January 17, 2003

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

SUBJECT: JAMES A. FITZPATRICK NUCLEAR POWER PLANT - NRC INSPECTION REPORT 50-333/02-08

Dear Mr. Sullivan:

On December 28, 2002, the NRC completed an inspection at your James A. FitzPatrick Nuclear Power Plant. The enclosed report documents the inspection findings which were discussed on January 13, 2003, with Mr. O'Grady and other members of your staff.

The inspection examined activities 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, and interviewed personnel.

Based on the results of this inspection, no findings of significance were identified.

The NRC has increased the security requirements at the James A. FitzPatrick Nuclear Power Plant in response to terrorist acts on September 11, 2001. Although the NRC is not aware of any specific threat against nuclear facilities, the NRC has issued an Order and several threat advisories to commercial power reactors to strengthen licensees' capabilities and readiness to respond to a potential attack. The NRC continues to inspect Entergy's security controls and its compliance with the Order and current security regulations.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document

Room or from the Publicly Available Records (PARS) component of the NRC's document system (ADAMS). ADAMS is accessible from the NRC Web Site at http://www.nrc.gov/reading-rm.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Glenn W. Meyer, Chief
Projects Branch 3
Division of Reactor Projects

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

Enclosure: Inspection Report 50-333/02-03
Attachment: Supplemental Information

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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.:	50-333
License No.:	DPR-59
Report No.:	50-333/02-08
Licensee:	Entergy Nuclear Northeast
Facility:	James A. FitzPatrick Nuclear Power Plant
Location:	268 Lake Road Scriba, New York 13093
Dates:	October 1 to December 28, 2002
Inspectors:	L. M. Cline, Senior Resident Inspector D. A. Dempsey, Resident Inspector D. M. Silk, Senior Emergency Preparedness Inspector T. A. Moslak, Health Physicist L. Scholl, Senior Reactor Inspector, Electrical Branch A. Lohmeier, Reactor Inspector, Systems Branch P. Frechette, Security and Safeguards Inspector
Approved by:	Glenn W. Meyer, Chief Projects Branch 3 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000333-02-08; Entergy Nuclear Northeast; on 10/01/2002 -12/28/2002; James A. FitzPatrick Nuclear Power Plant. Routine Integrated Report.

The inspection was conducted by the resident inspectors, a senior emergency preparedness inspector, a health physicist, and regional specialist inspectors. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. Inspector Identified and Self- Revealing Findings

No findings of significance were identified.

B. Licensee Identified Violations

None.

Report Details

The reactor began the inspection period operating at full power. On October 2, 2002, an unplanned power reduction of greater than 20% occurred due to a failed reactor feed pump seal. On October 6 the reactor was shutdown for a refueling outage (RFO-15). Startup from RFO-15 commenced on October 30. On November 5 an unplanned power reduction of greater than 20% from 98% occurred due to smoldering lagging on a reactor feed pump. Following repairs, full power operation was attained on November 7 and continued for the remainder of the inspection period.

1. REACTOR SAFETY
Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, and Emergency Preparedness

1R01 Adverse Weather Protection

a. Inspection Scope

The inspector reviewed the implementation of site cold weather preparations. The review included a tour of outdoor facilities, a review of procedures used to test outdoor heat tracing circuits and intake de-icing heaters, and a verification of items included in the cold weather checklist contained in Entergy procedure AP-12.04, “Cold Weather Preparations.” The inspector also reviewed Entergy’s responses to NRC Bulletin 79-24, “Frozen Lines,” and Information Notice 98-02, “Nuclear Power Plant Cold Weather Problems And Protective Measures.”

b. Findings

No findings of significance were identified.

1R02 Evaluation of Changes, Tests or Experiments

a. Inspection Scope

The inspectors reviewed safety evaluations for the initiating events, barrier integrity, and mitigating systems cornerstones to verify that changes, tests, and experiments were reviewed and documented in accordance with 10 CFR 50.59 and when required, NRC approval was obtained prior to implementation. The sample included safety evaluations supporting design change packages, engineering calculations, and Updated Final Safety Analysis Report (UFSAR) changes. The inspectors assessed the adequacy of the safety evaluations through interviews with the plant staff and review of supporting information, such as calculations, engineering analyses, design change documentation, the UFSAR, and plant drawings. In addition, the inspectors reviewed the administrative procedures that control the screening, preparation, and issuance of safety evaluations to ensure that the procedures adequately implemented the requirements of 10 CFR 50.59, “Changes, Tests, and Experiments.”

The inspectors also reviewed a sample of changes that Entergy had screened and determined to be outside of the scope of 10 CFR 50.59. Full safety evaluations were not required for these changes. The inspectors performed this review to assess if

Entergy's conclusions with respect to 10 CFR 50.59 applicability were appropriate. The sample of issues that were screened out included design changes, temporary alterations, procedure changes and set point changes.

The inspectors also reviewed issues that had been entered into the corrective action program to determine if Entergy had been effective in identifying problems associated with the 10 CFR 50.59 safety evaluation process. A sample of these issues was selected for further review during which the inspectors assessed the adequacy of the corrective actions implemented for the selected issues.

