosion, engineering evaluations would be required to demonstrate that continued operation is acceptable. The acceptance criteria for justification of continued operation would be satisfying the requirements of the appropriate sections of the ASME Code. A structural evaluation would be performed of the potentially degraded joint in its projected worst state of degradation. The results of the evaluation would have to meet ASME Code requirements suspected and inspection is warranted, limited containment access is possible into the containment pipe chase, cold leg accumulator rooms, and upper pressurizer cavity. Containment can be made more accessible for inspection when system leakage is suspected by reducing power level. Such inspections are not routinely performed due to the elevated radiation levels.

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The technical basis for the acceptance criterion for piping at Catawba Nuclear Station is the ASME Code of record. Satisfying the ASME Code of record provides a sound basis for the structural integrity of mechanical joints in piping systems.

When evidence of leakage of borated water from a mechanical joint is discovered, the corrective action program and Fluid Leak Management Program are activated to evaluate and resolve the leakage. The Fluid Leak Management Program requires that procedural actions be taken to determine the source of the leakage and identify the complete leak path. Technical procedures require recording the path of the leak and clean up of any boron deposits. If evidence of corrosion damage exists, engineering evaluation is required to determine if repair is required.

If observed leakage is acceptable for continued operation, a plan for inspection/ monitoring to trend/evaluate changes in leakage is created as required by the Fluid Leak Management Program. When leakage is identified, the location is entered into a leak management database. An inspection frequency is established based on the specific characteristics, location, leak path and risk of collateral damage associated with the leak. Each inspection is documented in the leak management database. This program manages active leakage until the leak is repaired.

If observed leakage is determined not to be acceptable, corrective actions are implemented to identify the source of leakage, determine the leak path prior to removal of the evidence, and conduct an engineering evaluation if evidence of corrosion damage exists. The cause of the leak is resolved, and any collateral damage evaluated to assure compliance with the ASME code. The leak resolution and the results of engineering evaluations are documented within the corrective action program.

RAI 5

Explain the capabilities of your program to detect the low levels of reactor coolant pressure boundary leakage that may result from through-wall cracking in the bottom reactor pressure vessel head incore instrumentation nozzles. Low levels of leakage may call into question reliance on visual detection techniques or installed leakage detection instrumentation, but has the potential for causing boric acid corrosion. The NRC has had a concern with the bottom reactor pressure vessel head incore instrumentation nozzles because of the high consequences associated with loss of integrity of the bottom head nozzles. Describe how your program would evaluate evidence of possible leakage in this instance. In addition, explain how your program addresses leakage that may impact components that are in the leak path.

A water inventory balance program monitors reactor coolant system leakage with established limits for continued operation specified in technical specifications. More restrictive plant procedure limits have been established to perform an assessment of potential leak sources whenever unidentified system leakage exceeds 0.15 gallons per minute. If leakage exceeded the more restrictive limit and the leak source remained unidentified, plant management would be required to assess the need for a plant shutdown or what additional leak investigation activities may be required.

The area beneath the bottom reactor pressure vessel head is monitored by radiation monitors for airborne radioactivity. Increased radioactivity levels detected by the radiation monitor may also indicate an RCS pressure boundary leak.

Lower containment ventilation units (LCVU) and the incore instrumentation ventilation units are inspected every refueling outage. If boron deposits are identified in the air handling unit, the discovery will be evaluated through the corrective action program to determine the source of the boron. In addition, the area below the reactor vessel lower head including the incore instrument nozzles is inspected during the system leakage test prior to unit start-up. The inspection is conducted with the insulation on.

RAI 6

Explain the capabilities of your program to detect the low levels of reactor coolant pressure boundary leakage that may result from through-wall cracking in certain components and configurations for other small diameter nozzles. Low levels of leakage may call into question reliance on visual detection techniques or installed leakage detection instrumentation, but has the potential for causing boric acid corrosion. Describe how your program would evaluate evidence of possible leakage in this instance. In addition, explain how your program addresses leakage that may impact components that are in the leak path.

A water inventory balance program monitors reactor coolant system leakage with established limits for continued operation specified in technical specifications. This program will measure any leakage from components and small diameter nozzles as unidentified system leakage, which has an established operational limit of less than 1.0 gallons per minute. More restrictive plant procedures require an assessment of potential leak sources whenever unidentified system leakage exceeds 0.15 gallons per minute. If leakage exceeds the more restrictive limit and the leak source remains unidentified, plant management would assess what additional leak investigation activities may be required, or if plant shutdown is necessary.

If leakage increases based on the mass balance, limited walk downs at power would be performed to identify the source of the leakage. In addition, the many walkdowns described previously help identify leakage from other small diameter nozzles during shut downs and refueling outages. Evidence of leakage would be evaluated and resolved through the corrective action program and Fluid Leak Management Program.

The area beneath the bottom reactor pressure vessel head is monitored by radiation monitors for airborne radioactivity. Increased radioactivity levels detected by the radiation monitor may also indicate an RCS pressure boundary leak

Lower containment ventilation units (LCVU) and the incore instrumentation ventilation units are inspected every refueling outage. If boron deposits are identified in the air handling unit, the discovery will be evaluated through the corrective action program to determine the source of the boron.

RAI 7

Explain how any aspects of your program (e.g., insulation removal, inaccessible areas, low levels of leakage, evaluation of relevant conditions) make use of susceptibility models or consequence models.

A list of all locations containing Alloy 600/82/182 has been compiled. The locations have been ranked in susceptibility to PWSCC based on operating temperature. A comprehensive program to manage Alloy 600/82/182 is under development at Catawba Nuclear Station. The Alloy 600 program will utilize susceptibility and consequence models to determine the proper actions required to manage the Alloy 600/82/182 locations.

RAI 8

Provide a summary of recommendations made by your reactor vendor on visual inspections of nozzles with Alloy 600/82/182 material, actions you have taken or plan to take regarding vendor recommendations, and the basis for any recommendations that are not followed.

Westinghouse fabricated the Catawba Nuclear Station NSSS system. No vendor recommendations have been received. A December 13, 2002 letter from the Westinghouse Owners Group stated that Westinghouse has reviewed its databases and communications to determine what past recommendations were made to owners of Westinghouse NSSSs on visual inspections of Alloy 600/82/182 materials in the reactor coolant pressure boundary. The detailed review of this information that may have contained such recommendations did not identify any Westinghouse recommendations on visual inspections of Alloy 600/82/182 locations.

RAI 9

Provide the basis for concluding that the inspections and evaluations described in your responses to the above questions comply with your plant Technical Specifications and Title 10 of the Code of Federal Regulations (10 CFR), Section 50.55(a), which incorporates Section XI of the American Society of Mechanical Engineers (ASME) Code by reference. Specifically, address how your boric acid corrosion control program complies with ASME Section XI, paragraph IWA-5250 (b) on corrective actions. Include a description of the procedures used to implement the corrective actions.

10 CFR 50.55a Codes and Standards

Catawba Nuclear Station is in compliance with 10 CFR 50.55a and code compliance criteria in IWB-5250. Table IWB-2500-1 of Section XI examination category B-P requires a VT-2 examination of the RCS during the system leakage test. Technical procedures that control the system leakage test require corrective action in accordance IWA-5250 (b) to identify the source of leakage and evaluate areas of general corrosion whenever boric acid residues are detected on components. This is currently accomplished prior to startup. In addition other procedurally controlled activities conducted to detect boric acid leakage or corrosion require corrective actions that are consistent with the requirements of IWA-5250 (b).

Periodic inspections are also required and conducted in accordance with the ISI plan. These inspections are conducted in accordance with Table IWB-2500-1 of Section XI of the ASME Code.

Technical Specifications

At Catawba Nuclear Station, 3.4.13 RCS Operational LEAKAGE is limited by Technical Specification 3.4.13 to:

a. No pressure boundary LEAKAGE;
b. 1 gpm unidentified LEAKAGE;
c. 10 gpm identified LEAKAGE;
d. 576 gallons per day total primary to secondary LEAKAGE through all steam generators (SGs); and
e. 150 gallons per day primary to secondary LEAKAGE through any one SG.

These limits are applicable in operational modes 1 through 4.

The inspection and operational leakage programs outlined in the above responses provide reasonable assurance of reactor pressure boundary integrity and compliance with Technical Specification 3.4.13 limits.

Catawba Nuclear Station
Unit 1
Alloy 600/690 Locations

Component	Item (quantity)	Type of ISI[1,2]	Extent of Coverage	Frequency	Insulation Type[3]
PZR	Surge Nozzle	UT/PT	100%	Once per interval	Metal Reflective
PZR	Spray Nozzle	UT/PT	100%	Once per interval	Metal Reflective
PZR	Relief Nozzle	UT/PT	100%	Once per interval	Metal Reflective
PZR	Safety Nozzles (3)	UT/PT	77.3%	Once per interval	Metal Reflective
PZR	Surge Nozzle Thermal Sleeve	N/A	N/A	N/A	N/A
PZR	Spray Nozzle Thermal Sleeve	N/A	N/A	N/A	N/A
PZR	Manway Insert	N/A	N/A	N/A	N/A
RV	Primary Outlet Nozzles (4)	UT/PT	100%	Once per interval	Blanket
RV	Primary Inlet Nozzles (4)	UT/PT	100%	Once per interval	Blanket
RV	78 CRDM nozzles (53 CRDM, 20 unused and 5 thermocouple nozzles)	Visual (see RAI #1)	Complete	To be determined	Metal Reflective
RV	Upper CRDM Welds [nozzle to adapter] (78)	Visual (see RAI #1)	Complete	To be determined	Metal Reflective
RV	RPV Head to UHI tube [lower] (4)	UT/PT	100%	Once per interval	Metal Reflective
RV	RPV Head to UHI tube [upper - nozzle to tube] (4	UT/PT	100%	Once per interval	Metal Reflective
RV	BMI Tubes [nozzles] (58)	None	None	None	Metal Reflective
RV	Inner Monitor Tube	None	None	None	Metal Reflective
RV	Outer Monitor Tube	None	None	None	Metal Reflective
RV	Head Vent	Visual (see RAI #1)	Complete	To be determined	Metal Reflective
SG	Primary Inlet Nozzles (4)	UT/PT	75% - Limited[5]	Once per interval	Blanket
SG	Primary Outlet Nozzles (4)	UT/PT	75% - Limited[5]	Once per interval	Blanket
SG	Primary Manway Cover Plate/Diaphragms (4)	N/A	N/A	N/A	N/A
SG	Primary Manway Cover Plate/Diaphragms (4)	N/A	N/A	N/A	N/A
SG	Handhole Diaphragms (4)	N/A	N/A	N/A	N/A
SG	Small Nozzles [3/4"] (8)	None	None	None	Blanket
SG	3" recirculation nozzle/ blowdown nozzles (4)	None	None	None	Blanket
SG	Main Feedwater Nozzle Safe Ends (4)	None	None	None	Blanket
SG	Auxiliary Feedwater Nozzle Safe End s (4)[4]	UT/PT	75%	Once per interval	Blanket
RCS	Cold leg Accumulator Nozzles (4)	None	None	None	Metal Reflective
RCS	Cold leg Accumulator Sample Taps (4)	None	None	None	Metal Reflective

Notes

1 Entire RCS pressure boundary is subject to a system leakage test prior to startup.

2 Personnel qualifications - Prior to May 2000 personnel were qualified and certified in accordance with ASNT SNT-TC-1A 1984 Edition. After May 2000, ANSI/ASNT CP-189, 199 Edition has been used.

