ENCLOSURE 2

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket No.:	50-397
License No.:	NPF-21
Report No.:	50-397/99-13
Licensee:	Energy Northwest
Facility:	WNP-2
Location:	Richland, Washington
Dates:	October 17 through November 27, 1999
Inspectors:	G. D. Replogle, Senior Resident Inspector G. A. Pick, Senior Project Engineer P. A. Goldberg, Reactor Inspector
Accompanied By:	L. M. Willoughby, 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. 50-397/99-13

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

Operations

- The conduct of operations was professional and safety conscious. The coordination and control of the plant startup was generally good. Operators were consistently knowledgeable of important plant issues and properly anticipated plant operations (Section O1.1).

Maintenance

- Maintenance and surveillance activities were conducted in a thorough and professional manner. Three maintenance and two surveillance activities were observed (Sections O2.1 and M1.1).

Engineering

- The licensee failed to restore compliance within a reasonable time frame after a violation of 10 CFR 50.59 was identified. In 1998, the NRC identified that the licensee inappropriately downgraded the reactor core isolation cooling (RCIC) system to nonsafety status via the 10 CFR 50.59 process. The licensee took corrective action to upgrade the system, but left some of the system in a nonsafety-status based on inadequate engineering analysis. Specifically, the RCIC system keepfill pump and barometric condenser level switch were maintained as nonsafety components based on inadequate water hammer and flooding calculations, respectively. This resulted in a continuing violation of 10 CFR 50.59. These inadequate corrective measures were cited as a violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action" (Section E1.1).

- The inspectors identified that reactor building flooding protection was not consistent with Final Safety Analysis Report commitments. The Final Safety Analysis Report states that all reactor building pump enclosure rooms are watertight, but the RCIC and the control rod drive pump rooms were connected via an unisolable equipment drain line. Further, this condition was not consistent with the licensee's flooding analysis, which assumed that flooding could not occur in both rooms simultaneously. The licensee maintained that even with the unisolable connection, safe shutdown could be achieved for design basis floods in these rooms. Pending further review of the licensee's revised flooding analysis and risk evaluation to determine the safety significance of the design deficiency, this is an unresolved item (Section E1.2).

- Thermolag-related corrective measures were completed consistent with commitments to the NRC (Section E8.2).

<u>Plant Support</u>

- During routine plant tours the inspectors verified that the emergency preparedness facilities were properly maintained and on-shift staffing was consistent with the Emergency Plan. No problems were found (Section P2.1).

- During routine tours, the inspector 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 (Section S2.1).

Report Details

Summary of Plant Status

At the beginning of the inspection period, the plant was shut down for Refueling Outage R14. On October 22, the plant entered Mode 2 and achieved criticality. The plant transitioned to Mode 1 and synchronized to the grid on October 24; on October 28, 100 percent power was achieved. The plant essentially maintained 100 percent power for the remainder of the inspection period.

I. Operations

O1 Conduct of Operations

O1.1 General Comments (71707)

The inspectors used Inspection Procedure 71707 to conduct frequent reviews of ongoing plant operations. Operators were generally knowledgeable of important plant parameters and problems and were appropriately focused on safety. The plant startup on October 22 was conducted in a thorough and methodical manner.

O2 Operational Status of Facilities and Equipment

O2.1 Engineered Safety Feature System Walkdowns

a. Inspection Scope (71707, 71750)

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
- RCIC
- Division I, II, and III emergency diesel generators
- Standby service water system, Trains A, B, and C

b. Observations and Findings

The systems were found to be properly aligned for the plant conditions and generally in good material condition. The 6-month detailed walkdown, specified by Inspection Procedure 71707, was performed this period on the RCIC system and is discussed in Section O2.2.

O2.2 RCIC System Walkdown

a. Inspection Scope (71707)

The inspectors verified the system lineup for the RCIC system in accordance with plant procedures. Additionally, the inspectors reviewed corrective maintenance performed on

RCIC system components and evaluated inservice test data on selected RCIC system valves and the pump. The RCIC system was one of the more risk-important systems at WNP-2.

b. Observations and Findings

During the system walkdown, the inspectors identified no concerns with the material condition of RCIC components and found excellent housekeeping in the RCIC pump room. The inspectors found a couple of valves with missing labels, and the licensee initiated label requests.

The inspectors reviewed Procedures OSP-RCIC/IST-Q701, "RCIC Operability Test," Revision 10, and OSP-RCIC/IST-Q702, "RCIC Valve Operability Test," Revision 6, for the pump and selected valves, respectively. The inspectors found that the licensee had established appropriate test acceptance criteria for the safety functions of the valves and the pump. The inspectors noted that the components had met all of their acceptance criteria. From review of the maintenance history, the inspectors identified no significant trend or component degradation.

c. Conclusions

The inspectors identified no concerns with the testing, maintenance, material condition, or standby lineup of the RCIC system.

