Florida Power & Light Company, 9760 SW 344 St., Fla City, FL 33035

FPL

L-2001-76
10 CFR 54

APR 19 2001

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

Re: Turkey Point Units 3 and 4
Docket Nos. 50-250 and 50-251
Response to Request for Additional Information for the Review of the Turkey Point Units 3 and 4 License Renewal Application

By letter dated February 2, 2001, the NRC requested additional information regarding the Turkey Point Units 3 and 4 License Renewal Application (LRA). Attachment 1 to this letter contains the responses to the Requests for Additional Information (RAIs) associated with Section 3.2, Reactor Coolant Systems of the LRA.

Should you have any further questions, please contact E. A. Thompson at (305)246-6921.

Very truly yours,

R. J. Hovey
Vice President - Turkey Point

RJH/EAT/hlo

Attachment

A084

cc: U.S. Nuclear Regulatory Commission, Washington, D.C.

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Turkey Point Units 3 and 4
Docket Nos. 50-250 and 50-251

Response to Request for Additional Information for the Review of the Turkey Point Units 3 and 4, License Renewal Application

STATE OF FLORIDA)
) ss
COUNTY OF MIAMI-DADE)

R. J. Hovey being first duly sworn, deposes and says:

That he is Vice President - Turkey Point of Florida Power and Light Company, the Licensee herein;

That he has executed the foregoing document; that the statements made in this document are true and correct to the best of his knowledge, information and belief, and that he is authorized to execute the document on behalf of said Licensee.

R. J. Hovey

Subscribed and sworn to before me this
19th day of April, 2001.

Olga Hanek

Olga Hanek
Name of Notary Public (Type or Print)

OLGA HANEK
MY COMMISSION # CC 926970
EXPIRES: June 18, 2004
Bonded Thru Notary Public Underwriters

R. J. Hovey is personally known to me.

ATTACHMENT 1
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION DATED FEBRUARY 2, 2001 FOR THE REVIEW OF THE TURKEY POINT UNITS 3 AND 4, LICENSE RENEWAL APPLICATION

SECTION 3.2 REACTOR COOLANT SYSTEMS

RAI 3.2-1:
Identify all RCS components and subcomponents that are fabricated from either Alloy 600 base metal or weld metal fabricated using INCO 182/82, and are exposed to primary water. Describe the aging management programs used to manage the cracking due to stress corrosion cracking (SCC), in particular primary water SCC (PWSCC), in these items during the license renewal period, or provide the basis for not requiring management of this aging effect.

FPL RESPONSE:
All Reactor Coolant Systems components and subcomponents fabricated from either Alloy 600 base metal or weld metal fabricated using Alloy 82/182 and exposed to treated water - primary are listed in LRA Table 3.2-1 under "Internal Environment" for "Reactor Vessels" (page 3.2-67), "Reactor Vessel Internals" (page 3.2-76) and "Steam Generators" (page 3.2-86). Aging Management programs used to manage cracking of these components are also identified in LRA Table 3.2-1 under "Program/Activity" and include the Chemistry Control Program, the ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program and the Reactor Vessel Head Alloy 600 Penetration Inspection Program. These programs are described in Appendix B, Subsections 3.1.6, 3.2.1 and 3.2.4 of the LRA, (pages B-21, B-27 and B-47, respectively).

SECTION 3.2.1 REACTOR COOLANT PIPING

RAI 3.2.1-1:
Section 2.3.1.2.2 of the application indicates that the inner reactor vessel flange O-ring leak detection line tubing, fittings and valves, and the reactor vessel head vent piping, fittings and valves downstream of the restricting orifices are non-Class 1 reactor coolant components requiring aging management. Describe the environment of these components during an operational cycle, including refueling outages. Could these items have cyclic exposure to an aqueous environment followed by drying and resultant accumulation of corrosive products? If so, describe aging management of stress corrosion cracking for these items during the license renewal period.

FPL RESPONSE:
The inner reactor vessel flange O-ring leak detection systems (see drawings 3/4-RCS-01) discharge to the Reactor Coolant Drain Tanks (RCDTs) via valve 3/4-502, which is open except during refueling. The RCDTs gas space is maintained pressurized with nitrogen at approximately 2 psig. Therefore, except during refueling, the internal environment of the inner reactor vessel flange O-ring leak detection piping/tubing, fittings and valves is nitrogen. During refueling, valve 3/4-502 is closed, therefore, the internal environment of the inner reactor vessel flange O-ring leak detection piping/tubing, fittings and valves (up to and including the isolation valve) is containment air prior to filling the refueling cavity, and treated water - borated while the refueling cavity is filled. After completion of refueling, valve 3/4-502 is reopened, the borated water remaining in the line drains to the RCDTs, except for a short (approximately 4 feet) section of pipe at a low point in the system. As the water drains or evaporates, the inert nitrogen environment is restored. The borated water that enters the leak detection systems is subject to very stringent chemistry requirements resulting in low levels of contaminants, therefore, the residue left within the piping/tubing, fittings and valves has a low contaminant level. Additionally, since the equipment is maintained under a nitrogen environment with no oxygen present, the aging management review for these components determined that there were no aging effects requiring management as shown in LRA Table 3.2-1, page 3.2-58, under "Reactor Coolant Piping - Non-Class 1". This conclusion is supported by plant operational and maintenance history review, which did not identify any instances of degradation of this piping/tubing/fittings.

The Reactor Head Vent systems (see drawings 3/4-RCS-01 and 3/4-RCS-02) piping/tubing, fittings and valves are internally exposed to treated water - primary in modes 1 - 5. The primary water in the Reactor Head Vent systems is subject to very stringent chemistry requirements resulting in low levels of contaminants. During mode 6, internal surfaces of the Reactor Head Vent systems are exposed to containment air. If any residue is left within the piping/tubing, fittings and valves it would have a low contaminant level. Following refueling, the Reactor Head Vent system is used to fill and vent the reactor. During the fill and vent process treated water - primary flows through the Reactor Head Vent system, thus flushing out any residue that might have deposited in the system. The aging management review for these components determined that cracking was the only aging effect requiring management and that the Chemistry Control Program would manage this aging effect (see LRA Table 3.2-1, page 3.2-58, "Reactor Coolant Piping - Non-Class 1 Components").

SECTION 3.2.2 REGENERATIVE AND EXCESS LETDOWN HEAT EXCHANGERS

RAI 3.2.2-1:
For the excess letdown heat exchangers, Section 3.2.2.2.2 of the LRA indicates that loss of material and loss of mechanical closure integrity of the bolting due to aggressive chemical attack will be managed by the boric acid wastage surveillance program. Describe any incidents to date where loss of material or mechanical closure integrity due to aggressive chemical attack have occurred for the excess letdown heat exchangers.

FPL RESPONSE:
There have been a few occurrences (3 on each unit) of minor leakage of borated water at the tube sheet flange gasket of the excess letdown heat exchangers. Inspections performed as part of the Boric Acid Wastage Surveillance Program identified this leakage. The leakage was characterized by boric acid residue or the presence of wetness on the exterior surfaces of the heat exchanger cover. Therefore, the leakage did not affect the intended function of the heat exchangers. Corrective actions to address this leakage included replacement of the gaskets and inspection and replacement of fasteners, as required. Based on the timely identification of this borated water leakage, no enhancements to the Boric Acid Wastage Surveillance Program were deemed necessary. No leakage from the excess letdown heat exchangers has been reported since 1995. In order to address this potential for loss of material and loss of mechanical closure integrity due to aggressive chemical attack, periodic inspections performed under the Boric Acid Wastage Surveillance Program are credited for managing these aging effects.

RAI 3.2.2-2:
Appendix C, page C-22 of the LRA indicates that high yield stress materials and contaminants such as lubricants containing MoS_2 have caused cracking of bolting in the industry. Address how yield strength and elimination of contaminants will be addressed during the period of extended operation.

FPL RESPONSE:
High stress in conjunction with an aggressive environment can cause cracking of certain bolting materials due to stress corrosion cracking (SCC). As identified in NRC IE Bulletin 82-02 and Generic Letter 91-17, cracking of bolting in the industry has occurred due to SCC. These instances of SCC have been primarily attributed to the use of high yield strength bolting materials, excessive torquing of fasteners, and contaminants, such as the use of lubricants containing molybdenum disulfide (MoS_2). In response to NRC IE Bulletin 82-02, Turkey Point verified that:

(a) Specific maintenance procedures were in place that address bolted closures of the Reactor Coolant pressure boundary with a nominal diameter of 6 inches or greater.

(b) The procedures in use addressed detensioning and retensioning practices and gasket installation and controls.

(c) Threaded fastener lubricants used in the reactor coolant pressure boundary have specified maximum allowable limits for chloride and sulfur content to minimize susceptibility to SCC environments.

(d) Maintenance crew training on threaded fasteners is performed.

In order for SCC to occur, three conditions must exist: a susceptible material, high tensile stresses, and a corrosive environment. At Turkey Point, the potential for SCC of fasteners is minimized by utilizing ASTM A193, Gr. B7 bolting material and limiting contaminants such as chlorides and sulfur in lubricants and sealant compounds. Additionally, sound maintenance bolt torquing practices are used to control bolting material stresses. The use of ASTM A193, Gr. B7 bolting specifies a minimum yield strength of 105 Ksi, which is well below the 150 Ksi threshold value specified in EPRI NP-5769, "Degradation of Bolting in Nuclear Power Plants", April 1988. Bolting fabricated in accordance with this standard could be expected to have yield strengths less than 150 Ksi. However, since the maximum yield strength is not specified for this bolting material, absolute assurance can not be provided that the yield strength of the bolting would not exceed 150 Ksi. For these cases, the combination of specifying ASTM A193 Gr. B7 bolting material,

control of bolt torquing, and control of contaminants will ensure that SCC will not occur. These actions have been effective in eliminating the potential for SCC of bolting materials. The results of a review of the Turkey Point condition report (1992 through 2000) and metallurgical report (1987 through 2000) databases support this conclusion in that no instances of bolting degradation due to SCC were identified. Additionally, review of NRC generic communications did not identify any recent bolting failures attributed to SCC. Therefore, cracking of bolting material due to SCC is not considered an aging effect requiring management at Turkey Point.

