FPL

Florida Power & Light Company, P. O. Box 14000, Juno Beach, FL 33408-0420

JAN 3 1 2003

L-2003-007
10 CFR 50.4

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington DC 20555

RE: Florida Power and Light Company
St. Lucie Units 1 and 2
Docket Nos. 50-335 and 50-389
Turkey Point Units 3 and 4
Docket Nos. 50-250 and 50-251

FPL Energy Seabrook, LLC
Seabrook Station
Docket No. 50-443

NRC Bulletin 2002-01
Request for Additional Information Response

On November 22, 2002, the NRC issued a request for additional information (RAI) regarding Bulletin (NRCB) 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity." Florida Power & Light Company (FPL), the licensee for the St. Lucie Nuclear Plant, Units 1 and 2, and the Turkey Point Nuclear Plant, Units 3 and 4, and FPL Energy Seabrook, LLC (FPLE Seabrook) the licensee for Seabrook Station hereby submit their responses to the RAI.

Attachment 1 provides the FPLE Seabrook response, and Attachments 2 and 3 provide the St. Lucie and Turkey Point site responses, respectively. As discussed in more detail in the Attachments, FPL and FPLE Seabrook continue to comply with plant Technical Specifications. Additionally, FPL and FPLE Seabrook continue to comply with 10 CFR 50.55a, which incorporates requirements of Section XI of the American Society of Mechanical Engineers (ASME) Code by reference. The NRC staff concluded that a comprehensive boric acid corrosion control program would exceed the current ASME Code requirements. The FPL and FPLE Seabrook responses demonstrate that the current programs both meet and exceed the ASME Code requirements for boric acid corrosion control.

FPL and FPLE Seabrook continue to work closely with the industry and will routinely review their boric acid corrosion control programs in light of plant-specific and industry experience.

Please contact us if you have any additional questions regarding these programs.

Very truly yours,

Rajiv S. Kundalkar
for J. A. Stall
Senior Vice President, Nuclear
And Chief Nuclear Officer

Attachments (3)

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ATTACHMENT 1

FPLE Seabrook Station
NRC Bulletin 2002-01
Response to NRC Request for Additional Information

REQUESTED INFORMATION

In response to NRC Bulletin 2002-01 request for additional information, FPLE Seabrook provides the following:

NRC Question 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).

Response:

Procedures described in this response (NRC Question 1) provide information on inspection technique scope, extent of coverage, frequency of inspections, personnel qualifications, and degree of insulation removal for examination of Alloy 600 pressure boundary material, dissimilar metal Alloy 82/182 welds, and connections in the Reactor Coolant Pressure Boundary (RCPB). The technical basis for adequacy of examination conduct is provided in the last paragraph.

Procedure MA10.1, *Station Leakage Programs,* provides overview of leakage programs at Seabrook Station. The boric acid component of this procedure includes direction to plant staff in the identification, processing, cleaning, and inspection/evaluation of boric acid deposits.

Procedure EX1801.002, *Leakage Reduction Surveillance,* requires periodic inspection and measurement of leakage from systems outside containment that may contain primary coolant. This surveillance is performed every eighteen (18) months by personnel familiar with observing leakage. A checklist in the procedure provides location and description of inspections. When a boric acid deposit is discovered, it is cleaned and a qualified individual makes initial metal loss determination. Should metal loss be apparent, a VT-2 qualified individual performs a documented evaluation. Insulation removal is not required for this inspection. Insulation removal may be necessary to determine leakage/boric acid point of origin or to fully evaluate metal loss.

Procedure EX1801.006, *Containment Leakage Reduction Program Surveillance,* requires periodic inspection and measurement of leakage from systems inside containment that contain primary coolant. This surveillance is performed each refueling outage by personnel familiar with observing leakage. A checklist in the procedure provides location and description of inspections. Locations include but are not limited to the area around the reactor vessel head, pressurizer, steam generator manways, reactor coolant pump cubicles, and reactor coolant system valves. Alloy 600 material and Alloy 82/182 welds are included within these locations. When a boric acid deposit is discovered, it is cleaned and a qualified individual makes initial metal loss determination. Should metal loss be apparent, a VT-2 qualified individual performs a documented evaluation. Insulation removal is not required for this inspection. Insulation removal may be necessary to determine leakage/boric acid point of origin or to fully evaluate metal loss.

Procedure EX1810.101, *Class 1 RC System ISI System Leakage Test*, is utilized to perform the system leakage examination at Normal Operating Pressure (NOP) as required by ASME Section XI, 1995 Edition through 1996 Addenda. This test is performed prior to plant startup following each reactor refueling outage and addresses those locations inside containment listed in Table A that are part of the RCPB (inclusive of Alloy 600 material and Alloy 82/182 welds). A VT-2 examination is performed after a 4-hour hold time without removal of insulation. Insulation removal may be necessary to determine leakage/boric acid point of origin or to fully evaluate metal loss.

In addition to the periodic inspections and examinations described above, System Engineers perform system walkdowns that include observing boric acid leakage /deposits. These walkdowns are performed on a quarterly basis with observations documented. Typically, System Engineers are certified VT-2 examiners to enhance their qualifications in this area.

FPLE Seabrook's Boric Acid Leakage Reduction Program is effective in identifying leaks from Alloy 600 pressure boundary material and dissimilar metal Alloy 82/182 welds and connections in the RCPB. The periodic inspections of borated systems described above, and the documented quarterly walkdowns performed by System Engineers, provide sufficient measures to detect and control boric acid deposition, to preclude wastage, and maintain integrity of the RCPB.

NRC Question 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.

Response:

FPLE Seabrook does not routinely remove insulation at locations that could be susceptible to primary water stress corrosion cracking (PWSCC). PWSCC requires the presence of high temperatures along with pure primary water, and tensile stress. PWSCC is a thermally activated process that follows an Arrhenius relationship where an increase in temperature results in a decrease in time to initiation of cracking or failure.

The following instances document insulation removal to inspect susceptible locations. During refueling outage OR07 in November 2000, insulation was removed from each of four (4) RCS hot leg nozzles to inspect the Alloy 82/182 safe end welds. These welds are enclosed with reflective metal type insulation. No evidence of leakage was observed. Although each of these welds was successfully exposed for inspection, access is available only through the reactor cavity seal ring, which is a high radiation area. Limitations in laydown space for insulation and personnel access also exist. During refueling outage OR08 in May 2002, a bare RPV head inspection was performed to assess the head surface and Alloy 600 penetrations. Panels in the reflective stepped insulation were removed to provide access for remote visual inspection equipment. With exception of peripheral CRDM penetrations, this access allowed robotic equipment to perform inspection of the RPV head surface and nozzles. Peripheral CRDM nozzles were inspected using a flexible boroscope fitted through a gap in the head shroud. Essentially 100% of the bare head was inspected with no indication of boric acid.

Alloy 600 and Alloy 82/182 welds depicted in Table A are located inside containment within the bio-shield enclosure wall. Radiation levels do not permit inspection during plant operation. As stated in the response to NRC Question 1, procedures EX1801.006, *Containment Leakage Reduction Program Surveillance,* and EX1810.101, *Class 1 RC System ISI System Leakage Test* inspect these susceptible locations without removal of insulation unless necessary to determine leakage/boric acid point of origin or to fully evaluate metal loss.

Plant personnel have little difficulty utilizing the procedures listed above in detecting boric acid leaks in quantities as small as a few ounces of boric acid. With insulation present, boric acid deposits may have to be more than a few ounces for detection. However, boric acid leakage that could potentially cause wastage, would be readily detectable because it needs to be present in sufficient quantity over a period of time. These quantities would most likely be identified from boric acid deposits around the insulation. FPLE Seabrook considers these inspection procedures adequate in ensuring cracking or failures at these locations would be identified well before they become a gross leakage concern.

