December 27, 2005

Mr. Theodore Sullivan
Site Vice President
Entergy Nuclear Northeast
James A. FitzPatrick Nuclear Power Plant
Post Office Box 110
Lycoming, NY 13093

SUBJECT: JAMES A. FITZPATRICK - NRC SPECIAL INSPECTION REPORT 05000333/2005009

Dear Mr. Sullivan:

On October 27, 2005, the NRC completed a Special Inspection at the James A. FitzPatrick Nuclear Generating Station. The enclosed report documents the inspection results which were discussed on October 27, 2005, with Mr. David Wallace and other members of your staff.

The Special Inspection Team (SIT) examined activities related to the June 27, 2005 discovery of a through-wall leak in the torus shell, which subsequently resulted in the plant being shutdown on June 30, 2005. The SIT also examined activities associated with a leak in the shutdown cooling (SDC) suction pipe wall of the residual heat removal (RHR) system that was identified after plant shutdown. The activities inspected by the SIT were those conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, inspected plant components, and interviewed personnel.

Based on the results of this inspection, the report documents two self-revealing findings of very low safety significance (Green). One of these findings was determined to involve a violation of NRC requirements. However, because of the very low safety significance and because it is entered into your corrective action program, the NRC is treating this violation of a requirement as a non-cited violation (NCV) consistent with Section VI.A of the NRC Enforcement Policy. If you contest the NCV or the Green finding without an NCV in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN.: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator Region I; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at FitzPatrick.

In accordance with 10 CFR 2.390 " Public Inspections, exemptions, requests for withholding," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publically Available Records (PARS) component of the NRC's document management system (ADAMS). ADAMS is accessible from the NRC website at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/
A. Randolph Blough, Director
Division of Reactor Safety
Region I

Docket No. 50-333
License No. DPR-59

Enclosure: Inspection Report 05000333/2005009
w/Attachments

cc w/encl:
G. Taylor, CEO, Entergy Operations
M. Kansler, President, Entergy
J. Herron, Sr, VP and Chief Operating Officer
C. Schwarz, VP, Operations Support
K. Mulligan, General Manager, Plant Operations
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J. Sniezek, PWR SRC Consultant
M. Lyster, PWR SRC Consultant
S. Lousteau, Treasury Department
INPO

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DATE	12/27/05	12/23/05 corres)			

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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No:	50-333
License No:	DPR-59
Report No:	05000333/2005009
Licensee:	Entergy Nuclear Northeast
Facility:	James A. FitzPatrick Nuclear Generating Station
Location:	Scriba, New York
Dates:	June 27 to October 27, 2005
Team Leader:	E. H. Gray, Senior Reactor Inspector, Plant Support Branch 2, DRS
Inspectors:	Hansraj Ashar, NRR Technical Reviewer D. Dempsey, Resident Inspector M. Modes, Senior Reactor Inspector, Engineering Branch 1, DRS L. Cline, Senior Resident Inspector B. Fuller, Resident Inspector, Nine Mile Point Simon Chia-Fu Sheng, NRR Technical Reviewer W. Schmidt, Senior Reactor Analyst, DRS C. Cahill, Senior Reactor Analyst, DRS
Approved by:	John R. White, Chief Plant Support Branch 2 Division of Reactor Safety

SUMMARY OF FINDINGS

IR 05000333/2005009; 06/27/2005 - 10/27/2005; James A. FitzPatrick Nuclear Generating Station; Special Inspection Report.

The inspection was conducted by region-based and resident inspectors, with support by NRC Office of Nuclear Reactor Regulation (NRR) Technical Reviewers and regional Senior Reactor Analysts. The inspection was implemented in response to the through-wall leak in the torus shell of approximately 4.6 inches in length, and a through-wall leak in the shutdown cooling suction line of the residual heat removal (RHR) system. This inspection also provided facts to support NRR's determination of possible generic implications of the torus leak.

Two Green findings, one of which was an NCV, were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using IMC 0609 "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after Management review. The NRC's program for overseeing the safe operation of commercial nuclear reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

NRC-Identified and Self-Revealing Findings

1. Cornerstone: Mitigating System

Torus shell crack and leak

(Green). A Green self-revealing finding was identified for failure to consider the relevant factors in conducting the initial engineering evaluation of the flaw in the torus shell. Specifically, the initial evaluation of the cracked torus and through-wall leakage did not consider the proximity of the HPCI steam exhaust to the degraded area of the torus shell. The issue was documented in the licensee's corrective action program as CR - JAF-2005-02735, "HPCI Line not Considered in Initial Evaluation of Torus Operability".

This self-revealing finding was of more than minor safety significance because the location of the high pressure coolant injection (HPCI) exhaust line resulted in unanalyzed hydrodynamic loads that resulted in torus cracks and minor leakage. Although this condition placed the torus outside of its design limits, subsequent structural and material analyses of the condition demonstrated that the torus would have been able to perform its mitigating safety function for all design basis transients and accidents. The finding was determined to be Green (very low safety significance) based on IMC 0609, Appendix A, Phase 1 SDP worksheet for at-power situations. The inspectors determined that the finding represented a design deficiency that did not result in a loss of function per Generic Letter 91-18, Revision 1.

2. Cornerstone: Initiating Events

Shutdown Cooling (SDC) Pipe Leak

(Green). A Green self-revealing non-cited violation (NCV) was identified for failure to comply with 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings." As a result of not adequately implementing a pipe support inspection procedure, a through-wall crack and leakage developed in the common residual heat removal (RHR) shutdown cooling (SDC) system suction pipe. After the leak was found, Entergy identified a 1/32 - inch gap between the pipe and adjacent pipe support PFSK-2084. Because of the gap, PFSK-2084 was not bearing its design load or adequately resisting normal pipe movement during system operation. This resulted in low stress, high cycle fatigue cracking of the pipe.

While the leakage was self-revealing, a performance deficiency existed in that a gap between pipe support PFSK-2084 and the SDC pipe was not identified during an examination in 1985 or in the interval leading up to the fatigue failure in 2005. The finding was more than minor because it affected the initiating events cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown operations. The leakage resulting from the crack would not have resulted in the loss of the residual heat removal (RHR) system or adversely impacted other mitigating systems. Since the finding did not require a quantitative assessment, it was determined to be Green (very low safety significance) based on Figure 1 of IMC 0609, Appendix G.