SECTION 3.2.3 PRESSURIZERS

RAI 3.2.3-1:
Discuss how the plant-specific water chemistry control programs provide for a sufficient level of hydrogen overpressure to support the conclusion in WCAP-14574 that hydrogen overpressure in the RCS will minimize the adverse effects of oxygen in the coolant and provide adequate protection against crevice corrosion in creviced geometries on the internal surfaces of the pressurizer.

FPL RESPONSE:
Hydrogen concentrations in the Turkey Point Reactor Coolant Systems (RCS) are strictly maintained within specified limits by measurement of hydrogen concentrations in periodic RCS samples, and adjusting hydrogen overpressure in the volume control tanks accordingly. The hydrogen concentration limits established for the RCS ensure that loss of material due to crevice corrosion is not significant for the internal surfaces of the Turkey Point pressurizers as well as other Class 1 components. The Chemistry Program is described in LRA Appendix B, Subsection 3.2.4 (page B-47). Also, refer to the response to RAI 3.9.4-1 (FPL letter L-2001-65).

RAI 3.2.3-2:
In order to take credit for the analysis in EPRI Report No. NP-5769 and the conclusion in WCAP-14574 that SCC is not an aging effect that needs to be managed for the SA193, Grade B7 low-alloy steel bolting materials, confirm that the yield strengths of record for the quenched and tempered, SA-193, Grade B7, low-alloy steel bolting materials in the Turkey Point pressurizers are within the range of 105-150 ksi.

FPL RESPONSE:
High stress in conjunction with an aggressive environment can cause cracking of certain bolting materials due to stress corrosion cracking (SCC). As identified in NRC IE Bulletin 82-02 and Generic Letter 91-17, cracking of bolting in the industry has occurred due to SCC. These instances of SCC have been primarily attributed to the use of high yield strength bolting materials, excessive torquing of fasteners, and contaminants, such as the use of lubricants containing molybdenum disulfide (MoS_2). In response to NRC IE Bulletin 82-02, Turkey Point verified that:

(a) Specific maintenance procedures were in place that address bolted closures of the Reactor Coolant pressure boundary with a nominal diameter of 6 inches or greater.

(b) The procedures in use addressed detensioning and retensioning practices and gasket installation and controls.

(c) Threaded fastener lubricants used in the reactor coolant pressure boundary have specified maximum allowable limits for chloride and sulfur content to minimize susceptibility to SCC environments.

(d) Maintenance crew training on threaded fasteners is performed.

In order for SCC to occur, three conditions must exist: a susceptible material, high tensile stresses, and a corrosive environment. At Turkey Point, the potential for SCC of fasteners is minimized by utilizing ASTM A193, Gr. B7 bolting material and limiting contaminants such as chlorides and sulfur in lubricants and sealant compounds. Additionally, sound maintenance bolt torquing practices are used to control bolting material stresses. The use of ASTM A193, Gr. B7 bolting specifies a minimum yield strength of 105 Ksi, which is well below the 150 Ksi threshold value specified in EPRI NP-5769, "Degradation of Bolting in Nuclear Power Plants", April 1988. Bolting fabricated in accordance with this standard could be expected to have yield strengths less than 150 Ksi. However, since the maximum yield strength is not specified for this bolting material, absolute assurance can not be provided that the yield strength of the

bolting would not exceed 150 Ksi. For these cases, the combination of specifying ASTM A193 Gr. B7 bolting material, control of bolt torquing, and control of contaminants will ensure that SCC will not occur. These actions have been effective in eliminating the potential for SCC of bolting materials. The results of a review of the Turkey Point condition report (1992 through 2000) and metallurgical report (1987 through 2000) databases support this conclusion in that no instances of bolting degradation due to SCC were identified. Additionally, review of NRC generic communications did not identify any recent bolting failures attributed to SCC. Therefore, cracking of bolting material due to SCC is not considered an aging effect requiring management at Turkey Point.

RAI 3.2.3-3:
In order to support the conclusion in WCAP-14574 that SCC would not be a problem in welded Type 304 stainless steel pressurizer supports if a reasonable justification could be made that the associated welds were not in the sensitized state, describe how the implementation of Turkey Point or FPL plant-specific procedures and quality assurance criteria for the welding and testing of austenitic welds, if any, provides a reasonable assurance that sensitization has not occurred in these welds.

FPL RESPONSE:
The cladding material (308L) used to protect the pressurizer alloy steel shell and the weld material (309L) used to join the pressurizer internal supports and the pressurizer cladding were selected to have sufficiently low carbon content to minimize the likelihood of sensitization of these welds. Additionally, welding processes were performed to minimize the likelihood of sensitization of stainless steels. However, the possibility cannot be precluded that sensitized areas may exist in 304 stainless steel supports or their welds, therefore, per LRA Table 3.2-1 (page 3.2-65), cracking due to stress corrosion is an aging effect requiring management for welded type 304 stainless steel pressurizer supports.

The Chemistry Control Program, as described in LRA Appendix B, Subsection 3.2.4 (page B-47), is credited for managing this aging effect. Control of oxygen and chlorides provides an essentially benign environment which has been shown to be effective in preventing stress corrosion cracking (SCC) in laboratory experiments and years of operating experience. The Chemistry Control Program precludes SCC in the pressurizer internal attachment welds for the period of extended operation regardless of weld sensitization.

RAI 3.2.3-4:
In WCAP-14574, the WOG was not clear whether or not loss of material by erosion was a plausible aging effect for pressurizer surge nozzle thermal sleeves, surge nozzle safe ends, spray nozzle thermal sleeves, and spray nozzle safe ends in Westinghouse-designed plants. Analyze and discuss whether or not loss of material by erosion is a plausible aging effect for these components. If the analysis supports the conclusion that erosion is plausible within any of these components and that the corresponding components are within the scope of license renewal, modify Table 3.2-1 appropriately and propose an aging management program to manage this aging effect within the proposed extended operating terms for the Turkey Point units.

FPL RESPONSE:
As summarized in LRA Subsection 3.2.3 (pages 3.2-18 through 3.2-22), a Turkey Point aging management review of the pressurizers was performed which evaluated the materials and environments of pressurizer components including pressurizer surge nozzle thermal sleeves, surge nozzle safe ends, spray nozzle thermal sleeves, and spray nozzle safe ends. This aging management review concluded that stainless steel materials are considered to be resistant to erosion, and that loss of material due to erosion was not an aging effect requiring management. The results of the aging management review for pressurizer components is provided in LRA Subsection 3.2.3 and Table 3.2-1 (pages 3.2-63 through 3.2-66).

RAI 3.2.3-5:
Propose an AMP to verify whether or not thermal fatigue-induced cracking in the pressurizer cladding has propagated through the clad into the ferritic base metal or weld metal materials beneath the clad.

FPL RESPONSE:
There is no industry experience to suggest that cracks initiating at the clad inner surfaces of the Reactor Coolant System (RCS) primary pressure boundary components fabricated of ferritic steel will propagate into the underlying base metal or weld metal.

Operating experience provides examples of observed indications in primary system components. These observed flaws were monitored for an extended period of time, and no significant change in size was observed. In 1990, several indications were discovered in the pressurizer cladding at the Connecticut Yankee Plant during a camera inspection of the bottom head and surge nozzle region. Ultrasonic inspection confirmed that the indications did not penetrate into the ferritic base metal, and therefore, in accordance with ASME Section XI, the indications were acceptable without repair. A surveillance program was initiated, and after two follow-up inspections that showed no change, the surveillance program was discontinued with NRC approval. In several of the cases of observed cracking, fracture mechanics analyses were performed and demonstrated that the cladding indications will not compromise the integrity of the primary system components.

In addition, at temperatures greater than 180°F, the cladding has virtually no impact on fracture behavior. This is the very low end of the plant operating temperature range. ASME Section XI flaw evaluation rules require that the effects of cladding must be considered in any structural integrity evaluation, especially for postulated flaws which penetrate the cladding into the base metal. The actual impact of the cladding on such an evaluation is negligible.

The Turkey Point pressurizer shell design considers fatigue usage throughout its operating lifetime and includes adequate margin. This is expected to preclude the formation of fatigue cracks in the cladding material. The fracture mechanics evaluations performed for actual observed cracks indicate that the cracks do not grow significantly over the plant lifetime. Therefore, a specific aging management program to address cracking of the pressurizer cladding due to fatigue is not required.

Although a specific aging management program is not required for the pressurizer cladding, as noted in the response to RAI 4.3.1-4, the ASME Section XI, Subsections IWB, IWC and IWD Inservice Inspection Program is credited for managing the potential for crack initiation and growth for the pressurizer surge lines. The pressurizer surge nozzles are the limiting pressurizer location from a fatigue usage perspective, and are included in the ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program described in LRA Appendix B, Subsection 3.2.1.1 (page B-27).

