November 2, 2001

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

SUBJECT: FITZPATRICK NUCLEAR POWER PLANT - NRC INSPECTION REPORT 50-333/2001-006

Dear Mr. Sullivan:

On September 27, 2001, the NRC completed a team inspection at the James A. FitzPatrick Nuclear Power Plant. The enclosed report documents the inspection findings which were discussed on September 27, 2001, with you and 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.

No findings of significance were identified.

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Sincerely,

/RA/

Lawrence T. Doerflein, Chief
Systems Branch
Division of Reactor Safety

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

Enclosure: Inspection Report 50-333/2001-006
Attachment: Supplemental Information

cc w/encl:
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M. Colomb, General Manager, Entergy Nuclear Operations
J. Knubel, VP Operations Support
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J. Fulton, Assistant General Counsel
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OFFICE	RI/DRS		RI/DRS		RI/DRP					
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DATE	10/24/01		11/01/01		10/26/01					

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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.:	50-333
License No.:	DPR-59
Report No.:	50-333/2001-06
Licensee:	Entergy Nuclear Northeast Post Office Box 110 Lycoming, NY 13093
Facility:	James A. FitzPatrick Nuclear Power Plant
Location:	268 Lake Road Scriba, New York 13093
Dates:	September 10 - 27, 2001
Inspectors:	M. Modes, Team Leader P. Kaufman, Sr. Reactor Inspector A. Lohmeier, Reactor Inspector F. Arner, Reactor Inspector O. Mazzoni, Contractor
Approved by:	L. Doerflein, Chief Systems Branch Division of Reactor Safety

SUMMARY OF FINDINGS

IR 05000333/2001-006, on 09/10 - 09/28/2001; Entergy Nuclear Northeast, James A. FitzPatrick Nuclear Power Plant, Engineering Team Report.

No findings of significance 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 are indicated by “No Color” or by the severity level of the applicable violation. The NRC’s program for overseeing the safe operation of commercial nuclear power reactors is described at its Reactor Oversight Process website at http://www.nrc.gov/NRR/OVERSIGHT/index.html.

A. Inspector Identified Findings

None

B. Licensee Identified Findings

None.

Report Details

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R02 Evaluations of Changes, Tests, or Experiments (IP71111.02)

a. Inspection Scope

The team reviewed a sample of safety evaluations (SEs) required by 10 CFR 50.59 for changes to facility systems, structures, and components or procedures as described in the J. A. Fitzpatrick Updated Final Safety Analysis Report (USFAR). The SEs were selected from a list of changes implemented during the last year. The review was conducted to verify that the changes to the facility or procedures as described in the UFSAR, and test and experiments not described in the UFSAR, were reviewed and documented by the licensee in accordance with 10 CFR 50.59. The team also verified that the changes, tests, and experiments did not require prior NRC approval or a license amendment.

The team also reviewed a sample of changes and tests for which Entergy determined that a safety evaluation was not required. This review was performed to verify that Entergy's threshold for performing safety evaluations was consistent with the requirements of 10 CFR 50.59. Lastly, the team verified that the problems identified with the implementation of the safety evaluation program were entered into the corrective action program.

b. Findings

No findings of significance were identified.

1R21 Safety System Design and Performance Capability (IP 71111.21)

a. Inspection Scope

The team reviewed the design and performance capability of the residual heat removal service water (RHRSW) system. The RHRSW is a system used to mitigate an accident by removing heat from containment during accident and transient conditions. Inspection criteria were based on the system performance requirements derived from the Fitzpatrick UFSAR, Technical Specifications (TS), probabilistic risk assessment (PRA), and the NRC's Significant Determination Process (SDP), Fitzpatrick Phase 2 Worksheets. To determine the system performance mitigation requirements the team selected and reviewed accident and transient analyses assumptions for sequences associated with RHRSW system components.