<u>Report Details</u>

Background

<u>Summary of Plant Event</u>

On June 27, 2005, technicians working in the torus area observed a small amount of liquid leakage from the torus shell. Follow-up examination revealed that the source of the liquid was a through-wall defect in the torus shell of approximately 4.6 inches in length. Upon initial evaluation of the defect and review of the ASME Code, Entergy established a Reasonable Expectation of Operability (REO) that concluded the degraded condition would not impact the functionality and operability of the primary containment boundary. However, upon further engineering review, Entergy determined that it's initial evaluation did not consider all stress intensity factors that affected the area of concern. Specifically, Entergy failed to recognize that the HPCI steam exhaust line was in close proximity of the crack and, as a result, did not consider the stress intensity that would be generated during HPCI operation. Subsequently, Entergy determined that, considering that HPCI system exhausted to the area of interest, primary containment integrity could not be assured. Accordingly, per the B 3.6.1.1 Primary Containment Technical Specification plant shutdown was initiated on June 30, 2005, about 80 hours after the identification of a through-wall crack in the torus but within about 30 minutes of the determination that containment integrity could not be assured. (reference Event Notification # 41815).

On July 4, 2005, subsequent to the plant shutdown and cooldown, Entergy also identified a through- wall defect in the shutdown cooling (SDC) suction line of the residual heat removal (RHR) system. Although the RHR shutdown cooling subsystems were declared inoperable per the Technical Specifications, Entergy performed an evaluation that determined that the two RHR subsystems would remain functional for shutdown cooling. Subsequently, the SDC flaw was repaired and confirmed to meet the ASME Code integrity requirements by nondestructive testing, including radiography.

This Special Team Inspection (SIT) was initiated on July 7, 2005, in accordance with NRC Management Directive 8.3, "NRC Incident Investigation Program." The decision to perform this special inspection was based on deterministic criteria in Management Directive 8.3 and the initial risk assessment, since the conditions may have involved a major deficiency in design, construction, or operation having potential generic safety implications. The initial risk assessment characterized the conditional core damage probability to be approximately low to mid-E-6. Accordingly, regional management determined that a special inspection was appropriate for this circumstance, and established the Special Team Inspection Charter (attached) to define the objectives of the inspection.

The special inspection independently developed facts about the licensee's investigation, root cause evaluation, evaluation of the structural integrity of the torus and the shutdown cooling line; corrective actions, and extent of condition analysis. The special inspection also assessed the adequacy of repair of the affected systems, and the process for restoring the affected systems to normal operation; and characterized the risk significance posed by these events posed. The special inspection also addressed regulatory compliance and performance deficiencies that may have contributed to these events.

4. OTHER ACTIVITIES [OA]

4OA3 Event Follow-up (93812)

.1 Torus Shell Crack/Leak and Shutdown Cooling (SDC) Pipe Leak Investigation

a. Inspection Scope

The events were reviewed in accordance with the scope and objectives identified in the Special Team Inspection Charter. The inspectors made physical observations of the affected areas and reviewed plant conditions and applicable design requirements. Additionally, applicable plant procedures, including EN-OP-104, Rev 0, Operability Determinations, were reviewed against current regulatory requirements. Condition Reports, describing the material degradation issues (i.e., leakage due to through-wall cracking), cause analysis, and corrective actions were also examined.

Modification and repair activities performed to return the affected components to full operability were examined in accordance with applicable ASME Code and regulatory requirements. Non-destructive examination (NDE) activities were reviewed to confirm the adequacy of the applied methods relative to ASME Code specifications. The inspectors also reviewed and confirmed the adequacy of Entergy's post-repair surveillance procedure for monitoring the affected torus area by NDE methods following each HPCI system actuation (i.e., such as periodic HPCI surveillance testing).

Analysis of the torus cracked condition and the relation of the torus material properties in response to the cracking included extensive structural analysis by Entergy and its contractors. The NRC NRR Engineering Staff reviewed this analytical work to confirm its adequacy, and to confirm the validity of the conclusion that the affected systems had maintained structural integrity and operational capability.

The Regional Senior Reactor Analysts evaluated the initial leak conditions to provide input on the suitability of performing a Special Inspection, and subsquentially reviewed the factors relating to the as-found torus and SDC degraded conditions to determine the potential risk to plant safety for various plant operational conditions within the plant design basis.

b. Findings

2. Torus shell crack and leak

Introduction:

A Green self-revealing finding was identified for failure to consider the relevant factors in conducting the initial engineering evaluation of the flaw in the torus shell. Specifically the initial evaluation of the cracked torus and minor through-wall leakage did not consider the proximity of the HPCI steam exhaust to the degraded area of the torus shell.

Description:

On June 27, 2005, the plant was operating at full power. Entergy technicians, involved with an in-service inspection activity associated with the RCIC system, noted a small amount of water on the floor that appeared to emerge from a location on the torus shell. Further examination by NDE methods revealed a through-wall crack through the torus shell of approximately 4.6 inches in length. Entergy then initiated an evaluation to determine the operability of the torus in this condition.

To address the condition, Entergy performed an operability determination that considered the guidance in Generic Letter 90-05 and ASME Code Case N–513, and developed a Reasonable Expectation of Operability based on its engineering evaluation that determined that the torus could continue to perform its safety function as a containment boundary with a maximum flaw length of 16.8 inches. However, this conclusion was based on an assessment of stress intensity factors that did not consider the affect of high pressure coolant injection (HPCI) turbine steam exhaust in the near vicinity of the flaw.

On June 29, 2005, Entergy recognized that the HPCI turbine exhaust line was located near the flaw, estimated that the stress intensity factor would be considerably higher than originally calculated, and concluded that torus operability could no longer be reasonably assured. Entergy initiated a plant shutdown and achieved Mode 4, cold shutdown, on July 1, as required by TS 3.6.1.1.

Analysis:

After the torus leak was identified the licensee had an initial structural integrity analysis performed. The input for the initial analysis was incomplete because it did not include the driving force (prime causal factor) for the cracking, the discharge into the torus pool of the HPCI pump turbine steam exhaust. While the licensee initiated additional analysis after realizing the contribution of the HPCI discharge to the torus degradation, the delay in recognizing this contributing factor was unwarranted.