SECTION 3.2.4 REACTOR VESSELS

RAI 3.2.4-1:
In Table 3.2-1 of the LRA, the applicant indicated that cracking of the core support lugs will be managed by the Chemistry Control Program and ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection (ISI) Programs (Examination Category B-N-2). The staff does not believe that the VT-3 examinations are sufficient to detect cracking of the core support lugs. Therefore, the staff requests that the applicant provide details of a plant specific aging management program to detect cracking of the core support lugs.

FPL RESPONSE:
Primary water stress corrosion cracking (PWSCC) of Alloy 600 core support pads has not been observed to date. Additionally, review of industry operating experience has not identified any PWSCC of Alloy 600 components normally exposed to cold leg temperatures. However, the occurrence of PWSCC in Alloy 600 CRDM housings has created a concern for the possibility of similar degradation. The core support lugs undergo stress relief with the reactor vessel as part of the fabrication process. Because of their location, at the bottom of the vessel and attached to the wall, they also operate at lower temperature levels (approximately 50°F) than the CRDM housings. The core support lugs do not support the internals but act as guides as well as providing anti-rotation resistance to the internals, and are therefore lightly loaded. As a result, the core support lugs are far less susceptible to PWSCC than CRDM housings. Accordingly, visual examination in accordance with ASME Section XI, Examination Category B-N-2 (for integrally welded core support structures on the interior of Reactor Pressure vessels) was determined to be effective for managing PWSCC and wear of core support lugs [reference below]. The Turkey Point ASME Section XI Subsections IWB, IWC, and IWD Inservice Inspection Program currently performs an enhanced VT-3 visual examination on the core support lugs. This enhanced visual examination employs the same resolution requirements as that required by ASME Section XI for VT-1 examinations. For the period of extended operation, the ASME Section XI Subsections IWB, IWC, and IWD Inservice Inspection Program will be enhanced to require ASME Section XI VT-1 examinations of the core support lugs.

Ref. Nuclear Management and Resource Council, Inc., PWR Reactor Pressure Vessel License Renewal Industry Report, NUMARC Report No. 90-04

SECTION 3.2.5 REACTOR VESSEL INTERNALS

RAI 3.2.5-1:
In Section 3.2.5, FPL indicates that the Turkey Point RVI components with fluence greater than 10^{21} n/cm^2 do not include the lower support casting. The staff requests that FPL provide the maximum fluence expected for the lower support casting during the extended period of operation and the basis for that expectation.

FPL RESPONSE:
The subject components in this RAI have been identified as forgings for Turkey Point and will likely be exposed to a fluence in excess of 10^{21} n/cm^2 at the end of the extended period of operation. This is expected to produce some reduction in fracture toughness relative to unirradiated material, as well as increased susceptibility to irradiation assisted stress corrosion cracking (IASCC). LRA Subsection 3.2.5.2.2 (page 3.2-31) will be revised to indicate that the only cast austenitic stainless steel (CASS) components in the reactor vessel internals are the lower support columns, the bottom mounted instrumentation columns, and the upper support column bases. LRA Subsection 3.2.5.2.2 (page 3.2-32) will be revised to include the lower support forgings in the list of components that are potentially susceptible to reduction in fracture toughness due to irradiation embrittlement.

Stress levels in the lower core support forgings are generally sufficiently low, so as not to support crack initiation or growth. The geometry of the lower support forgings, with a regular pattern of large through-holes, provides natural locations for stress relief. Cracking in the lower support forging has not been reported in any operating plant to date.

In addition, the presence of a secondary core support structure between the lower core support forging and the reactor vessel lower head provides an independent, backup mechanism to assure core support functions are performed by the lower support forgings for the period of extended operation.

Reduction in fracture toughness does not lead to crack initiation. However, in the presence of an existing flaw, the likelihood of flaw growth is increased relative to an unirradiated material. For the lower support forgings, the only other possible mechanism affecting wrought stainless steel is IASCC. Over the expected lifetime of the plant, the lower support forgings will be exposed to sufficient fluence to render them susceptible to this mechanism. However, actual crack formation also requires a relatively high local stress and an adverse environment. Therefore, cracking due to IASCC of the lower support forgings is managed by the Chemistry Control Program.

The Chemistry Control Program, as described in LRA Appendix B, Subsection 3.2.4 (page B-47), maintains the chemistry of the RCS including control of oxygen and chlorides. The effectiveness of the Chemistry Control Program has been demonstrated by both laboratory experiments and years of operating experience.

RAI 3.2.5-2:
The baffle assembly contains three categories of baffle bolts that are designated as, former/baffle bolts, barrel former/bolts and baffle/baffle bolts. The staff requests that FPL clarify or provide the basis for not including the baffle/baffle bolts in the baffle assembly bolting described in Subsections 3.2.5.2.2, 3.2.5.2.4 and Table 3.2-1

FPL RESPONSE:
The Turkey Point baffle/baffle bolts (edge bolts) perform no structural function, and are not required to remain functional to maintain the intended functions identified in LRA Table 3.2-1 (page 3.2-77) for the baffle and former plates.

WCAP-14577, Revision 1, "License Renewal Evaluation: Aging Management for Reactor Internals," submitted to the NRC by the Westinghouse Owners Group (WOG) on October 9, 2000, and accepted by NRC Final Safety Evaluation, dated February 10, 2001 identifies that the edge bolts perform no intended function, and are not included in the aging management review for reactor vessel internals.

Additionally, a methodology was developed as part of the Westinghouse Owners Group (WOG) Baffle Bolt Program (refer to response to RAI 3.8.6-1), to evaluate potential bolting patterns in the baffle/former/barrel regions. Applications of this methodology have identified acceptable baffle/former bolting patterns to protect the fuel during LOCA conditions without taking credit for edge bolts.

RAI 3.2.5-3:
In Subsection 3.2.5.2.1, FPL indicates that; susceptibility has been observed at fluences as low as $1x10^{21}$ n/cm^2 in laboratory studies on Type 304 stainless steel in PWR environments, Type 316 stainless steel is less susceptible, and field information suggests that greater exposures are required for the development of susceptibility. The staff requests that FPL identify the field information that suggests that greater exposures are required for the development of susceptibility.

FPL RESPONSE:
In operating PWRs, IASCC has been observed in baffle to former bolts. In very low oxygen environments, representative of PWR environments, the available literature data demonstrate that the lowest fluence at which IASCC susceptibility is observed is 10^{21} n/cm^2 (E>1MeV) (References 1, 2, and 3). Figures 1, 2, and 3, from the aforementioned references show this conclusion and are discussed below.

Figure 1 (Reference 1), shows the threshold type behavior in boiling water reactor (BWR) type environments with field data for observed cracking being at a somewhat higher threshold than laboratory data. The figure also shows that when the oxygen level is reduced to very low levels, closer to PWR conditions, the threshold for the laboratory observation of IASCC moves to greater than 10^{21} n/cm^2.

The data shown in Figure 2 (Reference 2), is for a range of conditions and also shows no susceptibility to IASCC at fluences less than 10^{21} n/cm^2.

Figure 3 (Reference 3), shows the effect of fluence and test temperature on the threshold for laboratory testing of cold worked 316 stainless steel. The lower solid line in Figure 3 is the highest fluence at which zero susceptibility was observed and the higher solid line is the lowest fluence at which susceptibility could be measured.

For temperatures relevant to the observed cracking in PWRs (Reference 4), approximately 600°F (316°C), it can been seen that cracking due to IASCC was not observed until fluences in excess of 10^{21} n/cm^2 were reached in laboratory tests. While gamma heating is important to the overall considerations of material performance, the field failures (Reference 4) have been just under the head of the bolt, at the head to shank fillet, where the fluence and stress are optimized, and the temperature is close to coolant temperature.

The tests that are used to define the susceptibility are very severe and typically involve the slow strain rate test (SSRT). This technique forces failure and the least departure from a 100% dimpled ductile failure is recorded as susceptibility. Thus, it is not surprising that field experience is that no failures have been observed until beyond this fluence level, and then only in a few plants. In France, where the phenomenon was first observed to a significant extent, industry data indicate that the larger number of failures has occurred in those plants that have experienced a larger number of major load cycles, which would be more akin to a series of rising load tests. Plants in the United States, including Turkey Point, tend to be base loaded and do not have the same number of cycles.

Inspections to date in the United States have revealed limited cracking in two plants with annealed Type 347 stainless steel fasteners. Both plants had fluences in the region of 10^{22} n/cm^2. In one plant, all 728 bolts were inspected, 55 had ultrasonic testing (UT) indications and nine were confirmed to be non-functional. In the other plant, 639 bolts were inspected, 59 had UT indications and five were confirmed to be non-functional. UT indications were examined on several bolts from both plants and were shown to be false indications. The presently used inspection techniques appear to be conservative. Limited information on crack growth rates in IASCC indicates that the rates are high. This is supported by the observation of only two or three partially cracked bolts from these plants. It appears that for this component cracking progresses to failure relatively quickly and that very few partially cracked bolts should be present. This means that the inspections give an accurate assessment of the condition of the baffle/former assembly.

At a third plant, with cold worked Type 316 stainless steel fasteners and a fluence in the region of 7 X 10^{21} n/cm^2, 1086 fasteners were UT inspected and no indications were found. In addition, no cracking has been observed in locking devices, baffle plates or several other components that are examined during certain outages.

The laboratory data show a threshold for the first signs of IASCC at 10^{21} n/cm^2. The testing techniques used ensure that this is a conservative measurement. In addition, the limited cracking observed in the field at higher fluences demonstrate that this estimate is also conservative, even for the more highly stressed components.

REFERENCES

1. P.L. Andersen, F. P. For, S. M. Murphy, J. M. Perks; pp 1-83, Proceedings of 4th International Symposium on Environmental Degradation of Materials in Nuclear Power Systems - Water Reactors, Aug. 1989. NACE, 1990.