The safety evaluations and screenings were selected based on the safety significance of the affected structures, systems and components. A listing of the safety evaluations, safety evaluation screens, and other documents reviewed is provided in Attachment 1.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

.1 Partial Walkdown

a. Inspection Scope

The inspectors performed three partial system walkdowns. To evaluate the operability of the redundant train for the selected systems while the affected train was inoperable, the inspectors compared system lineups to system operating procedures, system drawings, and the applicable chapters in the Final Safety Analysis Report. The inspectors also verified the operability of critical system components by observing component material condition during the system walkdown and reviewing of the maintenance history for each component. The inspectors performed the partial walkdowns on the following systems:

- Residual heat removal system (RHR) train B, including RHR pumps B and D, while RHR pump was out of service for seal replacement on November 19, 2002.
- High pressure coolant injection (HPCI)/automatic depressurization systems (ADS) during performance of reactor core isolation cooling (RCIC) system quarterly surveillance procedure ST-24J, "RCIC Flowrate and Inservice Test" during the week of November 8, 2002.
- Standby gas treatment (SGT) system train B and reactor building ventilation during corrective maintenance on the train SGT fan A during the week of November 8, 2002.

b. Findings

No findings of significance were identified.

1R05 Fire Protection - Fire Area Tours

a. Inspection Scope

The inspectors evaluated plant areas important to reactor safety for: (1) control of transient combustibles and ignition sources; (2) material condition, operational status, and operational lineup of fire protection systems, equipment and features; and (3) fire barriers used to prevent fire damage or fire propagation. The inspectors used administrative procedure AP-14.01, "Fire Protection Program."

The areas inspected included:

- Area IX/Zone RB-1A, Reactor Building east - elevation 272 feet
- Area IX/Zone RB-1B, Reactor Building west - elevation 272 feet
- Area IX/Zone SG-1, Standby gas treatment filter room - elevation 272 feet
- Area VI/Zones EG-3,4,6, B and D emergency diesel generator spaces and switchgear room - elevation 272 feet
- Area XVII/Zone RB-1E, Crescent area east - elevations 227 and 242 feet
- Area VII/Zone CR-1, Main control room and control room heating and ventilation equipment rooms - elevation 300 feet
- Area IA/Zone AD-6, Administration Building - elevation 300 feet
- Area IA/Zone MG-1, Reactor water recirculation pump motor generator set room

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures

a. Inspection Scope

The inspector reviewed the assumptions and analyses concerning internal flooding events included in the individual plant examination and the updated final safety analysis report. The inspector assessed the validity of these assumptions during a walkdown of areas where flooding would have the greatest impact on risk: the relay room, the battery rooms, and the reactor building crescent rooms.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities

a. Inspection Scope

The inspectors reviewed the following aspects of the in-service inspection (ISI) program to verify the effectiveness of activities that monitor degradation of the reactor coolant system boundary, risk significant piping system boundaries, and the containment boundary.

- Component nondestructive examinations performed during the refueling outage following cycle 15 (RFO-15) to satisfy the 10-year ASME Section XI ISI program B.
- Relief requests 18 and 19 and the subsequent NRC approval that justified delay of the reactor pressure vessel vertical shell weld and the shell to flange welds until RFO-16.
- 43 Examination report results from ultrasonic testing, penetrant testing, magnetic testing, and visual testing of reactor coolant system components. These examinations included reactor pressure vessel feedwater nozzle welds, recirculation system safe-end and piping welds, support skirt welds, and emergency water system piping.
- 24 Condition reports for deficiencies identified during refueling outage in-service inspections. In particular, the inspectors discussed with cognizant personnel the discovery of foreign material in the reactor lower plenum drain nozzle during the in-vessel visual inspections (IVVI) examination of the reactor lower head plenum in RFO-14, and reviewed the adequacy of corrective actions initiated in response to this issue.
- 14 Examination reports from the RFO-14 and RFO-15 IVVI, including core shroud, and core spray header piping defects that were monitored for crack propagation over past refueling outage periods. For these cases, the inspectors observed the video tape recordings of the critical internals locations with reportable indications.
- Summaries of the inspections, repairs, replacements, and re-inspections for components within the scope of the boiling water reactor visual inspection program.
- Engineering Standard CES-9, Revision 0, which documented the requirements of the visual containment inspection procedure, and the pre-evaluated IWE inspection criteria, ASME B&PV Code, 1998 Edition, 1998 Addenda, Subsection IWE, Examination Category E-A, Item No. E1-10, which together assessed the general condition of the containment and any degradation affecting the structural integrity or leak tightness of the containment.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program

a. Inspection Scope

On November 14, 2002, the inspector observed licensed operator simulator training to assess operator performance for a scenario involving a loss of coolant accident and a failure to scram. The inspectors evaluated the performance of risk significant operator actions including implementation of emergency operating procedure EOP-3, “Failure to Scram.” The inspectors assessed the clarity of communications, the implementation of appropriate actions in response to alarms, the performance of timely control board operation and manipulation, and the oversight and direction provided by the shift supervisor. The inspectors also reviewed simulator fidelity to evaluate the degree of similarity to the actual control room.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness

a. Inspection Scope

The inspectors reviewed performance-based problems involving selected in-scope structures, systems, or components (SSCs) to assess the effectiveness of the maintenance program. Reviews focused on: (1) proper maintenance rule scoping, in accordance with 10 CFR 50.65; (2) characterization of failed SSCs; (3) safety significance classifications; (4) 10 CFR 50.65 (a)(1) and (a)(2) classifications; and (5) the appropriateness of performance criteria for SSCs classified as (a)(2), and goals and corrective actions for SSCs classified as (a)(1). The inspectors reviewed the most recent system health reports and system functional failures of the last two years. The following SSCs were reviewed:

- Recirculation Flow Control System
- Neutron Monitoring System
- RHR shutdown cooling valve interlocks

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

For the selected work listed below, the inspectors verified: (1) risk assessments were performed in accordance with administrative procedure AP-10.10, "On-line Risk Assessment;" (2) risk of scheduled work was managed through the use of compensatory actions; and (3) applicable contingency plans were properly identified in the integrated work schedule. During the maintenance the inspectors toured the work areas to assure that the scope of the work was consistent with the maintenance plans and that no additional systems were adversely impacted.