NRC Question 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.

Response:

RCS leakage walkdowns are performed at the beginning of each refueling outage and prior to entering Mode 2. Areas of the RCPB are accessible and inspected during the walkdowns. These walkdowns are conducted during refueling outages as radiation exposure concerns preclude walkdowns during power operation.

During plant operations, Technical Specifications require operable leakage detection systems, which include radiation monitors, a sump level detection system, and periodic inventory balance capable of providing indication of primary system leakage prior to loss of RCS structural integrity. The reactor vessel head and flange are inspected during refueling operations. The head flange has its own leakage detection system consisting of a double O-ring seal monitored by temperature instrumentation. During power operation, the leakoff from the seals is monitored and will actuate a high temperature alarm in the control room to alert operators to the presence of leakage past the inner seal.

Technical Specifications require the performance of an RCS inventory balance every 72 hours. This inventory balance is automated and normally performed by the Main Plant Process Computer. An updated leak rate calculation is provided every 15 minutes. The calculated RCS unidentified leakage rate at Seabrook is typically less than 0.1 gpm and is prominently featured in the plant Daily Status Report by Operations. Should the leakage rate increase by a small fraction, an early warning alarm is provided and actions up to and including a walkdown of the RCS inside containment would be performed to identify the source of the increased leakage by the System Engineer and/or Operations.

FPLE Seabrook considers RCS walkdowns performed during refueling outages and leakage detection methods employed during power operation provide assurance of RCPB integrity in inaccessible areas.

NRC Question 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 were 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.

Response:

The Condition Reporting program is used to document conditions adverse to quality, such as significant material loss due to boric acid corrosion resulting from system leakage. This program typically requires determination of probable cause, corrective actions and actions to prevent recurrence. Leakage identified during the performance of an ASME Section XI pressure test is also documented on the Visual Examination Record VT-2. Whenever possible, the leakage amount is quantified for evaluation. A Work Order is also generated for corrective action.

If boric acid residue is detected on a component, the leakage source (point of origin) and any areas of material metal loss are identified as required by procedure MA 10.1, *Station Leakage Programs*. As stated in the VT-2 visual examination procedure, components with local areas of general corrosion that reduce the wall thickness by more than 5 percent are evaluated and documented in a Condition Report to determine whether the component is acceptable for continued service, or whether repair or replacement is required.

In accordance with ISI Request for Relief No. 2AR-03, Code case N-566-1, Corrective Action for Leakage Identified at Bolted Connections, when leakage is discovered at an ASME Class 1, 2, or 3 bolted connection by VT-2 visual examination during a system pressure test, a Condition Report is initiated and the bolting and component material are evaluated for joint integrity. If leakage is not corrected, the joint will be evaluated in accordance with ASME Code, Section XI, 1995 Edition through the 1996 Addenda, paragraph IWB-3142.4, Acceptance by Analytical Evaluation. This evaluation includes the following considerations to determine the susceptibility of the bolting to corrosion and failure.

- The number and service age of the bolting
- Bolt and component material
- Corrosiveness of process fluid
- Leakage location and system function
- Leakage history at connection or other system components
- Visual evidence of corrosion at connection (while connection is assembled)

Seabrook Station has implemented ASME Code Case N-616 (Relief Request 2AR-04) for performance of VT-2 visual examination at locations where corrosion resistant bolting is installed without removal of the insulation. If evidence of leakage is detected at locations where corrosive resistant bolting material is used, either by discovery of active leakage or evidence of boric acid crystals, the insulation will be removed and the bolted connection will be reexamined. If the evaluation determines that further examination is required, the bolt closest to the leak will be removed and VT-1 examined. The bolt will be evaluated in accordance with IWB-3517.1 of the ASME B&PV Code Section XI, 1995 Edition through 1996 Addenda.

In addition to the above program, Seabrook station's Team Management Manual (STMM) Chapter 2, section 7.7, *System Leakage Standard,* states that active leaks shall be corrected before plant startup from Mode 5 or shall be accepted by the Station Director. The policy ensures that structures, systems and components important to safety are in proper working order for safe operation.

NRC Question 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.

Response:

Procedures described previously, which comprise the boric acid program, perform detailed walkdowns of areas that include under the reactor vessel. VT-2 qualified System Engineers typically perform these walkdowns. Insulation on the bottom of the reactor vessel is reflective panel type, which runs flat across the bottom. Leakage would most likely be indicated by staining at panel seams or insulation bulging. Procedure MA10.1, *Station Leakage Program,* requires point of origin determination and insulation removal as necessary. If any metal loss or loss of pressure boundary exists, an evaluation is documented in a Condition Report and corrective actions performed by Work Order. Impact of components in the potential leak path are minimal as the only components in this area are the incore instrument tubes, which are fabricated of type 304 stainless steel.

NRC Question 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.

Response:

As described above, the FPLE Seabrook boric acid program is effective in identifying leaks in welds and connections of the RCPB. Low levels of leakage in small diameter nozzles can be detected through RCS walkdown inspections conducted under full system pressure. Point of origin determination will be pursued if evidence of leakage is observed.

RCS leakage and indications of leakage identified during performance of the RCS walkdowns and bolted joint inspections are identified in Condition Reports. Condition Reports document the point of origin of the leakage and/or deposit, potential pathways and affected components. System Engineering performs an evaluation to assess the effects of the leakage and recommend corrective actions. For bolting, the evaluation is also conducted in accordance with approved ISI Program relief requests discussed previously.

The RCS walkdown is performed to identify degradation or wastage that could impact component integrity or strength and to determine if any additional inspections or corrective actions are required. The following requirements are specifically noted in the RCS Leak Test procedures:

- Visually inspect each mechanical joint/component and record the leakage status. Indicate for each component whether boric acid residue is present or not.

- Leakage should be quantified whenever possible.
- Locations shall be identified by their component tag number and a clear description of the leakage source/point of origin.
- All points found with boric acid residue shall be documented and a work order generated.

This procedural guidance, and importance of identifying boric acid leakage paths and the affected components, is stressed during the pre-job briefs associated with inspection walkdowns.

NRC Question 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.

Response:

Seabrook does not use a susceptibility or consequence model for the boric acid leakage reduction program. If a boric acid deposit or leak is found it is addressed and corrected.

NRC Question 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.

Response:

FPLE Seabrook participated in a Westinghouse Owners Group (WOG) program to have Westinghouse review applicable databases and communications to determine what recommendations Westinghouse had made to its owners on visual inspections of Alloy 600/82/182 materials in the reactor coolant system pressure boundary. Westinghouse Owners Group reports did not contain recommendations for inspections.

NRC Question 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.

Response:

Procedures that comprise FPLE Seabrook's boric acid program comply with plant Technical Specification requirements and associated action statements regarding RCS pressure boundary leakage. If through-wall leakage or unacceptable indications are found, then the defect must be repaired before the plant returns to power operations. During plant operation, if a through-wall pressure boundary leak develops to a point that the leak is detected by the on-line leak detection systems or visual inspections, the leak must be evaluated per the specified Technical Specification acceptance criteria, and the plant shut down if the leak is determined to be a non-isolable RCS pressure boundary leakage (i.e., component body, pipe wall or vessel wall). Plant Technical Specification requirements continue to be met.

Title 10 of the Code of Federal Regulations, Part 50.55a requires that inservice inspection and testing be performed per the requirements of the ASME Boiler and Pressure Code, Section XI, Inservice Inspection of Nuclear Plant Components."