The inspectors concluded that a performance deficiency existed because Entergy did not apply a complete understanding of the HPCI design and plant system configuration to the initial torus leakage Engineering Evaluation, a condition that was reasonably within its ability to accomplish. Traditional enforcement does not apply because the issue did not have an actual safety consequence or a potential for impacting the NRC's regulatory function, and it was not the result of any willful violation of NRC requirements. This self-revealing finding was of more than minor safety significance because it was associated with the design control attribute of the mitigating systems cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, Entergy initially failed to effectively determine the operabilty of the torus as a result of insufficient system knowledge, i.e., the licensee's organization initially did not recognize that the HPCI system exhausted to the affected torus bay, and consequently did not initially consider the hydrodynamic loads that would be generated in the area upon HPCI initiation.

Subsequent to the shutdown, structural and material analyses of the condition showed that the torus would have been able to perform its mitigating safety function for all design basis transients and accidents; and though the torus also serves as a containment boundary, analysis of the conditions found leakage would have been relatively small and any radiological releases would have been minimal. Consequently, the leakage would not have increased the large early release potential. The finding was determined to be Green (very low safety significance) based on IMC 0609, Appendix A, Phase 1 SDP worksheet for at-power situations. The inspectors determined that the finding represented a design deficiency that did not result in a loss of function.

Enforcement:

Although a performance deficiency existed because Entergy did not apply an adequate understanding of the then current plant HPCI design and plant system configuration to the initial torus leakage Engineering Evaluation, the performance deficiency did not involve an explicit violation of regulatory requirements. The issue was considered a finding of very low safety significance **(FIN 05000333/2005009-01**). The issue was documented in the licensee's corrective action program as CR -JAF-2005-02735, "HPCI Line not Considered in Initial Evaluation of Torus Operability".

3. Shutdown Cooling (SDC) Pipe Leak

Introduction:

A Green self-revealing NCV was identified for failure to comply with 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings." A through-wall crack developed in the common residual heat removal (RHR) shutdown cooling (SDC) system suction pipe rendering the SDC system inoperable. This crack was the result of not adequately implementing a pipe support inspection procedure.

Description:

On July 4, 2005, Entergy identified a small (approximately 70 drops per minute) leak on the 20 inch diameter common SDC suction line coming from a 6.5 inch long crack at the toe of the trunion-to-pipe weld of support PFSK-2285. Entergy declared the SDC system inoperable in accordance with the technical specifications, but kept the system in service since it remained capable of performing its cooling function. Subsequently, Entergy identified a 1/32 inch gap between the pipe and adjacent pipe support PFSK-2084. Because of the gap, PFSK-2084 was not bearing its design load or adequately resisting normal pipe movement during system operation. This resulted in low stress high cycle fatigue cracking of the pipe.

Upon detection, Entergy immediately mitigated the condition at PFSK-2084 by installing a shim plate and an additional temporary pipe support to enhance SDC system functionality, which caused the leak rate to be substantially reduced. Subsequently, Entergy repaired the affected pipe by initially establishing a temporary repair by welding over the cracked area. Following that step, Entergy established plant conditions such that the shutdown cooling system was not required to be in service, then drained the pipe, removed the cracked area, and performed a full pipe wall penetration weld as necessary to return the pipe to full wall thickness in accordance with applicable ASME code requirements. Radiography was performed to confirm the repair met ASME code specifications. The SDC crack was repaired in accordance with design code requirements on July 15, 2005.

Pipe support PFSK-2084 is included in Entergy's Inservice Inspection Program and was last inspected in 1985 using procedure IP-PS-01, "Pipe Support Inspection Procedure." That inspection did not identify that the support failed to conform to the system design by not engaging the pipe that it was intended to support. The inspectors concluded that failure to identify the nonconformance in 1985 established the conditions that resulted in the pipe crack on July 4, 2005.

To assess the extent of condition, Entergy conducted detailed examinations of 118 additional supports on large bore RHR, core spray, high pressure coolant injection and reactor core isolation cooling system pipes. The examinations included supports that had been inspected in 1985. No other degraded conditions were identified.

Analysis:

The inspectors concluded that a performance deficiency existed because a gap between pipe support PFSK-2084 and the SDC pipe was not identified during an examination in 1985 or in the interval leading up to the fatigue failure in 2005. This issue was reasonably within Entergy's ability to identify and correct. Traditional enforcement does not apply because the issue did not have an actual safety consequence or a potential for impacting the NRC's regulatory function, and it was not the result of any willful violation of NRC requirements. The finding was more than minor because it affected the initiating events cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown operations. Subsequent analysis of the condition, assuming the worst case degradation, determined that the leakage resulting from the crack would not have resulted in the loss of RHR or adversely impacted other mitigating systems. Since the finding did not require a quantitative assessment, it was determined to be Green (very low safety significance) based on Figure 1 of IMC 0609, Appendix G.

Enforcement:

10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires activities affecting quality to be prescribed by documented instructions, procedures, or drawings of a type appropriate to the circumstances and to be accomplished in accordance with these instructions, procedures, or drawings. Contrary to the above, in April 1985, the Inservice Inspection of safety-related SDC system pipe support PFSK-2084 was not accomplished in accordance with the prescribed procedure and drawing. Because this failure to comply with Criterion V of 10 CFR 50, Appendix B, is of very low risk significance and has been entered into Entergy's corrective action program as CR-2005-02749, this violation is being treated as a NCV consistent with Section VI.A of the NRC Enforcement Policy: **NCV 05000333/2005009-02, Failure to Adequately Inspect Safety-Related Pipe Support**.

3. Review of Fracture Mechanics Evaluations for the Torus Cracking

a. Scope of Review

The licensee's flaw evaluation for the detected flaws in the torus shell includes five elements: (1) flaw sizing, (2) the applied stress intensity factor ($K_{applied}$) calculation and the associated crack growth evaluation, (3) a driving force evaluation for the final flaw size using elastic-plastic fracture mechanics (EPFM), (4) a failure resistance evaluation, and (5) a stability evaluation using ASME Code structural factors (SFs).

Flaw sizing was reported in Document FITZ-05Q-305, which showed a rather complex configuration for the detected flaw. To analyze the flaw analytically, the detected flaw was decomposed into an axial flaw and a circumferential flaw and analyzed separately. This practice is allowed by Section XI of the ASME Code and was noted to be acceptable for this application. The Entergy determination of the location and crack lengths of the axial and circumferential flaws were conservative because the flaws were assumed to be through-wall and the decomposed axial and circumferential flaws were located approximately along the maximum stress lines.

