January 28, 2002

Mr. Robert M. Bellamy
Site Vice President
Entergy Nuclear Generation Company
Pilgrim Nuclear Power Station
600 Rocky Hill Road
Plymouth, Massachusetts 02360-5599

SUBJECT: PILGRIM NUCLEAR POWER STATION - NRC INSPECTION REPORT 50-293/01-08

Dear Mr. Bellamy:

On December 29, 2001, the NRC completed an inspection at your Pilgrim reactor facility. The enclosed report documents the inspection findings which were discussed on January 17, 2002, with Mr. P. Dietrich 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, the inspectors identified a finding describing an adverse performance trend related to the cross-cutting issue of human performance, and one issue of very low safety significance (Green). This latter issue was determined to involve a violation of NRC requirements. However, because of it's safety significance and because the issue was entered into your corrective action program, the NRC is treating the issue as a Non-Cited Violation, in accordance with Section VI.A.1 of the NRC's Enforcement Policy, issued May 1, 2000, (65FR25368). If you deny this NCV, 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 1; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-001; and the NRC Resident Inspector at the Pilgrim facility.

Immediately following the terrorist attacks on the World Trade Center and the Pentagon, the NRC issued an advisory recommending that nuclear power plant licensees go to the highest level of security, and all promptly did so. With continued uncertainty about the possibility of additional terrorist activities, the Nation's nuclear power plants remain at the highest level of security and the NRC continues to monitor the situation. This advisory was followed by additional advisories, and although the specific actions are not releasable to the public, they generally include increased patrols, augmented security forces and capabilities, additional security posts, heightened coordination with law enforcement and military authorities, and more limited access of personnel and vehicles to the sites. The NRC has conducted various audits of your response to these advisories and your ability to respond to terrorist attacks with the capabilities of the current design basis threat (DBT). From these audits, the NRC has concluded that your security program is adequate at this time.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosures 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/

Clifford Anderson, Chief
Projects Branch 5
Division of Reactor Projects

Docket No. 50-293
License No. DPR-35

Enclosure: Inspection Report 50-293/01-08

Attachment: Supplemental Information

cc w/encl:
M. Krupa, Director, Nuclear Safety & Licensing
W. Riggs, Director, Nuclear Assessment Group
D. Tarantino, Nuclear Information Manager
B. Ford, Regulatory Affairs Department Manager
J. Fulton, Assistant General Counsel
R. Hallisey, Department of Public Health, Commonwealth of Massachusetts
The Honorable Therese Murray
The Honorable Vincent deMacedo
Chairman, Plymouth Board of Selectmen
Chairman, Duxbury Board of Selectmen
Chairman, Nuclear Matters Committee
Plymouth Civil Defense Director
D. O'Connor, Massachusetts Secretary of Energy Resources
J. Miller, Senior Issues Manager
Office of the Commissioner, Massachusetts Department of Environmental Quality Engineering
Office of the Attorney General, Commonwealth of Massachusetts
Chairman, Citizens Urging Responsible Energy
S. McGrail, Director, Commonwealth of Massachusetts, SLO Designee
Electric Power Division
J. Perlov, Secretary at the Executive Office of Public Safety
R. Shadis, New England Coalition Staff
Distribution w/encl (**VIA E-MAIL**):
H. Miller, RA
J. Wiggins, DRA

T. Bergman, RI EDO Coordinator
C. Anderson, DRP
R. Summers, DRP
P. Bonnett, DRP
E. Adensam, NRR (ridsnrrdlpmlpdi)
D. Starkey, PM, NRR
R. Pulsifer, Backup PM, NRR
R. Laura, SRI - Pilgrim
R. Junod, DRP
Region I Docket Room (with concurrences)

DOCUMENT NAME: G:\BRANCH5\2-PG\INSPECT\PIL0108.WPD
After declaring this document "An Official Agency Record" it **will/will not** be released to the Public. **To receive a copy of this document, indicate in the box:** "**C**" = Copy without attachment/enclosure "**E**" = Copy with attachment/enclosure "**N**" = No copy