2. E.D. Eason, E. E. Nelson; pp 1067 - 1079, Proceedings of 7th International Symposium on Environmental Degradation of Materials in Nuclear Power Systems - Water Reactors, Aug. 1995. NACE, 1995.

3. J.F. Williams et al.; pp 725-733, Proceedings of 8th International Symposium on Environmental Degradation of Materials in Nuclear Power Systems - Water Reactors, Aug. 1997. NACE, 1997.

4. R. Cauvin et al.; Proceedings of Conference, "Contribution of Materials Investigation to the Resolution of Problems Encountered in Pressurized Water Reactors," Sept. 1994. pp 54-65, SFEN/ENS 1994.

FIGURE 1

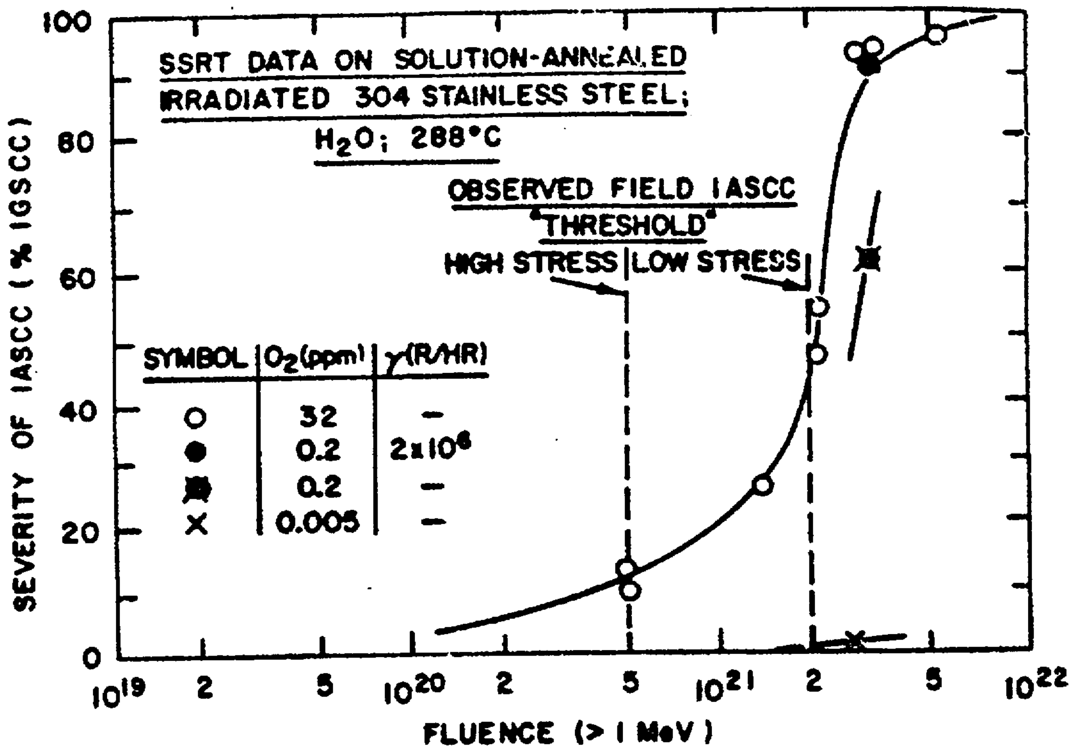

FIGURE 1. Effect of Fluence and Oxygen Level on the IASCC Susceptibility of 304 S.S. (Reference 1)

FIGURE 2

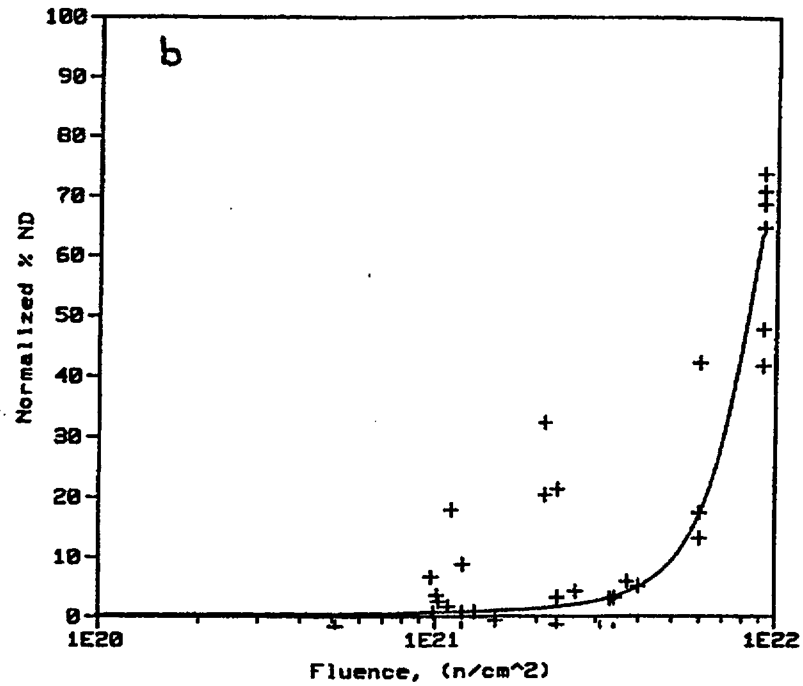

FIGURE 2. IASCC Susceptibility of 300 Series Stainless Steels in Various Environments as a Function of Fluence (Ref. 2

FIGURE 3

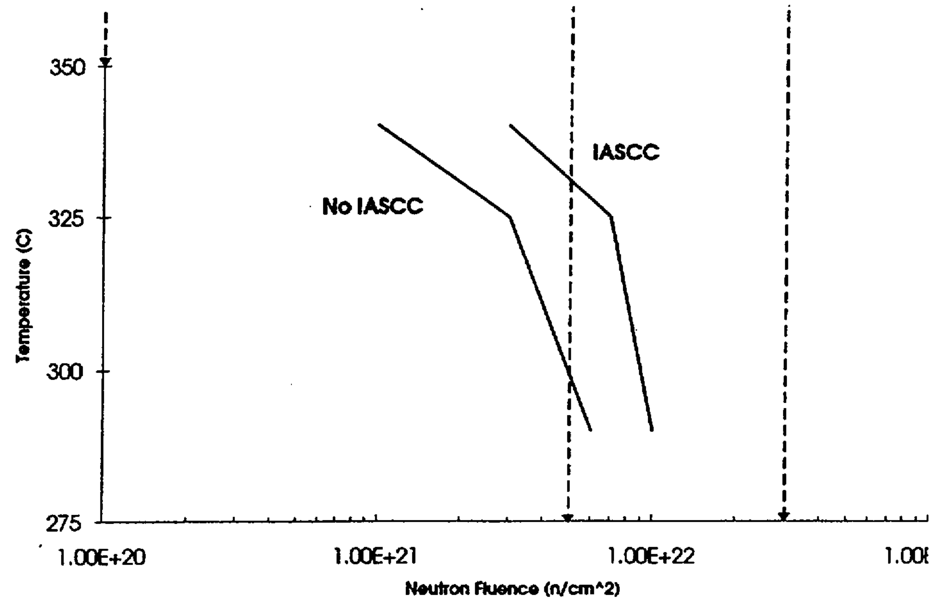

FIGURE 3. Effect of Fluence and Temperature on the IASCC of CW 316 S.S. (Reference 3)

RAI 3.2.5-4:
In Subsection 3.2.5.2.4, FPL indicates, in part, that significant data, information and industry experience relative to the aging of baffle bolting is provided in WCAP-14577 (Reference 3.2-7) and is not duplicated in the Subsection. Reference 3.2-7 is WCAP-14577, " License Renewal Evaluation: Aging Management for Reactor Internals," Revision 0, dated June 1997. This edition of WCAP-14577 predates NRC Information Notice 98-11, "Cracking of Reactor Vessel Internal Baffle Former bolts in Foreign Plants" and the WOG activities that developed the significant data, information and industry experience with regard to the baffle bolt cracking issue. Subsequent to the topical report submittal, the WOG had periodic meetings and interactions with the staff to present information, data and industry experience with regard to its ongoing baffle bolt program. The WOG program and activities included the development of an analytical methodology for minimum baffle bolting distributions under faulted conditions and plant baffle bolting inspections and bolting replacement activities.

The staff issued several RAIs (by letter from Raj K. Anand (NRC) to Roger A, Newton (WOG) dated June 14, 1999) with regard to updating WCAP-14577 Revision 0 and the WOG's plans for use of the results of the technical progress, the WOG's and licensee's commitments to participation and use of the industry's PWR Materials Reliability Project (MPR) initiatives with regard to the RVI aging management issues, conclusions and recommendations. WOG responses to the RAIs are contained in a letter from Roger A Newton (WOG) to Raj K. Anand (NRC) dated November 24, 1999

The staff requests that FPL review the staff RAIs, the associated WOG responses, and address the RAIs and their applicability and inclusion with regard to the Turkey Point Units 3 and 4 license renewal application. In addition, FPL&L is requested to provide responses to each of the renewal applicant action items provided in the final safety evaluation report issued by the staff for WCAP-14577 (these are repeated below for convenience):

Renewal Applicant Action Items from FSER for WCAP-14577

(1) To ensure applicability of the results and conclusions of WCAP-14577 to the applicant's plant(s), the license renewal applicant is to verify that the critical parameters for the plant are bounded by the topical report. Further, the renewal applicant must commit to programs described as necessary in the topical report to manage the effects of aging during the period of extended operation on the functionality of the reactor vessel components. Applicants for license renewal will be responsible for describing any such commitments and proposing the appropriate regulatory

controls. Any deviations from the aging management programs described in this topical report as necessary to manage the effects of aging during the period of extended operation and to maintain the functionality of the reactor vessel internal components or other information presented in the report, such as materials of construction, must be identified by the renewal applicant and evaluated on a plant-specific basis in accordance with 10 CFR 54.21(a)(3) and (c)(1).