3 There are no locations on the primary coolant system at CatawbaNuclear Station where insulation cannot be removed when an inspection is required.

4 1 of 4 required for examination per ASME Section XI.

5 Examinations conducted in a previous interval that did not achieve 100% of the required inspection were noted as "Limited." The exact "extent of condition" was not readily available.

Catawba Nuclear Station
Unit 2
Alloy 600/690 Locations

Component	Item (quantity)	Type of ISI[1,2]	Extent of Coverage	Frequency	Insulation Type[3]
PZR	Surge Nozzle	UT/PT	100%	Once Per Interval	Metal Reflective
PZR	Spray Nozzle	UT/PT	100%	Once Per Interval	Metal Reflective
PZR	Relief Nozzle	UT/PT	100%	Once Per Interval	Metal Reflective
PZR	Safety Nozzle (3)	UT/PT	100%	Once Per Interval	Metal Reflective
PZR	Surge Nozzle Thermal Sleeve	N/A	N/A	N/A	N/A
PZR	Spray Nozzle Thermal Sleeve	N/A	N/A	N/A	N/A
PZR	Manway Insert	N/A	N/A	N/A	N/A
RV	Primary Outlet Nozzles (4)	UT/PT	100%	Once Per Interval	Blanket
RV	Primary Inlet Nozzles (4)	UT/PT	100%	Once Per Interval	Metal Reflective
RV	78 CRDM nozzles (53 CRDM, 20 unused and 5 thermocouple nozzles)	Visual (see RAI #1)	Complete	To be determined	Metal Reflective
RV	RPV Head to UHI tube [lower] (4)	UT/PT	100%	Once Per Interval	Metal Reflective
RV	RPV Head to UHI tube [upper - nozzle to tube] (4)	UT/PT	100%	Once Per Interval	Metal Reflective
RV	Upper CRDM Welds [nozzle to adapter] (78)	Visual (see RAI #1)	Complete	To be determined	Metal Reflective
RV	BMI Tubes [nozzles] (58)	None	None	None	Metal Reflective
RV	Inner Monitor Tube	None	None	None	Metal Reflective
RV	Outer Monitor Tube	None	None	None	Metal Reflective
RV	Head Vent	Visual (see RAI #1)	Complete	To be determined	Metal Reflective
SG	Primary Chamber Drain Nozzle and Coupling (3)	Visual	Complete	Each RFO	Metal Reflective
RCS	Cold leg Accumulator Nozzles (4)	None	None	None	Metal Reflective
RCS	Cold leg Accumulator Sample Tap (4)	None	None	None	Metal Reflective

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Enclosure II
McGuire Nuclear Station
Response to NRC Bulletin 2002-01 RAI

RAI 1

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McGuire Nuclear Station relies on multiple methods to determine the condition of Alloy 600 pressure boundary material, including base metal and Alloy 82/182 welds that connect to the reactor coolant pressure boundary. These methods consist of code examinations to detect degradation, various walkdowns for the purpose of detecting borated water leakage, and specific inspections to address recent industry issues. The following paragraphs discuss these primary methods.

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McGuire Nuclear Station Unit 1 implemented a Risk Informed ISI Program in May of 2001 in accordance with Request for Relief 01-005. Under the Risk Informed Program, the expert panel changed all Alloy 600 locations to high safety significant due to potential PWSCC concerns.

Other connections containing Alloy 600/82/182 that the code does not require to be inspected volumetrically are inspected by area surveillance conducted as part of the ASME Code required VT-2 that occurs at operating pressure and temperature during plant startup. All Class 1 piping is within the inspection scope, including the reactor vessel upper head, the incore instrument nozzles on the lower reactor vessel head, and other areas subject to recent maintenance. A VT-2 examination method in accordance with ASME Section XI is used during the system leakage test, which does not require that insulation be removed during inspection.

McGuire Nuclear Station, as part of its developing Alloy 600 Program, has compiled a list of locations in the primary coolant system that contain Alloy 600 or Alloy 690, either as base metal or weld metal. These locations are listed in spreadsheets attached to this RAI. The spreadsheets detail the inspection techniques utilized for ISI at each location and the extent of coverage achieved during the last examination. The frequencies of the exams are dictated by ASME code requirements. The inspection techniques dictate whether insulation removal was necessary to provide access to the bare metal surface. Inspections for the McGuire Nuclear Station ISI plan are conducted per applicable procedures under the Duke Power Quality Assurance program. The program and applicable procedures govern the qualification requirements of the inspectors. Prior to May 2000, personnel were qualified and certified in accordance with ASNT SNT-TC-1A 1984 Edition. After May 2000, ANSI/ANST CP-189, 1991 Edition has been used.

Reactor Vessel Closure Head Bare Metal Inspection

The McGuire Nuclear Station reactor vessel closure heads were inspected in March 2002 (Unit 2) and September 2002 (Unit 1) in response to NRC Bulletin 2002-01. During both inspections any insulation that obscured a clear view of the vessel head was removed prior to inspection. The inspections were performed in accordance with EPRI document # 1006899 *Visual Inspection for Leakage of PWR Reactor Head Penetrations on Top Head* using video cameras whenever possible and direct visual inspection for areas not accessible with a video camera. The scope of the inspections included the entire head surface area as well as the annular area 360 degrees around each penetration. Note that at one penetration location on the Unit 1 head and several penetration locations on the Unit 2 head, the annular area could not be observed because insulation support clamps around the penetration are located close to or on the head surface at these locations. In these cases the entire circumference of the clamps was inspected for evidence of boron. Both heads were found to be free of boron deposits with no evidence of penetration leakage, corrosion, or wastage. Due to the low susceptibility of the McGuire reactor vessels to penetration cracking, head inspections are currently not planned for the next refueling outages. McGuire Nuclear Station is working on developing future visual and non-visual inspection plans for the reactor vessel heads in accordance with NRC Bulletin 2002-02.

Other Walkdowns

In parallel with unit shutdown, a containment walkdown is performed. The purpose is to systematically inspect inside containment for identification of boric acid leaks. This walkdown is performed at approximately operating pressure and temperature, during which insulation is not removed. The scope of this walkdown inspection includes lower containment inside the crane wall, the pipe chase, and pressurizer cavity. Engineering personnel perform this walkdown. Any identified system leakage and boric acid corrosion issues are addressed by the corrective action program.

RAI 2

Provide the technical basis for determining whether or not insulation is removed to examine all locations where conditions exist that could cause high concentrations of boric acid on pressure boundary surfaces or locations that are susceptible to primary water stress corrosion cracking (Alloy 600 base metal and dissimilar metal Alloy 82/182 welds). Identify the type of insulation for each component examined, as well as any *limitations to removal of insulation. Also include in your response actions involving removal of insulation required by your procedures to identify the source of leakage when relevant conditions (e.g., rust stains, boric acid stains, or boric acid deposits) are found.*

Alloy 600 locations located on the Reactor Pressure Vessel closure head have been examined per technical procedure. Specific instructions for insulation removal were available to support procedure implementation.

Alloy 600/82/182 locations on piping and components included in the ISI Program have been volumetrically examined. Examinations were conducted per ISI methods and frequency as required by the applicable ASME Code. Insulation was removed when access to the bare metal surface was required to perform NDE. There have been no ISI inspection locations waived due to difficulties from insulation removal.

Relevant conditions indicative of boric acid leakage are investigated. The Fluid Leak Management Program requires that insulation be removed to reveal possible high concentrations of boric acid on pressure boundary surfaces that are susceptible to corrosion. However, insulation is not always removed when the source leak has been examined, and the leak is determined to be benign, and the pressure boundary surfaces where boron is deposited are stainless steel.

McGuire Nuclear Station piping containing borated water is insulated with metal reflective and fiberglass blanket insulation inside the polar crane wall, metal reflective only elsewhere in containment, and calcium silicate outside containment. The Reactor Pressure Vessel is insulated with metal reflective type insulation. The Reactor Pressure Vessel closure head is insulated with metal reflective and fiberglass blanket insulation. The pressurizer has metal reflective and fiberglass blanket insulation. The steam generator head has fiberglass blanket insulation.

There are no locations on the primary coolant system at McGuire Nuclear Station where insulation cannot be removed when an inspection is required. Although accessible, access to certain limited locations may not be practical for inspection due to time, difficulty and potential extensive dose associated with removal of such insulation.

The type of insulation for the primary coolant system components at McGuire Nuclear Station is provided in the attached spreadsheet.

RAI 3

Describe the technical basis for the extent and frequency of walkdowns and the method for evaluating the potential for leakage in inaccessible areas. In addition, describe the degree of inaccessibility, and identify any leakage detection systems that are being used to detect potential leakage from components in inaccessible areas.

