UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
ARLINGTON, TEXAS 76011-8064

March 17, 2000

Mr. J. V. Parrish (Mail Drop 1023)
Chief Executive Officer
Energy Northwest
P.O. Box 968
Richland, Washington 99352-0968

SUBJECT: NRC INSPECTION REPORT NO. 50-397/00-04

Dear Mr. Parrish:

This refers to the inspection conducted on January 9 through February 19, 2000, at the WNP-2 facility. The enclosed report presents the results of this inspection.

Based on the results of this inspection, the NRC has determined that two Severity Level IV violations of NRC requirements occurred. The violations are being treated as noncited violations, consistent with Section VII.B.1.a of the Enforcement Policy. The noncited violations are described in the subject inspection report. If you contest the violations or severity level of the noncited violations, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, 611 Ryan Plaza Drive, Suite 400, Arlington, Texas 76011; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the WNP-2 facility.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response, if requested, will be placed in the NRC Public Document Room.

Should you have any questions concerning this inspection, we will be pleased to discuss them with you.

Sincerely,

/RA/

Linda Joy Smith, Chief
Project Branch E
Division of Reactor Projects

Docket No.: 50-397
License No.: NPF-21

Enclosure:
NRC Inspection Report No. 50-397/00-04

cc w/enclosure:
Chairman
Energy Facility Site Evaluation Council
P.O. Box 43172
Olympia, Washington 98504-3172

Rodney L. Webring (Mail Drop PE08)
Vice President, Operations Support/PIO
Energy Northwest
P.O. Box 968
Richland, Washington 99352-0968

Greg O. Smith (Mail Drop 927M)
Vice President, Generation
Energy Northwest
P.O. Box 968
Richland, Washington 99352-0968

D. W. Coleman (Mail Drop PE20)
Manager, Regulatory Affairs
Energy Northwest
P.O. Box 968
Richland, Washington 99352-0968

Albert E. Mouncer (Mail Drop 1396)
General Counsel
Energy Northwest
P.O. Box 968
Richland, Washington 99352-0968

Paul Inserra (Mail Drop PE20)
Manager, Licensing
Energy Northwest
P.O. Box 968
Richland, Washington 99352-0968

Thomas C. Poindexter, Esq.
Winston & Strawn
1400 L Street, N.W.
Washington, D.C. 20005-3502

Bob Nichols
State Liaison Officer
Executive Policy Division
Office of the Governor
P.O. Box 43113
Olympia, Washington 98504-3113

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ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket No.:	50-397
License No.:	NPF-21
Report No.:	50-397/00-04
Licensee:	Energy Northwest
Facility:	WNP-2
Location:	Richland, Washington
Dates:	January 9 through February 19, 2000
Inspectors:	G. D. Replogle, Senior Resident Inspector J. P. Rodriguez, Resident Inspector J. F. Melfi, Project Engineer
Approved By:	Linda Joy Smith, Chief, Project Branch E, Division of Reactor Projects

ATTACHMENT: Supplemental Information

EXECUTIVE SUMMARY

WNP-2
NRC Inspection Report No. 50-397/00-04

This information covers a 6-week period of resident inspection.

Operations

- The conduct of operations was professional and safety conscious. Operators were consistently knowledgeable of important plant issues and properly anticipated plant operations. Equipment was properly aligned (Sections O1.1 and O2.1).

Maintenance

- In general, most maintenance and surveillance activities were performed in a thorough and effective manner. In particular, the Division II, 125 Vdc battery charger work was exceptionally well planned and executed. The work involved entering a 2-hour Technical Specification shutdown action statement. Management involvement and oversight were excellent (Sections M1.1 and M1.2).