In the area of structural integrity analysis, the licensee's crack growth evaluation using the $K_{applied}$ followed the ASME Code Section XI fatigue crack growth law for ferritic steels in water environments. The NRR staff determined that the licensee's $K_{applied}$ results were suitable for the crack growth calculation in support of the root cause evaluation.

The projected crack growth due to fatigue was supported by laboratory investigation documented in Technical Report No. 05-0479-TR-002. NRC review of the referenced structural integrity calculations confirmed that the critical flaw size (determined to be 70 inches) would not be reached by any design basis HPCI system extent of operation, and that any resultant leakage would not exceed the torus room drain pump capacity of 200 gpm.

b. Findings

No findings of significance were identified.

4. Corrective Actions

a. Inspection Scope

The inspectors reviewed the corrective actions identified in the licensee's root cause evaluation reports to determine whether they addressed the causal factors. Additionally, the inspectors reviewed whether the corrective actions had been prioritized with consideration of the risk significance. The inspectors reviewed procedures and quality records related to both the torus cracking and the SDC cracked pipe. The inspectors monitored the licensee's inspection and corrective actions of these components to assure the extent of condition was captured.

b. Findings

No findings of significance were identified.

5. NRC Risk Assessment of the Torus Shell and Shutdown Cooling Pipe Leaks

a. Inspection Scope

– Initial Risk Evaluation

The NRC Senior Reactor Analyst (SRA) performed a conditional risk assessment of the torus crack, identified on June 27, 2005, and the shutdown cooling line crack that was identified a few days later to determine if a special inspection team was the appropriate NRC inspection response to the event. During the course of the special inspection, the SRA performed additional risk evaluation that included input from the Entergy calculations and cause analysis.

The initial condition assessment was conducted using the FitzPatrick Standardized Plant Analysis Risk (SPAR) model, Revision 3.11, and Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE), Version 7.0, Revision 24, for the torus crack and IMC 0609, Appendix G, for the shutdown cooling line crack. This initial risk assessment was based upon the following assumptions about the as-found material degradations.

RHR Shutdown Cooling Suction Line Crack

- The defect on the common suction line has a failure probability of 0.1. This estimate includes the failure due to loss of pipe integrity and air entrainment causing a loss of all RHR pumps.
- Given a failure of the common suction line, no credit was provided for recovery of RHR shutdown cooling.
- A credit of 3 was given for Manual Injection and RCS Pressure Control. Though the baseline credit in the SDP is 2, given the availability of an essentially inexhaustible water supply (RHR SW and Fire Protection), an additional credit was assigned.
- The base case credit of 1 was applied for Manual High Pressure Injection at Pressure.

Torus Crack (Assuming the condition existed for 1 year)

- Initiating events and transients that require the operation of HPCI may cause a torus failure. Without fracture mechanics evaluations of the condition, it is assumed that the events that required HPCI to run longer (such as MLOCA, SLOCA, and TRANS) would result in a torus failure.

- Given that the torus fails, level will decrease causing operators to depressurize in accordance with the EOP's.

- Given that the torus fails, the torus pressure will equalize with the reactor building.

- Given that the torus fails, the decreased torus pressure and increase in torus water temperature will result in a loss of net positive suction head (NPSH) for the CS and RHR pumps.

- There is no recovery credit for the failed torus.

- HPCI and RCIC will lose the torus as a water source, and will not be available for make-up once the reactor is depressurized.

Applying the above assumptions, the initial estimate of the increase in conditional core damage probability (ICCDP) for the crack in the torus and shutdown cooling line was approximately mid-E-6. The dominant accident sequences for the condition involving the torus were: a medium loss of coolant accident (MLOCA) where the operator fails to start alternate injection failure; and a transient in which 2 SRVs fail to close and the power conversion system (PCS) is unavailable. The dominant accident sequence for the shutdown cooling line was a loss of RHR with failure of manual injection and RCS pressure control along with a failure of manual high pressure injection at pressure.

– <u>Risk Evaluation Followup</u>

The Senior Reactor Analyst (SRA) performed an additional risk assessment of the torus and shutdown cooling line cracked conditions that considered the Entergy calculations and evaluations that were subsequently developed to assess actual risk for these conditions.

Subsequent fracture mechanics evaluations of the torus, with inputs including the maximum HPCI run times to address credible events, determined that the extent of cracking would result in leakage rates that would not reduce the torus level below that required for mitigating functions, exceed torus room drain capabilities or result in a release that would contribute to the large early release frequency (LERF). Therefore, the issue resulted in essentially no increase in the probability of core damage or large early release.

Additional analysis of the RHR shutdown cooling line determined that the line would maintain its structural integrity and be capable of performing its shutdown cooling function. Therefore, this issue resulted in essentially no increase in core damage probability.

b. <u>Findings</u>

No findings of significance were identified.

6. <u>Observations, Conclusions and Assessment based on the Special Team Inspection Charter</u>

The purpose of this section of the SIT report is to provide observations, conclusions and assessment, as applicable, for each identified Item specified in the Special Team Inspection Charter. As specified in the Charter, Items 1 through 5 are applicable to conditions prior to plant startup. Items 6 through 9 are applicable to conditions post-startup. While this section contains some discussion that may be redundant to other description in this report, such information is repeated only to establish completeness relative to the specific Charter Item.

Based on the results of this inspection, as previously described, the team identified two (Green) findings, one of which was a non-cited violation (NCV). Additionally, the team determined that the corrective actions established and implemented for the findings were appropriate for the circumstances. The inspectors confirmed that the licensee conducted a thorough extent-of-condition review, as applicable for these conditions.

1. **Review the adequacy of the new procedure for alternate decay heat removal during shutdown and the licensee's methodology for verification of its heat removal capability.**

Upon identifying the cracked pipe, Entergy declared both trains of shutdown cooling inoperable per TS 3.4.8. By repairing pipe support PFSK-2084 and installing a temporary pipe support, leakage was reduced from about 70 drops per minute to 2 drops per minute, and pipe vibration was dampened. Both trains of shutdown cooling remained functional. Alternate means of removing decay heat, such as RHR pump and safety relief valves, also were available but were not used.

To provide an alternate method of decay heat removal per TS 3.4.8, Entergy developed procedure TOP-353, “Alternate Shutdown Cooling,” which relied on an RHR pump and a combination of SRVs to circulate reactor coolant and remove decay heat through an RHR heat exchanger. Entergy also developed TOP-354, “Cooling Using Main Steam Line Drains While Shutdown.” The procedure was tested on July 8, but proved to be inadequate to remove the existing decay heat load.