O8 Miscellaneous Operations Issues (92901)

O8.1 (Closed) Inspection Followup Item 50-397/98005-06: weakness in operator requalification training program. This item was initiated to ensure NRC review of the discrepancy between licensed operator requalification training and actual response in the control room during events. Specifically, during requalification training the licensee used three personnel in the simulator throughout the scenario; however, during actual events and exercises, only two operators were available since the third operator delays entry into the control room because of other duties in the operations support center.

During this inspection, the inspectors confirmed that the licensee had initiated Problem Evaluation Request 298-0348 to document this discrepancy. The licensee performed training during Requalification Cycle 98-04 that included practicing with less than the full control room complement and starting a scenario with the shift technical advisor absent. In addition, in June 1999, the NRC evaluated corrective actions and activities related to operator control board awareness and critical parameter identification and trending. The inspectors concluded, at that time, that the deficiencies were being addressed. The inspectors found these actions to be sufficient.

M1 Conduct of Maintenance

M1.1 General Comments - Maintenance

a. Inspection Scope (61726, 62707)

The inspectors inspected the following maintenance and surveillance activities:

- Procedure OSP-ELEC-S703, "HPCS Diesel Generator Semi-Annual Operability Test," Revision 9
- Procedure ISP-APRM-S303, "APRM Channel C Modes 1 and 2 Channel Check," Revision 5
- Work Order 01002552, Vibration and Motor Current Signature Analysis for Diesel Generator 3 Fans
- Work Order 01002551, Diesel Generator Lube Oil Pump Vibration Analysis
- Work Order MVB3, Reactor Recirculation System Sensing Line Weld Repair

b. Observations and Findings

The inspectors identified no concerns during implementation of the above activities.

III. ENGINEERING

E1 Conduct of Engineering

E1.1 RCIC System Component Evaluations

a. Inspection Scope (37551)

During discussions with plant personnel, the inspectors were informed that certain RCIC system components were not upgraded to safety status with the remainder of the system. The inspector reviewed the licensee's justification for maintaining the RCIC keepfill pump and barometric condenser in a nonsafety status.

b. Observations and Findings

In 1997, an NRC engineering team identified that in 1985 the RCIC system had been downgraded to a nonsafety-related status in apparent violation of 10 CFR 50.59 (NRC Inspection Report 50-397/97-13). In Enforcement Action (EA) 97-573, dated June 1, 1998, the NRC subsequently concluded that the licensee was in violation of 10 CFR 50.59. The licensee had failed to perform an adequate safety evaluation, when the RCIC system was downgraded from a safety-related system to a nonsafety-related system without NRC approval, and as a result, they had failed to identify that this downgrade constituted an unreviewed safety question. The licensee was required to upgrade the system or seek NRC approval for the downgrade. The licensee chose to upgrade the system, but left some of the system in a nonsafety status based on engineering analysis.

During this inspection period, the inspector identified two instances where the licensee provided inadequate justification for maintaining the nonsafety-related status of RCIC components (the RCIC keepfill pump and the RCIC barometric condenser level switch).

RCIC Keepfill Pump: On October 27, 1997, during the upgrade process, the licensee utilized Technical Memo (TM) 2071 (dated October 6, 1994) to justify maintaining the nonsafety status of the RCIC keepfill pump. The keepfill pump maintains the RCIC injection pathway full of water during periods when the RCIC is not injecting, including during RCIC safety operation. Failure of the pump would result in a water hammer on the RCIC system.

TM 2071 was based on guidance provided in Electric Power Research Institute (EPRI) NP-6766, Volume 5, Part 1, "Water Hammer Prevention, Mitigation, and Accommodation," July 1992. The TM concluded that a water hammer event would result in a stress less than the ASME limits for a faulted condition on RCIC Valve 65 seismic supports. The calculation determined that the impact force was 5372 pounds and the faulted limit was 5897 pounds (about 500 pounds of margin was still available). Based on the calculation, the licensee concluded that damage was acceptable and the status of the pump could remain nonsafety. No 10 CFR 50.59 evaluation was performed.

The inspector reviewed TM 2071 and the guidance provided in EPRI NP-6766. The inspector concluded that TM 2071 was inadequate and did not appropriately consider all of the guidance provided by EPRI NP-6766. The inspector concluded that the licensee did not have appropriate justification for maintaining the RCIC keepfill pump in a nonsafety status. Specific problems related to TM 2071 included:

- Only one water hammer was assumed in the calculation. During a design basis accident, the RCIC system is required to start and stop repeatedly, so multiple water hammers should have been assumed.