During refueling outages, various walkdowns are performed by station personnel. Engineering performs a walkdown of lower containment, the pipe chase and the pressurizer cavity as part of the shutdown process. These walkdowns are controlled by technical procedure and the results documented per procedural requirements. These are general area walkdowns with special emphasis being given to the ECCS sump. Boric acid deposits are documented and entered into the Fluid Leak Management Program for corrective action. Any corrosion damage of carbon steel resulting from the leakage is evaluated and if necessary, repaired as dictated by the corrective action program.

Prior to start up, engineering performs a walkdown of lower containment, and the pressurizer cavity. Operations personnel also perform a walkdown of containment prior to start up. In addition to the code required VT-2, QA performs a walkdown of containment which includes the incore instrumentation nozzles on the lower reactor vessel head at full temperature and pressure. All leaks identified are corrected or evaluated prior to the unit proceeding with power operation.

During power operation at McGuire Nuclear Station, most of the primary system located inside the crane wall is considered inaccessible. For these inaccessible areas, a leakage calculation based on mass balance is used to identify the presence of leakage. When leakage is suspected, walkdowns are conducted in accessible areas to identify the source of the leakage.

If leakage cannot be located by the limited walkdowns of accessible areas, further actions are required to diagnose the cause of the leakage indications. These actions may include shutdown of the operating unit before the leakage level reaches technical specification limits. During a forced outage, if no cooldown is required, lower containment is inspected at full temperature using a controlling procedure. The pressurizer cavity is normally also inspected at this time. The specific areas inspected and extent of walkdowns is based on the magnitude of the leak indicated by the reactor coolant (NC) system leakage calculations and other indicators.

If warranted based on the leakage calculation and other indicators, limited containment walkdowns can be performed with the plant at power. Examples of inspections of areas that can occur at power include a pressurizer cavity inspection, pipe chase inspections, and various locations in the auxiliary building.

RAI 4

Describe the evaluations that would be conducted upon discovery of leakage from mechanical joints (e.g., bolted connections) to demonstrate that continued operation with the observed leakage is acceptable. Also describe the acceptance criteria that was established to make such a determination. Provide the technical basis used to establish the acceptance criteria. In addition,

a. if observed leakage is determined to be acceptable for continued operation, describe what inspection/monitoring actions are taken to trend/evaluate changes in leakage, or

b. if observed leakage is not determined to be acceptable, describe what corrective actions are taken to address the leakage.

Upon discovery of leakage of borated water from a mechanical joint, the typical action would be to repair the leak and assess the leak path for corrosion damage. These activities are conducted through the Fluid Leak Management Program. Should continued operation be necessary in the presence of leakage resulting in corrosion, engineering evaluations would be required to demonstrate that continued operation is acceptable. The acceptance criteria for justification of continued operation would be satisfying the requirements of the appropriate sections of the ASME Code. A structural evaluation would be performed of the potentially degraded joint in its projected worst state of degradation. The results of the evaluation would have to meet ASME Code requirements in order to justify continued operation. The evaluation would be documented in the station corrective action program.

The technical basis for the acceptance criterion for piping at McGuire Nuclear Station is the ASME Code of record. Satisfying the ASME Code of record provides a sound basis for the structural integrity of mechanical joints in piping systems.

When evidence of leakage of borated water from a mechanical joint is discovered, the corrective action program and Fluid Leak Management Program are activated to evaluate and resolve the leakage. The Fluid Leak Management Program requires that procedural actions be taken to determine the source of the leakage and identify the complete leak path. Technical procedures require recording the path of the leak, and clean up of any boron deposits. If evidence of corrosion damage exists, engineering evaluation is required to determine if repair is required.

If observed leakage is acceptable for continued operation, a plan for inspection/ monitoring to trend/evaluate changes in leakage is created as required by the Fluid Leak Management Program. When leakage is identified, the location is entered into a leak management database. An inspection frequency is established based on the specific characteristics, location, leak path and risk of collateral damage associated with the leak. Each inspection is documented in the leak management database. This program manages active leakage until the leak is repaired.

If observed leakage is determined not to be acceptable, corrective actions are implemented to identify the source of leakage, determine the leak path prior to removal of the evidence, and conduct an engineering evaluation if evidence of corrosion damage exists. The cause of the leak is resolved, and any collateral damage evaluated to assure compliance with the ASME code. The leak resolution and the results of engineering evaluations are documented within the corrective action program.

RAI 5

Explain the capabilities of your program to detect the low levels of reactor coolant pressure boundary leakage that may result from through-wall cracking in the bottom reactor pressure vessel head incore instrumentation nozzles. Low levels of leakage may call into question reliance on visual detection techniques or installed leakage detection

instrumentation, but has the potential for causing boric acid corrosion. The NRC has had a concern with the bottom reactor pressure vessel head incore instrumentation nozzles because of the high consequences associated with loss of integrity of the bottom head nozzles. Describe how your program would evaluate evidence of possible leakage in this instance. In addition, explain how your program addresses leakage that may impact components that are in the leak path.

McGuire Nuclear Station uses mass balance to detect low levels of reactor coolant leakage that may not be visible or otherwise detectable. The NC system leakage calculation performs a mass balance calculation to determine the total, identified and unidentified leakage. All of the leakage detection methods are required to be capable of detecting a 1.0 gpm leak rate within one hour. The mass balance calculation is capable of detecting much lower levels of leakage with steady state conditions. More restrictive plant procedure limits have been established to perform an assessment of potential leak sources whenever unidentified system leakage exceeds 0.15 gallons per minute.

The area beneath the bottom reactor pressure vessel head is monitored by EMFs for airborne radioactivity. Increased radioactivity levels detected by the EMF may also indicate an RCS pressure boundary leak.

The area below the reactor vessel lower head including the incore instrument nozzles is inspected during the system leakage test prior to unit start-up. The inspection is conducted with the insulation on. In addition, McGuire Nuclear Station is evaluating inspection requirements for the reactor vessel bottom heads and associated instrumentation nozzles as part of the developing Alloy 600 Program.

RAI 6

Explain the capabilities of your program to detect the low levels of reactor coolant pressure boundary leakage that may result from through-wall cracking in certain components and configurations for other small diameter nozzles. Low levels of leakage may call into question reliance on visual detection techniques or installed leakage detection instrumentation, but has the potential for causing boric acid corrosion. Describe how your program would evaluate evidence of possible leakage in this instance. In addition, explain how your program addresses leakage that may impact components that are in the leak path.

McGuire Nuclear Station uses mass balance to detect low levels of reactor coolant leakage that may not be visible or otherwise detectable. The NC system leakage calculation performs a mass balance calculation to determine the total, identified and unidentified leakage. All of the leakage detection methods are required to detect a 1.0 gpm leak rate within one hour. The mass balance calculation is capable of detecting much lower levels of leakage with steady state conditions. More restrictive plant procedure limits have been established to perform an assessment of potential leak sources whenever unidentified system leakage exceeds 0.15 gallons per minute.

If leakage increases based on the mass balance, limited walkdowns at power are performed to identify the source of the leakage. Examples where limited walkdowns in containment have allowed McGuire Nuclear Station personnel to identify the source of leakage include a NC system safety valve, PORV block valve packing leaks and vent and drain valves not fully closed.

The area beneath the bottom reactor pressure vessel head is monitored by EMFs for airborne radioactivity. Increased radioactivity levels detected by the EMF may also indicate an RCS pressure boundary leak.

In addition, the many walkdowns described previously help identify leakage during shut downs and refueling outages. These walkdowns assist in identifying leakage from other small diameter nozzles. Evidence of leakage is evaluated and resolved through the corrective action program and Fluid Leak Management Program previously described.

RAI 7

Explain how any aspects of your program (e.g., insulation removal, inaccessible areas, low levels of leakage, evaluation of relevant conditions) make use of susceptibility models or consequence models.

A list of all locations containing Alloy 600/82/182 has been compiled. The locations have been ranked in susceptibility to PWSCC based on operating temperature. A comprehensive program to manage Alloy 600/82/182 is under development at McGuire Nuclear Station. The Alloy 600 program will utilize susceptibility and consequence models to determine the proper actions required to manage the Alloy 600/82/182 locations.

RAI 8

Provide a summary of recommendations made by your reactor vendor on visual inspections of nozzles with Alloy 600/82/182 material, actions you have taken or plan to take regarding vendor recommendations, and the basis for any recommendations that are not followed.

Westinghouse fabricated the McGuire Nuclear Station NSSS system. No vendor recommendations have been received. A December 13, 2002 letter from the Westinghouse Owners Group stated that Westinghouse has reviewed its databases and communications to determine what past recommendations were made to owners of Westinghouse NSSSs on visual inspections of Alloy 600/82/182 materials in the reactor coolant pressure boundary. The detailed review of this information that may have contained such recommendations did not identify any Westinghouse recommendations on visual inspections of Alloy 600/82/182 locations.

RAI 9

Provide the basis for concluding that the inspections and evaluations described in your responses to the above questions comply with your plant Technical Specifications and Title 10 of the Code of Federal Regulations (10 CFR), Section 50.55(a), which *incorporates Section XI of the American Society of Mechanical Engineers (ASME) Code by reference. Specifically, address how your boric acid corrosion control program complies with ASME Section XI, paragraph IWA-5250 (b) on corrective actions. Include a description of the procedures used to implement the corrective actions.*

10 CFR 50.55a Codes and Standards

McGuire Nuclear Station is in compliance with 10 CFR 50.55a and code compliance criteria in IWB-5250. Table IWB-2500-1 of Section XI examination category B-P requires a VT-2 examination of the RCS during the system leakage test. Technical procedures that control the system leakage test require corrective action in accordance IWA-5250 (b) to identify the source of leakage and evaluate areas of general corrosion whenever boric acid residues are detected on components. This is currently accomplished prior to startup. In addition procedurally controlled activities conducted to detect boric acid leakage or corrosion require corrective actions that are consistent with the requirements of IWA-5250 (b).

Periodic inspections are also required and conducted in accordance with the ISI plan. These inspections are conducted in accordance with Table IWB-2500-1 of Section XI of the ASME Code.

Technical Specifications

At McGuire Nuclear Station RCS Operational LEAKAGE is limited by Technical Specification 3.4.13 to:

a. No pressure boundary LEAKAGE;
b. 1 gpm unidentified LEAKAGE;
c. 10 gpm identified LEAKAGE;
d. 576 gallons per day total primary to secondary LEAKAGE through all steam generators (SGs); and
e. 150 gallons per day primary to secondary LEAKAGE through any one SG.