- Emergency service water loop A system outage from December 10 to 12, 2002.
- Standby gas treatment fan A high vibration corrective maintenance performed the week of November 8, 2002.
- Reactor feed pump A speed control corrective maintenance performed the week of December 20, 2002.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed operability determinations to assess the accuracy of the evaluations, the use and control of compensatory measures if needed, and compliance with technical specifications. The review included a verification that the operability determinations were made as specified by Entergy's administrative procedure AP-03.11, "Operability Determinations." The technical adequacy of the determinations was reviewed and compared to technical specifications, the final safety analysis report, and associated design basis documents. The following evaluations were reviewed:

- HPCI turbine steam supply stop valve 23HOV-1 excessive stroke time
- Reactor water cleanup system high energy line break calculation error
- Recirculation jet pump flow out of specification
- Control room ventilation exhaust fan B rotating backwards
- HPCI operability with turbine steam supply isolation valve 23MOV-14 only partially open

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds

a. Inspection Scope

Prior to and after the October 2002 refueling outage the inspectors reviewed corrective and elective maintenance backlog items to assess Entergy's screening and prioritization of each item, and the effect of each item on the functional capability of the affected system and the ability of an operator responding to an event. The review included plant tours, discussions with plant operators, and a review of surveillance procedure ST-99H, "Operator Work-Arounds Assessment." The inspector also used plant procedures, the updated final safety analysis report, and the technical specifications and bases to assess the cumulative effect of operator workarounds which existed prior to and after the October 2002 refueling outage on the operator's ability to implement abnormal and emergency operating procedures.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications

a. Inspection Scope

The inspectors reviewed permanent plant modification packages, calculations, set point changes and engineering evaluations to verify that the design and licensing bases, and the performance capability of risk significant structure, systems, and components (SSCs) were not degraded through plant modifications.

Plant changes were selected for review based on risk insights for the plant and included structures, systems and components associated with the initiating events, barrier integrity and mitigating systems cornerstones. The inspectors walked down selected plant systems and components, interviewed plant staff, and reviewed applicable documents including procedures, calculations, modification packages, engineering evaluations, drawings, corrective action documents, the updated final safety analysis report (UFSAR), technical specifications, and system design basis documents (DBDs).

The inspectors verified that selected attributes were consistent with the design and licensing bases. These attributes included component safety classification, energy requirements supplied by supporting systems, seismic qualification, instrument set-points, uncertainty calculations, electrical coordination, electrical loads analysis, and equipment environmental qualification. Design assumptions were reviewed to verify that they were technically appropriate and consistent with the UFSAR. For each modification the 50.59 screenings or evaluations were reviewed as described in section 1R02 of this report. The inspectors verified that procedures, DBDs, and the UFSAR were properly updated with revised design information and operating guidance. The inspectors also verified that the as-built configuration was accurately reflected in the design documentation and that post-modification testing was adequate to ensure that the SSCs were operable.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing

a. Inspection Scope

The inspectors reviewed post-maintenance test procedures and associated testing activities for selected risk significant mitigating systems to assess whether: (1) the effect of testing on the plant had been adequately addressed by control room and engineering personnel; (2) testing was adequate for the maintenance performed; (3) acceptance criteria were clear and adequately demonstrated operational readiness, consistent with design and licensing basis documents; (4) test instrumentation had current calibrations, range, and accuracy for the application; (5) tests were performed, as written, with applicable prerequisites satisfied; and (6) that equipment was returned to the status required to perform its safety function. Post-maintenance testing for the following work was reviewed.

- Corrective maintenance on the SGT system fan A motor foundation
- Corrective maintenance on RHR pump A and B discharge check valves, 10RHR-42A/B
- Diagnostic testing of startup feedwater control valve 34FCV-137
- Corrective maintenance performed on the no. 11 cylinder fuel injector of emergency diesel generator B

- Spring pack, body guide, and bonnet modifications for the inboard main steam isolation valve A

b. Findings

No findings of significance were identified.

1R20 Refueling and Other Outage Activities

a. Inspection Scope

The inspector observed and reviewed selected refueling outage activities to verify that operability requirements were met and that risk, industry experience, and previous site specific problems were considered.

- Outage Plan: The inspector reviewed detailed outage schedules and procedures to verify that technical specification required safety system availability was maintained, that risk was considered, and that contingency plans existed for restoring key safety functions such as electrical power and primary coolant system makeup.

- Plant shutdown and cooldown: The inspector observed portions of the plant shutdown and cooldown on October 6 and 7, 2002, and verified that the TS cooldown rate limits were satisfied.

- During the course of the refueling outage, the inspector observed selected reactor disassembly activities and walked down clearances to verify that tagouts were properly hung and that equipment was properly configured. Through plant tours, the inspector verified that Entergy maintained and adequately protected electrical power supplies to safety-related equipment and that technical specification requirements were met.

- The inspector periodically verified proper alignment and operation of the shutdown cooling and decay heat removal systems. The verification also included reactor cavity and fuel pool makeup paths and water sources, and administrative control of drain down paths.

- The inspector reviewed procedures RAP-7.1.04B, “Refueling Procedure,” and RAP-7.1.04C, “Neutron Instrument Monitoring During In-Core Fuel Handling,” and the results of refueling platform interlock functional tests to ensure that the technical specification requirements for fuel movement were met. The inspector also verified through review of procedure ST-39D, “Secondary Containment Leak Test,” that containment requirements for refueling activities were met.