OFFICE	RI/DRP		RI/DRP	
NAME	RLaura/CA for		CAnderson/CA	
DATE	01/28/02		01/28/02	

OFFICIAL RECORD COPY

U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No:	50-293
License No:	DPR-35
Report No:	50-293/01-08
Licensee:	Entergy Nuclear Generation Company
Facility:	Pilgrim Nuclear Power Station
Location:	600 Rocky Hill Road Plymouth, MA 02360
Dates:	November 18, 2001, through December 29, 2001
Inspectors:	R. Laura, Senior Resident Inspector R. Arrighi, Resident Inspector J. Furia, Senior Health Physicist
Approved By:	Clifford Anderson, Chief Projects Branch 5 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000293-01-08; on 11/18 - 12/29/2001; Entergy Nuclear Generation Company; Pilgrim Nuclear Power Station, Personnel Performance During Non-routine Plant Evolutions, Human Performance issues.

The inspection was conducted by resident inspectors and a health physicist. The inspection identified one Green finding, which was a Non-Cited Violation and one finding, which was characterized as no color. 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

Cornerstone: Mitigating Systems

- GREEN. The inspector identified a non-cited violation for inadequate corrective actions associated with the December 27, 2001, reactor vessel water level instrumentation spiking. To date, reactor vessel level spiking has been experienced on two other occasions since the April 2001 refueling outage (April 21 and August 13, 2001).

  The finding was of very low safety significance. The significance was determined by comparing it to a Phase 3 SDP risk evaluation that was conducted for the April 13, 2001, reactor vessel water level notching event (reference NRC Inspection Report 50-293/01-06). The April 2001 spiking event affected both channels of reactor vessel level instrumentation and was concluded to be of very low safety significance (Green). Since this event only affected the "B" level instrumentation, this condition is also determined to be of very low safety significance. The inspectors determined that this issue involved a human performance causal factor since the instrument rack purge location was not properly identified, which could have resulted in entrapped gasses. (Section 1R14)

Cornerstone: Cross-cutting Issues: Human Performance

- No Color. The inspector noted development of an apparent trend in human performance, attributed mostly with engineering involvement related to procedure development, performance, and corrective action implementation that resulted in findings in multiple cornerstones. The following human performance deficiencies have occurred within the past 12 months:

  1. In February 2001, engineering personnel failed to ensure prompt corrective actions to resolve continued problems with the 125 VDC swing bus automatic transfer switch, Y-10 relay, that had the potential to render the low pressure coolant injection (LPCI) system inoperable (NCV 50-293/00-11-01);

2. In March 2001, the licensee failed to identify that certain relays in the 480 volt emergency load center transfer scheme (B-6) did not meet procurement specifications that could result in LPCI function being lost under certain conditions (NCV 50-293/01-02-01);

3. In May 2001, engineering personnel failed to control testing on a reactor recirculation system sample valve that prompted the need for a manual reactor scram during plant startup from the cycle 13 refueling outage (NCV 50-293/01-03-05);

4. In August 2001, the licensees failure to adequately develop a logic system functional test procedure for the A5 electrical emergency bus resulted in the loss of both reactor recirculation pumps and a reactor scram (NCV 50-293/01-05-03);

5. In December 2001, licensee radiation protection personnel failed to properly post a high radiation area boundary. (Section 4OA7);

6. In December 27, 2001, the NRC identified ineffective corrective actions associated with reactor vessel water level instrumentation spiking involving an inadequate instrument rack purge. (Section 1R14) (NCV 50-293/01-08-01).