These limits are applicable in operational modes 1 through 4.
The inspection and operational leakage programs outlined in the above responses provide reasonable assurance of reactor pressure boundary integrity and compliance with Technical Specification 3.4.13 limits.

McGuire Unit 1 Alloy 600/690 Locations

Component	Item (quantity)	Type of ISI[1,2]	Extent of Coverage	Frequency	Insulation Type[3]
PZR	Surge Nozzle	UT/PT	100%	Once per interval	Blanket
PZR	Spray Nozzle	UT/PT	100%	Once per interval	Blanket
PZR	Relief Nozzle	UT/PT	100%	Once per interval	Blanket
PZR	Safety Nozzles (3)	UT/PT	100%	Once per interval	Blanket
PZR	Surge Nozzle Thermal Sleeve	N/A	N/A	N/A	N/A
PZR	Spray Nozzle Thermal Sleeve	N/A	N/A	N/A	N/A
PZR	Manway Insert	N/A	N/A	N/A	N/A
RV	Primary Outlet Nozzles (4)	UT/PT	100%	Once per interval	Metal Reflective
RV	Primary Inlet Nozzles (4)	UT/PT	100%	Once per interval	Metal Reflective
RV	53 CRDM nozzles 8 part-length nozzles 12 capped 5 thermocouple nozzles	Visual (see RAI #1)	Complete	To Be Determined	Metal Reflective
RV	Upper CRDM Welds [nozzle to adapter] (78)	Visual (see RAI #1)	Complete	To Be Determined	Uninsulated
RV	RPV Head to UHI Tubes [lower] (4)	UT/PT	89.1%	Once per interval	Metal Reflective
RV	RPV Head to UHI Tubes [upper - nozzle to tube] (4)	UT/PT	82.7%	Once per interval	Metal Reflective
RV	BMI Tubes [nozzles] (58)	None	None	None	Metal Reflective
RV	Inner Monitor Tube	None	None	None	Uninsulated
RV	Outer Monitor Tube	None	None	None	Uninsulated
RV	Vent Tube	Visual (see RAI #1)	Complete	To Be Determined	Metal Reflective
SG	Primary Inlet Nozzles (4)	UT/PT	48.6% - 75.0%	Once per interval	Blanket
SG	Primary Outlet Nozzles (4)	UT/PT	47.3%-75.0%	Once per interval	Blanket
SG	Primary Manway Cover Plate/Diaphragms (4)	N/A	N/A	N/A	N/A
SG	Primary Manway Cover Plate/Diaphragms (4)	N/A	N/A	N/A	N/A
SG	Handhole Diaphragms (4)	N/A	N/A	N/A	N/A
SG	Small Nozzles [3/4"] (8)	None	None	None	Blanket
SG	3" Recirculation Blowdown Nozzles (4)	None	None	None	Blanket
SG	Main Feedwater Nozzle Safe Ends (4)	None	None	None	Blanket
SG	Auxiliary Feedwater Nozzle Safe Ends (4)	UT	Not scheduled	Once per interval	Blanket
RCS	Cold leg Accumulator Nozzles (4)	None	None	None	Uninsulated
RCS	Cold leg Sample Taps (4)	None	None	None	Blanket

Notes

1 Entire RCS pressure boundary is subject to system pressure test prior to start-up.

2 Personnel qualifications - Prior to May 2000 personnel were qualified and certified in accordance with ASNT SNT-TC-1A 1984 Edition. After May 2000, ANSI/ASNT CP-189, 1991 Edition has been used.

3 There are no locations on the primary coolant system at McGuire Nuclear Station where insulation cannot be removed when an inspection is required.

McGuire Unit 2 Alloy 600/690 Locations

Component	Item (quantity)	Type of ISI[1,2]	Extent of Coverage	Frequency	Insulation Type[3]
PZR	Surge Nozzle	UT/PT	100%	Once Per Interval	Blanket
PZR	Spray Nozzle	UT/PT	100%	Once Per Interval	Blanket
PZR	Relief Nozzle	UT/PT	100%	Once Per Interval	Blanket
PZR	Safety Nozzles (3)	UT/PT	100%	Once Per Interval	Blanket
PZR	Surge Nozzle Thermal Sleeve	N/A	N/A	N/A	N/A
PZR	Spray Nozzle Thermal Sleeve	N/A	N/A	N/A	N/A
PZR	Manway Insert	N/A	N/A	N/A	N/A
RV	Primary Outlet Nozzles (4)	UT/PT	100%	Once Per Interval	Metal Reflective
RV	Primary Inlet Nozzles (4)	UT/PT	97.9%	Once Per Interval	Metal Reflective
RV	53 CRDM nozzles 8 part-length nozzles 12 capped 5 thermocouple nozzles	Visual (see RAI#1)	Complete	To Be Determined	Metal Reflective
RV	Upper CRDM Welds [nozzle to adapter] (78)	Visual (see RAI #1)	Complete	To Be Determined	Uninsulated
RV	RPV Head to UHI Tubes [lower] (4)	UT/PT	100%	Once Per Interval	Metal Reflective
RV	RPV Head to UHI Tubes [upper - nozzle to tube] (4)	UT/PT	100%	Once Per Interval	Metal Reflective
RV	Vent Tube	Visual (see RAI #1)	Complete	To Be Determined	Metal Reflective
RV	BMI Tubes [nozzles] (58)	None	None	None	Metal Reflective
RV	Inner Monitor Tube	None	None	None	Uninsulated
RV	Outer Monitor Tube	None	None	None	Uninsulated
SG	Primary Inlet Nozzles (4)	UT/PT	75% - Limited[4]	Once Per Interval	Metal Reflective
SG	Primary Outlet Nozzles (4)	UT/PT	75% - Limited[4]	Once Per Interval	Metal Reflective
SG	Primary Manway Cover Plate/Diaphragms (4)	N/A	N/A	N/A	N/A
SG	Primary Manway Cover Plate/Diaphragms (4)	N/A	N/A	N/A	N/A
SG	Handhole Diaphragms (4)	N/A	N/A	N/A	N/A
SG	Small Nozzles [3/4"] (8)	None	None	None	Blanket
SG	3" Recirculation Blowdown Nozzles (4)	None	None	None	Blanket
SG	Main Feedwater Nozzle Safe Ends (4)	None	None	None	Blanket
SG	Auxiliary Feedwater Nozzle Safe Ends (4)[5]	UT/PT	Not Sched	Once per Interval	Blanket
RCS	Cold leg Accumulator Nozzles (4)	None	None	None	Uninsulated
RCS	Cold leg Accumulator Sample Taps (4)	None	None	None	Blanket

Notes:

1 Entire RCS pressure boundary is subject to system pressure test prior to start-up.

Personnel qualifications - Prior to May 2000 personnel were qualified and certified in accordance with ASNT SNT-TC-1A 1984 Edition.

2 After May 2000, ANSI/ASNT CP-189, 1991 Edition has been used.

3 There are no locations on the primary coolant system at McGuire Nuclear Station where insulation cannot be removed when an inspection is required.

Examinations conducted in a previous interval that did not achieve 100% of the required inspection were noted as "Limited." The

4 specific "extent of condition" was not readily available.

5 These welds were not included in sample selected for code %s.

Enclosure III
Oconee Nuclear Station
Response to NRC Bulletin 2002-01 RAIs

RAI 1

Provide detailed information on, and the technical basis for, the inspection techniques, scope, extent of coverage, and frequency of inspections, personnel qualifications, and degree of insulation removal for examination of Alloy 600 pressure boundary material and dissimilar metal Alloy 82/182 welds and connections in the reactor coolant pressure boundary (RCPB). Include specific discussion of inspection of locations where reactor coolant leaks have the potential to come in contact with and degrade the subject material (e.g., reactor pressure vessel (RPV) bottom head).

Oconee Nuclear Station relies on multiple methods to determine the condition of Alloy 600 pressure boundary material, including base metal and Alloy 82/182 welds that connect to the reactor coolant pressure boundary. These methods consist of code examinations to detect degradation, various walkdowns for the purpose of detecting borated water leakage, and specific inspections to address recent industry issues. The following paragraphs discuss these primary methods.

ASME Code In-service Inspection

Oconee Nuclear Station conducts In-service Inspections (ISI) in accordance with the applicable ASME Section XI code. As part of a comprehensive ISI program, those connections with the RCPB that contain Alloy 600/82/182 and are greater than 1" in diameter for Class 1 receive a surface examination once per ten-year interval. Class 1 piping equal to or greater than 4" in diameter receives a surface and volumetric examination once per ten-year interval.

Oconee Nuclear Station performs augmented inspections on several Alloy 600 primary system components. Currently, augmented surface inspections are performed once per interval for the pressurizer level tap and sample tap safe ends and the hot leg and cold leg RTE mounting bosses. The HPI Nozzle to Safe-End Welds (both Normal and Emergency Injection) are UT inspected every other outage.

There are no elective inspections performed for Alloy 600 components.

Other connections containing Alloy 600 that the code does not require to be inspected volumetrically are inspected by area surveillance conducted as part of the ASME code required VT-2 that occurs at operating pressure and temperature during plant startup.

Oconee Nuclear Station, as part of its developing Alloy 600 Program, has compiled a list of locations in the primary coolant system that contains Alloy 600. These locations are listed in the attached spreadsheets. The spreadsheets detail the inspection techniques utilized for each location and the extent of coverage achieved during the last examination. The frequencies of the exams are dictated by ASME code requirements. The inspection techniques dictate whether insulation was removed to provide access to the bare metal surface. Inspections for the Oconee Nuclear Station ISI plan are conducted per applicable procedures under the Duke Power Quality Assurance program. The program and applicable procedures govern the qualification requirements of the inspectors. Prior to May 2000, personnel were qualified and certified in accordance with ASNT SNT-TC-1A 1984 Edition. After May 2000, ANSI/ASNT CP-189, 1991 Edition has been used.