- The inspector observed portions of the reactor startup following the outage, and verified through plant walkdowns, control room observations, and surveillance test reviews that the safety-related equipment required for mode change was operable, that containment integrity was set, and that reactor coolant boundary leakage was within technical specification limits.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors witnessed performance of surveillance test procedures and reviewed test data of selected risk-significant systems, structures, and components (SSCs) to assess whether the SSCs satisfied technical specifications, updated final safety analysis report, technical requirements manual, and Entergy procedure requirements. The inspectors assessed whether the testing appropriately demonstrated that the SSCs were operationally ready and capable of performing their intended safety functions. The following tests were witnessed:

- ST-4N, "HPCI Quick Start, Inservice, and Transient Monitoring Test"
- ST-6M, "Standby Liquid Control Recirculation and Injection Test"
- ST-18, "Main Control Room Emergency Fan and Damper Operability Test"
- ST-2AM, "RHR Loop B Quarterly Operability Test"
- ST-23C, "Jet Pump Operability Test For Two Loop Operation"
- ST-9C, "Emergency AC Power Load Sequencing Test and 4KV Emergency Power System Voltage Relays Instrument Functional Test"

b. Findings

No findings of significance were identified.

1EP4 Emergency Action Level (EAL) and Emergency Plan Changes

a. Inspection Scope

During an in-office inspection on November 13, 2002, the inspector reviewed recent changes to emergency plan documents to determine if the changes resulted in a decrease of effectiveness of the emergency plan. A thorough review was performed on aspects of the plan related to the risk significant planning standards, such as classifications, notifications and protective action recommendations. A cursory review was performed for the non-risk significant planning standards portions. These changes were reviewed against 10 CFR 50.54(q) to ensure that the changes did not decrease the effectiveness of the plan, and that the changes made did meet the standards of 10 CFR 50.47(b) and the requirements of Appendix E. These changes are subject to future inspections to ensure that the impact of the changes continues to meet NRC regulations. A list of the reviewed documents is in Attachment 1.

a. Findings

No findings of significance were identified.

1EP6 Drill Evaluation

a. Inspection Scope

The inspectors observed simulator activities associated with licensed operator requalification training on November 14, 2003. The inspectors verified that emergency classification declarations and notification activities were properly completed.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY
Cornerstone: Occupational Radiation Safety

2OS2 ALARA Planning and Controls

a. Inspection Scope

From October 21 to October 25, 2002, the inspector performed the following activities to verify that Entergy properly implemented operational, engineering, and administrative controls for maintaining personnel exposure as low as is reasonably achievable (ALARA) during the refueling outage. These controls were reviewed against the criteria contained in 10 CFR 20, applicable industry standards, and Entergy's procedures.

- The inspector reviewed pertinent information regarding cumulative exposure history, current exposure trends, ALARA reviews, in-progress ALARA reviews, and radiation work permits (RWP) to assess effectiveness in establishing goals and keeping actual personnel exposure ALARA when performing outage work activities. The inspector also reviewed the results of Entergy's efforts to reduce plant source terms through system flushing, component decontamination, temporary/permanent shielding installations, and use of the Wet Lift system for transferring contaminated reactor components.

- The inspector performed independent radiation surveys in radiologically controlled areas including the drywell, reactor building, turbine building, and motor-generator set room to confirm the accuracy of posted survey results and assess the adequacy of RWPs and associated controls.

- The inspector reviewed the exposure controls specified in ALARA reviews (AR) for all work activities in which actual cumulative exposure exceeded five person-rem and also exceeded the originally estimated dose. The review included AR 02-025 for snubber inspection/testing, AR 02-033 for control rod drive changeout, and AR 02-057 for drywell permanent shielding installation. In-progress ALARA reviews for these jobs and ALARA committee meeting minutes were reviewed to assess Entergy's subsequent actions in controlling dose for these tasks and methods of forecasting dose estimates.

- The inspector observed radiologically significant jobs-in-progress including: (1) exposure controls specified in radiation work permits, and (2) pre-job briefings for RWP 02-0504, removal of drywell shielding, and RWP 02-0700, reactor cavity decontamination. For these tasks the inspector interviewed selected workers on their knowledge of the relevant radiation work permit, electronic dosimetry setpoints, and job-site radiological conditions.

- The inspector reviewed various records regarding monitoring of personnel exposure, including dose records for contractors and a declared pregnant worker, dose control limit extensions, whole body counter results, air sample results, and contamination survey records.

b. Findings

No findings of significance were identified.

3. SAFEGUARDS
Cornerstone: Physical Protection

3PP1 Response to Contingency Events

a. Inspection Scope

The inspectors reviewed the status of security operations and assessed Entergy's implementation of the protective measures in place as a result of the current, elevated threat environment.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification

a. Inspection Scope

The inspectors compared performance indicator (PI) data reported for the below listed cornerstones to the guidance set forth in NEI 99-02, "Regulatory Assessment Performance Indicator Guidance," to verify the accuracy of the PI data reported between January 2002 and December 2002.

Initiating Events and Mitigating Systems Cornerstones

- Unplanned scrams per 7,000 critical hours PI
- Scrams with a loss of normal heat removal PI
- Unplanned power changes per 7,000 critical hours PI
- Safety system functional failures PI

The inspector reviewed operator logs, monthly operating reports, licensee event reports, and plant computer data to determine the number of scrams, scrams with loss of normal heat removal, unplanned power changes greater than 20%, and safety system functional failures reported by licensee event reports that occurred between January 2002 and December 2002.

Occupational Radiation Safety Cornerstone

- Occupational exposure control effectiveness PI

The inspector reviewed condition reports and associated documents to determine the number of occurrences involving locked high radiation areas, very high radiation areas, and unplanned personnel exposures between September and October 2002.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems

.1 Problem Identification and Resolution Sample Inspection, DER 01-04688, Momentary three to four megawatt power reduction without operator action

a. Inspection Scope

The inspector reviewed the identification and resolution of problems involving spurious control rod block alarms and core power indication anomalies as documented in DER 01-04688. The inspector also reviewed the adequacy of the root cause analysis to verify that the corrective actions were appropriate.

b. Findings

No findings of significance were identified.