ASME Section XI, paragraph IWA-5250 (b) on corrective action requirements is stated in FPLE Seabrook relief request 2AR-03 using Code Case N566-1 as follows:

a. The leakage shall be stopped, and the bolting and component material shall be evaluated for joint integrity as described in (c) below.
b. If the leakage is not stopped, the joint shall be evaluated in accordance with IWB-3142.4 for joint integrity. This evaluation shall include the considerations listed in (c) below.
c. The evaluation of (a) and (b) above is to determine the susceptibility of bolting to corrosion and failure. This evaluation shall include the following:
    1. The number and service age of the bolts;
    2. Bolt and component material;
    3. Corrosiveness of process fluid;
    4. Leakage location and system function;
    5. Leakage history at the connection or other system components;
    6. Visual evidence of corrosion at the assembled connection.

If the evaluation determines that examination is required, the bolt closest to the leak will be removed and VT-1 examined. The bolt will be evaluated in accordance with IWB-3517.1 of the ASME B&PV Code Section XI, 1995 Edition through the 1996 Addenda.

As stated in response to Question 4, components with local areas of general corrosion that reduce the wall thickness by more than 5 percent are evaluated and documented in a Condition Report, procedure OE3.6. If a determination of repair or replacement is made, a Work Order is generated per procedure WM8.4.

TABLE A
FPLE Seabrook Station
NRC Bulletin 2002-01
Response to NRC Request for Additional Information

Component	Inspection Techniques	Personnel Qualifications	Extent of Coverage	Frequency	Degree of Insulation Removal/Insulation Type	Corrective Action
Reactor Vessel RC Loop Piping Safe End Welds	System Leakage Test	VT-2	General Area	18 month (refueling outage)	Left in place unless evidence of leakage/Removable metal reflective or fiberglass blanket	In accordance with OE3.6 Condition Reports
Reactor Vessel Top and Bottom Head	System Leakage Test	VT-2	General Area	18 month (refueling outage)	Left in place unless evidence of leakage/Removable metal reflective or fiberglass blanket	In accordance with OE3.6 Condition Reports
Steam Generator RC Loop Piping Safe End Welds	System Leakage Test	VT-2	General Area	18 month (refueling outage)	Left in place unless evidence of leakage/Removable metal reflective or fiberglass blanket	In accordance with OE3.6 Condition Reports
Pressurizer Surge Nozzle	System Leakage Test	VT-2	General Area	18 month (refueling outage)	Left in place unless evidence of leakage/Removable metal reflective or fiberglass blanket	In accordance with OE3.6 Condition Reports
Pressurizer Spray Nozzle	System Leakage Test	VT-2	General Area	18 month (refueling outage)	Left in place unless evidence of leakage/Removable metal reflective or fiberglass blanket	In accordance with OE3.6 Condition Reports

TABLE A
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Component	Inspection Techniques	Personnel Qualifications	Extent of Coverage	Frequency	Degree of Insulation Removal/Insulation Type	Corrective Action
Pressurizer Safety and Relief Nozzles	System Leakage Test	VT-2	General Area	18 month (refueling outage)	Left in place unless evidence of leakage/Removable metal reflective or fiberglass blanket	In accordance with OE3.6 Condition Reports

ATTACHMENT 2

St. Lucie Units 1 and 2
NRC Bulletin 2002-01
Response to NRC Request for Additional Information

REQUESTED INFORMATION

In response to NRC Bulletin 2002-01 request for additional information, St. Lucie Units 1 and 2 provide the following:

NRC Question 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).

Response:

St. Lucie Units 1 and 2 Table A provides the inspection scope and detailed information requested. The St. Lucie Plant Boric Acid Wastage Surveillance Program is effective in identifying leaks from Alloy 600 pressure boundary material and dissimilar metal Alloy 82/182 welds and connections in the reactor coolant pressure boundary (RCPB). This program has identified leakage from Alloy 600 locations in the pressurizer in 1993 and 1994 at St. Lucie Unit 2, and the reactor coolant system (RCS) hot leg piping in St. Lucie Unit 2 in 1995 and St. Lucie Unit 1 in 2001. In each case the leakage was identified before any measurable boric acid wastage could occur, repairs were implemented, and the integrity of the RCS pressure boundary was restored.

The walkdowns performed in accordance with Operating Procedures OP 1[2]-0120022 are conducted at the beginning of refueling outages and for all plant heatups prior to entering Mode 2. These procedures also contain the instructions to inspect both general locations and specific locations including all the small bore Alloy 600 nozzles in the pressurizer, the pressurizer heater sleeves and RCS hot leg piping each refueling after cooldown. These procedures require insulation removal when necessary to facilitate the inspection of the small bore hot leg and pressurizer instrument nozzles. The gaps in the blanket type insulation of the pressurizer heaters, along with the uninsulated vertical extension of the heater sleeve on the bottom of the vessel, provide acceptable access without insulation removal. Deposits or discoloration that may indicate evidence of leakage would be easily visible. Inspections of pressurizer and hot leg penetrations that have been replaced with Alloy 690 are performed as part of the general area walkdown, without removal of insulation. The inspection teams normally consist of inservice inspection, system and component engineers, and operations personnel. Each inspection typically includes personnel that are VT2-qualified.

Additionally, examinations are performed in accordance with Operations Support Engineering Procedures 1[2]-ISP-01.01, as required by ASME Section XI and plant Technical Specifications. These walkdowns are performed by VT2 qualified personnel after the RCS has been pressurized for a minimum of four hours to allow time for potential leakage to be detected outside of insulation. These procedures identify the specific areas to perform inspections including the reactor vessel head area (above the insulation), reactor vessel head O-ring seating surface, reactor coolant gas vent system, control rod drive mechanisms, in core instrument (ICI) flanges and the general area around the reactor vessel. These inspections support Technical Specifications and ASME Section XI requirements, and are also used to meet post maintenance leakage testing requirements. Some boric acid leaks are also identified from other walkdowns and activities performed by system and design engineers, and maintenance personnel. Based on the walkdowns described above, and other outage activities that identify boric acid deposits, it is unlikely that boric acid leakage would not be detected.

The St. Lucie reactor pressure vessels have solid reactor pressure vessel bottoms with no penetrations; therefore the St. Lucie Plant has no concern with leakage at that location. Leakage potentially affecting other locations is examined during the RCS system walkdowns. When evidence of reactor coolant leakage is found, the surrounding areas are examined to locate the source of the leak, and to identify any additional components/piping that may have boric acid deposits. The importance of identifying each boric acid leakage path and the affected components is stressed during the pre-job briefs associated with the inspection walkdowns.

Bare metal visual inspections of the Reactor Vessel Upper Head (RVUH) penetrations have been performed on both Unit 1 in 2002 and Unit 2 in 2001, as reported in FPL letters L-2002-061 and L-2002-233, with no indication of any leakage at that time. FPL is working with the EPRI Material Reliability Program (MRP), the ASME and the industry to determine the appropriate frequency for future bare metal visual head inspections.

NRC Question 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.

Response:

Insulation is removed from locations known to be most susceptible to primary water stress corrosion cracking (PWSCC) based on temperature and previous industry failure experience. For St. Lucie, the most susceptible locations are the small bore Alloy 600 nozzles in the pressurizer and RCS loop piping. Insulation is moved or removed as required to facilitate the inspection of the small bore hot leg and pressurizer instrument nozzles. The gaps in the blanket type insulation of the pressurizer heaters along with the uninsulated vertical extension of the heater sleeve on the bottom of the vessel, provide acceptable access without insulation removal since deposits or discoloration that may indicate evidence of leakage would be easily visible. Inspections of pressurizer and hot leg penetrations that have been replaced with Alloy 690 are performed as part of the general area walkdown without removal of insulation since the Alloy 690 material has not shown a susceptibility to the same PWSCC mechanism.