These individual issues have a related cause in that they represent human performance errors. They also have a direct impact on safety, increase the frequency of initiating events and affect the reliability, operability and functionality of mitigating equipment. This performance trend is considered a cross-cutting issue and is a finding characterized as "no color." (Section 4OA4)

B. Licensee Identified Findings

- Violations of very low safety significance which were identified by the licensee have been reviewed by the inspectors. Corrective actions taken or planned by the licensee appear reasonable. (Section 4OA7)

Report Details

SUMMARY OF PLANT STATUS

Pilgrim Nuclear Power Station began the period at 100 percent core thermal power. Power was reduced to 97% on November 28 and 30 for feedwater level control and control rod drive (CRD) 06-11 testing respectively. On December 13, 2001, power was reduced to 55 percent to perform a thermal backwash of the main condenser. The unit returned to 100 percent power on December 16, 2001. On December 27, 2001, the unit automatically scrammed from 100 percent power during anticipated transient without scram (ATWS) testing due to the calibration unit malfunctioning. This caused both reactor recirculation pump drive motor breakers to trip and resulted in a flow bias scram. The unit was brought to cold shutdown to restore reactor recirculation pump flow. At the end of the period, the mode switch was in shutdown with the plant in a cold shutdown condition.

1. REACTOR SAFETY
(Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity)

1R01 Adverse Weather

a. Inspection Scope

A review was performed to determine the plant readiness for adverse weather due to winds and rough sea conditions that occurred between November 18, 2001, through December 29, 2001. The inspector reviewed Entergy procedure 2.1.37, "Coastal Storm," and 5.2.2, "High Winds (Hurricanes)," for site preparations for adverse weather. The inspector toured the intake structure, SBO diesel, and the protected area to verify adequate protections for adverse weather.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

a. Inspection Scope

The inspector conducted a partial system walk down of the reactor building closed cooling water and low pressure coolant injection systems. This included reviewing applicable plant and information drawings, and normal operating procedures. The inspector reviewed valve static mimics in the control room and walked down accessible portions of the system to ensure proper system alignment. The inspector confirmed that the system was properly aligned to support normal and emergency plant operations.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

a. Inspection Scope

The inspector reviewed the performance of an operating crew in the simulator on December 18, 2001. The scenario involved a failed mechanical seal with high leakage on the "B" reactor recirculation pump. Operators showed proficiency using EOP-1, "Reactor Pressure Vessel Control," and EOP-2, "Primary Containment Control". The inspector verified proper use of the Emergency Plan and also confirmed that the post scenario critique discussed any relevant lessons learned. There was an extra focus on procedural usage by the operating crew personnel.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule

a. Inspection Scope

The inspector reviewed problems involving selected in-scope systems, structures, and components (SSCs) to assess the effectiveness of the maintenance rule (10 CFR 50.65) program. The review focused on proper characterization of failed SSCs as related to the following:

- Proper classification of equipment failures for the post accident sample system (PASS). The inspector reviewed problem reports (PR) issued within the last two years and reviewed the PASS maintenance rule basis document. Problem reports reviewed included PR 00.1006 (unable to draw PASS sample).

- Proper classification of equipment failures for the reactor building closed cooling water (RBCCW) system. The inspector reviewed problem reports issued within the last two years and reviewed the RBCCW maintenance rule basis document. Problem reports reviewed included PR 00.9031 (low pressure alarm for "B" RBCCW) and 01.5063 (RBCCW pump "F" running hot to touch).

- Proper classification of equipment failures for the core spray (CS) system within the last twelve months.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

The inspector reviewed the following work activities to evaluate the licensee’s risk assessment process. The inspector reviewed work plans and packages against the criteria contained in procedures 1.5.21, “Integrated Scheduling Activities and Guidelines,” and 1.5.22, “Risk Assessment Process.” The inspection included a review of the risk assessments and contingencies that were established, and verification that the effect on plant risk and protected equipment was discussed during briefings and shift turnovers.

- Replacement of the anticipated transient without scram (ATWS) calibration unit.

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Non-routine Plant Evolutions

a. Inspection Scope

The inspector reviewed the integrated plant equipment and human performance following an automatic reactor scram that occurred on December 27, 2001. The unit automatically scrammed from 100 percent power during anticipated transient without scram (ATWS) testing due to the calibration unit malfunctioning. The calibration unit caused a false simulated low-low reactor vessel water level to be sensed on two ATWS trip channels, resulting in activation of the trip logic, which caused both reactor recirculation pump drive motor breakers to trip and the resultant flow bias scram. During the plant cool down and subsequent restart of the “A” reactor recirculation pump, the plant experienced reactor vessel level spiking (notching) on the “B” train reactor water level indicators.