(2) A summary description of the programs and activities for managing the effects of aging and the evaluation of TLAAs must be provided in the license renewal FSAR supplement in accordance with 10 CFR 54.21(d).

(3) For the holddown spring, applicants for license renewal are expected to address intended function, aging management review, and appropriate aging management program(s).

(4) The license renewal applicant must address aging management review, and appropriate aging management program(s), for guide tube support pins.

(5) The license renewal applicant must explicitly identify the materials of fabrication of each of the components within the scope of the topical report. The applicable aging effect should be reviewed for each component based on the materials of fabrication and the environment.

(6) The license renewal applicant must describe its aging management plans for loss of fracture toughness in cast austenitic stainless steel RVI components, considering the synergistic effects of thermal aging and neutron irradiation embrittlement in reducing the fracture toughness of these components.

(7) The license renewal applicant must describe its aging management plans for void swelling during the license renewal period.

(8) Applicants for license renewal must describe how each plant-specific AMP addresses the following elements: (1) scope of the program, (2) preventative actions, (3) parameters monitored or inspected, (4) detection of aging effects, (5) monitoring and trending, (6) acceptance criteria, (7) corrective actions, (8) confirmation process, (9) administrative controls, and (10) operating experience.

(9) The license renewal applicant must address plant-specific plans for management of cracking (and loss of fracture toughness) of RVI components, including any plans for augmented inspection activities.

(10) The license renewal applicant must address plant-specific plans for management of age-related degradation of baffle/former and barrel/former bolting, including any plans for augmented inspection activities.

(11) The license renewal applicant must address the TLAA of fatigue on a plant-specific basis.

FPL RESPONSE:
FPL reviewed and addressed the NRC RAIs and the Westinghouse Owners Group (WOG) responses in the Turkey Point aging management review performed on the reactor vessel internals. Applicable information regarding the RAIs and their responses was included in the Turkey Point LRA. Specific references to the RAI responses are included as References 2.3-9 on page 2.3-43 and 3.2-8 on page 3.2-53 of the LRA. FPL has reviewed the NRC Final Safety Evaluation, and specific responses to each Applicant Action Item are provided below.

(1) LRA Subsections 2.3.1.6 (page 2.3-10) and 3.2.5 (page 3.2-29) provides a summary of the comparison of the critical parameters and attributes of Turkey Point to WCAP-14577 and describe the WCAP applicability to Turkey Point.

(2) Programs necessary to manage the effects of aging for the Turkey Point reactor vessel internals are the Reactor Vessel Internals Inspection Program, the ASME Section XI, Subsection IWB, IWC, and IWD Inservice Inspection Program, and the Chemistry Control Program. Summary descriptions of these programs are provided in the LRA FSAR Supplement, Appendix A, Subsections 16.1.6 (page A-34), 16.2.1 (page A-34), and 16.2.4 (page A-36), respectively. As stated in LRA Subsection 3.2.5 (page 3.2-29), the only TLAA applicable to the Turkey Point reactor vessel internals is fatigue. A summary description of the fatigue TLAA evaluation is provided in the LRA FSAR Supplement, Appendix A, Subsection 16.3.2 (page A-44).

(3) The information on the reactor vessel internals holddown springs is provided in LRA Subsection 3.2-5 (pages 3.2-29 through 3.2-36) and in Table 3.2-1 (page 3.2-78).

(4) The information on the reactor vessel internals guide tube support pins is provided in LRA Subsection 3.2-5 (pages 3.2-29 through 3.2-36) and in Table 3.2-1 (page 3.2-77).

(5) Upon further review of the plant-specific reactor vessel internals materials and environments, FPL has identified the following:

- The lower support castings identified in LRA Table 3.2-1 (page 3.2-78) are forgings.
- The bottom mounted instrumentation columns identified in LRA Table 3.2-1 (page 3.2-76) are cast stainless steel.
- The lower support columns identified in LRA Table 3.2-1 (page 3.2-76) are cast stainless steel.
- The upper support column bases (new line item for LRA Table 3.2-1 on page 3.2-77) are cast stainless steel but not exposed to a fluence greater than 10^{21} n/cm^2.
- The lower support forgings will be exposed to a fluence in excess of 10^{21} n/cm^2 as discussed in the response to RAI 3.2.5-1.

With the exception of the changes discussed above, the specific materials of fabrication and environments for all parts of the Turkey Point reactor vessel internals which require aging management review are provided in LRA Subsection 3.2.5.1 (page 3.2-30) and in Table 3.2-1 (pages 3.2-76 through 3.2-79). Changes to Table 3.2-1 as a result of the above are included in the following tables.

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TABLE 3.2-1 (continued) REACTOR COOLANT SYSTEMS					
COMPONENT/ COMMODITY GROUP	**INTENDED FUNCTION**	**MATERIAL**	**ENVIRONMENT**	**AGING EFFECTS REQUIRING MANAGEMENT**	**PROGRAM/ACTIVITY**
REACTOR VESSEL INTERNALS					
Lower support columns	Core support	Stainless steel - cast	Treated water - primary	Cracking	Chemistry Control Program
				Cracking Reduction in fracture toughness	ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program • Examination Category B-N-3 Reactor Vessel Internals Inspection Program

LRA Page 3.2-77

TABLE 3.2-1 (continued) REACTOR COOLANT SYSTEMS					
COMPONENT/ COMMODITY GROUP	**INTENDED FUNCTION**	**MATERIAL**	**ENVIRONMENT**	**AGING EFFECTS REQUIRING MANAGEMENT**	**PROGRAM/ACTIVITY**
REACTOR VESSEL INTERNALS					
Upper support column bases	Guide and support RCCAs	Stainless steel - cast	Treated water - primary	Cracking	Chemistry Control Program
				Reduction in fracture toughness	ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program • Examination Category B-N-3 Reactor Vessel Internals Inspection Program

LRA Page 3.2-78

TABLE 3.2-1 (continued) REACTOR COOLANT SYSTEMS					
COMPONENT/ COMMODITY GROUP	**INTENDED FUNCTION**	**MATERIAL**	**ENVIRONMENT**	**AGING EFFECTS REQUIRING MANAGEMENT**	**PROGRAM/ACTIVITY**
REACTOR VESSEL INTERNALS					
Bottom mounted instrumentation columns	Guide and instrumentation support	Stainless steel - cast	Treated water - primary	Cracking	Chemistry Control Program
				Cracking Reduction in fracture toughness	ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program • Examination Category B-N-3 Reactor Vessel Internals Inspection Program
Upper instrumentation columns	Guide and instrumentation support	Stainless steel	Treated water - primary	Cracking	Chemistry Control Program
Lower support forging	Core support Flow distribution	Stainless steel	Treated water - primary	Cracking	Chemistry Control Program
				Cracking Reduction in fracture toughness	ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program • Examination Category B-N-3 Reactor Vessel Internals Inspection Program

(6) Considering the response to item (5) above, the only cast austenitic stainless steel (CASS) reactor vessel internals components within the scope of license renewal are the lower support columns, the bottom mounted instrumentation columns, and the upper support column bases. Of these components, only the lower support columns will be subjected to fluences of greater than 10^{21} n/cm^2. Accordingly, synergistic effects of thermal aging and irradiation embrittlement in reducing the fracture toughness will be a consideration for the lower support columns. As noted in item (5) above and in LRA Table 3.2-1 (pages 3.2-76 through 3.2-79), reduction in fracture toughness will be managed by the Reactor Vessel Internals Inspection Program, as described in LRA Appendix B, Subsection 3.1.6 (page B-21).

(7) Aging management plans regarding dimensional change due to void swelling of the Turkey Point reactor vessel internals are discussed in LRA Subsection 3.2.5.2.6 (page 3.2-33). These plans are included in the Reactor Vessel Internals Inspection Program which is described in LRA Appendix B, Subsection 3.1.6 (page B-21).

(8) The programs necessary to manage the effects of aging of the Turkey Point reactor vessel internals are the Reactor Vessel Internals Inspection Program, the ASME Section XI, Subsection IWB, IWC, and IWD Inservice Inspection Program, and the Chemistry Control Program. The descriptions of these programs, provided in LRA Appendix B, Subsections 3.1.6 (page B-21), 3.2.1.1 (page B-27), and 3.2.4 (page B-47), respectively, address the 10 elements identified (two elements, corrective action and administrative controls are common to all programs and are described in LRA Appendix B Section 2.0 (page B-5)).

(9) Aging management plans to address cracking and reduction in fracture toughness of the Turkey Point reactor vessel internals are discussed in LRA Subsections 3.2.5.2.1 (page 3.2-30) and 3.2.5.2.2 (page 3.2-31), respectively. The programs necessary to manage cracking and reduction in fracture toughness are the Reactor Vessel Internals Inspection Program, the ASME Section XI, Subsection IWB, IWC, and IWD Inservice Inspection Program, and the Chemistry Control Program. The descriptions of these programs are provided in LRA Appendix B, Subsections 3.1.6 (page B-21), 3.2.1.1 (page B-27), and 3.2.4 (page B-47), respectively. The Reactor Vessel Internals Inspection Program includes inspection activities for cracking and reduction in fracture toughness.