Subsequently, Entergy, with NRC approval, restored the affected SDC pipe compliance with the ASME code by performing a temporary repair that allowed a plant mode change to an operational condition that did not require SDC to be in-service. Subsequently, the SDC was repaired in a manner that fully met the ASME Code design.

Based on review, the inspectors did not identify any significant issues relative to procedure development. The licensee’s method to mitigate the condition and effect repair was determined to be safe and effective.

2. **Review the adequacy of Entergy's repairs to the leaks, and the effectiveness of their initial extent-of-condition reviews for the torus shell through-wall defect identified on June 27, 2005, and the shutdown cooling pipe through-wall defect identified on July 4, 2005.**

Both the torus shell area and shutdown cooling pipe leak area were repaired to meet their applicable Code requirements. The torus repair included adding a cover plate over the leak area with a full penetration weld and removal of the cracked area by underwater cutting. The cut edge of the torus was ground to remove thermally affected material. Calculations were performed to quantify the HPCI steam turbine discharge on the repaired torus shell configuration. Additionally, a surveillance procedure was developed to perform NDE on the repaired area following any condition that required HPCI initiation. The objective of the surveillance procedure was to verify the integrity of the repair and the absence of any new cracking in the area.

Relative to the SDC pipe leak, Entergy immediately mitigated the condition at PFSK-2084 by installing a shim plate and an additional temporary pipe support to enhance SDC system functionality, which caused the leak rate to be substantially reduced. Subsequently, Entergy repaired the affected pipe by initially establishing a temporary repair by welding over the cracked area. Following, Entergy established plant conditions such that the shutdown cooling system was not required to be in service, then drained the pipe, removed the cracked area, and performed a full pipe wall penetration weld as necessary to return the pipe to full wall thickness in accordance with applicable ASME Code requirements. Radiography was performed to confirm the repair met ASME Code specifications. The SDC crack was repaired in accordance with design code requirements on July 15, 2005.

The inspectors evaluated the extent-of-condition reviews for the torus shell and the shutdown cooling pipe through-wall leaks. For the torus, the Entergy staff work included an inspection of the torus for evidence of other leakage and NDE of questionable areas. Also, the other system interactions that stress the torus including SRV and RCIC turbine exhaust into the torus were evaluated as driving forces for torus degradation. No other areas of torus degradation were identified.

For the shutdown cooling pipe leak, the inspector reviewed the actions taken to determine the extent of condition of the residual heat removal suction line crack. A review was performed of the licensee’s list of all large bore piping supports of a similar configuration including condensate storage, high pressure coolant injection, and reactor core isolation cooling. The criteria used to perform the inspection, such as observation of free-play in the inspected support or evidence of leaking, was reviewed. The results of the inspection was reviewed. As noted in Licensee Event Report 05-004-00, no additional degraded or non-conforming conditions were identified in the extent of condition sample.

Based on review, the inspectors did not identify any significant issues associated with the repair of the conditions or extent of condition review.

3. **Review the adequacy of Entergy's initial causal evaluations and any interim compensatory actions necessary for restart of the reactor.**

The physical causal factor for the torus leakage was the presence of the HPCI steam turbine 24" diameter pipe un-attenuated (i.e., without sparger) discharge into the torus pool near the area that developed cracking. The original installation and modification of the HPCI discharge line in the 1980's, did not include a sparger to dissipate the steam discharge forces. At that time, a sparger was not an NRC design consideration. While the un-attenuated HPCI discharge was an old design issue, the existing condition did allow the torus to be susceptible to stress cracking, and was a primary cause identified by the licensee.

Entergy recognized that they initially failed to effectively determine the operabilty of the torus in a timely manner as a result of insufficient system knowledge, i.e., the licensee's organization did not recognize that the HPCI system exhausted to the affected torus bay, and consequently did not initially consider the hydrodynamic loads that would be generated in the area upon HPCI initiation. This issue was entered into the site Problem Identification and Resolution Program as Condition Report JAF-2005-02735.

For the SDC pipe fatigue cracking, Entergy identifed that a primary causal factor was inadequate pipe support due to a lack of support engagement as required by plant design. The condition allowed the pipe to vibrate, generating fatigue, and resulting in eventual cracking. This condition was not identified during a pipe support walkdown inspection in the 1980's or during the time interval up to when the SDC leakage was identified in 2005. As part of the extent of condition review to the SDC leak, the Entergy staff conducted an extensive review of other mitigating system pipe supports and initiated corrective actions for the few minor problems identified.

Based on review, the inspectors did not identify any significant causal factors that were not identified or recognized by the licensee. Compensatory actions taken by the licensee in response to this condition were reviewed and determined to be adequate.

4. **Identify any potential generic issues that may require prompt action.**

NRC is continuing to review the findings and observations associated with this matter for generic implications, and will issue generic communications as appropriate. Notwithstanding, Entergy has taken action to inform other BWR operators of this operating experience. Other generic issues that may pertain include the following:

- HPCI discharge pipe condensing spargers are an effective way to adequately distribute operational hydrodynamic loads and ensure stresses are maintained within acceptable limits.

- Design changes, including past and planned modifications, that affect the torus imposed operational and accident load characteristics should be subject to design control review to ensure critical areas meet the acceptance criteria of applicable design specifications.

- The combined operation of the HPCI system and safety relief valve discharges during the Northeast grid disturbance of August 2003 may have may have initiated the torus cracking, or contributed significantly to the condition. The Fitzpatrick HPCI system operated over 14 hours and the safety relief valves lifted five times over a period of 28 hours. Augmented inspections may be warranted after such stress inducing events or excessive fatigue cycling conditions.

5. **Gather additional information to refine NRC's initial risk assessment and evaluate potential changes to the inspection approach.**

The Senior Reactor Analyst (SRA) performed a risk assessment as part of the process to establish if a Special Inspection was the appropriate response to the torus and SDC leaks. Refer to Part 4OA3.5 of this report for discussion of the pre-inspection risk analysis.

For the risk evaluation followup, during the inspection, the Senior Reactor Analyst (SRA) performed a additional risk assessment of the torus and shutdown cooling line cracked conditions that considered the Entergy calculations and evaluations that were developed in the weeks after June 27, 2005.