- The use of the ASME faulted load limits (which assume some, but limited, component damage) was inappropriate for this application. The faulted load limits were intended for a 1-time occurrence only. With multiple water hammers, the faulted load limits do not apply.

- Other water hammer mechanisms were neglected and should have been evaluated. For example, potential water hammers in pipe elbows and in horizontal piping runs should have been considered.

- The effects of the subsequent pressure pulse through the system was not considered. Following the water hammer, a sizable pressure pulse is reflected back through piping and components. This pressure pulse, on its own, can cause damage to plant components, especially sensitive instruments such as flow meters.

The licensee agreed with the inspector's assessment and also noted that the conclusions derived by TM 2071 were inconsistent with the WNP-2 informal "no water-hammer" policy.

The RCIC keepfill pump contained some parts that were not safety grade and was not being tested per the licensee's ASME inservice testing program, as required for safety-related pumps. The licensee evaluated these deficiencies and determined that the RCIC keepfill pump was operable but nonconforming. The pump remained reliable for the inspection period and the inspector had no operability concerns. The licensee planned to either provide appropriate justification for the nonsafety status or upgrade the pump to meet safety-related requirements. The issue was entered into the licensee's corrective action program as Problem Evaluation Request 299-2257.

Barometric Condenser Level Switch: The RCIC barometric condenser level switch trips on high barometric condenser level to operate the barometric condenser pump. Failure of the switch, during RCIC operation, would result in overfilling the barometric condenser during RCIC operation, lifting the condenser relief valve and spilling condensate into the RCIC pump room. The licensee completed calculation EQ-02-97-01, dated August 20, 1997, and determined that flooding during the 5-hour expected system response time would not affect RCIC operability. The licensee took credit for the equipment drain piping between the RCIC and control rod drive (CRD) pump rooms, which minimized the maximum level derived in the calculation. The flooding rate was 16 gpm from the barometric condenser alone, and the maximum water level was found to be 2.4 inches. RCIC would be rendered inoperable at a level of 6 inches.

The inspector reviewed the calculation and found it inadequate. Specifically, the calculation only assumed leakage from the barometric condenser and neglected normal leakage from other sources. The RCIC room normally drained to the equipment drain system Sump 5, in the CRD pump room, and there was no isolation mechanism between the rooms. Further, leakage from a substantial number of components in the reactor building also drained to equipment drain system Sump 5, including drywell-identified leakage. Leakage from drywell-identified leakage, alone, was permitted by Technical Specifications to reach up to 25 gpm.

The licensee agreed with the inspector's observations and initiated Problem Evaluation Request 299-2516 to address the problem.

NRC Assessment: The continued failure to perform an adequate safety evaluation for a RCIC System component downgrade as required by 10 CFR 50.59 was determined to be a violation of 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Action,” (50-397/99013-01). This criterion requires the licensee to take prompt corrective measures in response to conditions adverse to quality, including nonconformances. The inappropriate RCIC safety classification, identified in 1998, was a nonconformance and the licensee had inadequate justification to support the continued nonsafety status of the RCIC keepfill pump and the barometric condenser level switch.

c. Conclusions

The licensee failed to restore compliance within a reasonable time frame after a violation of 10 CFR 50.59 was identified. In 1998, the NRC identified that the licensee inappropriately downgraded the reactor core isolation cooling (RCIC) system to nonsafety status via the 10 CFR 50.59 process. The licensee took corrective action to upgrade the system, but left some of the system in nonsafety status based on inadequate engineering analysis. Specifically, the RCIC system keepfill pump and barometric condenser level switch were maintained as nonsafety components based on inadequate water hammer and flooding calculations, respectively. This resulted in a continuing violation of 10 CFR 50.59. These inadequate corrective measures were cited as a violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action."

E1.2 Reactor Building Flood Protection

a. Scope (37551)

During review of plant documents, the inspector observed that an unisolable equipment drain line connected the RCIC and CRD pump rooms. The rooms share a common sump, located in the control rod drive room. The inspector checked the adequacy of this configuration against the licensee's Final Safety Analysis Report (FSAR) commitments with respect to flood protection.

b. Observations and Findings

The inspector observed that the equipment drain line was depicted in FSAR Figure 9.3-9, which was referenced by the FSAR system description. However, the inspector identified that the design and configuration of the unisolable 3-inch diameter equipment drain line was not consistent with commitments for flooding protection which were also contained in the FSAR. The WNP-2 FSAR has the following statements:

1. The primary or safety-related functions fulfilled by the reactor building include the capability to withstand the effects of flooding from internal sources . . . The pump enclosure rooms in the reactor building basement are made water resistant between and including the top of the foundation mat at elevation 422 . . . and elevation 468 . . . (Section 3.4.1.5.2).