(10) Aging management plans to address loss of mechanical closure integrity of the Turkey Point baffle/former and barrel/former bolting are discussed in LRA Subsection 3.2.5.2.4 (page 3.2-33). Note that these plans also consider information provided in WCAP-14577, Revision 1, "License Renewal Evaluation: Aging Management for Reactor Internals," submitted to the NRC by the Westinghouse Owners Group (WOG) on October 9, 2000. The program necessary to manage loss of mechanical closure integrity of this bolting is the Reactor Vessel Internals Inspection Program. The description of this program is provided in LRA Appendix B, Subsection 3.1.6 (page B-21). The Reactor Vessel Internals Inspection Program includes augmented inspection activities as they apply to loss of mechanical closure integrity of the baffle/former and barrel/former bolting.

(11) A description of the plant specific fatigue TLAA evaluation performed for Turkey Point is provided in LRA Section 4.3 (pages 4.3-1 through 4.3-13). Also, refer to response to RAI 3.2.5-7.

RAI 3.2.5-5:
In Subsection 3.2.5.2.6, FPL discusses RVI material dimensional changes and cites references that indicate the material may be subject to various levels of dimensional changes resulting from void swelling under certain conditions. One reference cited in the discussion concludes that at the approximate RVI end-of-life dose of 100 displacements per atom, swelling would be less than 2% at irradiation temperatures between 572°F and 752°F. In the discussion FPL indicates that, field service experience in PWR plants have not shown any evidence of swelling, and at present there have been no indications from the different reactor vessel internals bolt removal programs, or from any of the other inspection and functional evaluations (e.g., refueling), that there are any discernible effects attributable to swelling.

The staff requests that FPL identify specific examples of field service experience, bolt removal programs, and other inspections and functional evaluations with detailed descriptions of the examinations, inspections and evaluations that have been performed to support the conclusion that there is not any evidence of, or any discernible effects attributable to swelling.

The staff requests that FPL describe the change in loading on the baffle bolt and its impact on the bolt integrity that would occur if the thickness of the baffle material located under the bolt head were subjected to a 2% or less dimensional change due to swelling.

FPL RESPONSE:
Field service experience is derived from refueling outages and inservice inspections performed on plants since their startup. The effect of void swelling would be the closure of gaps and physical deformation caused by localized dimensional increases. The absence of observable physical deformation is indicative of the absence of significant swelling.

Fuel assemblies at Turkey Point and other nuclear plants have been unloaded, shuffled, and replaced for over 20 years with no indication of interference within the baffle envelop which would indicate deformation due to void swelling. Over the same time period, reactor vessel internals have been removed from the vessel to allow inspections, then replaced without incident, indicating no general or localized dimensional deformation which would be the result of void swelling. No closure of interface gaps has been apparent, based on the continued ability to remove and replace the reactor vessel internals.

Data of swelling are currently being evaluated as part of industry programs. Baffle/former bolts have been removed from several Westinghouse plants due to isolated failures (presently attributed to IASCC). Several of the bolts were subjected to detailed hot-cell micrographic examination and instances of void formation were observed. The measured volumetric changes were less than 0.03%. The swelling was located in the threaded region of the bolt where fluence and gamma heating provided the requisite conditions. Swelling was not observed in the head/shank transition of these bolts where the fluence or temperature conditions were less. It is also important to note that the bolts were readily removed from the baffle assembly. It is possible that similar isolated spots of combined high temperature/high fluence will exist in the baffle plates.

The issue of bolt integrity following a 2% change in metal thickness is clarified as follows: (This information was provided to Westinghouse Electric Company by F.A. Garner.)

> The stresses developed by void formation will be limited by irradiation creep. Void swelling and irradiation creep are intimately connected through their relationship to the local stress state. Initially irradiation creep exists prior to the onset of swelling, and it will relieve any applied or thermally-induced stresses. Once swelling begins, a new much larger component of creep develops that is directly proportional to the instantaneous swelling rate. Therefore, any swelling-induced stress (arising from gradients of swelling or differential swelling of two components) will be relaxed at a rate proportional to the swelling rate. This leads to a maximum stress well below 200 MPa, regardless of the local swelling rate. Thus, the yield stress can never be exceeded for a typical bolt application. The stress is maintained as long as the swelling rate difference is minimal.
>
> Irradiation creep always attempts to move mass via dislocation movement in a direction that will relieve the shear component of the stress state. In a bolt shank, the primary stress is tensile along the bolt axis, but the resolved shear stress moves the dislocation in such a way as to lengthen the bolt shank and reduce its diameter. The same process will operate in a clamped plate with the result that the plate will exhibit some preferred lateral growth rather than just extension of the bolt.

The field service experience, plus the hot cell-evaluations of baffle/former bolts, indicate that localized void swelling is presently much less than 2% and reasonable extrapolations to the end of life suggest that it will remain small.

RAI 3.2.5-6:
The application uses 1 x 10^{21} n/cm^2 (E > 0.1 MeV) as a fluence threshhold for neutron embrittlement of stainless steel used to fabricate internal components. Provide data to support this position, or revise the application to expand the list of potentially susceptible components to include those at lower fluences.

FPL RESPONSE:
In 10 CFR 50 Appendices G and H, the fluence level of 10^{17} n/cm^2 appears to originate from the data for the onset of embrittlement in ferritic pressure vessel steels. In Appendix G, the introduction and scope refers to "fracture toughness requirements for ferritic materials of pressure retaining components of the reactor coolant pressure boundary of light water nuclear power reactors." In Appendix H, the introduction refers to "changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region of light water nuclear power reactors which result from exposure of these materials to neutron irradiation and the thermal environment." Additionally, Appendix G defines a ferritic material as "carbon and low-alloy steels, higher alloy steels, including all stainless alloys of the 4xx series, and maraging and precipitation hardening steels with a predominantly body-centered cubic crystal structure." The Turkey Point reactor vessel internals components do not match the definitions of Appendices G or H.

The fluence threshold for neutron embrittlement of stainless steel is expected to be much higher than 10^{17} n/cm^2 in wrought stainless steel, stainless steel welds, and cast stainless steels. The primary mechanism for the embrittlement in ferritic steels is the precipitation of a copper rich phase with possible contributions from nickel and phosphorous. The mechanism of thermal embrittlement below 500°C in the delta ferrite of CASS or of austenitic welds is primarily due to the spinodal decomposition of the chromium rich ferrite to produce variations in chromium content in the ferrite which is referred to as alpha prime embrittlement. There is no copper in these materials. The mechanisms for embrittlement are different for the two types of ferrite. In addition, the wrought 300 series stainless steels under consideration have no or very little ferrite content and the cast components have less than 50% ferrite. Thus, both materials have a predominantly face centered cubic microstructure and do not in any way fall under the definition of a ferritic material. The face centered cubic austenite has greater toughness and continues to behave in a ductile manner until greater fluences are achieved.

When considering the embrittlement of stainless steel, it is important to define the different types of stainless steel that are to be evaluated. For a PWR, the important structural materials that receive the highest fluence are the wrought, austenitic 300 series stainless steels, usually 304, 316, and 347. The wrought stainless steels are not susceptible to thermal embrittlement of the type considered for the cast stainless steels, so only the effect of neutrons is considered for these materials. There is relatively little data on the effect of neutron irradiation on the fracture toughness of these steels that is directly relevant to PWR applications. Much of the data is at higher temperatures.

The fracture toughness of unirradiated stainless steels is sufficiently high at >200MPa$\sqrt{m}$ (Reference 5) that there has been little concern for such fracture in these materials. After irradiation, the fracture toughness does exhibit reductions accompanied by reductions in tearing modulus. However, this reduction does not become apparent until significant dose has been accumulated. It has been shown that there is no loss of toughness for a Type 316H stainless steel at fluences to 1 displacement per atom (dpa) (7×10^{20} n/cm^2 for a PWR spectrum) at 370°C (Reference 6). Data for 304 and 316 show that there is no significant effect of irradiation toughness until fluences of 2.41 (1.68×10^{21} n/cm^2) and 1.6 dpa (1.12×10^{21} n/cm^2), respectively (Reference 7). Other data (Reference 8), showed some loss of toughness at 3 dpa, but it remained relatively high at >150MPa$\sqrt{m}$, although much of this data was on small specimens and possibly not valid for the saturation toughness value (K_{JC}) toughness measurements. The data of Mills (Reference 9) and a subsequent review (Reference 10) support the presence of a threshold for toughness of approximately 10^{21} n/cm^2, see Figure 4. After the onset of embrittlement, there is a decline until a saturation value is reached at approximately 10 dpa (7×10^{21} n/cm^2) , see Figure 4. For the wrought austenitic materials, K_{JC} is in the region of 70 MPa$\sqrt{m}$, see Table 1, (References 9 and 10). The toughness values were calculated from $K_{JC} = \sqrt{EJ_{IC}}$

TABLE 1 FRACTURE TOUGHNESS OF STAINLESS STEEL AFTER IRRADIATION					
MATERIAL	**DOSE (dpa)**	**JC (kJ/m)**	**DJ/da (MPa)**	**T**	**KJC (MPa√m)**
316	10 to 17	31	19	8	71
304	15	28	13	5	68
308 weld	14	11	17	8	43

The toughness of the baffle to former bolts removed from US plants was demonstrated by the tensile testing of the bolts. Industry data indicated that all bolts exhibited ductile failures with significant elongation to fracture. The tensile properties for the annealed 347 stainless steel bolts from one plant are presented in Table 2. Data from the other 347 plant was similar.