PWSCC requires the presence of high temperature along with pure primary water, and tensile stress. The higher the temperature the more susceptible Alloy 600 is to PWSCC. PWSCC is a thermally activated process that follows an Arrhenius relationship where an increase in temperature results in a decrease in time to initiation of cracking or failure. PWSCC cracks have occurred in over 98 Combustion Engineering (CE) built pressurizer and RCS Alloy 600 nozzle penetrations. The problem has been well documented in Combustion Engineering Owners Group (CEOG) reports. The pressurizer, which operates at the highest RCS temperature of 653°F, had the majority of the early penetration leaks followed much later by occurrences of leaks in hot leg nozzles. As discussed above, insulation is removed to perform inspections of Alloy 600 small bore penetrations at locations in the pressurizer (except heater sleeves). Insulation is also removed to perform inspections of Alloy 600 small bore penetrations on the RCS hot leg that operates at approximately 600°F. No PWSCC has been observed in Alloy 600 nozzle or weld material at locations operating at known RCS cold leg temperatures of approximately 550°F. Inspections of Alloy 600 penetrations at these locations are performed with the insulation in place.

It is reasonable to assume that all Alloy 600 weld material may have some susceptibility to PWSCC. In order to identify potential locations, the CEOG has identified the locations of Alloy 600 weld metal (safe end) pressure boundary joints in CEOG Report CENPSD 1211P (included in St. Lucie Plant Table A). Each of these locations is inspected during the RCS walkdowns with insulation installed, however many of the locations in the pressurizer and hot leg locations are within a few feet of the small bore penetrations being inspected with insulation removed or in areas like the top of the pressurizer where maintenance is performed, with the insulation removed, every refueling outage. RCS walkdowns usually have little difficulty in detecting boric acid leaks in quantities as small as a few ounces of boric acid. If insulation is present, boric acid deposits may have to be more than a few ounces for detection. However, boric acid leakage that would be expected to result in wastage would need to be present over a period of time and in sufficient quantity which would allow detection. These quantities would be identified from boric acid deposits around the insulation. FPL considers the RCS leakage walkdown procedures adequate in ensuring cracking or failures at these locations would be identified well before they become a gross leakage concern.

When evidence of reactor coolant leakage is found, the surrounding areas are examined to locate the source of the leak, as well as to identify any additional components/piping that may have boric acid deposits. There are no limitations to insulation removal. Removable metal reflective or fiberglass thermal insulation is on all weld areas of the reactor coolant system as indicated in St. Lucie Plant Table A. The following requirements are specifically noted in the RCS Leak Test Operating Procedures, OP 1[2]-0120022:

> All pressure-retaining components of the RCS pressure boundary shall be visually examined for evidence of reactor coolant leakage. This examination

(which need not require removal of insulation) shall be performed by inspecting (a) the exposed surface and joints of insulation and (b) the floor area (or equipment) directly underneath these components. During this inspection, particular attention shall be given to the insulated areas of components constructed of ferrous steels to detect evidence of boric acid residues resulting from reactor coolant leakage. However, specific sections of insulation shall be removed as necessary to determine the exact location and source (valve packing, cracked weld, pipe crack, etc.) of any symptoms of leakage (steam wisps, water drips, boric acid residue, etc.).

This procedural guidance, and the importance of identifying each boric acid leakage path and the affected components, is stressed during the pre-job briefs associated with the inspection walkdown.

The RVUH insulation consists of closely conforming, multi-panel, metal insulation. As described above a bare head metal examination has been performed on both units with no indication of leakage.

NRC Question 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.

Response:

The RCS leakage walkdowns are performed at the beginning of each refueling outage and during all heatups prior to entering Mode 2. The basis for this frequency is the same as all pressure test inspections performed as part of the ASME Section XI Code. All areas of the RCPB are accessible and inspected during the walkdowns with the exception of the reactor vessel bottoms. The St. Lucie reactor pressure vessels have a solid reactor pressure vessel bottom with no penetrations; therefore the St. Lucie Plant has no concern with leakage at that location. The reactor vessel head flange is inspected during refueling operations and also has its own leakage detection system consisting of a double O-ring seal monitored by a local pressure gauge and pressure switch. The pressure between the seals is monitored and will actuate a high-pressure alarm in the control room to alert the presence of leakage past the inner seal.

During plant operations Technical Specifications require operable leakage detection systems, which include radiation monitors, a sump level detection system, and a periodic inventory balance surveillance that are capable of providing indication of primary system leakage prior to a loss of RCS structural integrity.

Periodic test procedures require the performance of a daily RCS pressure boundary leakage calculation. The purpose of this surveillance is to verify that RCS leakage is within Technical Specification limits. The calculated RCS leakage rate at the St. Lucie Plant is typically less than 0.1 gpm and is prominently featured in the plant Daily Status Report by Operations. Should the leakage rate increase by a small fraction of the Technical Specification limit for pressure boundary leakage (a few tenths of a gpm) actions up to and including a walkdown of the RCS inside containment are typically performed to identify the source of the increased leakage.

NRC Question 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.

Response:

Administrative Procedure ADM-07.02, Condition Reports, is used to document non-conformances and conditions adverse to quality, such as significant material loss due to boric acid corrosion resulting from system leakage. This procedure requires determination of probable cause, corrective actions, and actions to prevent recurrence. Leakage identified during the performance of an ASME Section XI pressure test is also documented on the Visual Examination Record VT-2. All RCS leakage and indications of leakage identified during performance of the RCS walkdown procedures and bolted joint inspections are identified in condition reports. The condition reports document the source of leakage, potential pathways, and affected components. Whenever possible, the leakage amount is quantified for evaluation. A work request is also generated for corrective action.

In accordance with ISI Relief Request (R/R) No. 04 for St. Lucie Unit 1 and R/R No. 24 for St. Lucie Unit 2, when leakage is discovered at an ASME Class 1, 2, or 3 bolted connection by VT-2 visual examination during a system pressure test, a condition report is initiated and the bolting and component material are evaluated for joint integrity. If leakage is not stopped, the joint will be evaluated in accordance with ASME Code, Section XI, 1989 Edition, paragraph IWB-3142.4, Acceptance by Analytical Evaluation. This evaluation includes the following considerations to determine the susceptibility of the bolting to corrosion and failure. This evaluation will, at a minimum, consider the following conditions:

- The number and service age of the bolting
- Bolt and component material
- Corrosiveness of process fluid
- Leakage location and system function
- Leakage history at connection or other system components
- Visual evidence of corrosion at connection (while connection is assembled)

When the pressure test is performed with the system in service or the system is required by the Technical Specifications to be operable, and the bolting is susceptible to corrosion, the evaluation shall address the connection's structural integrity until the next component/system outage of sufficient duration. If the evaluation concludes that the system can perform its safety related function, removal of the bolt closest to the leakage to perform a VT-1 visual examination and evaluation of the bolt will be performed when the system/component is taken out of service during an outage of sufficient duration. If the bolt shows evidence of unacceptable degradation, additional bolting shall be removed and VT-1 examined.

For bolting that is susceptible to corrosion, and when the initial evaluation indicates that the connection can not conclusively perform its safety function until the next system/component outage of sufficient duration, the bolt closest to the source of leakage will be removed. A VT-1 visual examination shall be performed and evaluated. When the removed bolt shows evidence of unacceptable degradation, additional affected bolting shall be removed, VT-1 examined, and evaluated, or the affected bolting shall be replaced.