The team verified that: (1) the system design bases were in accordance with the licensing commitments and regulatory requirements; and (2) the design documents, such as drawings and design calculations, were correct. The documents reviewed included engineering analyses, calculations, piping and instrumentation (P&IDs), electrical schematics, and instrument setpoint documentation. The mechanical design review focused on the capability of the RHRSW system to supply adequate water to the RHR heat exchangers under design and transient conditions. Emphasis was placed on the RHRSW pumps and heat exchangers. Selected valves with active safety functions were reviewed to ensure they could support design and risk significant functions. Specifically, the RHRSW heat exchanger outlet valves and crosstie valves were reviewed to ensure their ability to support the success criteria flow rates for the containment heat removal function and late injection function defined in the individual plant examination (IPE), SDP, FSAR and technical specifications.

Additionally, the current performance and test criteria for the RHRSW pumps and heat exchangers were reviewed to ensure consistency between allowable component performance and minimum allowable capabilities assumed in the accident analyses and associated design basis calculations. The team performed independent calculations utilizing the most recent RHRHX test data and best estimate RHRSW flow data to verify heat exchanger performance remained above minimum assumed performance in accident and transient analyses. These analyses included the design basis accident loss of coolant accident (DBA-LOCA), net positive suction head (NPSH) evaluations for the RHR and core spray systems, and NUREG-0783 (steam condensing during safety relief valve discharge) events.

The team walked down system components and supporting systems including service water to verify associated functions such as keep-fill of the RHRSW piping. The team verified that normal, abnormal, and emergency operating procedures were consistent with system design bases and PRA, SDP operating assumptions. As part of this review, the team reviewed system interfaces (instrumentation, controls, and alarms) available to operators to support operator decision making. The team also reviewed the ability of procedures in place to respond to anomalous conditions and complete activities outside the design basis, but risk significant, such as late vessel injection via the RHRSW system crosstie to the residual heat removal loops.

b. Findings

No findings of significance were identified.

C. OTHER ACTIVITIES (OA)

4OA6 Meetings

.1 Exit Meeting Summary

On September 27, 2001, the team presented the inspection results to Mr. T. A. Sullivan and other members of the licensee’s staff. The team verified this inspection report does not contain proprietary information.

KEY POINTS OF CONTACT

Licensee

B. Drain	Manager, Project Management
G. Brownell	Licensing Engineer
T. Herrmann	Response Team Lead
A. Holliday	Licensing Manager
G. Thomas	Director Design Engineering
A. Zaremba	Director of Safety Assurance
W. Maquire	Maintenance Manager
S. Bono	Manager, Corrective Action

NRC

L. Doerflein, Chief, Systems Branch
R. Rasmussen, Senior Resident Inspector

LIST OF ITEMS OPEN, CLOSED, AND DISCUSSED

Opened/Closed

None

LIST OF ACRONYMS

DBA-LOCA	Design Basis Accident - Loss of Coolant Accident
IPE	Individual Plant Examination
NPSH	Net Positive Suction Head
PI&D	Piping and Instrumentation Drawing
PRA	Probabilistic Risk Assessment
RHRSW	Residual Heat Removal Service Water
SDP	Significant Determination Process
SE	Safety Evaluation
UFSAR	Updated Final Safety Evaluation Report

DOCUMENTATION REVIEWED

50.59 Documents

Modification Control Manuals

MCM-4.1 Rev 0	10 CFR 50.59 Screen
MCM-4.2 Rev 0	10 CFR 50.59 Evaluation

10 CFR 50.59 Screens

OP-20 Rev 28	Standby Gas Treatment System
OP-24A Rev 36	Off Gas System
OP-48 Rev 24	Solid Radwaste System
JAF-RPT-SWS-04335	Deletion of IST Program Requirements for Valve Closure
JAF-CALC-CRC-04276	Max Allowable Air Handling Units Tube Plugging Limit
FSAR Section 13.2	Organizational Structure and Responsibilities
SP-03.05	Steam Jet Ejector and Recombiner Sampling and Analysis
ACT-99-43527	JAF 345 KV Bus Operation in FSAR Section 8.6.6b
DES Change JD-01-031	Condensate Pump Materials Documentation
DCR-01-157	20 TK -159 Drawing Discrepancies Documentation
NuAP 5.6	Void NuAP 5.6 - Fire Protection Program FSAR Change
FSAR Change 01-019	Clarification of Reactor Seal Pump Type