Engineering

- The inspectors identified a vulnerability with the use of the reactor core isolation cooling system during a station blackout event. The system is risk significant for station blackout and is vulnerable to repetitive water hammers, which may challenge system operability, the integrity of the reactor coolant pressure boundary, or the integrity of the primary containment. The system keep-fill pump fails during a station blackout, and cycling of the primary pump, as designed, likely causes repetitive water hammer. Also, the Individual Plant Evaluation did not consider the potential challenges to system operability under station blackout conditions. This is an unresolved item pending further NRC review of the risk implications of the current reactor core isolation cooling system design, after considering the results of the planned water hammer analysis (Section E2.1).

- The inspectors identified a 10 CFR 50.55a violation in that certain valves in containment bypass lines had specific leakage limits but were not being leak-rate tested. 10 CFR 50.55a requires that safety-related valves be tested in accordance with the ASME Code. The Code requires leak-rate testing for valves that are assigned specified leakage limits. This Severity Level IV violation is being treated as a noncited violation, consistent with Section VII.B.1.a of the NRC Enforcement Policy. The problem is in the licensee's corrective action program as Problem Evaluation Request 298-0928 (Section E8.1).

Plant Support

- The inspectors identified a violation of Technical Specification 5.7.1.b, which requires radiation work permit controls for work in high radiation areas. Limit switch work for reactor water cleanup system Valve RWCU-V-437A was inadvertently performed on Valve RWCU-V-433. As a result, maintenance craftsmen worked on a valve that was not covered by a radiation work permit. The work did not adversely affect plant operations; however, workers received approximately 100 millirem of additional dose. Several departments failed to properly communicate. This Severity Level IV violation is being treated as a noncited violation, consistent with Section VII.B.1.a of the NRC Enforcement Policy. The problem is in the licensee's corrective action program as Problem Evaluation Request 200-0109 (Section R1.1).

- Emergency preparedness facilities were properly maintained and on-shift staffing was consistent with the Emergency Plan (Section P2.1).

- Protected area illumination levels, maintenance of the isolation zones around protective area barriers, and security power supply equipment were properly maintained (Section S2.1).

Report Details

Summary of Plant Status

At the beginning of the inspection period, the plant operated at 100 percent power where it remained for most of the inspection interval. Power was reduced briefly on January 22, 2000, to 75 percent and on February 19 to 85 percent to establish a new control rod pattern, perform surveillance testing, and accomplish selected maintenance tasks in high radiation areas.

I. Operations

O1 Conduct of Operations

O1.1 General Comments (71707)

Operators were knowledgeable of important plant parameters and problems and were appropriately focused on safety.

O2 Operational Status of Facilities and Equipment

O2.1 Engineered Safety Feature System Walkdowns

a. Inspection Scope (71707)

The inspectors walked down accessible portions of the following safety-related systems:

- High pressure core spray
- Low pressure core spray
- Residual heat removal, Trains A, B, and C
- Reactor core isolation cooling (RCIC)
- Divisions I, II, and III emergency diesel generators
- Standby liquid control system
- Standby gas treatment system, Trains A and B

b. Observations and Findings

The inspectors found the systems properly aligned for the plant conditions and generally in good material condition.

II. MAINTENANCE

M1 Conduct of Maintenance

M1.1 General Comments - Maintenance

a. Inspection Scope (61726, 62707)

The inspectors inspected the following maintenance activities:

- Work Order 01005292, Division II 125 Vdc Charger Repair
- Work Order RGV-02, Valve RWCU-V-437A, Limit Switch Adjustments (event-related review)
- Work Order 29008391, Division II 125 Vdc Battery Cell Replacement
- Surveillance ESP-B12-Q101, "Quarterly Battery Testing 125 VDC E-B1-2"
- Work Order 01007866, RCIC Keep-fill Pump Repair

b. Observations and Findings

Maintenance and surveillances were generally conducted in a thorough and professional manner. Good performance associated with 125 Vdc charger work is discussed in Section M1.2. Problems with craftsmen working on the wrong reactor water cleanup system valve are discussed in Section R1.1.