Reactor Vessel Closure Head Bare Metal Inspection

The technical basis for the reactor vessel closure head bare metal inspection is that degradation (i.e., through wall cracking) of the Oconee Nuclear Station CRD nozzle to reactor head Alloy 600 weld or base metal will typically exhibit itself as boric acid deposits on the surface of the head at the nozzle penetration. Visual inspection is an effective method of identifying possible leakage from these locations. Any identified leakage would then be further evaluated by the appropriate means, which could include NDE. These evaluations are conducted and documented under the corrective action program to conclusively identify the source of leakage as well as to characterize any flaws found in base or weld material in accordance with Oconee commitments to the Alloy 600 CRDM NRC Bulletins.

Visual inspection of the reactor head and CRD nozzles is performed during reactor shutdown conditions. This inspection is controlled by a technical procedure. Access to these areas is through specifically designed and installed openings in the reactor head service structure. The openings provide line of sight access to the areas of interest, allowing direct visual inspection. Additional optical and electronic tooling aids are used to access those limited areas that are not directly visible due to intervening nozzles.

The entire outer surface of the upper reactor head is inspected, including all CRD nozzles and their penetrations through the head. Extent of coverage is 100 percent of the head and nozzle penetrations. Bare metal inspections of the reactor vessel closure head are performed each refueling outage.

Reactor head inspections are currently performed by personnel qualified to perform Level II Visual Inspections in accordance with the ASME Code and assisted by experienced plant engineering personnel. The individuals performing the inspections are familiar with the construction of the reactor vessel head / CRD interface and are knowledgeable of and familiar with the symptoms of borated water leakage, as well as the detrimental effects of such leakage.

The insulation internal to the head service structure provides sufficient clearance for complete visual inspection. There is no other insulation affecting access to the head surface.

The most recent reactor vessel closure head bare metal inspection, conducted on Oconee Nuclear Station Unit 2 during the Fall 2002 RFO, was supplemented by volumetric NDE.

Control Rod Drive Mechanism Flange Inspections

The technical basis for the control rod drive mechanism flange inspections is that deterioration of the CRDM to CRD nozzle flange joint will exhibit itself as boric acid deposits at the flanged connection. Visual inspection is an effective method of determining whether leakage has occurred through the mechanical joint. This inspection looks for evidence of leakage through the flange faces, around the threads of the bolts and split ring, and around the inside diameter of the split ring. Any leakage at the bi-metallic weld would also be evident. If discovered, the leakage is repaired, or alternately for minor flange leakage, evaluated for acceptability for continued service. The evaluations include the size and extent of the boron build-up; whether leaking was previously present and has sealed itself off during system operation or is repetitive or increasing; and whether leakage is onto carbon or alloy steel components. Results of the flange inspection are entered into the corrective action program. Any joints determined to be not acceptable for continued service will be repaired. Leaks evaluated as acceptable for continued service will be re-inspected during the following refueling outage, when they would again be evaluated for acceptability for service

Remote visual inspection of the CRDM to CRD nozzle joint is performed during reactor shutdown conditions. Access to the areas of interest is obtained by specially designed optical tooling inserted through the working surface of the service structure platform.

The scope of the inspection consists of inspecting all CRDM to CRD nozzle flange joints for leakage. The outside diameter of the flanged joint is inspected, as is the visible portions of the underside of the nozzle flange, split ring and bolts. This inspection is performed each refueling outage.

The CRDM to CRD flange inspections are performed using experienced plant maintenance and engineering personnel. The procedure used for the inspection specifies the areas to be inspected and reporting requirements. The individuals performing the inspection are familiar with the construction of the CRDM to CRD nozzle flange joint and are knowledgeable of and familiar with the of symptoms of borated water leakage, as well as the detrimental effects of such leakage.

There is no insulation present on the CRDM to CRD nozzle flange joints that prevents access to the areas of interest.

Other Walkdowns

Engineering performs walkdowns during refueling outages whenever selected Alloy 600/82/182 locations are made accessible by outage activities. During a refueling outage various walkdowns and inspections are performed in the reactor building looking for any indications of leakage. Although these inspections focus primarily on identifying boric acid leaks, special attention is given to the attachments to the reactor coolant system (RCS) pressure boundary where the Alloy 600/82/182 is generally located. These walkdowns are documented as part of the Alloy 600 Program. These inspections are conducted in the following manner.

1. Any component with the insulation removed due to other outage activity is inspected looking for leakage and corrosion damage to the component. This inspection focuses on looking for boric acid leakage onto carbon steel.
2. Any accessible opening in the insulation is inspected looking for boron accumulations or flow paths left by dried boron. Any boron deposits found are evaluated to determine the source of the boron. Insulation may be removed to facilitate the inspection of the underlying component for damage.
3. For any leak identified, a work request is written to have the boron cleaned up and the leak repaired. Leaks that cannot be repaired immediately are evaluated to assure the component and surrounding equipment will remain structurally sound and that the leakage rate will be acceptable until such time as the leak can be repaired.

RAI 2

Provide the technical basis for determining whether or not insulation is removed to examine all locations where conditions exist that could cause high concentrations of boric acid on pressure boundary surfaces or locations that are susceptible to primary water stress corrosion cracking (Alloy 600 base metal and dissimilar metal Alloy 82/182 welds). Identify the type of insulation for each component examined, as well as any limitations to removal of insulation. Also include in your response actions involving removal of insulation required by your procedures to identify the source of leakage when relevant conditions (e.g., rust stains, boric acid stains, or boric acid deposits) are found.

At Oconee Nuclear Station, insulation is removed from the components or piping whenever access to the bare metal surface is required. Access may be required to perform NDE, inspect for leakage or a leak path, or determine the extent of condition whenever PWSCC of Alloy 600 base metal or Alloy 82/182 weld metal has been identified.

The insulation on Oconee Nuclear Station RCS pressure boundary components is metal reflective. The investigative procedure is activated through the Fluid Leak Management program and requires an inspection of each leak source as well as target components of leakage. The procedure requires an inspection of each component for signs of corrosion and pitting. Specific instruction for insulation removal is not given in the procedure. However, the insulation would usually have to be removed to satisfy the intent of the procedure. If boron can be seen coming from openings in the insulation, the insulation will be removed to identify the source of the leak and to determine if the component under the insulation has been damaged. If the insulation fits loosely enough, the insulation may not require removal to determine that the piping beneath is free of boron accumulations.

There are no locations on the primary coolant system at Oconee Nuclear Station where insulation cannot be removed when an inspection is required.

RAI 3

Describe the technical basis for the extent and frequency of walkdowns and the method for evaluating the potential for leakage in inaccessible areas. In addition, describe the degree of inaccessibility, and identify any leakage detection systems that are being used to detect potential leakage from components in inaccessible areas.

Accessibility to certain areas is dependent on the operational mode of the plant and the associated radiological conditions. The more inaccessible areas are limited to inspection during plant shutdowns. Walkdowns and visual inspections are the primary method for determining specific sources of leakage identified by the RCS leakage calculations, as well as for identifying leakage below detectable limits. Routine reactor building entries and walkdowns occur during plant operation and plant shutdowns.

Reactor building entries during operation are scheduled quarterly, but may occur more frequently if indicators such as sump level/pump-out frequency, reactor coolant leakage rate calculation, or area radiation monitors provide reason to believe a leak exists. The areas inspected during these reactor building entries are outside the secondary shield wall and selected areas in the reactor building basement. The inspection from the reactor building basement focuses on looking for evidence of leakage in the steam generator cavities. Areas that are inaccessible during plant operation include the steam generator cavities themselves, the area around and under the reactor vessel, and the reactor vessel head. Radiological conditions cause these areas to be inaccessible.

Multiple reactor building entries for inspection purposes occur during refueling outages.

After the reactor is shutdown station personnel perform a walkdown of the entire reactor building, except the area around and under the reactor vessel. This area is excluded due to radiation dose. This inspection is looking for any indications of leaks, i.e. boron deposits or rust stains. The results of this inspection are documented in the controlling walkdown procedure. Any evidence of boric acid leakage is documented for entry into the Fluid Leak Management Program and the corrective action program for evaluation and resolution.

Prior to startup multiple inspections are performed by station personnel of the reactor building, except the area around and under the reactor vessel. This area is excluded due to radiation dose. Some of the inspections are controlled by procedure for direction and documentation. However any indication of leakage is entered into the Fluid Leak Management Program and the corrective action program for evaluation and resolution.

Inaccessible Areas

Due to physical obstacles and radiological concerns, the bottom of the reactor vessel is considered an inaccessible area for inspection. Limited access to the bottom of the reactor vessel is possible, but only during certain shutdown conditions. Radiation dose rates and contamination levels are major concerns. Access to the bottom of the reactor vessel is also limited by many physical obstructions.

The bottom the reactor vessel is totally inaccessible during plant operation. This is also true during shutdown conditions when the Incore Monitoring System instrumentation is withdrawn (the normal condition during refueling outages), since both these operational modes create Very High Radiation Area (> 500 Rad/hr) conditions beneath the reactor vessel.

Any leakage which could occur in these inaccessible areas is monitored by and included within the RCS leakage calculations.

RAI 4

Describe the evaluations that would be conducted upon discovery of leakage from mechanical joints (e.g., bolted connections) to demonstrate that continued operation with the observed leakage is acceptable. Also describe the acceptance criteria that was established to make such a determination. Provide the technical basis used to establish the acceptance criteria. In addition,

a. if observed leakage is determined to be acceptable for continued operation, describe what inspection/monitoring actions are taken to trend/evaluate changes in leakage, or

b. if observed leakage is not determined to be acceptable, describe what corrective actions are taken to address the leakage.

Upon discovery of leakage of borated water from a mechanical joint, the typical action would be to repair the leak and assess the leak path for corrosion damage. These activities are conducted through the Fluid Leak Management Program. Should continued operation be necessary in the presence of leakage resulting in corrosion, engineering evaluations would be required to demonstrate that continued operation is acceptable. The acceptance criteria for justification of continued operation would be satisfying the requirements of the appropriate sections of the ASME Code. A structural evaluation would be performed of the potentially degraded joint in its projected worst state of degradation. The results of the evaluation would have to meet ASME Code requirements in order to justify continued operation. The evaluation would be documented in the station corrective action program.

The technical basis for the acceptance criterion for piping at Oconee Nuclear Station is the ASME Code of record. Satisfying the ASME Code of record provides a sound basis for the structural integrity of mechanical joints in piping systems.