TABLE 2 TENSILE DATA SUMMARY FOR 347 SS BOLTS AFTER IN-PLANT IRRADIATON IN FLUENCES APPROACHING 10^{22} n/cm^2	
PROPERTY	**IRRADIATED VALUE**
Yield Strength	80 - 160 ksi
Ultimate Strength	110 - 176 ksi
Elongation	60% - 30%

It is apparent that these fasteners exhibit excellent elongation to failure, even at the high yield strengths measured. Similar good properties were found for the cold worked 316 bolts removed from another plant. The typical properties for bolts in the active length of the core are listed in Table 3.

TABLE 3 TENSILE DATA SUMMARY FOR 316 SS BOLTS AFTER IN-PLANT IRRADIATON IN FLUENCES APPROACHING 7 X 10^{21} n/cm^2 (188 bolts tested)	
PROPERTY	**IRRADIATED VALUE**
Yield Strength	104 - 154 ksi
Ultimate Strength	117 - 154 ksi
Elongation	33% - 20%

Ductile behavior was also obtained for an irradiated, cold worked 316 stainless steel bolt which had been saw cut through 50% of the section at the head to shank fillet.

The tensile data generated from the small specimens removed from the bolts gave smaller elongations to fracture with uniform elongations of less than 10%. This data coupled with the bolt data suggest a size effect in the tensile testing and that the actual bolts perform significantly better than the small specimen data would suggest.

Thus, for wrought austenitic stainless steels, the laboratory and recent field data does not indicate any degradation in toughness or any other sign of embrittlement until fluences beyond 10^{21} n/cm^2. Significant toughness exists at higher fluences when a saturation value is reached.

There appears to be a similar reduction in toughness at fluences in the region of 10^{21} n/cm^2 for cast austenitic stainless steels, such as CF-8 and CF-8M, and for welds such as 308 and 309. The data for many welds irradiated at intermediate temperatures (370°C to 430°C), even those with high molybdenum, show that "exposures up to 1 dpa have no significant effect on fracture resistance (Reference 7)." The literature data on castings is limited (Reference 10). That which is available indicates that the fracture toughness of an SA351 CF8 casting with 15% delta ferrite behaves in a similar manner to a 308 stainless steel weldment with 7% delta ferrite when irradiated (Reference 10). The data is less than that observed for the wrought material (see Table 1) suggesting that the saturation value is less, probably due to the ferrite in the structure. The austenitic phase will continue to fail in a ductile manner, as shown previously. The threshold before signs of embrittlement are observed appears to be similar to that for the wrought austenitic materials, see Figure 4.

The literature data is somewhat limited on the subject of neutron embrittlement of the wrought and cast stainless steels used for PWR core structures. Existing data shows that significant reductions in toughness are not experienced until fluences of 10^{21} n/cm^2 are reached. The field data for the tensile testing of annealed 347 stainless steel and cold worked 316 stainless steel fasteners has demonstrated that there is a large strain to fracture in these components even after in core exposure of fluences of 7 X 10^{21} n/cm^2 and greater.

REFERENCES

5. W.H. Bamford, A. J. Bush; pp 553 - 577, ASTM STP 668, 1979.

6. J. Bernard, G. Verzeletti, pp 619 - 641, ASTM STP 870, 1985.

7. J. Dufresne, B. Henry, H. Larsson; pp 511-528, ASTM STP 683, 1979.

8. J.E. Pawel, Journal of Nuclear Materials, pp 212 - 215, 442, 1994.

9. W.J. Mills, Nuclear Technology, pp 290 - 303, V.82, Sept. 1988.

10. W.J. Mills, International Materials Review, pp 45 - 84, V.42, #2, 1997.

11. J. A. Begley, J. D. Landes; ASTM STP 514, 1974.

FIGURE 4

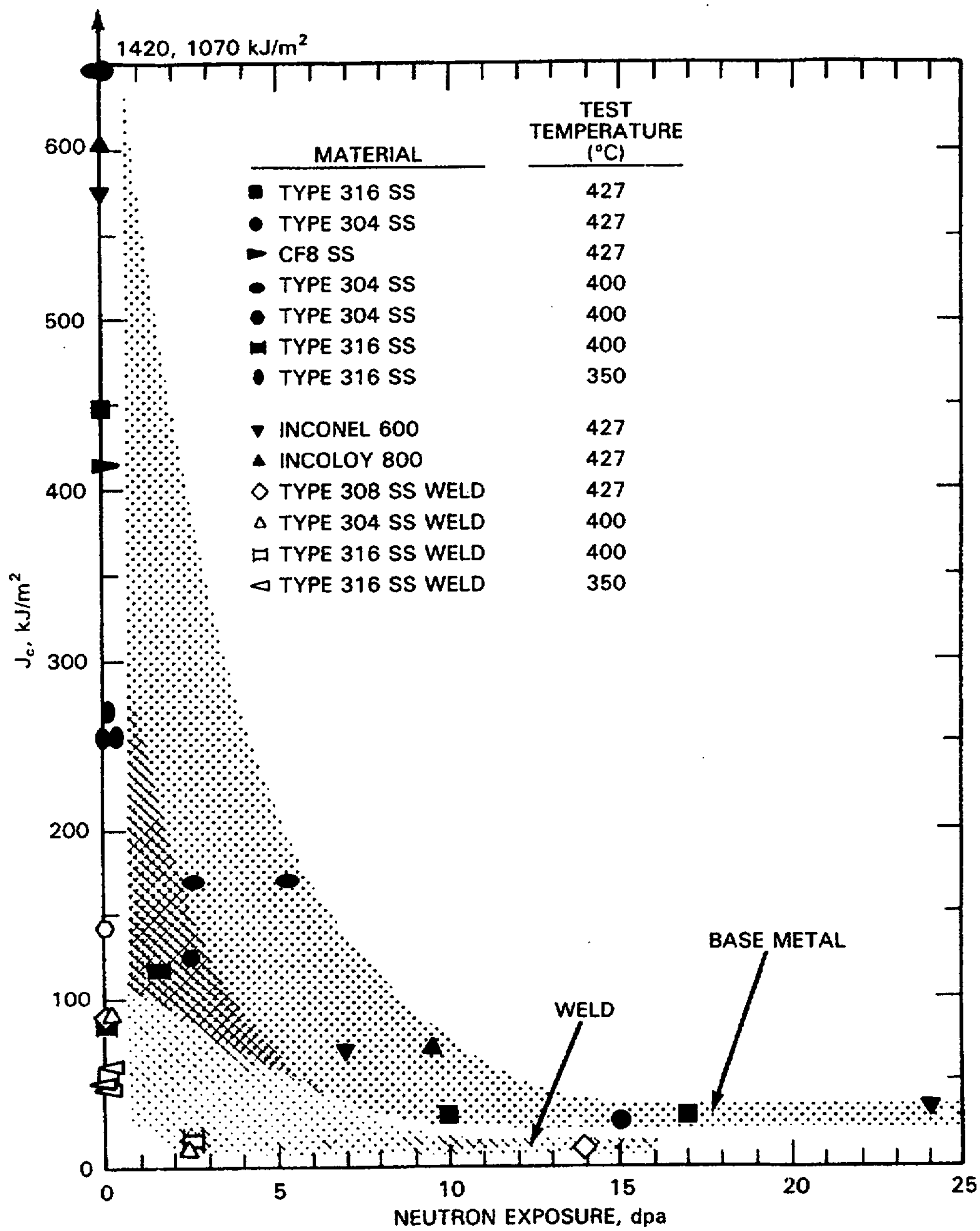

FIGURE 4. Fracture Toughness Behavior as a Function of Neutron Dose (Reference 10)

RAI 3.2.5-7:
In Section 3.2.5 of the application, FPL states that, "Turkey Point's TLAA identification effort also identified fatigue as the only TLAA applicable to the reactor vessel internals. Fatigue of the reactor vessel internals is addressed in Subsection 4.3.1."

The staff requests that FPL provide a list of the TLAAs associated with fatigue used in verifying that the structural integrity of the reactor vessel internals were evaluated and determined to remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

FPL RESPONSE:
As required by 10 CFR 54.21(c), an extensive review of the Turkey Point current licensing basis was performed to identify TLAAs requiring evaluation for license renewal. This review is documented in a detailed engineering evaluation that includes a description of the TLAA identification process, evaluation results, and summary tables. This evaluation is available on site for NRC review. In accordance with 10 CFR 54.21(c)(1), a list of TLAAs is included in Table 4.1-1 (page 4.1-5) of the Turkey Point LRA. Metal fatigue of ASME Section III Class 1 components is identified as a TLAA in LRA Table 4.1-1 and, accordingly, metal fatigue of the reactor vessel internals is evaluated in LRA Subsection 4.3.1 (page 4.3-2).

A fatigue evaluation was performed on the Turkey Point reactor vessel internals in support of the thermal power uprating of the units in the mid-1990s (Reference - Turkey Point Units 3 and 4 Operating License Amendment 191/185, issued September 26, 1996). A specific fatigue analysis was performed on the lower core plate because the temperature differences between components of the lower support assembly induce significant thermal stresses in the lower core plate. This analysis determined that the maximum fatigue usage factor in the lower core plate was 0.55 based on the 40 year design cycles for Turkey Point. This value is well within the ASME Code acceptance limit of 1.0. Additionally, simplified assessments performed on the other parts of the reactor vessel internals as part of the same uprating effort determined that the parts were structurally adequate for the new Reactor Coolant System conditions. As indicated in LRA, Subsection 4.3.1 (page 4.3-2) and in the response to RAI 4.3.1-1, the existing 40 year design cycles and cycle frequencies were determined to be conservative and bounding for the period of extended operation. Accordingly, the analysis associated with verifying the structural integrity of the reactor vessel internals has been evaluated and determined to remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

SECTION 3.2.7 STEAM GENERATORS

RAI 3.2.7-1:
In Section 3.2.7.2.2 (Loss of Material) of the license renewal application, the aging mechanisms that can cause loss of material for the steam generators are listed. However, industry operating experience indicated that erosion (aging mechanism) could cause the loss of section thickness (aging effect) of a component, and this aging effect is not addressed in the application. One example of this aging effect is the loss of section thickness of the feedwater impingement plate supports in the Harris Nuclear Plant steam generators. Provide the plant specific aging management program for this aging effect in general for the steam generators and other components in the plant within the scope of license renewal for the period of extended operation.