The licensee initiated a post trip review per procedure 1.3.37 and conducted an investigation to identify the root cause and necessary corrective actions prior to plant restart. The inspector attended the post trip review debrief, interviewed the I&C technicians that performed the ATWS surveillance, and reviewed the post trip review report. The inspector also discussed with plant engineering the cause of the vessel level notching.

b. Findings

Green. The inspectors identified a non-cited violation of 10 CFR Part 50 Appendix B, Criterion XVI, for inadequate corrective actions for the failure to resolve reactor vessel water level notching. Required back flush activities for the instrument racks following an August 2001 scram event were not implemented appropriately.

Operators responded in accordance with plant procedures to stabilize plant conditions and place the unit in cold shutdown. No significant human performance issues were identified with operator response to the event or I&C technician performance during the

ATWS surveillance. Reactor vessel level notching was experienced during the plant cool down and again when the “A” reactor recirculation pump was started.

The “B” reactor vessel water level instruments experienced level notching of a two inch magnitude during the plant cool down at a frequency of 6 per hour. Upon restart of the “A” reactor recirculation pump, a rapid pressure decrease resulted in notching of a 20 inch magnitude on the “B” water level instrumentation, generating a Group 1 isolation. No notching was experienced on the “A” level instrumentation. The Group 1 isolation signal did not result in any equipment changing state, other than the PCIS logic, since the main steam isolation valves were already closed.

The licensee investigation identified air entrapped in instrument rack C2206. The investigation of the source of air was still in progress at the close of this inspection period. A contributing cause of the event involved the I&C back flush procedure that allowed the instrument rack C2206 to be purged from a point which did not include the reference leg header. Although the instrument rack was back flushed following the August 13, 2001, reactor scram as part of the corrective actions taken in response to that event, it was not flushed from the appropriate location (the system low point). Failure to effectively implement this corrective action is a violation of 10 CFR Part 50 Appendix B Criterion XVI. Also, another contributor was a design that included a negative slope of the reference leg header on the C2206 rack, which could trap any air that is introduced. This condition had been previously identified and the licensee had developed a plant design change (PDC) in 1992 to change the slope of the instrument racks. However, the licensee subsequently determined to not implement this PDC based on a cost benefit analysis.

This finding was greater than minor because it could have an actual impact on safety in that false level indication during severe transients or a design basis accident could prevent automatic safety system functions or cause operators to take inappropriate actions. Therefore, this deficiency directly affects the Mitigation System cornerstone of the NRC significance determination process in that it could adversely affect the availability or reliability of core decay heat removal systems.

This condition was evaluated under the Phase 1 of the NRC’s significance determination process (SDP). Under mitigation systems, the condition was confirmed not to result in a loss of safety function per Generic Letter 91-18. Although automatic trip functions were unaffected, the system’s indication function was impaired. Since Phase 2 of the SDP does not readily apply to the condition, the safety significance of the condition was determined by comparing it to a similar Phase 3 SDP risk evaluation that was conducted for the April 13, 2001, reactor vessel water level notching event (reference NRC Inspection Report 50-293/01-06). The April 2001 notching event affected both channels of reactor vessel level instrumentation and was concluded to be of very low safety significance (Green). Since this event only affected the “B” level instrumentation, this condition is also determined to be of very low safety significance.

Reactor vessel level notching has been experienced on two other occasions since the April 2001 refueling outage (April 21 and August 13, 2001). To date, the licensee’s corrective actions have been ineffective in preventing recurrence of reactor vessel water level instrument spiking. As a result, the reliability of safety-related reactor water level instruments was degraded. The failure to ensure that conditions adverse to quality are identified and corrected is a violation of 10 CFR Part 50 Appendix B, Criterion XVI, "Corrective Actions." This violation is being treated as a Non-Cited Violation (**NCV 01-08-01**) consistent with Section VI.A.1. of the NRC Enforcement Policy (NUREG 1600). This issue is not considered to be a repetitive violation of the prior two level notching events in that it is not a condition that could have reasonably been prevented by the licensee's prior corrective actions. The cause of the level notching events do not involve the same set of conditions. The April 2001 event was attributed to inadequate venting after maintenance activities, and the August 2001 event was attributed to a design deficiency of the reference leg backfill system. This issue is documented in the licensee's corrective action program as PR01.8152.