If boric acid residue is detected on components, the leakage source and any areas of general corrosion are identified. Components with local areas of general corrosion that reduce the wall thickness by more than 10 percent are evaluated by a condition report to determine whether the component is acceptable for continued service, or whether repair or replacement is required.

The St. Lucie Units 1 and 2 Technical Specifications include requirements and associated action statements addressing RCS pressure boundary leakage. The limits for reactor coolant pressure boundary leakage are 1 gallon per minute (gpm) for unidentified leakage, 10 gpm for identified leakage, and no leakage through a non-isolable fault in a RCS component body, pipe wall or vessel wall. If through-wall leakage or unacceptable indications are found, then the defect must be repaired before the plant returns to power operations. During plant operation, if a through-wall pressure boundary leak is detected, the leak must be evaluated per the specified technical specification acceptance criteria, and the plant shut down if the leak is determined to be non-isolable RCS pressure boundary leakage (i.e., component body, pipe wall or vessel wall).

In addition to Technical Specifications, FPL has a Nuclear Division Policy, NP-910, that requires the nuclear plant Site Vice President to personally review and approve any return to operation of a unit with known leakage from the reactor coolant system. The policy recognizes that plant technical specifications are bounding, and will ensure that structures, systems, and components important to safety are in proper working order for safe operation. However, the policy provides additional guidance and consideration to ensure maximum unit reliability by avoiding operation of the plant where there is a high likelihood of a future forced shutdown.

NRC Question 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.

Response:

The St. Lucie reactor pressure vessels have a solid reactor pressure vessel bottom with no penetrations. The reactor pressure vessel head incore instrumentation nozzles are located at the top of the reactor vessel head. Therefore, the St. Lucie Plant has no concern with leakage at the bottom of the reactor pressure vessel.

NRC Question 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.

Response:

As described above, the St Lucie Boric Acid Wastage Surveillance Program is effective in identifying leaks in welds and connections of the reactor coolant pressure boundary (RCPB). Through the RCS walkdown inspections under full system pressure, the Alloy 600 nozzle inspections of the small bore pressurizer and hot leg nozzles susceptible to PWSCC, and the inspection of all bolted joint locations during outages, low levels of leakage can be detected.

Periodic test procedures require the performance of an RCS pressure boundary leakage calculation daily. The purpose of this surveillance is to verify that RCS leakage is within Technical Specification limits. The calculated RCS leakage rate at the St. Lucie Plant is typically less than 0.1 gpm and is prominently featured in the plant Daily Status Report by Operations. Should the leakage rate increase by a small fraction of the Technical Specification limit for pressure boundary leakage (a few tenths of a gpm) actions up to and including a walkdown of the RCS inside containment are typically performed to identify the source of the increased leakage.

All RCS leakage and indications of leakage identified during performance of the RCS walkdown procedures and bolted joint inspections are required to be identified in condition reports. The condition reports document the source of leakage, potential pathways, and affected components. Engineering performs an evaluation to assess the effects of the leakage and recommend corrective actions. For bolting, the evaluation is also conducted in accordance with approved ISI Program relief requests discussed above.

When evidence of reactor coolant leakage is found, the surrounding areas are examined to locate the source of the leak, as well as to identify any additional components/piping that may have boric acid deposits. Upon discovery of leakage, Engineering Quality Instruction ENG-QI 2.3, "Operability Determinations," requires that an operability determination shall be conducted for the degraded component and must include the effects of the leakage on other components and materials.

NRC Question 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.

Response:

Since PWSCC is a thermally activated process, St. Lucie uses the principles of the Arrhenius relationship and prior industry leak experience as a guide to prioritize insulation removal to facilitate inspections. Insulation removal is performed on the hot leg and pressurizer Alloy 600 penetration locations as noted above while lower temperature applications are inspected with the insulation in place. Alloy 600 weld safe end walkdown inspections are performed with the insulation in place since the few industry events of leakage as a result of PWSCC have been transverse to the weld (axial to the run of the pipe) and well below a critical flaw size. There is no propagation mechanism into the adjoining stainless or carbon steel pipe material. To date the few safe end weld leaks that have occurred have all been adequately identified by visual inspection with the insulation in place. In addition, all of the Alloy 600 safe end welds at St. Lucie Units 1 and 2 are accessible.

FPL will continue to evaluate industry guidance on these boric acid walkdown inspections through the owners groups, EPRI, and the ASME Code and modify the boric acid inspection program as appropriate.

NRC Question 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.

Response:

FPL participated in a CEOG program to have Westinghouse review Combustion Engineering and ABB-CE databases and communications to determine what recommendations had been made to CE plant owners on visual inspections of Alloy 600/82/182 materials in the reactor coolant pressure boundary. This detailed review indicated that several visual inspection recommendations were made as the result of leakage caused by PWSCC in Alloy 600 pressurizer heater sleeves and instrumentation nozzles in a CE NSSS in 1989. Most of the recommendations were in CEOG reports and were for Alloy 600 in pressurizers, the component where most of the early leakage events occurred. In summary, the recommendations to CE plant owners were to:

(1) inspect pressurizer small diameter nozzles and heater sleeves during each refueling outage for signs of primary coolant leakage,

(2) inspections could be with insulation in-place or removed. The presence of boric acid deposits or corrosion products should be assumed to be an indication of primary coolant leakage until proven otherwise and appropriate actions taken to stop the leakage,
(3) inspect low Alloy steel components exposed to boric acid and promptly repair primary coolant leaks.

A conclusion from one of the CEOG reports reviewed that is relevant to the Alloy 600 management program was that visual inspection is the best method of detecting a leaking nozzle or heater sleeve or for detecting damage to the pressurizer shell as a result of boric acid corrosion.

FPL implemented all of the above vendor recommendations at the time they were issued and has incorporated them into the boric acid walkdown inspection procedures identified above. FPL also tracks CE NSSS utility experience with Alloy 600 reported leaks and has modified its inspection program or replacement plans according to those identified events. As a result, FPL now removes the insulation on the Alloy 600 RCS hot leg penetrations since some leaks have been identified at these locations.

NRC Question 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 10 CFR 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.

Response:

St Lucie's Boric Acid Wastage Surveillance Program complies with plant Technical Specification requirements and associated action statements regarding RCS pressure boundary leakage. The operability determinations required by plant condition reports initiated when leakage is observed ensure compliance with Technical Specifications. Specifically, Nuclear Engineering Quality Instruction, ENG-QI 2.3, Operability Determinations, Section 5.9 discusses the requirement to inform plant management upon discovery of leakage from Class 1, 2 or 3 components and to comply with the Technical Specification leakage restrictions. If a through-wall leak or unacceptable indication is found, then the defect must be repaired before the plant returns to power operations. During plant operation, if a through-wall pressure boundary leak is detected, the leak must be evaluated per the specified technical specification acceptance criteria, and the plant shut down if the leak is determined to be non-isolable RCS pressure boundary leakage (i.e., component body, pipe wall or vessel wall). Plant Technical Specifications requirements continue to be met.

The Plant Technical Specifications also require that the Inservice Inspection of ASME Code Class 1, 2 and 3 components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Code and applicable addenda as required by 10 CFR 50.55a. The requirements for the pressure and leakage testing of ASME Class 1, 2 and 3 components are outlined in St. Lucie Plant Quality Instruction Procedure QI 11-PR/PSL-8. The corrective action requirements specified in ASME Section XI, paragraph IWA-5250 (b) are required by the ISI Program at the St. Lucie Plant and restated in Procedure QI 11-PR/PSL-8 as follows.