Subsequent fracture mechanics evaluations of the torus, with inputs including the maximum HPCI run times to address credible events, determined that the extent of cracking would result in leakage rates that would not reduce the torus level below the that required for mitigating functions, exceed torus room drain capabilities or result in a release that would contribute to LERF.

Additional analysis of the RHR shutdown cooling line determined that the line would maintain its structural integrity and be capable of performing its shutdown cooling function. Accordingly, the inspectors confirmed that the original line failure assumption for the passive component was sufficiently conservative.

6. **Examine Entergy's corrective actions for the shutdown cooling pipe through-wall defect at seismic support PFSK-2285 discovered on July 4, 2005, including the licensee's investigation and root cause evaluation. Specifically, address (a) the conformance of the as-built support with its intended design; (b) the adequacy of design; (c) the availability of procedural guidance for alternate decay heat removal under cold shutdown conditions and (d) the adequacy of the surveillance program for the support system**.

(a) The pipe was not engaged at the adjacent pipe support in accordance with that pipe support design, i.e., the pipe support was not in-conformance with the expected design. Entergy’s corrective actions and extent of conditon review were acceptable to address this old design issue.

(b) The designs of the integral pipe support where the fatigue leak occurred and the adjacent pipe support were adequate. The causal factor fault was that the gap between the adjacent pipe support and the pipe allowed pipe vibration which in turn resulted in fatigue failure at the integral pipe support was in excess of that specified in the design.

(c) While the licensee did develop procedures for alternate decay heat removal to be support repairs options, in the final analysis, the normal decay heat removal method was employed. Item 1 discussion pertains.

(d) The current surveillance program for pipe supports was found to be in established and implemented as required by the ASME Code and current In-service Inspection Program requirements. As of the time of this finding, he specific non-conforming support was never selected from the total population as a sample to be inspected.

This inspection did identify a finding related to the fact that this non-conforming condition was specifically inspected in a formal pipe support walkdown inspection conducted in 1985, but was not identified or otherwise noted, based on licensee inspection documentation. The failure to find a condition contrary to the plant design is a non-cited violation of 10CFR50, Appendix B, Criterion V, Inadequate Inspection of Safety-Related Pipe Support.

7. **Examine Entergy's corrective actions for the 4.6 inch torus shell through-wall defect identified on June 27, 2005, including the licensee's investigation and root cause evaluation. Specifically, address (a) adequacy of the initial operability determination for the primary containment, relative to structural integrity and the impact of leakage into secondary containment, (b) Entergy's lack of consideration of the presence and location of the High Pressure Core Injection turbine exhaust line in their initial evaluation of primary containment operability, and (c) the effectiveness of the surveillance program for the torus shell.**

The torus leak was identified at 9:55 am on June 27 while performing ST-24H, "RCIC Class 2 and 3 Piping Functional Test (ISI)." Entergy initially focused its efforts on examining and characterizing the leak and inspecting the rest of the torus (particularly the similar torus shell-to-column junctions. A reasonable expectation of operability for the torus was developed by late evening of June 27 that considered radiological consequences of the leakage (10 CFR 50, Appendix J), ECCS room flooding, and torus shell integrity. Structural Integrity Associates (SIA) was contracted to perform the shell evaluation. Entergy provided SIA with design stress information from the Teledyne Mark 1 Stress Report and the GE Load Definition Report. Neither of the reports included local stress due to the HPCI turbine exhaust. The location of the HPCI turbine exhaust near the torus shell crack location was not identified until June 29, and subsquently Entergy concluded that torus operability could no longer be reasonably assured. Upon recognition that HPCI initiation could create significant stress impingement on an already degraded torus shell, Entergy commenced a plant shutdown in accordance with Technical Specifications.

The crack was located in the "P" torus bay. Due to inadequate system knowledge at the time, Entergy's engineering staff were not able to correctly identify what equipment and components were present in the bay when the condition was initially evaluated on June 27. The torus drawing that was being used for reference at that time was 6.60-65, which shows the location of suction strainers, but not the HPCI turbine steam exhaust. On June 28, an engineering turnover action to confirm the proximity of the HPCI and RCIC steam spargers was distributed to several plant engineers. However, the action was only one of several and was not assigned any particular urgency. Additionally, the inspectors learned that the torus design analyses of record did not consider stresses due to HPCI or RCIC operation.

Entergy performed the visual examinations of the torus and torus supports in compliance with ASME Code Section XI, Subsection IWE and 10 CFR 50, Appendix J. The last Code examination was conducted in October 2004. The examinations were conducted by appropriately qualified personnel.

8. **Evaluate Entergy's interpretation of the American Society of Mechanical Engineers Code and application of existing regulatory guidance in their initial evaluation of the torus shell through-wall defect and development of an operability determination for primary containment integrity on June 28.**

The inspectors confirmed that NRC policy allows for licensee's to apply ASME Section XI, Paragraph IWE-3122.2, "Acceptance by Engineering Evaluation," for the purpose of determining the ability of a BWR Mark I torus to perform its safety function, with the condition of a through-wall leak in the pressure retaining boundary. IWE-3122 provides three methods for establishing acceptability of identified degradation in a component: Acceptance by Examination, Acceptance by Corrective Measures or Repair/Replacement, or Acceptance by Engineering Evaluation. IWE-3122 permits the option of an engineering evaluation to support continued operation of a torus (without repair or replacement) provided that the flaw or area of degradation is not structural in nature, or has no effect on the structural integrity of the containment; or in the case of reduced thickness of the base metal, the component can be shown by analysis to satisfy the requirements of the applicable Design Specification.

In the case of the torus, the design specification is indicated in Fitzpatrick's Updated Final Safety Analysis Report (UFSAR). Table 16.7-2 of the UFSAR identifies the ASME Code allowable stress value of 17.5 ksi at 309 EF. While the UFSAR does not specify any fatigue stress limit, the licensee's report SIR-05-234, "Failure and Operability Determination of the Torus Cracking at JAFNPP," calculated the stress intensity of 10.3 ksi in the vicinity of the cracking due to the applied cyclic loads associated with the HPCI discharge loads, without the consideration of any stress concentration factors. Applying the ASME Code fatigue curves fo the calculated cycles of 1.25E7, the licensee identified the crack initiation stress as 22 ksi, with a corresponding stress concentration factor of 2.1. Accordingly, the licensee's analysis confirmed that the torus in its through-wall cracked condition did not meet the criteria of the Design Specification. That is, the fact that a through-wall crack developed was sufficient to demonstrated that the torus had exceeded the applicable Design Specification, and in accordance with the provisions of IWE-3122, was not acceptable for continued operations.