M1.2 Division II, 125 Vdc Charger Work

The Division II, 125 Vdc charger has operated erratically and the reliability of the unit was questionable. For example, on December 29 and 30, 1999, at several different times the charger amps inexplicably increased from 90 to 130 amps and then decreased to about 30 amps before returning to normal. The licensee considered the erratic charger operation a significant concern because charger failure would place the plant in a 2-hour Technical Specification shutdown action statement. Following the 2 hours, operators would have an additional 12 hours to shut down the plant. Accordingly, the inspectors determined that on-line troubleshooting and repairs were justified.

The maintenance was well planned. Prior to work, electricians performed the work steps on a charger mockup. During the job, several components were checked and replaced, and work proceeded in an uneventful manner. However, the cause of the erratic operation was not conclusively determined. The job took approximately 6 hours, as expected, and a plant shutdown was not initiated. Planning and management oversight were considered excellent and appropriate to the circumstances. The performance of the charger was stable following the maintenance.

c. Conclusions

The Division II, 125 Vdc battery charger work was exceptionally well planned and executed. The work involved entering a 2-hour Technical Specification shutdown action statement. Management involvement and oversight were excellent.

III. ENGINEERING

E2 Engineering Support of Facilities and Equipment

E2.1 RCIC System Contribution to Station Blackout

a. Inspection Scope (37551)

The inspectors reviewed the licensing and design requirements associated with the RCIC system during a station blackout event.

b. Observations and Findings

Background: The RCIC system is a safety-related system that was credited for the control rod drop accident and is relied upon during a station blackout (complete loss of ac power).

A station blackout is the most risk-significant event at WNP-2 and the RCIC system is risk-important to the plant response. The risk achievement worth of the RCIC pump at WNP-2 is in the range of 2 to 3 and the Individual Plant Evaluation states that the baseline core damage frequency is about 1.7E-05/yr. Using these numbers, the increase in core damage frequency because of guaranteed RCIC failure during station blackout would be about 2.0E-05 to 3.0E-05 per year. The risk is relatively high because only two systems (RCIC and high pressure core spray) are available to mitigate the event.

System injection is controlled by starting and stopping the primary system pump. In automatic, the RCIC pump stops at reactor vessel Level 8 and restarts at Level 2. The barometric condenser is an atmospheric condenser that receives RCIC turbine drains and gland seal steam leak-off and main steam valve stem leak-off. During RCIC operation, some RCIC pump discharge flow is bypassed through the lube oil cooler and then provides cooling spray for the barometric condenser. The condensate is then directed to the RCIC condensate tank (a barometric condenser vacuum tank). The condensate is pumped to the suction of the RCIC pump or to an equipment drain sump. Under station blackout conditions, the automatic controls for the barometric condenser pump do not work, as they rely on ac power, so the pump would have to be manually operated or condensate would spill from a relief valve and onto the RCIC room floor.

As documented in NRC Inspection Report 50-397/99-13, the inspectors had identified that a RCIC water hammer analysis, utilized to justify the nonsafety status of the RCIC keep-fill pump, was inadequate. Accordingly, a potential water hammer on the RCIC system was not analyzed.

On February 2, the licensee reported, per 10 CFR 50.72 (Event 36653), that the plant was operated outside the design basis. Specifically, the licensee reported that a random failure of the keep-fill pump at any time during RCIC operation could result in an unanalyzed water hammer on the RCIC system, RCIC containment isolation valves, and RCIC piping that is part of the reactor coolant pressure boundary. The licensee further indicated that the problem was not significant because operators were alerted to the condition by a control room alarm and had sufficient time to secure RCIC. Subsequent to a loss-of-fill condition, the system would only be used, if needed, while implementing emergency operating procedures. The report did not address, however, the guaranteed loss of the keep-fill pump during station blackout conditions.