FPL RESPONSE:
The feedwater impingement plate design at the Harris Nuclear Plant is not present in the Turkey Point Plant steam generators. The Turkey Point steam generator tube support system is stainless steel and is not susceptible to erosion. While other steam generator components are inspected for loss of material due to erosion as part of the steam generator integrity program, (e.g., steam separators including associated supports and internal decking, the feedring and J-tubes, and internal blowdown piping) they are not within the license renewal scope because they do not perform a license renewal intended function. Loss of material due to erosion is not an aging effect requiring management for steam generator components within the scope of license renewal.

As discussed in Section 5.1 of Appendix C (page C-17) of the LRA, general erosion occurs under high velocity conditions, turbulence, and impingement. Geometric factors are extremely important. Systems and components are designed to preclude erosion mechanisms such as liquid impingement, flashing, and cavitation. Erosion mechanisms would typically be discovered early in a component's life. The only components identified through the aging management review process as subject to loss of material due to erosion are the Emergency Containment Coolers (ECCs). The Emergency Containment Coolers Inspection as described in Appendix B, Subsection 3.1.3 (page B-14) of the LRA is credited for managing this aging effect.

RAI 3.2.7-2:
Table 3.2-1 states that the applicable AMP for steam generator internals (e.g., the steam generator tube support plates) is the chemistry control program. However, FPL's response to Generic Letter (GL) 97-06, "Degradation of Steam Generator Internals," indicates that significantly more activities are undertaken to manage the potential degradation of steam generator internal components. In addition, the scope of the steam generator integrity program AMP includes steam generator secondary-side integrity inspections. Identify any additional AMPs (e.g., steam generator integrity program) that are applicable to steam generator internals in Table 3.2-1.

FPL RESPONSE:
Steam generator internals are inspected as part of the current licensing basis via the Steam Generator Integrity Program. However, with the exception of the tubes and tube plugs, this program is not credited for managing aging effects associated with components within the scope of license renewal as described in response to RAI 3.9.14-1 (FPL letter L-2001-65). The Chemistry Control Program as described in Appendix B, Section 3.2.4 (page B-47) of the LRA, manages the aging effects of loss of material and cracking for internal components of the Steam Generators.

RAI 3.2.7-3:
FPL identified a "loss of mechanical closure integrity" as the aging effect requiring management for primary bolting. Section 3.2.7.2.3 of the LRA identifies stress relaxation and/or aggressive chemical attack as two potential causes of a loss of mechanical closure integrity. However, industry operating experience indicates that a loss of mechanical closure integrity can also result from stress corrosion cracking.

A) Section 5.4 of Appendix C of the LRA discusses the "loss of mechanical closure integrity" aging effect. The last paragraph of section 5.4 briefly discusses stress corrosion cracking. Describe, more thoroughly, the actions taken by FPL (e.g., the use of non-susceptible material and/or the use of non-aggressive lubricants) to prevent stress corrosion cracking in primary bolting. Operating experience has shown that some alloy steels with lower yield strengths are susceptible to stress corrosion cracking. Identify the range of yield strengths used at Turkey Point Units 3 and 4 and the susceptibility of those material strengths.

B) Several NRC generic communications (e.g., NRC IE Bulletin 82-02, "Degradation of Threaded Fasteners in the Reactor Coolant Pressure Boundary of PWR Plants" and NRC Generic Letter 91-17, "Bolting Degradation or Failure in Nuclear Power Plants") provide information on industry operating experience associated with the degradation of primary bolting, but are not referenced by FPL in Section 3.2.7.3.1 of the LRA. Explain why these generic communications were not identified as reference documents and whether the information contained within was assessed for Turkey Point Units 3 and 4.

FPL RESPONSE:
A) High stress in conjunction with an aggressive environment can cause cracking of certain bolting materials due to stress corrosion cracking (SCC). As identified in NRC IE Bulletin 82-02 and Generic Letter 91-17, cracking of bolting in the industry has occurred due to SCC. These instances of SCC have been primarily attributed to the use of high yield strength bolting materials, excessive torquing of fasteners and contaminants, such as the use of lubricants containing molybdenum disulfide (MoS_2). In response to NRC IE Bulletin 82-02, Turkey Point verified that:

(a) Specific maintenance procedures were in place that address bolted closures of the Reactor Coolant pressure boundary with a nominal diameter of 6 inches or greater.

(b) The procedures in use addressed detensioning and retensioning practices and gasket installation and controls.

(c) Threaded fastener lubricants used in the reactor coolant pressure boundary have specified maximum allowable limits for chloride and sulfur content to minimize susceptibility to SCC environments.

(d) Maintenance crew training on threaded fasteners is performed.

In order for SCC to occur, three conditions must exist: a susceptible material, high tensile stresses, and a corrosive environment. At Turkey Point, the potential for SCC of fasteners is minimized by utilizing ASTM A193, Gr. B7 bolting material and limiting contaminants such as chlorides and sulfur in lubricants and sealant compounds. Additionally, sound maintenance bolt torquing practices are used to control bolting material stresses. The use of ASTM A193, Gr. B7 bolting specifies a minimum yield strength of 105 Ksi, which is well below the 150 Ksi threshold value specified in EPRI NP-5769, "Degradation of Bolting in Nuclear Power Plants", April 1988. Bolting fabricated in accordance with this standard could be expected to have yield strengths less than 150 Ksi. However, since the maximum yield strength is not specified for this bolting material, absolute assurance can not be provided that the yield strength of the bolting would not exceed 150 Ksi. For these cases, the combination of specifying ASTM A193 Gr. B7 bolting material, control of bolt torquing and control of contaminants will ensure that SCC will not occur. These actions have been effective in eliminating the potential for SCC of bolting materials. The results of a review of the Turkey Point condition report (1992 through 2000) and metallurgical report (1987 through 2000) databases support this conclusion in that no instances of bolting degradation due to SCC were identified. Additionally, review of NRC generic communications did not identify any recent bolting failures attributed to SCC. Therefore, cracking of bolting material due to SCC is not considered an aging effect requiring management at Turkey Point.

B) NRC generic communications (NRC IE Bulletin 82-02, "Degradation of Threaded Fasteners in the Reactor Coolant Pressure Boundary of PWR Plants" and NRC Generic Letter 91-17, "Bolting Degradation or Failure in Nuclear Power Plants") were assessed for Turkey Point as described in response to Part A) above. However, these generic communications were inadvertently omitted from LRA Subsections 3.2.1.1.3, 3.2.2.3.1, 3.2.3.3.1 (GL 97-01 only), 3.2.4.3.1, 3.2.5.3.1 (GL 97-01 only), 3.2.6.3.1 and 3.2.7.3.1.

RAI 3.2.7-4:
Table 3.2-1 lists the ASME Section XI, Subsections IWB, IWC and IWD Inservice Inspection Program as an AMP for steam generator U-tubes. ASME Examination Categories B-Q and B-P are identified as applicable to U-tubes. Explain the type of inspections associated with examination categories B-P and B-Q that are applicable to steam generator U-tubes, plugs and sleeves (if installed in the future).

FPL RESPONSE:
Category B-Q volumetric examinations include eddy current and ultrasonic methods for steam generator U-tubes. Category B-Q does not explicitly address tube plugs and tube sleeves. Examination of tube plugs is discussed in the response to RAI 3.9.14-3 (FPL letter L 2001-65). Turkey Point does not have any sleeves installed at this time. In accordance with ASME Boiler and Pressure Vessel Code, Section XI, Category B-P applies to all reactor coolant pressure retaining components. Since the steam generator tubes are part of the reactor coolant pressure boundary, Category B-P is identified in LRA Table 3.2-1 (page 3.2-83). Although visual examination of tubes is not possible during pressure testing, any significant primary-to-secondary tube leakage would be detected by reactor coolant system leak rate calculations and the steam generator blowdown radiation monitors.

RAI 3.2.7-5:
NRC Information Notice (IN) 97-88, "Experiences During Recent Steam Generator Inspections," was not identified as a reference in Section 3.2.7.3.1 of the LRA. Discuss why the IN was not listed as a reference for the Turkey Point Units 3 and 4 LRA.

FPL RESPONSE:
NRC Information Notice (IN) 97-88, "Experiences During Recent Steam Generator Inspections" was inadvertently omitted in Subsection 3.2.7.3.1 (page 3.2-50) of the LRA. IN 97-88 was evaluated under the FPL operating experience feedback (OEF) program and the information contained in it has been addressed in the Steam Generator aging management review and the Steam Generator Integrity Program as described in Appendix B, Subsection 3.2.14 (page B-80) of the LRA.

RAI 3.2.7-6:
Feedwater nozzle safe ends and steam outlet nozzle safe ends were not identified in Table 3.2-1 as components requiring an aging management program. Explain why they were not identified.

FPL RESPONSE:
The feedwater nozzle safe ends and steam outlet nozzle ends were not identified in LRA Table 3.2-1 (page 3.2-83) because they do not exist at Turkey Point Units 3 and 4. The nozzle to pipe welds connecting the feedwater piping and the steam outlet piping to the steam generators are carbon steel to carbon steel full penetration butt welds.