.2 Other Identification and Resolution of Problems Items

a. Inspection Scope

The inspectors reviewed 14 items selected across the initiating events, mitigating systems, and barrier integrity cornerstones to determine if problems were being properly identified, prioritized, entered into the corrective action program, and corrected. The inspectors evaluated Entergy's threshold for problem identification and efforts to establish the scope of problems by reviewing pertinent logs, work requests, engineering evaluations, surveillance test results, and self-assessments. The following condition reports were reviewed:

- 02-04162, Local leak rate test failure - 15AOV-133B
- 02-04165, Stack automatic iodine and particulate sampling system failed surveillance test
- 02-04170, Local leak rate test failure - 14CSP-62A
- 02-04125, Standby liquid control failed post-work test ST-6M
- 02-04120, Battery charger A adjustment problems
- 02-03968, High vibration standby gas treatment fan A
- 02-04986, Reactor core isolation cooling piping design pressure exceeded during surveillance procedure ST-24J
- 02-04871, Reactor building cooling air-operated valves slow stroke during surveillance procedure ST-1R
- 02-05048, Adverse trend - secondary containment operability with a degraded or open reactor building penetration
- 02-05188, Control room fan 70FN-4B rotated backwards
- 02-02693, Off-gas dryer heater grounded
- 02-04351, Residual heat removal pump discharge check valve broken
- 02-05425, High pressure coolant injection turbine steam admission valve not full open
- 02-03968, Surveillance procedure ST-23C acceptance criteria not met for jet pump operability

The inspector also reviewed twelve condition reports and a quality assurance process monitoring report relating to the implementation of radiological controls for work performed in radiologically controlled areas. The inspector also reviewed twelve condition reports concerning ALARA planning and controls that were initiated during the refueling outage to evaluate Entergy's threshold for identifying, evaluating, and resolving ALARA program implementation. The review was conducted against the criteria contained in 10 CFR 20, technical specifications, and Entergy's procedures.

The inspectors reviewed corrective action documents associated with 10 CFR 50.59 and plant modification issues to ensure that Entergy was identifying, evaluating, and correcting problems in these areas and that corrective actions were appropriate. The inspectors also reviewed a quality assurance audit and several self-assessments related to 10 CFR 50.59 and plant modification activities.

4OA3 Event Follow-up

(Closed) LER 50-333/00-015-02: Containment Leakage Rate Exceeds Authorized Limits. This licensee event report (LER) revision provided an update of planned corrective actions to improve main steam isolation valve performance. The LER revision was reviewed by the inspector and no findings of significance were identified. The event was documented in Entergy's corrective action program as DER-00-05158. This LER revision is closed.

4OA5 Other

a. Inspection Scope

The inspectors completed an audit of Entergy's performance of the interim compensatory measures imposed by the NRC's Order Modifying License, issued on February 25, 2002, in accordance with the guidance of NRC Inspection Manual Temporary Instruction 2515/148, Revision 1, Appendix A, dated September 13, 2002.

b. Findings

No findings of significance were identified.

4OA6 Management Meetings

.1 Exit Meeting Summary

On January 13, 2003, the resident inspectors presented the inspection results to Mr. Brian O'Grady and members of the Entergy staff. No proprietary information was identified.

ATTACHMENT 1

SUPPLEMENTAL INFORMATION

a. Key Points of Contact

K. Pushee	Radiation Protection Manager
B. O'Grady	General Manager or Plant Operations
T. Sullivan	Site Executive Officer
A. Zaremba	Director, Safety Assurance
A. Halliday	Manager, Licensing
D. Johnson	Manager, Scheduling and Outages
O. Limpias	Director, Engineering
P. Russell	Operations Manager
N. Avrankatos	Emergency Preparedness Coordinator
D. Ruddy	Manager, CA&A
A. Khanifar	Manager, Design Engineering
S. Bono	Manager, Systems Engineering

b. List of Items Opened, Closed, or Discussed

Closed

50-333/03-15-02	LER	Containment Leakage Rate Exceeds Authorized Limits (Section 40A3.1)

c. List of Documents Reviewed

Section 1R01, Adverse Weather Protection

MP-71.5, "Outdoor Heat Tracing Inspection and Testing"
MST-71.6, "Intake De-icing Heaters Insulation Resistance Surveillance Test"
MST-71-17, "Intake De-icing Heaters Rated Current Surveillance Test"
ST-8G, "Intake Deicing Heaters Feeder Ammeters Test"

Section 1R04, Equipment Alignment

Operating Procedure OP-13, "Residual Heat Removal System"
Operating Procedure OP-13A, "Low Pressure Coolant Injection"
Operating Procedure OP-13B, "Containment Control"
Entergy Nuclear Northeast Drawing No. FM-20A, "Residual Heat Removal"

Section 1R06, Flood Protection Measures

ST-50, "Floor Drain Flow Test"
AOP-51, "Unexpected Fire Pump Start"

Section 1R08, Inservice Inspection Activities

Entergy Inservice Inspection Program for RFO-15

List of planned, deferred, or implemented inservice examinations for reactor vessel - class 1
James A. FitzPatrick Third Inspection Interval Relief Request 18, Reactor Vessel Vertical Shell Welds
James A. FitzPatrick Third Inspection Interval Relief Request 19, Shell to Flange Weld
JAF-ISI-0002, Third Interval Inservice Inspection Program, Revision 3
James A. FitzPatrick Appendix A Program Summary Tables - ASME Class 1 System and Components
James A. FitzPatrick RFO-15 Inservice Inspection Scope of Components Examined
Table IWB-2500-1, ASME Exam Category B-A Pressure Retaining Welds in Reactor Vessel