Vulnerability: The inspectors identified a vulnerability with the use of the RCIC system during a station blackout event. The system keep-fill pump relies on Division I ac power and fails during a station blackout. When the RCIC pump stops, the RCIC injection line drains to the barometric condenser at a rate of approximately 25 gpm. This results in a loss of fill condition in the RCIC injection line and any subsequent start of the RCIC pump would likely result in water hammer. Depending on the severity, the water hammers could breach the reactor coolant pressure boundary, render the RCIC system inoperable, or damage containment isolation valves.

In response to the inspectors' concern, the licensee noted that emergency operating procedures already require operators to take manual control of RCIC injection and prevent the system from tripping at Level 8. If the system did not have to be restarted, under station blackout conditions, a water hammer would be averted.

The inspectors checked historical documents and discussed RCIC operation with plant operators. The inspectors found that operators were not normally successful at controlling reactor level with RCIC and preventing the system from tripping at Level 8. For example, on September 18, 1999, during a normal plant shutdown, operators attempted to control vessel inventory with RCIC, but the system still tripped on Level 8. Additionally, following a March 11, 1998, scram with main steam isolation valve closure, the on-shift operator informed the inspectors that level control with RCIC was difficult and the system tripped on Level 8 approximately three times. Restart of the system at the time was not a problem because normal ac power was available and the keep-fill pump was operational.

In addition to the above, the inspectors determined that RCIC level control during a station blackout would be even more challenging for the following reasons:

- As discussed in NRC Inspection Report 50-397/98-11, Section 2.b.3, the RCIC condensate tank level switch also receives Division I AC power and, therefore, fails during a station blackout. Consequently, operators are tasked with manually operating the barometric condenser pump to prevent flooding in the RCIC room. Procedures instruct operators to run the pump for 4 minutes, 30 seconds and secure the pump for 30 seconds, during each 5-minute interval. This work-around would divert the operator's attention from reactor vessel level control.

- During a station blackout, the main steam isolation valves close, and operators manually control pressure by operating safety relief valves. This task diverts the operators’ attention from reactor vessel level control. Additionally, the reactor vessel level swells approximately 20 inches when a safety relief valve is opened, which makes precise level control with RCIC more difficult.

- Station blackout is a very challenging event, as most plant equipment fails from loss of power. Because of the event complexity, operators are not likely to attempt manual RCIC control for several minutes. During this time, while the RCIC system is operating in automatic, the system will likely trip at Level 8. For example, during the March 11, 1998, main steam isolation valve closure event, reactor water level reached Level 8, the high pressure core spray system tripped, and operators did not initially know that this had occurred.

Considering the preceding information, the inspectors concluded that there was not reasonable assurance that operators could successfully control water level with RCIC during a station blackout event. Accordingly, operators would likely be forced to secure RCIC and rely on high pressure core spray alone or initiate RCIC and bear the consequences of a system water hammer.

Risk Model Fidelity: The inspectors also observed that the WNP-2 Individual Plant Evaluation did not account for the noted RCIC operational problems. The RCIC system was modeled as if there were no potential RCIC operational challenges from the water hammer threat. The potential impact to the reactor coolant pressure boundary was also ignored. The inspectors determined that changes to the model were appropriate in order to more accurately reflect plant design and risk. The licensee agreed with the inspectors' observation and planned to make the necessary changes once the threat from water hammer was better understood. The planned actions were acceptable.

Station Blackout Licensing Basis: The WNP-2 Final Safety Analysis Report states, in part:

> The WNP-2 station blackout emergency response procedure provides guidance for responding to a station blackout including specific instructions for (1) providing for core cooling if HPCS and/or reactor core isolation cooling (RCIC) are available . . .
>
> The basis for the WNP-2 response to the station blackout rule is use of HPCS which is safety related and already adequately covered in TSs . . .

NOTE: The sole use of high pressure core spray for station blackout coping is inconsistent with the NRC safety evaluation on the issue.