When evidence of leakage of borated water from a mechanical joint is discovered, the corrective action program and Fluid Leak Management Program are activated to evaluate and resolve the leakage. The Fluid Leak Management Program requires that procedural actions be taken to determine the source of the leakage and identify the complete leak path. Technical procedures require recording the path of the leak and clean up of any boron deposits. If evidence of corrosion damage exists, engineering evaluation is required to determine if repair is required. Work orders are written to correct the cause of the leak.

If observed leakage is acceptable for continued operation, a plan for inspection/ monitoring to trend/evaluate changes in leakage is created as required by the Fluid Leak Management Program. When leakage is identified, the location is entered into a leak management database. An inspection frequency is established based on the specific characteristics, location, leak path and risk of collateral damage associated with the leak. Each inspection is documented in the leak management database. This program manages active leakage until the leak is repaired.

If observed leakage is determined not to be acceptable, corrective actions are implemented to identify the source of leakage, determine the leak path prior to removal of the evidence, and conduct an engineering evaluation if evidence of corrosion damage exists. The cause of the leak is resolved, and any collateral damage evaluated to assure compliance with the ASME code. The leak resolution and the results of engineering evaluations are documented within the corrective action program.

RAI 5

Explain the capabilities of your program to detect the low levels of reactor coolant pressure boundary leakage that may result from through-wall cracking in the bottom reactor pressure vessel head incore instrumentation nozzles. Low levels of leakage may call into question reliance on visual detection techniques or installed leakage detection instrumentation, but has the potential for causing boric acid corrosion. The NRC has had a concern with the bottom reactor pressure vessel head incore instrumentation nozzles because of the high consequences associated with loss of integrity of the bottom head nozzles. Describe how your program would evaluate evidence of possible leakage in this instance. In addition, explain how your program addresses leakage that may impact components that are in the leak path.

Pressure boundary leaks from the Oconee Nuclear Station RCS are monitored in several ways. The Operations group performs a reactor coolant leakage calculation on a daily basis. Typically this is done by running a leakage program on the plant computer for a duration of 2 to 3 hours each evening. The program performs mass calculations for the RCS, Pressurizer, Letdown Storage Tank and Quench Tank. RCS daily leakage values typically average between 0.05 and 0.10 gpm. The accuracy of the calculation has been demonstrated through testing to be on the order of 0.05 gpm when run for 2 hours or greater. Procedurally a second leakage calculation is required if the leakage rate was initially calculated to be greater than 0.2 gpm. If a leakage rate of greater than 0.2 gpm is verified by the second calculation the procedure initiates a primary system leak identification procedure.

RCS leakage may also be identified by the following activities:

1) Monitoring and trending of RCS leakage rate is performed by the RCS system engineer. Five day and 30 day averages are calculated.

2) Reactor Building Normal Sump (RBNS) rates are trended daily. An increase in the sump collection rate may indicate the presence of an RCS leak.

3) Frequent chemical analyses of the RBNS inventory may indicate an RCS pressure boundary leak if the boron concentrations begin to increase.

4) Reactor Building radiation monitors (RIA-47, 48 and 49) are trended weekly. Increasing radioactivity in the Reactor Building may also indicate a RCS pressure boundary leak.

Additionally, the Reactor Building Cooling Unit (RBCU) component engineer has been instructed to look for evidence of boric acid corrosion waste products. Boric acid corrosion products may coat the RBCU cooling coils, possibly resulting in a decrease in cooling efficiency. The RBCU component engineer has been made aware of these possibilities, and precautions have been added as appropriate to the affected operating/test procedures.

Any leakage in this area beneath the RV lower head would be collected and contained in the building sump drainage system. Identified leakage is evaluated and resolved through the corrective action program and Fluid Leak Management program. Usually RCS leakage > 0.2 gpm is found and corrected. Technical Specifications require any RCS pressure boundary leakage to be corrected.

RAI 6

Explain the capabilities of your program to detect the low levels of reactor coolant pressure boundary leakage that may result from through-wall cracking in certain components and configurations for other small diameter nozzles. Low levels of leakage may call into question reliance on visual detection techniques or installed leakage detection instrumentation, but has the potential for causing boric acid corrosion. Describe how your program would evaluate evidence of possible leakage in this instance. In addition, explain how your program addresses leakage that may impact components that are in the leak path.

Pressure boundary leaks from the Oconee Nuclear Station RCS are monitored in several ways. The Operations group performs a reactor coolant leakage calculation on a daily basis. Typically this is done by running a leakage program on the plant computer for a duration of 2 to 3 hours each evening. The program performs mass calculations for the RCS, Pressurizer, Letdown Storage Tank and Quench Tank. RCS daily leakage values typically average between 0.05 and 0.10 gpm. Procedurally a second leakage calculation is required if the leakage rate was initially calculated to be greater than 0.2 gpm. If a leakage rate of greater than 0.2 gpm is verified by the second calculation the procedure initiates a primary system leak identification procedure.

RCS leakage may also be identified by the following activities:

1) Monitoring and trending of RCS leakage rate is performed by the RCS system engineer. Five day and 30 day averages are calculated.

2) Reactor Building Normal Sump (RBNS) rates are trended daily. An increase in the sump collection rate may indicate the presence of an RCS leak.

3) Frequent chemical analyses of the RBNS inventory may indicate an RCS pressure boundary leak if the boron concentrations begin to increase.

4) Reactor Building radiation monitors (RIA-47, 48 and 49) are trended weekly. Increasing radioactivity in the Reactor Building may also indicate a RCS pressure boundary leak.

Additionally, the Reactor Building Cooling Unit (RBCU) component engineer has been instructed to look for evidence of boric acid corrosion waste products. Boric acid corrosion products may coat the RBCU cooling coils, possibly resulting in a decrease in cooling efficiency. The RBCU component engineer has been made aware of these possibilities, and precautions have been added as appropriate to the affected operating/test procedures.

Any leakage in the Reactor Building would be collected and contained in the building sump drainage system. Identified leakage is evaluated and resolved through the corrective action program and Fluid Leak Management program. Usually RCS leakage > 0.2 gpm is found and corrected. Technical Specifications require any RCS pressure boundary leakage to be corrected.

In addition, the many walkdowns described previously help identify leakage during shut downs and refueling outages. These walkdowns would assist in identifying leakage from other small diameter nozzles. Evidence of leakage would be evaluated and resolved through the corrective action program and Fluid Leak Management Program previously described.

RAI 7

Explain how any aspects of your program (e.g., insulation removal, inaccessible areas, low levels of leakage, evaluation of relevant conditions) make use of susceptibility models or consequence models.

The Oconee Nuclear Station Alloy 600 Program currently uses Primary Water Stress Corrosion Cracking (PWSCC) susceptibility models as the primary method to identify the component locations that are most susceptible to PWSCC attack. In addition, Oconee Nuclear Station is developing consequence models from the Program Plan for Alloy 600 PWSCC Life Cycle Management. The goal is to use this program to identify the consequences of specific plant component failures that are significant contributors to plant core melt frequency. Oconee Nuclear Station plans to use a combination of these two models to identify and rank the Alloy 600 component locations with the most overall risk to plant safety. The Alloy 600 Program will then use this combined risk model to determine the proper actions required to manage the Alloy 600 locations.

RAI 8

Provide a summary of recommendations made by your reactor vendor on visual inspections of nozzles with Alloy 600/82/182 material, actions you have taken or plan to take regarding vendor recommendations, and the basis for any recommendations that are not followed.

Framatome ANP provided a letter dated October 21, 2002, recommending that Oconee Nuclear Station conduct bare metal visual inspection of the 52 incore monitoring instrumentation nozzles on the bottom of the Oconee Nuclear Station Unit 2 reactor head during the Fall 2002 refueling outage. Oconee Nuclear Station personnel conducted a general visual inspection of the area underneath the vessels, without the removal of insulation, during the last Unit 1 and Unit 2 refueling outages. A complete visual inspection of the bottom surface of the lower head insulation, the incore instrument piping, and the penetrations of the piping through the insulation was performed on Unit 2. No evidence of leakage was discovered on either unit.

Oconee Nuclear Station elected to perform field walk downs, in a controlled manner, to determine the preparations required to perform such a bare metal visual inspection. Inspections of the bottom reactor pressure vessel head and work in this area involves a first of a kind evolution in a high dose field. Oconee is working to identify the necessary actions that will improve coverage of our future visual inspections.

RAI 9

Provide the basis for concluding that the inspections and evaluations described in your responses to the above questions comply with your plant Technical Specifications and Title 10 of the Code of Federal Regulations (10 CFR), Section 50.55(a), which incorporates Section XI of the American Society of Mechanical Engineers (ASME) Code by reference. Specifically, address how your boric acid corrosion control program complies with ASME Section XI, paragraph IWA-5250 (b) on corrective actions. Include a description of the procedures used to implement the corrective actions.

<u>10 CFR 50.55a Codes and Standards</u>

Oconee Nuclear Station is in compliance with 10 CFR 50.55a and code compliance criteria in IWB-5250. Table IWB-2500-1 of Section XI examination category B-P requires a VT-2 examination of the RCS during the system leakage test. Technical procedures that control the system leakage test require corrective action in accordance with IWA-5250 (b) to identify the source of leakage and evaluate areas of general corrosion whenever boric acid residues are detected on components. This is currently accomplished prior to startup. In addition, procedurally controlled activities conducted to detect boric acid leakage or corrosion require corrective actions that are consistent with the requirements of IWA-5250 (b).

Periodic inspections are also required and conducted in accordance with the ISI plan. These inspections are conducted in accordance with Table IWB-2500-1 of Section XI of the ASME Code.

Technical Specifications

The current limiting condition of operation (LCO) for Oconee Nuclear Station, Technical Specification 3.4.13, requires that RCS operational LEAKAGE be limited to no pressure boundary LEAKAGE;

- 1 gpm unidentified LEAKAGE;
- 10 gpm identified LEAKAGE;
- 300 gallon per day total primary to secondary LEAKAGE through all steam generators (SGs)
- and 150 gallon per day primary to secondary leakage through any one SG.

These limits are applicable in operational modes 1 through 4.