> If boric acid residue is detected on components, the leakage source and any areas of general corrosion are identified. Components with local areas of general corrosion that reduce the wall thickness by more than 10 percent are evaluated by a condition report to determine whether the component is acceptable for continued service, or whether repair or replacement is required.

TABLE A.
St. Lucie Units 1 and 2
NRC Bulletin 2002-01
Response to NRC Request for Additional Information

Component	Inspection Techniques	Personnel Qualifications	Extent of Coverage	Frequency	Degree of Insulation Removal/Insulation Type	Corrective Action
Pressurizer Surge Nozzle Alloy 600 Weld Safe End	System Leakage Test	VT-2	General Area	18 month (refueling outage)	Left in place unless evidence of leakage/Removable metal reflective or fiberglass blanket	In accordance with ADM –07.02, Condition Reports
Pressurizer Spray Nozzle Alloy 600 Weld Safe End	System Leakage Test	VT-2	General Area	18 month (refueling outage)	Left in place unless evidence of leakage/Removable metal reflective or fiberglass blanket	In accordance with ADM –07.02, Condition Reports
Pressurizer Safety Nozzle(3) Alloy 600 Weld Safe End	System Leakage Test	VT-2	General Area	18 month (refueling outage)	Left in place unless evidence of leakage/Removable metal reflective or fiberglass blanket	In accordance with ADM –07.02, Condition Reports
Control Element Drive Mechanism Motor Housings	System Leakage Test	VT-2	General Area	18 month (refueling outage)	Left in place unless evidence of leakage/Removable metal reflective or fiberglass blanket	In accordance with ADM –07.02, Condition Reports
Reactor Coolant Pipe Surge Nozzle Alloy 600 Weld Safe End	System Leakage Test	VT-2	General Area	18 month (refueling outage)	Left in place unless evidence of leakage/Removable metal reflective or fiberglass blanket	In accordance with ADM –07.02, Condition Reports
Shutdown Cooling Nozzle to Hot Leg Piping(2) Alloy 600 Weld Safe End	System Leakage Test	VT-2	General Area	18 month (refueling outage)	Left in place unless evidence of leakage/Removable metal reflective or fiberglass blanket	In accordance with ADM –07.02, Condition Reports

TABLE A.
St. Lucie Units 1 and 2
NRC Bulletin 2002-01
Response to NRC Request for Additional Information

Component	Inspection Techniques	Personnel Qualifications	Extent of Coverage	Frequency	Degree of Insulation Removal/Insulation Type	Corrective Action
Hot Leg Drain Alloy 600 Weld Safe End	System Leakage Test	VT-2	General Area	18 month (refueling outage)	Left in place unless evidence of leakage/Removable metal reflective or fiberglass blanket	In accordance with ADM –07.02, Condition Reports
Reactor Coolant Pump Suction to Pipe(4) Alloy 600 Weld Safe End	System Leakage Test	VT-2	General Area	18 month (refueling outage)	Left in place unless evidence of leakage/Removable metal reflective or fiberglass blanket	In accordance with ADM –07.02, Condition Reports
Reactor Coolant Pump Discharge to Pipe(4) Alloy 600 Weld Safe End	System Leakage Test	VT-2	General Area	18 month (refueling outage)	Left in place unless evidence of leakage/Removable metal reflective or fiberglass blanket	In accordance with ADM –07.02, Condition Reports
Safety Injection Nozzle(4) Alloy 600 Weld Safe End	System Leakage Test	VT-2	General Area	18 month (refueling outage)	Left in place unless evidence of leakage/Removable metal reflective or fiberglass blanket	In accordance with ADM –07.02, Condition Reports
Charging Inlet Nozzle(2) Alloy 600 Weld Safe End	System Leakage Test	VT-2	General Area	18 month (refueling outage)	Left in place unless evidence of leakage/Removable metal reflective or fiberglass blanket	In accordance with ADM –07.02, Condition Reports
Letdown & or Drain Nozzle(4) Alloy 600 Weld Safe End	System Leakage Test	VT-2	General Area	18 month (refueling outage)	Left in place unless evidence of leakage/Removable metal reflective or fiberglass blanket	In accordance with ADM –07.02, Condition Reports

TABLE A.
St. Lucie Units 1 and 2
NRC Bulletin 2002-01
Response to NRC Request for Additional Information

Component	Inspection Techniques	Personnel Qualifications	Extent of Coverage	Frequency	Degree of Insulation Removal/Insulation Type	Corrective Action
Pressurizer Bypass Spray Nozzle(2) Alloy 600 Tee	System Leakage Test	VT-2	General Area	18 month (refueling outage)	Left in place unless evidence of leakage/Removable metal reflective or fiberglass blanket	In accordance with ADM –07.02, Condition Reports
Pressurizer Instrument Nozzles- Alloy 600	Visual Inspection No evidence of leakage	VT-2	Vessel to Nozzle Interface	18 month (refueling outage)	Insulation moved or removed as required (See Note 1) / Removable metal reflective or fiberglass blanket	In accordance with ADM –07.02, Condition Reports
Pressurizer Heater Sleeves- Alloy 600	Visual Inspection No evidence of leakage	VT-2	Vessel to Nozzle Interface	18 month (refueling outage)	Insulation removal not required; gaps in blanket type insulation provide acceptable access	In accordance with ADM –07.02, Condition Reports
Reactor Coolant System Hot Leg Instrument Nozzles – Alloy 600	Visual Inspection No evidence of leakage	VT-2	Nozzle and RCS Loop Interface	18 month (refueling outage)	Insulation moved or removed as required(See Note 1)/ Removable metal reflective or fiberglass blanket	In accordance with ADM –07.02, Condition Reports
Reactor Coolant System Cold Leg Instrument Nozzles - Alloy 600	System Leakage Test	VT-2	General Area	18 month (refueling outage)	Left in place unless evidence of leakage/Removable metal reflective or fiberglass blanket	In accordance with ADM –07.02, Condition Reports

TABLE A.
St. Lucie Units 1 and 2
NRC Bulletin 2002-01
Response to NRC Request for Additional Information

Component	Inspection Techniques	Personnel Qualifications	Extent of Coverage	Frequency	Degree of Insulation Removal/Insulation Type	Corrective Action
Piping Fittings in the Pressurizer Spray Piping	System Leakage Test	VT-2	General Area	18 month (refueling outage)	Left in place unless evidence of leakage/Removable metal reflective or fiberglass blanket	In accordance with ADM –07.02, Condition Reports

Notes:
(#) Quantity of item
1) Inspections are performed with the insulation in place at instrument penetrations that have been replaced with Alloy 690 nozzles.

ATTACHMENT 3

Turkey Point Units 3 and 4
NRC Bulletin 2002-01
Response to NRC Request for Additional Information

REQUESTED INFORMATION

In response to NRC Bulletin 2002-01 request for additional information, Turkey Point Units 3 and 4 provide the following:

NRC Question 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).

Response:

Table A provides the inspection scope and detailed information requested. At Turkey Point Units 3 and 4, Alloy 600 and dissimilar metal Alloy 82/182 material are found only on the reactor vessel lower head (RVLH) and reactor vessel upper head (RVUH).