In the WNP-2 station blackout safety evaluations, the Agency did not require that RCIC remain operable for the entire duration of a station blackout event. The NRC Safety Evaluation Report, dated December 30, 1991, stated, in part:

> The licensee, however, stated that both RCIC and HPCS pumps will be available to maintain the RCS [reactor coolant system] inventory, and the RCIC pump will not be shut down. It is the staff's understanding that the licensee will use RCIC until it fails due to high temperature (no other failure is assumed). Since HPCS can support the functions provided by the RCIC pump, the staff concludes that RCIC failure is of no concern.

NOTE: The NRC supplemental safety evaluation dated June 26, 1992, made similar statements to those identified above.

NRC Assessment: While the NRC did not require the licensee to maintain the RCIC system operable for the entire duration of a station blackout event, the NRC did not have an opportunity to quantitatively consider risk at the time the station blackout coping strategy was approved. Further, the vulnerability to water hammer was not known and the consequences of such an event on the RCIC system, the reactor coolant pressure boundary, and the primary containment are still not known.

At the close of the inspection, the licensee continued to analyze the consequences of a RCIC water hammer. Considering the risk importance of RCIC to station blackout, more review is warranted to understand the risk implications of this analysis and to determine whether a backfit should be considered in accordance with 10 CFR 50.109. The potential backfit may require the licensee to modify RCIC to remain functional and not vulnerable to water hammer during a station blackout. This is an unresolved item pending further NRC review of the risk implications of the current RCIC system design after considering the results of the planned water hammer analysis (URI 50-397/00004-01).

c. Conclusions

The inspectors identified a vulnerability with the use of the RCIC system during a station blackout event. The system is risk significant for station blackout and is vulnerable to repetitive water hammers, which may challenge system operability, the integrity of the reactor coolant pressure boundary, or the integrity of primary containment. The system keep-fill pump fails during a station blackout, and cycling of the primary pump, as designed, likely causes repetitive water hammer. Also, the Individual Plant Evaluation did not consider the potential challenges to system operability under station blackout conditions. This is an unresolved item pending further NRC review of the risk implications of the current RCIC system design, after considering the planned water hammer analysis.

E8 Miscellaneous Engineering Issues (92903)

E8.1 (Closed) Unresolved Item 50-397/98015-04: failure to test bypass leakage valves.

Inspectors identified that the licensee had specified valve-specific leakage criteria for 19 potential bypass leakage pathways; however, the licensee had not leak-rate tested the associated valves in accordance with 10 CFR 50.55a and the ASME Code. Further, the documented leakage limits were so small that it was not reasonable for the licensee to assume that leakage was below the specified limits without testing.

At the time of that inspection, the licensee stated that a new analysis, utilizing General Electric Report 22A5718, "Mark III Containment Dose Reduction Study," had demonstrated that permissible leakage past the valves was about 100 gpm and could easily be observed during normal operation. The report, in part, reduced the iodine release fraction substantially from that originally utilized in the NRC's Standard Review

Plan. Accordingly, the licensee had planned to change the licensing basis to delete the need for leakage testing. The inspectors referred the analysis to the Office of Nuclear Reactor Regulation for further consideration.

The Office of Nuclear Reactor Regulation found that the General Electric report did not specify how the General Electric iodine partition coefficients were derived. The partition coefficients were an important factor in decreasing the assumed offsite iodine release rate. Sheet 37 of the report identified that the values were taken from a phone conversation, which was cited as Reference 13. The Office of Nuclear Reactor Regulation concluded that the General Electric report was not acceptable, and, unless the licensee provides some other technically justifiable method for determining leakage limits, WNP-2 leakage limits should be determined by methods found in Standard Review Plan 15.6.5. This type of justification is not acceptable to the NRC.