Condition Reports (CR-JAF-)

02-03766, 02-03767, 02-03825, 02-03826, 02-03870, 02-03882, 02-03901, 02-03943, 02-03946, 02-03949, 02-03980, 02-04141, 02-04207, 02-04373, 02-04399, 02-04408, 02-04477, 02-04512, 02-04513, 02-04643, 02-04678, 02-04900

CR-JAF-ICD-MULTI-04346, RFO-15 Basis for Inservice Inspection Weld and Support Selection, Revision 3
CR-JAF-ICD-MULTI-04306, RFO-15 Basis for Intragranular Stress Corrosion Cracking Weld Selection, Revision 1

NDT Examination Reports

02-UT-069, Ultrasonic Examination of N4C Nozzle to Shell Weld
02-UT-012, Ultrasonic Examination of Recirculation System Weld 22-02-2-63
02-UT-019, Ultrasonic Examination of Recirculation System Safe End Weld N-2F-SE
02-MT-006, Magnetic Particle Examination of Residual Heat Removal System Weld 24-10-992
Wesdyne, N4C-IR feed water system outer diameter automated ultrasonic nozzle IR exam summary sheet
Wesdyne, Ultrasonic feed water nozzle ultrasonic examination scope
List of 37 UT, MT, VT-1, VT-3, or PT results reviewed for system nos. 002, 010, 013, 029, and 034

In-vessel Visual Inspection

Reactor Internals Inspection History (Updated for RFO-15)
Summary of RFO-15 Inspections, Repairs, Replacements, Re-Inspections of Boiling Water Reactor Visual Inspection Program Scope
Summary of RFO-14 Inspections, Repairs, Replacements, Re-Inspections of Boiling Water Reactor Visual Inspection Program Scope
Historical summary of in-vessel visual inspection results for core spray sparger, core shroud, feedwater sparger, control rod guide tube, fuel support castings, in-core dry tubes, and lower plenum.
FPK-R-14-RPT-01, In-vessel Visual Inspection Data Sheet Core Spray Header Piping - Loop B
FPK-R-14-RPT-01, In-vessel Visual Inspection Data Sheet Core Shroud

FPK-R14-RPT-01GE, RFO-14 VT-3 of lower plenum sheet and VT of Reactor Pressure Vessel drain bore
JAF-R15-IVVI-01b, Examination Data Sheet for Core Spray Internal Piping Welds - Loop B
JAF-R15-IVVI-03, Examination Data Sheet for Core Shroud Welds
INR FPK-02-02, Shroud SV5B Inner and Outer Diameter Previous Indications
INR FPK- 02-01, CSB-12 Previous Indication Core Spray Header Pipe
IVVI RPT FPK-02-JLMC9, RFO-15 Examination Summary Report
IVVI Report No./ FPK-R14-RPT-01, RFO-14 Examination Summary Report

Containment Boundary Examination

CES-9, Visual containment Inspection Procedure and pre-evaluated IWE Inspection Criteria
JAF-RPT-PC-04486, ASME XI IWE Class MC Pressure Component Inspection, Rev. 0
02VT296, Southeast equipment hatch VT-1 inservice inspection examination of bolting

Foreign Material Exclusion Evaluation

AP-19.06, "Piping Support and Pressure Retaining Component ISI," Rev. 4
AP-05.06, "System Internal Cleanliness and Foreign Material Exclusion," Rev. 7
AP-05.01, "Job Briefings," Rev. 10
JAF-RPT-RPV - 04515, RFO15 Reactor Vessel Inspection, Foreign Material Evaluation
DER-98-03082, Foreign Material on core plate near control rod 06-15
DER-98-03141, Debris found during in-vessel visual inspection

Drawings

PA-MSK-3036, "Inservice Inspection Reactor Vessel Stretchout for Weld Designation," Rev. 3
ISI-SYS-02, "Reactor Water Recirculation System Piping Isometric," Rev. 11

Section 1R12, Maintenance Rule Implementation

Maintenance Rule Basis Document for System 007, Neutron Monitoring System
JTS-02-0440, Residual Heat Removal Shutdown Cooling Functional Failures, dated November 25, 2002
Maintenance Rule Basis Document for System 002-184, Recirculation Flow Control System
JENG-APL-02-005, Maintenance Rule Action Plan for the Recirculation Flow Control System

Section 1R13, Maintenance Risk Assessment and Emergent Work

Limiting condition for operation tracking and safety function determination form, No. 02-3137, dated October 3, 2002
ST-23C, "Jet Pump Operability Test For Two Loop Operation"
JAF-CALC-HPCI-05430, High Pressure Coolant Injection Steam Line Pressure Drop with Partially Open Steam Admission Valve, dated December 12, 2002

Limiting condition for operation tracking and safety function determination form, No. 02-3325, dated November 19, 2002
ST-18B, "Control Room Emergency Ventilation Air Supply Operability Test"
FB-45A, Flow Diagram - Control and Relay Rooms Heating and Ventilation System

Sections 1R02, Evaluations of Changes, Tests, or Experiments and 1R17, Permanent Plant Modifications

Commercial Grade Dedications

CG-01-006, Selector Switches
CG-01-011, EMI Filter

Permanent Plant Modifications

JD-01-170, Electro Hydraulic Control Scram Frequency Reduction
JD-01-102, Cycle 16 Core Reload
JD-01-204, Feedwater Pump Seal Replacement
JD-99-015, Power Supplies - 15MOV-102 & 103 Replacement
JD-01-106, Alleviate Voltage Drop Restraints on "A" Station Battery
JD-01-147, Main Steam Isolation Valve Bonnet & Spring Pack Upgrade
JE-00-115, Lamda LFS-39-48 and RWS-30A-48/A Power Supply Replacement