The Entergy Procedure EN-OP-104, Rev. 1, which describes the criteria and considerations for performing Operability Determinations appears generally consistent with the ASME Code. Paragraph 6.1 of that procedure establishes that the conditions of of minor fluid process leakage (packing glands, gaskets, and non-welded surfaces) that constitute containment boundary leakage that are considered inoperable. This does not include leakage through an ASME Code pressure component, that for a containment boundary is to be evaluated per the ASME Code Section XI, IWE-3122.

9. **Identify any potential generic safety concerns and provide information to support generic communication, if appropriate**.

The discussion in Item 4, above, pertains. An NRC Information Notice relative to this matter is currently being developed.

4OA6 Meetings, including Exit

On October 27, 2005, and in a subsequent phone discussion, the special inspection team leader presented the inspection results to Mr. D. Wallace and other members of the licensee's staff. During the inspection, the team reviewed some proprietary information which was returned to the licensee. The team verified that the inspection report does not contain proprietary information.

ATTACHMENT

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

S. Bono*	Director, Engineering
J. Costedio*	Manager, Regulatory Compliance
C. Boucher*	Superintendent, Chemistry
G . Brownell	Regulatory Compliance
C. Brown*	Manager, Quality Assurance
B. Drain*	Manager, Project Management
J. Gerety*	Manager, Design Engineering
P. Scanlan*	Supervisor, Engineering FIN
W. Rheaume *	Manager, Corrective Action and Assessment
D. Wallace*	Director, Nuclear Safety
D. Huwe*	Quality Assurance Auditor
T. Page*	Regulatory Compliance Specialist
T. Herrmann*	Root Cause Analysis Coordinator
J. Pechacek*	Manager, Engineering Support
D. Johnson*	Operations Manager
G. Lozier*	Superintendent, Maintenance Support
B. Sholler*	Manager, Plant Maintenance
L. VanHorn*	Manager, Security (Acting)

NRC Personnel

H. Gray*	Senior Reactor Inspector (Team Leader)
G. Hunegs*	Senior Resident Inspector

*attended exit meeting on 10/27/2005

LIST OF DOCUMENTS REVIEWED

Condition Reports CR-JAF-2005-02749 and CR JAF-2005-02593.

Structural Integrity Associates, Inc.

FITZ-04Q-303, “Load Prediction for Torus Cracking”

FITZ-04Q-304, “Design and Analysis of Torus Repair Plate”

FITZ-04Q-305, “Evaluation of Material Removal from the Torus Ring Girder Web and Gussets”

FITZ-05Q-305, “Fatigue Crack Growth and Leakage Evaluation”

SIR-05-234, “Failure and Operability Determination of the Torus Cracking at James A. FitzPatrick Nuclear Power Plant”

Automated Engineering Services Corporation

SA-05019.AES-100, “Torus Evaluation for HPCI Turbine Exhaust Load”

Altran Corporation

05-0479-TR-002, “Laboratory Investigation of Torus Wall Cracking”

Entergy

JAF-RPT-05-00115, “PSA Evaluation of JAF Torus and Shutdown Cooling Line Cracks”

JAF-CALC-05-00112, “Allowable Flaw Size, Crack Growth and Leakage Evaluation”

CR-JAF-2005-02593, “Root Cause Analysis Report - Torus Leak Discovered Near the Support between Bays ‘A’ and ‘P’”

CR-JAF-2005-02749, “Root Cause Analysis Report - Through Wall Leak Shutdown Cooling Line”

TOP-353, “Alternate Shutdown Cooling”

TOP-354, “Cooling Using Main Steam Line Drains While Shutdown”

TOP-355, “Transition To Mode 3 During RHR SDC Suction Line Repair”

TST-87, “Primary Containment Pressurization Test”

IP-PS-01, “Pipe Support Inspection Procedure”

EN-OP-104, Rev. 0. Procedure for Operability Determinations

EN # 41815. Event Notification of Plant Shutdown on 6/30/05 due to torus leak.

ENN-NDE-9.03, Rev. 0. Procedure for Ultrasonic Examination - ASME Section XI, Appendix III

ENN-NDE-9.28, Rev. 0. Procedure for Manual Ultrasonic Through Wall Sizing in pipe welds.

ENN-DC-185, Rev. 0. Procedure for evaluating Through-Wall Leaks in ASME Section XI, Class 3 Moderate Energy Piping Systems

ENN-EP-S-001, Rev. 0. Engineering Standard for IWE General Visual Containment Inspection

J A FitzPatrick FSAR Update, Chapter 5 for the Containment.

Fracture Mechanics Related Documents

(1) Calculation package file no. FITZ-05Q-304, “Critical and Allowable Flaw Sizes Determination.”

(2) Calculation package file no. FITZ-05Q-305, “Fatigue Crack Growth and Leakage Evaluation,” Revision 0 and Revision 1

(3) Technical report no. 05-0479-TR-002, “Laboratory Investigation of Torus Wall Cracking.”

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened - None

Opened and Closed

FIN 05000333/2005009-01	Green Finding -Failure to consider the relevant factors in conducting the initial engineering evaluation of the flaw in the torus shell
NCV 05000333/2005009-02	NCV Failure to Adequately Inspect Safety-Related Pipe Support

Closed - None

LIST OF ACRONYMS

ASME	American Society of Mechanical Engineers Code
CFR	Code of Federal Regulations
EN	Event Notification
EPFM	Elastic-plastic fracture mechanics
HPCI	High pressure coolant injection
IBA	Intermediate break accident
LERF	Large early release frequency
MLOCA	Medium loss of coolant accident
NCV	Non-cited violations
NPSH	Net positive suction head
NRC	Nuclear Regulatory Commission
PCS	Power conversion system
RHR	Residual heat removal
SAPHIRE	Systems Analysis Programs for Hands-on Integrated Reliability Evaluations
SDC	Shutdown cooling
SDP	Significance Determination Process
SIT	Special Inspection Team
SPAR	FitzPatrick Standardized Plant Analysis Risk Model
SRA	Senior Reactor Analyst

July 18, 2005

MEMORANDUM TO: John R. White, Manager
Special Team Inspection

E. H. Gray, Leader
Special Team Inspection

FROM: A. Randolph Blough, Director**/RA/**
Division of Reactor Safety

Brian E. Holian, Director **/RA/**
Division of Reactor Projects

SUBJECT: SPECIAL TEAM INSPECTION CHARTER -
JAMES A. FITZPATRICK NUCLEAR GENERATING STATION

A special inspection has been established to inspect and assess conditions that were discovered on June 27, 2005 and July 4, 2005 at the James A. FitzPatrick Nuclear Generating Station. Specifically:

On June 27, 2005, Entergy discovered a through-wall defect in the torus shell of approximately 4.6 inches in length, after technicians working in the area observed a small amount of liquid leakage from the torus shell. Entergy's initial evaluation of the defect and review of the ASME code concluded the degraded condition would not impact the operability of the primary containment boundary. However, upon continued engineering review, Entergy determined that operability of the primary containment was not assured and a shutdown was initiated on June 30, 2005 (reference EN# 41815).