In response to the NRC position, the licensee performed an operability determination and concluded that the valves remained operable. The licensee stated that most valve pathways were not of immediate concern because leakage past the valves is directed to sumps in the reactor building or the radwaste building. Each building is equipped with a charcoal filtration system to limit the release of fission products to the environment. For the remaining five pathways in the RCIC and high pressure core spray systems (each pathway has two closed valves), the licensee used the iodine partition coefficients specified in Standard Review Plan 15.6.5 and determined that the cumulative leakage limit for all the pathways totaled 2.4 gpm. A review of leakage test results for valves in a similar condition indicated that the expected leakage was about 0.2 gpm per valve, which, in aggregate, remained below the 2.4 gpm limit. Since each line contained two closed valves, the licensee believed that this provided reasonable confidence that leakage through any individual pathway was less than 0.2 gpm.

Additionally, the licensee referenced pressure decay data on the RCIC and high pressure core spray systems. The data indicated that the systems were reasonably leak-tight. Therefore, significant leakage across the boundary valves was not likely. The inspectors determined that the operability evaluation was acceptable and met the intent for an initial operability assessment per Generic Letter 91-18, "Information to Licensees Regarding Two NRC Inspection Manual Sections on Resolution of Degraded and Nonconforming Conditions and on Operability," Revision 1.

The failure to leak test the valves violated 10 CFR 50.55a. This requirement specifies, in part, that testing be performed in accordance with the ASME Code. The licensee was committed to ASME/ANSI Operations and Maintenance Standards, Part 10, "Inservice Testing of Valves in Light-Water Reactor Power Plants," OMa-1988 Addenda to Operations and Maintenance 1987 Edition. This Code specifies, in part, that Category A valves are those for which seat leakage is limited to a specific maximum amount in the closed position for fulfillment of their required functions. Additionally, the Code requires that these valves be seat-leakage tested to verify their leak-tightness integrity. Final Safety Analysis Report Section 6.2.3.2 specified that several lines penetrate outside of containment and certain valves in those lines were assigned leakage limits. Therefore, the valves were required to be classified as Category A valves and leak-rate tested. This Severity Level IV violation is being treated as a noncited violation, consistent with

Section VII.B.1.a of the NRC Enforcement Policy (50-397/00004-02). The problem is in the licensee's corrective action program as Problem Evaluation Request 298-0928.

E8.2 (Closed) Inspection Followup Item 50-397/98011-02: RCIC barometric condenser level control system design adequacy.

This issue was subsequently addressed in NRC Inspection Report 50-397/99-13, where a Notice of Violation was issued to address remaining problems (VIO 50-397/99013-01).

IV. Plant Support

R1 Radiological Protection and Chemistry Controls

R1.1 Craftsmen Get Extra Dose Working on the Wrong Reactor Water Cleanup Valve

a. Inspection Scope (62707)

On January 18, 2000, craftsmen worked on the wrong reactor water cleanup valve during maintenance. The inspectors investigated the event facts and consequences.

b. Observations and Findings

Backwash system Valve RWCU-V-437A had lost control room indication. On January 18, craftsmen attempted to adjust exterior limit switches to restore valve indication. The valve was located in a high radiation area.

While the craftsmen were supposed to adjust the limit switches on Valve RWCU-V-437A, they mistakenly adjusted the limit switches on Valve RWCU-V-433, which was abandoned in place and no longer utilized. The workers attempted to adjust the wrong limit switch components several times. The problem was identified when one of the craftsmen heard Valve RWCU-V-437A move during operational checks in a different location. The workers received approximately 100 millirem of additional exposure. The work did not cause any operational problems, as Valve RWCU-V-433 was not repositioned during the job.

Several departments were involved in the job preparation, including Operations and Radiation Protection. During the prejob brief, the workers were provided a picture of the wrong valve. A health physics technician initially identified the valve using the picture but did not check the tag. The craftsmen also failed to check the tag.

An Investigation Review Board was conducted and plant personnel were briefed on the event. Managers stressed self-checking techniques to all pertinent plant staff. The inspector found the licensee response to be acceptable.