10 CFR 50.59 Nuclear Safety Evaluations

JAF-SE-97-005, Rev 2	Feedwater Flow Ultrasonic Monitoring System (LEFM)
JAF-SE-98-013, Rev 3	RHR and Core Spray Suppression Pool Suction Strainer
JAF-SE-98-025, Rev 2	HPCI and Rx Core Isolation Cooling Suction Strainer
JAF-SE-98-039, Rev 0	Reactor Building Closed Loop Water Pump Motor Speed
JAF-SE-99-022, Rev 0	RHR Min Flow Bypass Valve Trip Setpoint Changes
JAF-SE-99-023, Rev 0	Reactor Vessel Spray Connection Mod
JAF-SE-00-023, Rev 0	ASTM E446 Acceptability for Radiography of Castings
JAF-SE-00-028, Rev 0	Alternate Feedwater Temp Inputs to Heat Balance Calc.
JAF-SE-01-012, Rev 0	FSAR change to EDG air start system description

Station Procedures

OP-13C, Revision 3	Residual Heat Removal (RHR) Service Water
OP-13, Revision 88	Residual Heat Removal System
EP-7, Revision 3	Primary Containment Flooding
EP-8, Revision 2	Alternate Injection Systems
EP-10, Revision 1	Fire Water Crosstie To RHR Service Water Loop A When Directed By EOP-4 or SAOGs
EOP-4, Revision 6	Primary Containment Control
EOP-7, Revision 6	Reactor Pressure Vessel Flooding
ISP-32, Revision 9	RHR & RHR Service Water Flow Loop A and B Calibration (IST)
ST-43I, Revision 6	Remote Shutdown Instrument Check
ST-2AL, Revision 15	RHR Loop A Quarterly Operability Test
ST-2AM, Revision 14	RHR Loop B Quarterly Operability Test
AOP-18, Revision 10	Loss Of 10500 Bus
AOP-19, Revision 19	Loss Of 10600 Bus
AOP-30, Revision 13	Loss Of Shutdown Cooling
AOP-28, Revision 11	Operation During Plant Fires
AOP-43, Revision 25	Plant Shutdown From Outside The Control Room
AOP-53,Revision 7	Loss Of Spent Fuel Storage Pool, Reactor Head Cavity well, Or Dryer Separator Storage Pit Water Level
AOP-55, Revision 5	Alternate Shutdown Cooling Due To Plant Fires

DERs

99-01246	00-04036	01-01503
99-02877	00-04754	01-01576
00-00159	00-04824	01-02508
00-01306	00-05230	01-02985
00-01308	00-05396	01-03476
00-01764	01-00106	

Tests

ST-2Y	RHR Heat Exchanger Performance Test, 10-7-00
ST-2Y	RHR Heat Exchanger Performance Test, 10-8-00
ISP-32	RHR and RHRSW Flow Loop A&B Calibration
ST-2XA	RHRSW Loop A Quarterly Operability Test
ST-2XB	RHRSW Loop B Quarterly Operability Test

Drawings

FM-20A Flow Diagram RHR
RHR Heat Exchangers TEMA Data Sheet 4.12-4
RHR Heat Exchangers TEMA Data Sheet 4.12-3
RHR Heat Exchangers TEMA Data Sheet 4.12-5

Calculations

JAF-CALC-RHR-02953	RHR heat exchanger K-value with reduced tube side FF
5012-006	Loop Uncertainty Calculation For RHR HX Outlet Temperature
5012-005	Loop Uncertainty Calculation For RHRSW Loops
10-13	RHRSW Pumps Required Head
JAF-CALC-RHR-02578	NPSH Calculation
JAF-CALC-RHR-01903	Instrument Uncertainty For RHR HX Performance Test

Miscellaneous

EOP/EP Non-Licensed Operator Qualification Standard Tasks
EP-7, 5.7 NLO-5EOP.205049 Fire Water Crosstie Injection

Heat Sink Performance, SR# 2236