10 CFR 50.59 Safety Evaluations

JAF-SE-00-001, Power Supplies - 15MOV-102(OP)/103(OP)
JAF-SE-00-050, Digital Upgrade - Replacement of Dry Well Equipment and Floor Drain Sump Level Recorders 20 LR-122A [B]
JAF-SE-97-039, Torus/Drywell Vacuum Breaker Alternate Test Method
JAF-SE-00-014, Safety Relief Valve Electric Lift/Anticipated Transient Without Scram Setpoint Change Project
JAF-SE-02-010, Final Feedwater Temperature Reduction
JAF-SE-02-002, Clarification of Control Room Emergency Ventilation Design
JAF-SE-02-011, Cycle 16 Core Reload

10 CFR 50.59 Safety Evaluation Screens

JD-01-147, Main Steam Isolation Valve Bonnet & Spring Pack Upgrade
JD-01-106, Alleviate Voltage Drop Restraints on "A" Station Battery
JD-01-204, Main Feedwater Pump Seal Replacement
JD-01-170, Electro Hydraulic Control Scram Frequency Reduction
JE-00-115, LFS-39-48 & RWS-30A-48/A Power Supply Replacement
TA-02-039, Temporary Alteration - Install Chart Recorder on 31MCU-1B
TM-02-003, Temporary Modification - Replace 12-4FT-75A Transmitter
S1-01-0021, 23PS-85 Setpoint Revision
EOP-3, "Failure to Scram," procedure temporary change
EOP-7, "RPV Flooding," procedure temporary change
EP-1, "EOP Entry and Use," procedure revision

Design References and Calculations

JAF-CALC-02609, 125 Vdc Station Battery "A" Sizing & Voltage Drop, Revision 1
JAF-ECAF-MCC152-OE3, Electrical Distribution System Coordination Adequacy Form - MCC152-OE3, Revision 0
JAF-ECAF-MCC162-OE3, Electrical Distribution System Coordination Adequacy Form - MCC162-OE3, Revision 0
99-123, Assessment of the Emergency Service Water System to Provide Minimum Required Safety Related Flows with the Valves 15MOV-102 & 15MOV-103 Open
9663-JAF-CALC-0001, 15MOV-102 & 103 Power Supplies, Revision 0
W-NYPA-78Q-303 thru -311, Reactor Pressure Vessel Component Stress/Fatigue Analysis
JAF-CALC-MISC-02875, Suppression Pool Temp Following Small Break Loss of Coolant Accident with High Pressure Coolant Injection Operation
JAF-CALC-RBC-04400, Wall Thinning and Remaining Service Life for Emergency Service Water Lines
JAF-CALC-SWS-04350, Bases For Inservice Testing Acceptance Criteria for Emergency Service Water Pumps
JAF-CALC-MISC-03116, Evaluate Impact of Core Reload and Revised Plant Specific Decay Heat Values on the Results of the Emergency Operating Procedures/Severe Accident Operation Guidelines Support Calculations
JAF-ICD-HPCI-03270, Improved Technical Specification Open Item 380

Procedures

AOP-11, "Loss of Reactor Building Closed Loop Cooling," Revision 13
AOP-28, "Operation During Plant Fires," Revision 12
AOP-43, "Plant Shutdown From Outside The Control Room," Revision 28
AOP-45," Loss of DC Power System A," Revision 8
EOP-3, "Failure to Scram," Revision 6
ENN-LI-100, "Process Applicability Determination," Revision 2
ENN-LI-101, "10CFR50.59 Review Process," Revision 1
EOP-7, "RPV Flooding," Revision 6
EP-1, "EOP Entry and Use," Revision 6
IMP-02-184.6, "Recirculation MG Set Scoop Tube Actuator Online High Speed Stop and Limit Adjustment," Revision 3
ISP-22-2, "High Pressure Coolant Injection System Loop Low Flow Bypass Valve Instruments," Revision 2
MP-029.01, "Main Steam Isolation Valve Maintenance," Revision 21
MP-029.02, "Main Steam Isolation Valve Actuator Maintenance," Revision 10
RAP-7.3.30, "Cycle Startup Reactor Physics Test Program," Revision 11
TST-105, "RWR Pump Scoop Tube Mechanical Stop Verification," Revision 3

Work Requests

97-01354-11, 01-10534-00, 01-10534-01, 97-01354-18, 01-10534-02, 01-00684-00, 01-10535-00, 01-04011-00, 01-10535-01, 01-04011-01, 01-10535-02, 01-09985-11

Drawings

1.11-282, Elementary Diagram - Electro Hydraulic Control System Alarm and Trip Circuits
1.11-464, Schematic Diagram Speed Control Logic
1.11-544, Schematic Wiring Diagram Valve Test Logic TSV-1,2,3&4
1.11-629, Schematic Power Load Unbalance Logic
FE-15, 600 V One Line Diagram - 71MCC-151, 152,161 & 162
FM-15B, Flow Diagram Reactor Building Cooling Water
FM-46B, Flow Diagram Emergency Service Water
FM-46E, Conduit Plan Reactor Building
FM-46F, Conduit Plan Reactor Building

Self-Assessments and Quality Assurance Audits

A02-03J, Quality Assurance Audit Report, Design Control
JENG-02-0037, 4th Quarter 2001 Reactor Engineering Critique/Self Assessment of Down Powers
JENG-02-0177, 1st Quarter 2002 Reactor Engineering Critique/Self Assessment of Reactivity Management
JENG-02-0201, 1st Quarter 2002 Engineering Roll-up Self Assessment
JENG-02-0329, Six Month Maintenance Effectiveness Review