The failure to work on the correct valve resulted in a Technical Specification 5.7.1.b violation. This Technical Specification requires radiation work permit controls for work in high radiation areas. Radiation Work Permit 30000058-02 was applicable to work on Valve RWCU-V-437A, not Valve RWCU-V-433. A radiation work permit was not generated for work on Valve RWCU-V-433. This Severity Level IV violation is being treated as a noncited violation, consistent with Section VII.B.1.a of the NRC Enforcement Policy (50-397/00004-03). The problem is in the licensee's corrective action program as Problem Evaluation Request 200-0109.

c. Conclusions

The inspectors identified a violation of Technical Specification 5.7.1.b, which requires radiation work permit controls for work in high radiation areas. Limit switch work for reactor water cleanup system Valve RWCU-V-437A was inadvertently performed on Valve RWCU-V-433. As a result, maintenance craftsmen worked on a valve that was not covered by an radiation work permit. The work did not adversely affect plant operations, but workers received approximately 100 millirem of additional dose. Several departments failed to properly communicate. This Severity Level IV violation is being treated as a noncited violation, consistent with Section VII.B.1.a of the NRC Enforcement Policy. The problem is in the licensee's corrective action program as Problem Evaluation Request 200-0109.

P2 Status of Emergency Preparedness Facilities, Equipment, and Resources

P2.1 General Comments (71750)

During routine plant tours, the inspectors verified that the emergency preparedness facilities were properly maintained and that the licensee maintained at least the minimum staffing required by their Emergency Plan. No problems were found.

S2 Status of Security Facilities and Equipment

S2.1 General Comments (71750)

During routine tours, the inspectors observed protected area illumination levels, maintenance of the isolation zones around protective area barriers, and the status of security power supply equipment. No problems were observed.

V. MANAGEMENT MEETINGS

X1 Exit Meeting Summary

The inspectors presented the inspection results to members of licensee management on February 17, 2000. The licensee acknowledged the findings presented. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

ATTACHMENT

SUPPLEMENTAL INFORMATION

PARTIAL LIST OF PERSONS CONTACTED

Licensee

J. V. Parrish, Chief Executive Officer
D. K. Atkinson, Engineering Manager
I. M. Borland, Radiation Protection Manager
S. A. Boynton, Quality Assurance Manager
J. W. Dabney, Outage Manager
P. J. Inserra, Licensing Manager
D. W. Martin, Security Manager
W. S. Oxenford, Operations Manager
D. J. Poirier, Maintenance Manager
G. O. Smith, Vice President - Generation/Nuclear Plant General Manager
R. L. Webring, Vice President - Operations Support

INSPECTION PROCEDURES USED

IP 37551:	Onsite Engineering
IP 61726:	Surveillance Observations
IP 62707:	Maintenance Observations
IP 71707:	Plant Operations
IP 71750:	Plant Support
IP 92903:	Engineering Followup

ITEMS OPENED AND CLOSED

Opened

50-397/00004-01	URI	Reactor core isolation cooling system vulnerability during station blackout (Section E2.1).

Opened and Closed

50-397/00004-02	NCV	Failure to local leak rate test containment bypass valves (Section E8.1).
50-397/00004-03	NCV	Failure to follow high radiation area radiation work permit (Section R1.1).

Closed

50-397/98015-04	URI	Failure to local leak rate test containment bypass valves (Section E8.1).
50-397/98011-02	IFI	Adequacy of the design of the RCIC barometric condenser level control (Section E8.2).

LIST OF ACRONYMS USED

ac	alternating current
ASME	American Society of Mechanical Engineers
ANSI	American Nuclear Standards Institute
CFR	Code of Federal Regulations
gpm	gallons per minute
IFI	Inspector Followup Item
NCV	noncited violation
NRC	U.S. Nuclear Regulatory Commission
RCIC	reactor core isolation cooling
URI	unresolved item
Vdc	volts, direct current