The Boric Acid Wastage Surveillance Program (PTN-ENG-LRAM-00-0028) covers all areas of the Alloy 600 and dissimilar metal Alloy 82/182 material, i.e. reactor vessel lower head (RVLH) and reactor vessel upper head (RVUH). The program credits several inspections as detailed in the following procedures:

Procedure OP-0206.7, Containment Visual Leak Inspection, specifies the required components to be inspected by providing an extensive matrix of locations and component descriptions. The purpose of this procedure is to inspect and report estimated leakage, structural distress or corrosion of any system or component located inside containment which could contribute to system leakage or component failure. This procedure requires documentation of all boric acid found, point of origin, evidence of structural distress or corrosion, and verification that the boric acid has not spilled onto other components, specifically carbon steel components. This inspection is performed each time the unit is placed in hot shutdown, unless performed within the previous 30 days. This inspection includes all of the RCPB, with the exception of the RVLH. Insulation removal is not required for this inspection to be conducted. There are no certifications or qualifications required for this inspection, however, system engineers and VT-2 certified personnel typically perform these inspections.

Procedure OSP-41.25, Class 1 RCS Overpressure Leak Testing, is utilized to perform the system leakage examination at a pressure slightly above normal operating pressure. This inspection includes all of the RCPB, including the RVLH. Insulation removal is not required, however, the inspection is performed following a four hour hold at pressure and temperature. This inspection is performed prior to plant startup following each reactor refueling outage. Certified VT-2 inspectors that are very familiar with the reactor coolant system perform these VT-2 inspections.

Bare metal visual inspections of the RVUH penetrations have been performed on both Units 3 and 4 in 2001 and 2002 respectively, as reported in the FPL response to Bulletin 2002-01 question 1.C (L-2002-061), with no indication of any leakage at that time. FPL is working with the EPRI Material Reliability Program (MRP), the ASME and the industry to determine the appropriate frequency for future bare metal visual head inspections.

NRC Question 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.

Response:

Inspections of the RCPB, which includes Alloy 600 and dissimilar metal Alloy 82/182 materials are performed under procedure OP-0206.7, Containment Visual Leak Inspection, and procedure OSP-41.25, Class 1 RCS Overpressure Leak Testing, as described above. These inspections are performed with insulation installed unless boric acid leakage or boric acid residue is found. In that case, insulation is removed as required to determine the point of origin, inspect the leak path, and inspect for general corrosion.

Reflective stainless steel insulation covers the lower portions of the reactor vessel. The RVLH has 50 incore instrumentation penetrations. Each penetrates through this insulation and has a 1/2 inch gap 360 degrees around. The insulation for the RVLH also has an air gap of approximately 1/2" to 5 7/8" between the insulation and the RVLH. Any leakage from an instrumentation penetration would be indicated by fluid, boric acid residue, or staining through the insulation gaps. If any of these indications were to be present, this would require the insulation to be removed and further inspections performed to determine the source.

The insulation on the RVUH consists of blanket type, multi layer insulation. As described above, RVUH bare head inspections have been performed on both Units 3 and 4, with no indication of any leakage. FPL is working with the EPRI Material Reliability Program (MRP), the ASME and the industry to determine the appropriate frequency for future bare metal visual head inspections.

The following procedure requirements are specifically noted in the OSP-41.25, Class 1 RCS Overpressure Leak Testing:

> All pressure-retaining components of the RCS pressure boundary shall be visually examined for evidence of reactor coolant leakage. This examination (which need not require removal of insulation) shall be performed by inspecting (a) the exposed surface and joints of insulation and (b) the floor area (or equipment) directly underneath these components. During this inspection, particular attention shall be given to the insulated areas of components constructed of ferrous steels to detect evidence of boric acid residues resulting from reactor coolant leakage. However, specific sections of insulation shall be removed as necessary to determine the exact location and source (valve packing, cracked weld, pipe crack, etc.) of any symptoms of leakage (steam wisps, water drips, boric acid residue, etc.).

NRC Question 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.

Response:

The RCS leakage walkdowns are performed at the beginning of each refueling outage and prior to entering mode 2 as described above. The basis for this frequency is the same as all pressure test inspections performed as part of the ASME Section XI Code. All areas of the RCPB are accessible and inspected.

During plant operations Technical Specifications require operable leakage detection systems, which include radiation monitors, a sump level detection system, and a periodic inventory balance surveillance that are capable of providing indication of primary system leakage prior to a loss of RCS structural integrity. The reactor vessel head and flange are inspected during refueling operations and also has its own leakage detection system consisting of a double O-ring seal monitored by temperature instrumentation. The leakoff from the seals is monitored and will actuate a high temperature alarm in the control room to alert the presence of leakage past the inner seal.

Reactor Coolant System Leak Rate Calculation, OSP-41.1 procedures require the performance of an RCS pressure boundary leakage calculation daily at least once per 24 hours. The purpose of this surveillance is to verify that RCS leakage is within Technical Specification limits Tech Spec Section 3.4.6, RCS Operational Leakage, and Section 4.4.6.2.1.c, RCS Water Inventory Balance. The calculated RCS leakage rate at Turkey Point is typically less than 0.1 gpm and is prominently featured in the plant Daily Plant Report by Operations.

Should the leakage rate increase by some fraction of the Technical Specification limit for pressure boundary leakage, an investigation is initiated which may lead to a containment walkdown to identify the source of the increased leakage.

NRC Question 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 were 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.

Response:

Administrative procedure 0-ADM-518, Condition Reports, is used to document non-conformances and conditions adverse to quality, such as significant material loss due to boric acid corrosion resulting from system leakage. This procedure requires determination of probable cause, corrective actions and actions to prevent recurrence. Leakage identified during the performance of an ASME Section XI pressure test is also documented on the Visual Examination Record VT-2. Evidence of pressure boundary leaks require a Condition Report for evaluation. Whenever possible, the leakage amount is quantified for evaluation. A Work Order is also generated for any corrective action required.

In accordance with ISI Request for Relief No.11 for Turkey Point Units 3 and 4, when leakage is discovered at an ASME Class 1, 2, or 3 bolted connection, by VT-2 visual examination during a system pressure test, a Condition Report is initiated and the bolting and component material are evaluated for joint integrity. The engineering evaluation includes the following considerations to determine the susceptibility of the bolting to corrosion and failure. This evaluation will, at a minimum, consider the following conditions:

- The service age of the bolting
- Bolt and component material
- Corrosiveness of process fluid
- Leakage history and system function
- Leakage history at the specific location
- Visual evidence of corrosion at connection (while connection is assembled)
- Physical configuration of the bolted connection

When the evaluation of the above criteria concludes that the leaking condition has not degraded the bolting, no further action is necessary. If the evaluation concludes that the bolting is degraded or is inconclusive in determining degradation, the bolt closest to the source of leakage shall be removed, VT-1 examined and evaluated in accordance with IWA-3100(a). When the removed bolting shows evidence of unacceptable degradation, all affected bolting shall be removed, VT-1 examined and evaluated in accordance with IWA-3100(a) or the bolting shall be replaced.

In addition to the above program, FPL Nuclear Division Policy NP-910 requires the nuclear plant Site Vice President to personally review and approve any return to operation of a unit with known leakage from the reactor coolant system. The policy recognizes that plant technical specifications are bounding, and will ensure that structures, systems and components important to safety are in proper working order for safe operation. However, the policy provides additional guidance and consideration to ensure maximum unit reliability by avoiding operation of the plant where there is a high likelihood of a future forced shutdown.

NRC Question 5

Explain the capabilities of your program to detect the low levels of RCPB leakage that may result from through-wall cracking in the bottom RPV 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 RPV 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.

Response:

The Boric Acid program entails detailed walkdowns that include the RVLH as well as the RCPB. The inspection is performed by VT-2 Level II certified inspectors. The only components that are located under the reactor vessel are the bottom mounted instrument tubes, which are made of SB-166 (Alloy 600).