Miscellaneous Documents

UFSAR Change Request No. 01-026, Revise Power Supply Swap of 15MOV-102 & 103
MO100149-01, Verification & Validation Report Series 2100 Log, Version 3.0A
Procedure Change Request for AOP-43
GE Technical Information Letter (TIL) 1212-2, Plant Scram Frequency Reduction Features for Boiling Water Reactor and Pressurized Water Reactor Nuclear Turbines with MK I or MK II Electro Hydraulic Controls
SCR-S1-01-0021, 23PS-85 Set Point Revision

Condition Reports:

CR-JAF-2001-01551, 03010, 03691, 05027, 00403, 00651, 00787, 00222, 03564
CR-JAF-2002-00363, 01301, 02524, 03392, 03559, 03697, 01160, 01176, 01187, 01693,
03395

Action/Commitment Tracking Forms

ACT-01-56115, 56883, 56972, 56974, 57288, 57290

ACT-02-57290, 63374, 64963, 64962

Section 1R20, Refueling Outage Activities

OP-65, "Startup and Shutdown Procedure"
OP-30A, "Refueling Water Level Control"
OP-65B, "Shutdown Operation"
ST-20G, "Refuel Bridge and Main Fuel Grapple Daily Checks"

ST-69A, "Main Fuel Grapple Fail Safe Test"
ST-20F, "Refuel Interlocks Functional Test"
ST-23F, "Control Rod "Full-In" Functional Test"
ST-5O, "SRM Functional Test"
ST-20N, "Control Rod Exercise/Timing/Stall Flow Test"
RAP-7.2.4, "Reactor Core Fuel Verification"
RAP-7.3.13, "Estimated Critical Positions and Reactor Startup/Neutron Monitoring"
RAP-7.4.09, "Shutdown Margin Check"
RAP-7.3.03, "Core Thermal Power Evaluation"
RAP-7.3.5, "Core Power Symmetry Analysis"
RAP-7.3.7, "Core Flow Evaluation and Indication Calibration"
RAP-7.3.29, "Determination of Rated Recirculation Flow"

Section 2OS2, ALARA Planning and Controls

Procedures:

AP-07.00, "Radiation Protection Program," Revision 6
AP-07.01, "Radiation Work Permit Program," Revision 8
AP-07.03, "ALARA Program," Revision 4
AP-07.05, "Exposure Monitoring and Radiological Controls for Site & RCA Access," Revision 7
AP-07.06, "High Radiation Area Control," Revision 11
RP-OPS-02.02, "Radiation Work Permit," Revision 7
RP-OPS-02.03, "High Radiation Area Access and Key Control," Revision 3
RP-OPS-02.04, "Personnel Radiological Hold," Revision 3
RP-OPS-03.01, "Radiological Survey Performance and Documentation," Revision 4
RP-OPS-03.03, "Radiological Posting and Labels," Revision 5
RP-OPS-03.05, "Refuel Floor and Drywell Radiological Controls," Revision 2
RP-OPS-08.01, "Routine Surveys and Inspections," Revision 10
RP-ALARA-01.01, "ALARA Review," Revision 4

ALARA Reviews/In-progress ALARA Reviews:

AR 02-004, Contract Services Activities
AR 02-006, Mechanical Maintenance Activities
AR 02-010, Radiation Protection Activities
AR 02-018, Inservice Inspection/Erosion/Corrosion
AR 02-020, Motor Operated Valve Maintenance/Testing
AR 02-022, Reactor Disassemble/Reassembly
AR 02-025, Snubbers Inspection and Testing
AR 02-033, Control Rod Drive Changeout
AR 02-036, Drywell Main Steam Isolation Valves
AR 02-057, Drywell Permanent Shielding Installation

Radiation work permits:

RWP 02-0504, Contract Services Support in Drywell
RWP 02-0505, Instrument and Control Support in Drywell
RWP 02-0507, Electrical Maintenance in Drywell

RWP 02-0508, Operations Support in Drywell

Condition reports:

CR-JAF-2002-04773, 04635, 04634, 04633, 04632, 04631, 04630, 04629, 04619, 04615, 04314, and 04106

Quality assurance process monitoring report:

RQQAPM No. 02-0140, Main Steam Isolation Valve Repairs

Other:

Outline of Cavity Decontamination Evolution

Section 4OA2, Identification and Resolution of Problems

Root Cause Evaluation of the Momentary Three to Four Megawatt Power Reduction Without Operator Action, DER-01-04688, dated March 28, 2002

JENG-APL-01-019, Control Rod Block Alarm Reduction Improvement Action Plan

c. List of Acronyms

ADS	Automatic depressurization system
ALARA	As low as is reasonably achievable
AR	ALARA review
ASME	American Society of Mechanical Engineers
CFR	Code of Federal Regulations
CR	Condition report
DBD	Design basis document
DER	Deviation/event report
EOP	Emergency Operating Procedure
HPCI	High pressure coolant injection
ISI	Inservice Inspection
IVVI	In-Vessel Visual Inspection
JAF	James A. FitzPatrick
LER	Licensee event report
NRC	Nuclear Regulatory Commission
PI	Performance indicator
RCA	Radiologically controlled area
RCIC	Reactor core isolation cooling
RFO	Refueling Outage
RHR	Residual heat removal
RWP	Radiation work permit
SSC	Structures, systems, and components
SGT	Standby Gas Treatment
TS	Technical specifications
UFSAR	Updated Final Safety Analysis Report