The inspection and operational leakage programs outlined in the above responses provide reasonable assurance of reactor pressure boundary integrity and compliance with Technical Specification 3.4.13 limits.

Oconee Unit 1 Alloy 600 Locations

Component	Item (quantity)	Type of ISI (UT, PT or WD) [1,2&3]	Extent of Coverage [4]	Frequency (Inter=Interval)	Degree of Insulation Removal [5]	Insulation Type [6]
RV	CRDM Nozzle	VT2/UT	100%	Each Outage	No removal necessary	metal, but not directly at joint
PZR	Pzr Heater Sleeve	WD	Note 3	Each Outage	Note 5	metal, but is missing several pieces
PZR	Pzr Level Tap Safe Ends (6)	PT WD	100% 100% Note 3	Augm't 1/Inter Each Outage	100% 100% Note 5	metal
PZR	Pzr Sample Tap Safe End	PT WD	100% Note 3	Augm't 1/Inter Each Outage	100% Note 5	metal
PZR	4" Spray Nozzle Safe End	PT/UT	100%	1/ Inter	100%	metal
PZR	Pzr Vent Nozzle	WD	100%	Each Outage	100%	metal, insulation removed each outage.
PZR	Vent Nozzle J-Groove Weld	None	None	N/A	Note 5	None, inside PZR
PZR	Spray Nozzle Extension Pin	None	None	N/A	N/A	None, connection inside pressurizer.
PZR	Pzr Diaphram Plate	None	None	N/A	N/A	metal over the manway which is on top of plate.
PZR	10 in Surge Nozzle Safe End Weld	PT/UT	100%	1/ Inter	100%	metal
PZR	2.5 in Pressure Relief Nozzle Welds (3)	PT	100%	1/ Inter	100%	metal
RCS	HL Flow Meter Nozzle Safe End	WD	Note 3	Each Outage	Note 5	metal
PZR	Pzr Vent Nozzle-Pipe FP Weld	WD	Note 3	Each Outage	Note 5	metal
PZR	Thermowell Nozzle	WD	Note 3	Each Outage	Note 5	metal
RCS	HL Pressure Tap Safe End	WD	Note 3	Each Outage	Note 5	metal
RCS	HL Vent Safe End	WD	Note 3	Each Outage	Note 5	metal
RCS	HL Temperature Connections	WD	Note 3	Each Outage	Note 5	metal
RCS	HL Flow Meter Ring	None	None	N/A	N/A	metal
RCS	HL Impulse Nozzle	None	None	N/A	N/A	metal
RCS	10 in Surge Nozzle Weld [butter & pipe to butter]	PT/UT (both welds)	100%/Limited 100%/96.09%	1/ Interval (both welds)	100% 100%	metal
RCS	HL RTE Mounting Bosses (6)	PT	100%	Augm't 1/Inter	100%	metal
RV	CRDM Nozzle Body Weld [below flange]	PT	100%	1 / Inter	100%	none
RCS	UCL Pressure Tap Safe End	WD	Note 3	Each Outage	Note 5	metal

Oconee Unit 1 Alloy 600 Locations

Component	Item (quantity)	Type of ISI (UT, PT or WD) [1,2 & 3]	Extent of Coverage [4]	Frequency (Inter=Interval)	Degree of Insulation Removal [5]	Insulation Type [6]
RCS	LCL Pressure Tap Safe End	WD	Note 3	Each Outage	Note 5	metal
RCS	LCL Drain Nozzle Safe Ends (3)	PT WD	100% Note 3	1 / Interval Each Outage	100% Note 5	metal, insulation removed each outage to place lead shielding.
RCS	Letdown Nozzle Safe End [Dwg says 2.5" Drain]	PT WD	100% Note 3	1/ Interval Each Outage	100% Note 5	metal, insulation removed each outage to place lead shielding.
RCS	LCL Temperature Connections	WD	Note 3	Each Outage	Note 5	metal
RV	Decay Heat Nozzle Weld [butter & pipe to butter]	PT/UT PT/UT	100%/90.25% 100%	1/ Interval 1/ Interval	100% 100%	metal, inspected if insulation is removed
RV	28 in Coolant Inlet Piping Welds (4)	PT/UT	100%	1/ Interval	100%	metal
RV	Modified 28 in Inlet Discharge Pipe Welds (4)	PT/UT	100%/Limited	1/ Interval	100%	metal
RCS	HPI/Emergency Nozzle Welds (2)	PT(Code), UT (Augmented) - both WD	100% 100% Note 3	Code I / Inter, Augm't UT every other outage (both)	100% 100% Note 5	metal, insulation removed each outage.
RCS	HPI/Makeup Nozzle Welds (2)	PT(Code), UT (Augmented) - both WD	100% 100% Note 3	Code I / Inter, Augm't UT every other outage (both)	100% 100% Note 5	metal, insulation removed each outage.
RV	Core Flood Nozzle (RV) Safe End Welds (2)	UT WD	86%-81% Note 4	1 / Interval Each Outage	100% Note 5	metal, visual from outside of Primary Shield Wall through 5 ft long penetration.
RV	Monitor Tap Weld	None	None	N/A	None	metal
RCS	LCL RTE Mounting Bosses (4)	PT	100%	Augm't 1 per Interval (all)	100%	metal
RV	Modified Instrument Penetration	WD	Note 3	Each Outage	Note 5	metal, but a short distance away
RV	Core Guide Lug	VT1	100%	1/ Interval	N/A	none
RCS	1 in Nozzle Safe End	None	N/A	N/A	N/A	none
RCS	2" Pressure Relief Nozzle Safe End	WD	100%	Each Outage	N/A	none
RCS	14 in Outlet Nozzle Weld	WD	100%	Each Outage	N/A	none
SG	1" Primary Drain Nozzle	WD	Note 3	Each Outage	No removal necessary	metal, none directly on drain line, SG support stand lined with metal insulation
SG	Tube-Tubesheet Weld	None	None	N/A	N/A	none

Oconee Unit 1 Alloy 600 Locations

Component	Item (quantity)	Type of ISI (UT, PT or WD) [1,2 & 3]	Extent of Coverage [4]	Frequency (Inter=Interval)	Degree of Insulation Removal [5]	Insulation Type [6]
Notes:						
1	The entire RCS pressure boundary is subject to a system leakage test prior to startup.					
2	Personnel qualifications - Prior to May 2000 personnel were qualified and certified in accordance with ASNT SNT-TC-1A 1984 Edition. After May 2000, ANSI/ASNT CP-189, 1991 Edition has been used.					
3	The Walk down is performed by knowledgeable engineer, very experienced in finding & identifying Boron Leakage. No extent of coverage details have been previously documented.					
4	For two or more examinations, the extent of coverage data (if different) is in the same order as the examination, i.e. if the examinations are "PT/UT", coverage data is PT first, UT second. The extent of coverage for WD examinations varies. Missing insulation and insulation gaps are used to directly view the component, when practical.					
5	Insulation is completely removed for Code Interval and Augmented ISI Examinations. Walkdowns take advantage of insulation removal, but also look inside insulation gaps with insulation installed.					
6	There are no locations on the primary coolant system at Oconee Nuclear Station where insulation cannot be removed when an inspection is required. Insulation within the Reactor Vessel Head Service Structure does not need to be removed to perform inspections on primary coolant system components.					

Oconee Unit 2 Alloy 600 Locations

Component	Item (quantity)	Type of ISI (UT, PT or WD) [1,2 & 3]	Extent of Coverage [4]	Frequency (Inter=Interval)	Degree of Insulation Removal [5]	Insulation Type [6]
RV	CRDM Nozzle	VT2/UT	100%	Each Outage	No removal necessary	metal, but not directly at joint
PZR	Pzr Level Tap Safe Ends (6)	PT WD[2]	100% Note 3	Augm't 1/Inter Each Outage	100% Note 5	metal
PZR	Pzr Sample Tap Safe End	PTWD	100% Note 3	Augm't 1/ Inter Each Outage	100% Note 5	metal
PZR	4" Spray Nozzle Safe End	PT/UT	100%/Limited	1/ Inter	100%	metal
PZR	Pzr Vent Nozzle	VT2	100%	Each Outage	100%	metal, insulation removed each outage.
PZR	Vent Nozzle J-Groove Weld	None	None	N/A	Note 5	None, inside PZR
PZR	Spray Nozzle Extension Pin	Visual - Video dated 2/12/88.	Inside Surface	N/A	N/A	none
PZR	10 in Surge Nozzle Safe End Weld	PT/UT	100%	1/ Inter	N/A	metal
PZR	2.5 in Pressure Relief Nozzle Welds (3)	PT	100%	1/ Inter	100%	metal
RCS	HL Flow Meter Nozzle Safe End	WD	Note 3	Each Outage	Note 5	metal
PZR	Thermowell Nozzle	WD	Note 3	Each Outage	Note 5	metal
RCS	HL Pressure Tap Safe End	WD	Note 3	Each Outage	Note 5	metal
RCS	HL Vent Safe End	WD	Note 3	Each Outage	Note 5	metal
RCS	HL Temperature Connections	WD	Note 3	Each Outage	Note 5	metal
RCS	HL Flow Meter Ring	None	None	N/A	N/A	metal
RCS	HL Impulse Nozzle	None	None	N/A	N/A	metal
RCS	10 in Surge Nozzle Weld [butter & pipe to butter]	PT/UT (both)	100% 100%	1/ Interval (both)	100% 100%	metal
RCS	HL RTE Mounting Bosses (6)	PT	100%	Augm't 1/Inter	100%	metal
RV	CRDM Nozzle Body Weld (below flange)	PT	100%	1 / Inter	100%	none
RCS	UCL Pressure Tap Safe End	WD	Note 3	Each Outage	Note 5	metal
RCS	LCL Pressure Tap Safe End	WD	Note 3	Each Outage	Note 5	metal
RCS	1.5" LCL Drain Nozzle Safe Ends (3)	PT WD[2]	100% Note 3	1 / Interval Each Outage	100% Note 5	metal, insulation removed each outage to place lead shielding.