Prior to plant restart after the December scram, the licensee had appropriately back flushed all reactor vessel water level instrument racks.

1R15 Operability Evaluations

a. Inspection Scope

The inspector reviewed the following operability evaluation to verify that continued operability was justified. The Pilgrim Updated Final Safety Evaluation Report (UFSAR), technical specifications, and licensee procedure 1.3.34.5, "Operability Evaluations," were used as references to assess the adequacy of the operability evaluations. The inspector also verified that the identified corrective actions to correct the degraded condition were adequate and scheduled in the licensee's work control process.

- OE 01-042, Drywell-to-torus vent line expansion bellows

b. Findings

No findings of significance were identified.

1R16 Operator Work-Arounds

a. Inspection Scope

The inspector reviewed the list of operator work-arounds, lifted lead and jumper log, and licensee procedure 1.3.34.4, "Compensatory Measures," for determining the aggregate effect of work-arounds on the operators ability to implement abnormal or emergency operating procedures. The inspector also verified that the licensee had entered the identified conditions in their corrective action program for resolution.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing

a. Inspection Scope

The inspector reviewed the following post-maintenance testing activities:

- MR 01110967, Vessel flange leak detection alarm, LS-261-19
- MR 01106122, Troubleshoot drywell floor drain sump flow controller alarm
- MR 01122921, Replace "B" emergency diesel generator governor
- MR 19700865, Replace reactor building closed cooling water flow transmitter, FT-6240, root valve
- MR 01119155, Tune feed water level master controller

The review included ensuring that the effect of the test on the plant had been evaluated adequately, verifying the test data meet the required acceptance criteria, and ensuring that the test activity was adequate to verify system operability and functional capability following maintenance.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

The inspector reviewed the results of the following surveillance tests:

- 8.5.3.2.1, "Salt Service Water Pump and Valve Operability Tests with Full Flow Test Conditions,"
- 8.5.3.1, "RBCCW Pump Operability and Flow Rate Test"
- 8.5.5.4, "RCIC Motor Operated Valve Quarterly Operability Test"

The inspector verified that the test acceptance criteria was consistent with technical specifications and Updated Final Safety Analysis Report requirements, that the test was performed in accordance with the written procedure, the test data was complete and meet procedural requirements, and that the system was properly returned to service following testing.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY
Cornerstone: Occupational Radiation Safety

2OS1 Access Control

a. Inspection Scope

The inspector identified exposure significant work areas (e.g., high radiation areas, and potential airborne radioactivity areas) in the turbine and reactor buildings, and reviewed associated controls and surveys of these areas to determine if the controls (i.e., radiological surveys, postings, barricades) were adequate to identify and control radiation exposures. Specific work activities observed included hydrolazing in the fuel pool skimmer overflow tanks and conduct of feedwater flow valve work with the reactor at power. For these areas, the inspector: reviewed radiological job requirements and attended job briefings; determined if radiological conditions in the work area were adequately communicated to workers through briefings and postings; verified the implementation of radiological job coverage and contamination controls; and verified the accuracy of surveys and applicable posting and barricade requirements. The inspector determined if prescribed radiation work permit (RWPs) controls were in-place, procedure and engineering controls were in place, whether licensee surveys and postings were complete and accurate, and whether air samplers were properly located. The inspector reviewed electronic pocket dosimeter alarm set points (both integrated dose and dose rate) for conformity with survey indications and plant policy. Plant technical specification (TS) 5.7 and the requirements contained in 10 CFR 20, Subpart G were utilized as the standard for necessary barriers.

The inspector also examined two Quality Assurance Surveillance Reports (QASR) [01-058 and 01-073] to evaluate the licensee's implementation of its program for self-evaluation, problem identification and resolution.