On July 4, 2005, subsequent to the plant shutdown and cooldown, Entergy identified a through-wall defect in the shutdown cooling suction line of the residual heat removal (RHR) system. Although the RHR shutdown cooling subsystems were declared inoperable per the Technical Specifications, the two RHR subsystems remained functional for shutdown cooling.

This special team inspection was initiated in accordance with NRC Management Directive 8.3, “NRC Incident Investigation Program.” The decision to perform this special team inspection was based on deterministic criteria in Management Directive 8.3 and the initial risk assessment. Specifically, the conditions may involve a major deficiency in design, construction, or operation having potential generic safety implications. The initial risk assessment characterized the conditional core damage probability to be approximately low to mid E-6, which is in the range for a special inspection.

The inspection will be performed in accordance with the guidance of NRC Inspection Procedure 93812, "Special Inspection," and the inspection report will be issued within 45 days following the final exit meeting for the inspection.

The special inspection will commence on July 7, 2005. The following personnel have been appointed to this effort:

Manager:	John White, Chief, Plant Support Branch 2
Team Leader:	E. Harold Gray, Sr. Reactor Inspector
Full Time Members:	Michael Modes, Sr. Reactor Inspector Douglas Dempsey, Resident Inspector, FitzPatrick
Part Time Members:	Christopher Cahill, Senior Reactor Analyst Leonard Cline, Sr. Resident Inspector, FitzPatrick Brian Fuller, Resident Inspector, Nine Mile Point

Attachment: Special Inspection Charter

Special Inspection Charter
James A. FitzPatrick Nuclear Generating Station
Torus and Residual Heat Removal Suction Piping Through-Wall Flaws with Leakage

Background:
The plant was operating at 100% on June 27, 2005. Entergy personnel discovered a Torus leak in the vicinity of a structural support between Bays A and P during an unrelated surveillance activity. The leak was observed and characterized as a slight weepage. A small puddle was found on the floor, below the leak. Subsequent non- destructive examination determined that the leakage was from a small “X” shaped through-wall crack. Entergy’s initial evaluation determined that there was reasonable expectation of operability, and an Operational Decision Making Instruction (ODMI) was prepared and implemented to monitor the leakage while a detailed operability determination was conducted. On June 30, 2005, upon consideration of other pertinent information, Entergy determined that operability of the primary containment was not assured. Subsequently, at 7:29 p.m. on that date, the Shift Manager declared Primary Containment Inoperable and entered Technical Specification Limiting Condition for Operation (LCO) 3.6.1.1 Condition A, Primary Containment Inoperable, and declared an Unusual Event under EAL 9.1.2; and commenced plant shutdown at 8:00 p.m, June 30, 2005.

On July 4, 2005, with the reactor in the Cold Shutdown condition, a plant operator noted a small accumulation of water leakage on the floor in the reactor building under a 20-inch diameter section of the residual heat removal (RHR) shutdown cooling (SDC) suction line. Entergy’s investigation identified a linear crack, about 6 inches long, in the SDC suction line adjacent to the trunion that was welded to the pipe to provide suspended support. The crack was observed to be leaking at several drops per minute. Entergy declared the two RHR SDC subsystems required by Technical Specification LCO 3.4.8 inoperable based on the presence of this through-wall defect on an ASME Class 2 pipe. Notwithstanding the TS inoperability, the RHR SDC system remained in service and functional for decay heat removal.

Objectives of the Special Inspection: The objectives of the special inspection are to evaluate the circumstances associated with the events described above. In the event that information is determined that the nature of these events are significantly different than currently understood, such that circumstances and conditions may be beyond the scope of a Special Inspection, the Team Leader will immediately inform the Special Inspection Manager. Otherwise, the inspection objectives are the following:

Near Term (prior to startup)

1. Review adequacy of new procedure for alternate decay heat removal during shutdown and the licensee's methodology for verification of its heat removal capability.

2. Review adequacy of Entergy's repairs to the leaks, and the effectiveness of their initial extent-of-condition reviews for the torus shell through-wall defect identified on June 27, 2005, and the shutdown cooling pipe through-wall defect identified on July 4, 2005.

3. Review the adequacy of Entergy's initial causal evaluations and any interim compensatory actions necessary for restart of the reactor.

4. Identify any potential generic issues that may require prompt action.

5. Gather additional information to refine NRC's initial risk assessment and evaluate potential changes to the inspection approach.

<u>Additional Scope (post startup)</u>

6. Examine Entergy's corrective actions for the shutdown cooling pipe through-wall defect at seismic support PFSK-2285 discovered on July 4, 2005, including the licensee's investigation and root cause evaluation. Specifically address (a) the conformance of the as-built support with its intended design; (b) the adequacy of design; (c) the availability of procedural guidance for alternate decay heat removal under cold shutdown conditions and (d) the adequacy of surveillance program for the support system.

7. Examine Entergy's corrective actions for the 4.6 inch torus shell through-wall defect identified on June 27, 2005, including the licensee's investigation and root cause evaluation. Specifically address (a) adequacy of the initial operability determination for the primary containment, relative to structural integrity and the impact of leakage into secondary containment, (b) Entergy’s lack of consideration of the presence and location of the High Pressure Core Injection turbine exhaust line in their initial evaluation of primary containment operability, and (c) the effectiveness of the surveillance program for the torus shell.

8. Evaluate Entergy's interpretation of the American Society of Mechanical Engineers code and application of existing regulatory guidance in their initial evaluation of the torus shell through-wall defect and development of an operability determination for primary containment integrity on June 28.

9. Identify any potential generic safety concerns and provide information to support generic communication, if appropriate.