Oconee Unit 2 Alloy 600 Locations

Component	Item (quantity)	Type of ISI (UT, PT or WD) [1,2 & 3]	Extent of Coverage [4]	Frequency (Inter=Interval)	Degree of Insulation Removal [5]	Insulation Type [6]
RCS	Letdown Nozzle Safe End	PT WD	100% Note 3	1 / Interval Each Outage	100% Note 5	metal, insulation removed each outage to place lead shielding.
RCS	LCL Temperature Connections	WD	Note 3	Each Outage	Note 5	metal
RV	Decay Heat Nozzle Weld	PT/UT PT/UT	100% 100%	1 / Interval 1/ Interval	100% 100%	metal, inspected if insulation is removed.
RV	28 in Coolant Inlet (Top) Piping Welds (4)	PT/UT	100%	1 / Interval	100%	metal
RV	28 in Coolant Inlet (45) Pipe Welds (4)	PT/UT	100%/Limited	1 / Interval	100%	metal
RCS	HPI/Emergency Nozzle Welds (2)	PT(Code), UT (Aug) - both WD	100% 100% Note 3	Code I / Inter, Augm't UT every other outage (both)	100% 100% -- Note 5	metal, insulation removed each outage.
RCS	HPI/Makeup Nozzle Welds (2)	PT(Code), UT (Aug) - both WD	100% 100% Note 3	Code I / Inter, Augm't UT every other outage (both)	100% 100% -- Note 5	metal, insulation removed each outage.
RV	Core Flood Nozzle Safe End Welds (2)	UT WD	100% Note 3	1 / Interval Each Outage	100% Note 5	metal, visual from outside of Primary Shield Wall through 5 ft long penetration.
RV	Monitor Tap Weld	None	None	N/A	None	metal
RCS	LCL RTE Mounting Bosses (4)	PT	100%	Augm't 1 per Interval (all)	100%	metal
RV	Modified Instrument Penetration	WD	Note 3	Each Outage	Note 5	metal, but a short distance away
RV	Core Guide Lug	VT1	100%	1 / Interval	N/A	none
RCS	Safe End on Pressure & Level Sensing Nozzle	None	N/A	N/A	N/A	none
RCS	1 in Nozzle Safe End	None	N/A	N/A	N/A	none
RCS	2" Pressure Relief Nozzle Safe End	WD	100%	Each Outage	N/A	none
RCS	14 in Outlet Nozzle Weld	WD	100%	Each Outage	N/A	none
SG	1" Primary Drain Nozzle	WD	Note 3	Each Outage	No removal necessary	metal, none directly on drain line, SG support stand lined with metal insulation
SG	Tube-Tubesheet Weld	None	None	N/A	N/A	none

Oconee Unit 2 Alloy 600 Locations

Component	Item (quantity)	Type of ISI (UT, PT or WD) [1,2 & 3]	Extent of Coverage [4]	Frequency (Inter=Interval)	Degree of Insulation Removal [5]	Insulation Type [6]
Notes:						
1	The entire RCS pressure boundary is subject to a system leakage test prior to startup.					
2	Personnel qualifications - Prior to May 2000 personnel were qualified and certified in accordance with ASNT SNT-TC-1A 1984 Edition. After May 2000, ANSI/ASNT CP-189, 1991 Edition has been used.					
3	The Walk down is performed by knowledgeable engineer, very experienced in finding & identifying Boron Leakage. No extent of coverage details have been previously documented.					
4	For two or more examinations, the extent of coverage data (if different) is in the same order as the examination, i.e. if the examinations are "PT/UT", coverage data is PT first, UT second. The extent of coverage for WD examinations varies. Missing insulation and insulation gaps are used to directly view the component, when practical.					
5	Insulation is completely removed for Code Interval and Augmented ISI Examinations. Walkdowns take advantage of insulation removal, but also look inside insulation gaps with insulation installed.					
6	There are no locations on the primary coolant system at Oconee Nuclear Station where insulation cannot be removed when an inspection is required. Insulation within the Reactor Vessel Head Service Structure does not need to be removed to perform inspections on primary coolant system components.					

Oconee Unit 3 Alloy 600 Locations

Component	Item (quantity)	Type of ISI (UT, PT or WD) [1,2 & 3]	Extent of Coverage [4]	Frequency (Inter=Interval)	Degree of Insulation Removal [5]	Insulation Type [6]
RV	CRDM Nozzle	VT2/UT	100%	Each Outage	No removal necessary	metal, but not directly at joint
PZR	Pzr Level Tap Safe Ends (6)	PT WD[2]	100% Note 3	Augm't 1/Inter Each Outage	100% Note 5	metal
PZR	Pzr Sample Tap Safe End	PT WD	100% Note 3	Augm't 1/ Inter Each Outage	100% Note 5	metal
PZR	4" Spray Nozzle Safe End	PT/UT	100%	1/ Inter	100%	metal, insulation removed each outage.
PZR	Pzr 1" Vent Nozzle	WD	100%	Each Outage	100%	metal
PZR	Vent Nozzle J-Groove Weld	None	None	N/A	Note 5	None, Inside PZR
PZR	Spray Nozzle Extension Pin	None	None	N/A	N/A	none
PZR	10 in Surge Nozzle Safe End Weld	PT/UT	100%	1/ Inter	100%	metal
PZR	2.5 in Pressure Relief Nozzle Welds (3)	PT	100%	1/ Inter	100%	metal
RCS	HL Flow Meter Nozzle	WD	Note 3	Each Outage	Note 5	metal
PZR	Thermowell Nozzle	WD	Note 3	Each Outage	Note 5	metal
RCS	HL Pressure Tap Nozzle	WD	Note 3	Each Outage	Note 5	metal
RCS	HL Vent Nozzle	WD	Note 3	Each Outage	Note 5	metal
RCS	HL Temperature Connections	WD	Note 3	Each Outage	Note 5	metal
RCS	HL Flow Meter Ring	None	None	N/A	N/A	metal
RCS	HL Impulse Nozzle	None	None	N/A	N/A	metal
RCS	10 in Surge Nozzle Weld [nozzle butter & surge line to pipe butter]	PT/UT (both)	100%/Limited 100%	1/ Interval (both)	100% 100%	metal
RCS	HL RTE Mounting Bosses (6)	PT	100%	Augm't 1/Inter	100%	metal
RV	CRDM Nozzle Body Weld (below flange)	PT	100%	1 / Interval	100%	none
RCS	UCL Level (or Pressure) Tap Nozzle	WD	Note 3	Each Outage	Note 5	metal
RCS	LCL Pressure Tap Safe End	WD	Note 3	Each Outage	Note 5	metal
RCS	1.5" LCL Drain Nozzles (3)	PT WD	100% Note 3	1 / IntervalEach Outage	100%Note 5	metal, insulation removed each outage to place lead shielding.
RCS	Letdown Nozzle Safe End	PT WD	100% Note 3	1 / Interval Each Outage	100% Note 5	metal, insulation removed each outage to place lead shielding.
RCS	LCL Temperature Connections	WD	Note 3	Each Outage	Note 5	metal
RV	Decay Heat Nozzle Weld	PT/UT	100%/75%	1 / Interval	100%	metal, inspected if insulation is removed.
RV	28 in Coolant Inlet (Top) Piping Welds (4)	PT/UT	100%	1 / Interval	100%	metal
RV	28 in Coolant Inlet (45) Pipe Welds (4)	PT/UT	100%/Limited	1 / Interval	100%	metal

Oconee Unit 3 Alloy 600 Locations

Component	Item (quantity)	Type of ISI (UT, PT or WD) [1,2 & 3]	Extent of Coverage [4]	Frequency (Inter=Interval)	Degree of Insulation Removal [5]	Insulation Type [6]
RCS	HPI/Emergency Nozzle Welds (2)	PT(Code), UT (Aug) - both WD	100% 100% Note 3	Code I / Inter, Augm't UT every other outage (both)	100% 100% -- Note 5	metal, insulation removed each outage.
RCS	HPI/Makeup Nozzle Welds (2)	PT(Code), UT (Aug) - both WD	100% 100% Note 3	Code I / Inter, Augm't UT every other outage (both)	100% 100% -- Note 5	metal, insulation removed each outage.
RV	Core Flood Nozzle Safe End Weld (2)	UT WD	100% Note 3	1 / Interval Each Outage	100% Note 5	metal, visual from outside of Primary Shield Wall through 5 ft long penetration.
RCS	Monitor Tap Weld	None	None	N/A	None	metal
RCS	LCL RTE Mounting Bosses (4)	PT	100%	Augm't 1 per Interval (all)	100%	metal
RV	Modified Instrument Penetration	WD	None	None to date	Note 5	metal, but a short distance away
RV	Core Guide Lug	VT1	100%	1 / Interval	N/A	none
RCS	1 in Nozzles (2, 13)	None	N/A	N/A	N/A	none
RCS	1 in Nozzles (4, 14)	None	N/A	N/A	N/A	none
RCS	2" Pressure Relief Nozzle	WD	100%	Each Outage	N/A	none
RCS	14 in Outlet Nozzle Weld	WD	100%	Each Outage	N/A	none
SG	1" Primary Drain Nozzle	WD	Note 3	Each Outage	No removal necessary	metal, none directly on drain line, SG support stand lined with metal insulation
SG	Tube-Tubesheet Weld	None	None	N/A	N/A	none
Notes						
1	The entire RCS pressure boundary is subject to a system leakage test prior to startup.					
2	Personnel qualifications - Prior to May 2000 personnel were qualified and certified in accordance with ASNT SNT-TC-1A 1984 Edition. After May 2000, ANSI/ASNT CP-189, 1991 Edition has been used.					
3	The Walk down is performed by knowledgeable engineer, very experienced in finding & identifying Boron Leakage. No extent of coverage details have been previously documented.					
4	For two or more examinations, the extent of coverage data (if different) is in the same order as the examination, i.e. if the examinations are "PT/UT", coverage data is PT first, UT second. The extent of coverage for WD examinations varies. Missing insulation and insulation gaps are used to directly view the component, when practical.					
5	Insulation is completely removed for Code Interval and Augmented ISI Examinations. Walkdowns take advantage of insulation removal, but also look inside insulation gaps with insulation installed.					
6	There are no locations on the primary coolant system at Oconee Nuclear Station where insulation cannot be removed when an inspection is required. Insulation within the Reactor Vessel Head Service Structure does not need to be removed to perform inspections on primary coolant system components.					