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Controls

a. Inspection Scope

The inspector reviewed ALARA job evaluations, reviewed exposure estimates and exposure mitigation requirements, and reviewed ALARA plans. The inspector conducted a review of: the integration of ALARA requirements into work procedures and RWP documents; the accuracy of person-hour estimates and person-hour tracking; and the generation of shielding requests including their effectiveness in dose rate reduction.

For the work areas identified in section 2OS1 (above), the inspector: evaluated the licensee's use of engineering controls to achieve dose reductions; determined if workers utilized the low dose waiting areas and were effective in maintaining their doses ALARA; determined if workers received appropriate on-the-job supervision to ensure ALARA requirements were met; and reviewed individual exposures of selected work groups.

The inspector conducted a review of actual exposure results versus initial exposure estimates including comparison of estimated and actual dose rates and person-hours expended; determination of the accuracy of estimations to actual results; and determination of the level of exposure tracking detail, exposure report timeliness and exposure report distribution. The review was against requirements contained in 10 CFR 20.1101(b).

b. Findings

No findings of significance were identified.

2OS3 Radiation Monitoring Instrumentation

a. Inspection Scope

The inspector reviewed field instrumentation utilized by health physics technicians and plant workers to measure radioactivity including; portable field survey instruments, friskers, portal monitors and small article monitors. The inspector conducted a review of instruments observed, specifically verification of proper function and certification of appropriate source checks for these instruments, which were utilized to ensure that occupational exposures were maintained in accordance with 10 CFR 20.1201.

The inspector examined a Quality Assurance Surveillance Report (QASR) 1-072 to evaluate the licensee's program for self-evaluation, problem identification and resolution.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES [OA]

4OA1 Performance Indicator Verification

a. Inspection Scope

Occupational Radiation Safety Cornerstone

The inspector reviewed a listing of licensee problem reports for the period April 1, 2001, through November 26, 2001, for issues related to the occupational radiation safety performance indicator. The information contained in these records was compared against the criteria contained in Nuclear Energy Institute (NEI) 99-02, Regulatory Assessment Performance Indicator Guideline, Revision 1, to verify that all conditions that met the NEI criteria were recognized, identified, and reported as a Performance Indicator.

Emergency AC Power System Unavailability

The inspector reviewed licensee event reports and NRC inspection reports for the period of January 2001 to December 2001 to determine the accuracy and completeness for the reported Pilgrim PI for emergency AC power system. The data was accurate and the indicator remained in the Green band for the specified period.

Residual Heat Removal (RHR) System Unavailability

The inspector reviewed licensee event reports and NRC inspection reports for the period of December 2000 to December 2001 to verify the accuracy and completeness for the reported Pilgrim performance indicator for RHR system unavailability. The data was accurate and the indicator remained in the Green band for the specified period.

Safety System Functional Failures

The inspector reviewed a sample of related problem reports, NRC inspection reports and licensee event reports for the period of December 2000 to December 2001 to verify the accuracy and completeness for the reported Pilgrim performance indicator for safety system functional failures. The data was found to be accurate and the indicator remained in the Green band.

b. Findings

No significant findings were identified.

4OA2 Identification and Resolution of Problems

A Non-Cited Violation of 10 CFR 50 Appendix B, part XVI, for inadequate corrective actions was issued for the failure to resolve reactor vessel level notching issues. (Section 71111.14)

4OA4 Cross -Cutting Issues

Human Performance Problems

a. Inspection Scope

The inspector reviewed human performance issues that have occurred at the site over the past 12 months.

c. Findings

No Color. A cross-cutting issue has been identified in the human performance area that affects multiple cornerstones.

Following the December 27, 2001, reactor scram, the “B” reactor vessel water level experienced vessel level “notching” during the plant cool down and depressurization; and upon starting of the “A” reactor recirculation pump. The licensee determined the root cause to be air entrapment within instrument rack C2206. This is the third occurrence of vessel level instrumentation spiking since the April 2001 refueling outage. The inspector determined that this issue involved a human performance causal factor since the instrument rack purge location was not properly identified contributing to air or passes being trapped in the rack.