Reflective stainless steel insulation covers the lower portions of the reactor vessel. The RVLH has 50 incore instrumentation penetrations. Each penetrates through this insulation and has a 1/2 inch gap 360 degrees around. The insulation for the RVLH also has an air gap between the insulation and the RVLH of approximately 1/2" to 5 7/8". Any leakage from an instrumentation penetration would be indicated by fluid, boric acid residue, or staining through the insulation gaps. If any of these indications were to be present, this would require the insulation to be removed and further inspections performed to determine the source.

When evidence of reactor coolant leakage is found, the path of the leakage is determined to locate the source of the leak, as well as to identify any additional components/piping that may have boric acid deposits. The inspection is performed to identify any degradation or wastage that could impact component integrity or strength and to determine if any additional inspections or corrective actions are required.

NRC Question 6

Explain the capabilities of your program to detect the low levels of RCPB 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.

Response:

As described above, the Turkey Point Boric Acid Wastage Surveillance Program is effective in identifying leaks in the RCPB. Through the RCS walkdown inspections under full system pressure, the inspections of bolted connections, and the reactor head inspections, low levels of leakage can be detected.

Periodic test procedures require the performance of an RCS pressure boundary leakage calculation daily. The purpose of this surveillance is to verify that RCS leakage is within Technical Specification limits. The calculated RCS leakage rate at the Turkey Point Plant is typically less than 0.1 gpm and is prominently featured in the plant Daily Status Report by Operations.

All reactor coolant pressure boundary leakage and indications of leakage identified during performance of the RCS walkdown procedures and bolted joint inspections are identified in Condition Reports. The Condition Reports document the point of origin of the leakage and/or deposit, potential pathways and affected components. Engineering performs an evaluation to assess the effects of the leakage and recommend corrective actions. For bolting, the evaluation is also conducted in accordance with approved ISI Program relief requests discussed above.

When evidence of reactor coolant leakage is found, the surrounding areas are examined to locate the source of the leak, as well as to identify any additional components/piping that may have boric acid deposits. Upon discovery of leakage, Engineering Quality Instruction ENG-QI 2.3, "Operability Determinations," requires that an operability determination shall be conducted for the degraded component and must include the effects of the leakage on other components and materials.

NRC Question 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.

Response:

The Turkey Point Plant was manufactured at a time when Alloy 600 weld safe ends were not being used as part of the construction of the RCS. The only use of Alloy 600 is in the reactor vessel top and bottom head penetrations. The top head penetrations are near the hot leg temperature of approximately 594°F and the bottom RV penetrations are at the cold leg temperature of approximately 547°F. Both locations are inspected each refueling outage but more recently the higher temperature upper head penetrations have been visually inspected with the insulation removed due to the higher susceptibility based on the higher temperature and the understanding that PWSCC is a thermally activated process that follows an Arrhenius relationship.

FPL will continue to evaluate industry guidance on these boric acid walkdown inspections through the owners groups, EPRI, and the ASME Code and modify our boric acid inspection program as appropriate.

NRC Question 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.

Response:

FPL participated in a Westinghouse Owners Group (WOG) program to have Westinghouse review applicable databases and communications to determine what recommendations Westinghouse had made to its owners on visual inspections of Alloy 600/82/182 materials in the reactor coolant system pressure boundary. Westinghouse Owners Group reports did not contain recommendations for inspections.

NRC Question 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, 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.

Response:

Plant Technical Specifications include requirements and associated action statements addressing RCS pressure boundary leakage. The limits for reactor coolant pressure boundary leakage (T/S 3.4.6.2) are as follows: No pressure boundary leakage, 1 gallon per minute (gpm) for unidentified leakage, and 10 GPM identified leakage. If through-wall leakage or unacceptable indications are found, then the defect must be repaired before the plant returns to power operations. During plant operation, if a through-wall pressure boundary leak develops to a point that the leakage is detected by the on-line leak detection systems or visual inspections, the leak must be evaluated per the specified technical specification acceptance criteria, and the plant shut down if the leak is determined to be a non-isolable RCS pressure boundary leakage (i.e., component body, pipe wall or vessel wall). Plant Technical Specifications requirements continue to be met.

Title 10 of the Code of Federal Regulations, Part 50.55a requires that in-service inspection and testing be performed per the requirements of the ASME Boiler and Pressure Vessel Code, Section XI, In-service Inspection of Nuclear Plant Components.

ASME Section XI, paragraph IWA-5250 (b) requirements for corrective action are included in procedure OSP-45.1, ASME Section XI Quality Group A Bolting Examination.

If leakage is discovered at a bolted connection by VT-2 examination during a system pressure test, either the bolt closest to the source of leakage will be removed and a VT-1 examination conducted and evaluated in accordance with IWA-3100(a) or an engineering evaluation will be performed to determine the susceptibility of the bolting to corrosion and assess the potential for failure. The following factors will be considered as applicable, when evaluating the acceptability of the bolting:

- The service age of the bolting
- Bolt and component material
- Corrosiveness of process fluid
- Leakage history and system function
- Leakage history at the specific location
- Visual evidence of corrosion at connection (while connection is assembled)
- Physical configuration of the bolted connection

When the evaluation of the above criteria concludes that the leaking condition has not degraded the bolting, no further action is necessary. If the evaluation concludes that the bolting is degraded or is inconclusive in determining degradation, the bolt closest to the source of leakage shall be removed, VT-1 examined and evaluated in accordance with IWA-3100(a). When the removed bolting shows evidence of unacceptable degradation, all affected bolting shall be removed, VT-1 examined and evaluated in accordance with IWA-3100(a) or the bolting shall be replaced.

Procedure 0-ADM-523, ASME Section XI Pressure Tests for Quality Group A, B, C Systems/Components, section 5.10 states the following:

> All leakage identified during the performance of an ASME section XI Pressure test shall be documented an a VT-2 data sheet, pressure test package, and reviewed by the Nuclear Plant Supervisor, Engineering, and the ISI supervisor to establish retest and/or any corrective actions required. In addition to the above, the following actions shall be performed, as applicable:
>
> 1. Quality group A, B and C pressure retaining boundary through-wall-leakage shall be corrected, repaired, or replaced prior to returning the effected portion of the system to service unless;

a. An analytical Evaluation is performed for Quality Group B and C pressure retaining piping or components as satisfactory results are obtained, as required by IWC-3000 or IWD-3000 respectively.
OR
b. Written relief for temporary repair is granted by the USNRC.

2. Pressure boundary through wall leakage shall be documented on a Condition Report and a Work Request shall be generated.

TABLE A.
Turkey Point Units 3 and 4
NRC Bulletin 2002-01
Response to NRC Request for Additional Information
Summary of Alloy 600 and Dissimilar Metal Alloy 82/182

Component	Inspection Techniques	Personnel Qualifications	Extent of Coverage	Frequency	Degree of Insulation Removal/Insulation Type	Corrective Action
Bottom Head Mounted Instrumentation Alloy 600 Weld	System Leakage Test	VT-2	General Area	18 month (refueling outage)	Left in place unless evidence of leakage/Removable metal reflective	In accordance with ADM –518, Condition Reports
CRDM Housing Tubes (Head adapters)Alloy 600 Weld	System Leakage Test	VT-2	General Area	18 month (refueling outage)	Left in place unless evidence of leakage/Removable fiberglass blanket	In accordance with ADM –518, Condition Reports
Reactor Head Vent Pipe Alloy 600 Weld	System Leakage Test	VT-2	General Area	18 month (refueling outage)	Left in place unless evidence of leakage/Removable fiberglass blanket	In accordance with ADM –518, Condition Reports