In addition to this most recent finding the inspectors noted the following human performance deficiencies contributing to findings or violations in multiple cornerstones within the past 12 months:

1. In February 2001, engineering personnel failed to ensure prompt corrective actions to resolve continued problems with the 125 VDC swing bus automatic transfer switch, Y-10 relay, that had the potential to render the low pressure coolant injection (LPCI) system inoperable (NCV 50-293/00-11-01);

2. In March 2001, the licensee failed to identify that certain relays in the 480 volt emergency load center transfer scheme (B-6) did not meet procurement specifications that could result in LPCI function being lost under certain conditions (NCV 50-293/01-02-01);

3. In May 2001, engineering personnel failed to control testing on a reactor recirculation system sample valve that prompted the need for a manual reactor scram during plant startup from the cycle 13 refueling outage (NCV 50-293/01-03-05);

4. In August 2001, the licensee failed to adequately develop a logic system functional test procedure for the A5 electrical emergency bus, resulted in the loss of both reactor recirculation pumps and a reactor scram (NCV 50-293/01-05-03);

5. In December 2001, licensee radiation protection personnel failed to properly post a high radiation area boundary (Section 4OA7).

These individual issues have a related cause in that they represent human performance errors; attributed mostly with engineering involvement in procedure development, performance, and corrective action implementation. They also have a direct impact on safety, increase the frequency of initiating events and affect the reliability, operability and functionality of mitigating equipment. This performance trend is considered a cross-cutting issue. and is a finding **(FIN 50-293/2001-08-02)** characterized as “no color”.

4OA6 Management Meetings

Exit Meeting Summary

The inspectors presented the inspection results to Mr. P. Dietrich, General Manager Plant Operations, and other members of licensee management at the conclusion of the inspection on January 17, 2002. The licensee acknowledged the findings presented.

The inspectors asked the licensee whether any materials examined during the inspection should be considered propriety. No propriety information was identified.

4OA7 Licensee Identified Violations

The following finding of very low significance was identified by the licensee and is a violation of NRC requirements which meets the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600 for being dispositioned as Non-cited Violation (NCV).

NCV Tracking Number	Requirement Licensee Failed to Meet
NCV 05000293/2001-008-03	Plant Technical Specification 5.7 sets forth the requirements for the posting, barricading and control of access to high radiation areas (dose rates in excess of 100 millirem per hour measured at 30 centimeters from the source of radiation). A lack of the proper posting and barricade for a high radiation area located on the 91' elevation of the reactor building in the skimmer corridor was identified by the licensee. The licensee placed this issue into its corrective action program as problem report (PR) 01.8070.

ATTACHMENT

SUPPLEMENTAL INFORMATION

a. Key Points of Contact

M. Christopher, Radiation Protection Supervisor
J. Henderson, Radiation Protection Supervisor
W. Lobo, Licensing Engineer
W. Mauro, Radiation Protection Manager (Acting)
J. McClellan, Senior Engineer - Nuclear
T. Tetzlaff, Radiation Protection Supervisor - Operations

b. List of Items Opened, Closed and Discussed

NCV 293-010008-01	Reactor vessel water level notching
NCV 293-010008-03	Non-posted high radiation area

c. List of Documents Reviewed

d. List of Acronyms

ALARA	As Low As is Reasonably Achievable
ATWS	Anticipated Transient Without Scram
CFR	Code of Federal Regulations
CRD	Control Rod Drive
CS	Core Spray
DBT	Design Basis Threat
EOP	Emergency Operating Procedure
I&C	Instrumentation and Controls
LPCI	Low Pressure Coolant Injection
MR	Maintenance Request
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
PASS	Post Accident Sampling System
PR	Problem Report
QASR	Quality Assurance Surveillance Report
RBCCW	Reactor Building Closed Cooling Water
RHR	Residual Heat Removal
RWP	Radiation Work Permit
SBO	Station Black Out
SDP	Significance Determination Process
SSC	Structures Systems and Components
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report