2. Separation of redundant features is the basic design criteria utilized to protect SSCs [structure, system, and components] important to safety from the dynamic effects of postulated pipe ruptures . . . The emergency core cooling pumps, as well as the three RHR [residual heat removal] and the RCIC pump, are located in individual water resistant compartments (Section 3.6.1.5.2).

3. To accommodate the safety-related systems, equipment and components housed therein, the reactor building is functionally designed and arranged to provide the following structural facilities: . . . At the foundation mat level, flood walls are provided to enclose rooms and isolate such equipment as RHR pumps, core spray pumps, and CRD pumps. Each room is water resistant for protection against flooding from internal sources (Section 3.8.4.1.1).

The inspector found that the CRD and RCIC rooms were not individual water resistant rooms, but were connected together through an unisolable 3-inch equipment drain line that runs between the two rooms.

The above commitments were made, in part, to demonstrate compliance with 10 CFR Part 50, Appendix A, General Design Criterion 4, which requires, in part, that equipment important to safety be protected from flooding. The rooms provide separation, which was one of the acceptable methods that may be utilized to meet General Design Criterion 4. The licensee had not demonstrated compliance with their commitment to this criterion prior to the close of the inspection.

Furthermore, Calculation 5.51.055, "Flooding Analysis," neglected the equipment drain line piping run and assumed that each room can flood separately but not at the same time. As such, the analysis was inadequate.

By following NRC flooding-related guidance, the licensee believed that a less conservative but acceptable evaluation would demonstrate that most of the emergency core cooling system trains would remain functional during and following a flood in the CRD or RCIC rooms and that safe shutdown during these design basis floods could be achieved.

Finally, since the RCIC and CRD pump rooms were not watertight, this condition increased the probability that a flood in the CRD room would damage the RCIC system, a system important to safety. Accordingly, the inspector was concerned that the condition may be an unreviewed safety question. The licensee stated that they did not believe the condition was an unreviewed safety question but had not completed the safety evaluation at the close of the inspection.

The inspector found that the licensee had an existing plant tracking item that pertained to verification of flood protection design basis information. The tracking item was initiated following a reactor building flood in June 1998. The licensee believed that this initiative may have identified this problem given enough time. However, the inspector also observed that the item had been deferred three different times since being initiated and was not currently scheduled to be completed until March 2000. Furthermore, very little effort was expended toward the completion of this item. Considering the safety significance of the June 1998 flooding event, the licensee's actions toward completing this particular corrective action item were not commensurate with the safety significance.

The licensee initiated Problem Evaluation Request 299-2499 to address the inspector's concerns. This issue is considered an unresolved item pending further NRC review of the licensee's revised analysis, their safety evaluation, and appropriate consideration of the risk impact, as necessary to determine the potential safety consequences of this design deficiency (50-397/99013-02).

c. Conclusions

The inspectors identified that reactor building flooding protection was not consistent with FSAR commitments. The FSAR states that all reactor building pump enclosure rooms are watertight, but the reactor core isolation cooling and the control rod drive pump rooms were connected via an unisolable equipment drain line. Further, this condition was not consistent with the licensee's flooding analysis, which assumed that flooding could not occur in both rooms simultaneously. The licensee maintained that even with the unisolable connection, safe shutdown could be achieved for design basis floods in these rooms. Pending further review of the licensee's revised flooding analysis and risk evaluation to determine the safety significance of the design deficiency, this is an unresolved item.

E8 Miscellaneous Engineering Issues (92903)

E8.1 (Closed) Licensee Event Reports 50-397/98-006-00 and -01: Discrepancies found in low voltage bus calculations during review of 10 CFR Part 50, Appendix R calculations. The licensee discovered discrepancies in the low voltage bus calculations during a review of the 10 CFR Part 50, Appendix R calculations for high impedance faults.

The inspectors reviewed Calculation Modification Record CMR-98-0164, dated April 22, 1999, which was prepared to incorporate changes to Section 1D, "High Impedance Fault Analysis," of the Appendix R postfire safe shutdown calculation. The calculation results disclosed that deficiencies resulted from underestimated or omitted loads that could contribute to the total high impedance fault load on a postfire safe shutdown bus not included in the original calculation. The licensee had revised the calculation methods, which had revealed this problem.

To correct the problem that was revealed through the use of the new methodology, the licensee revised procedures to require operators to monitor safe shutdown buses and/or remove nonsafe shutdown loads in the event that certain buses became overloaded because of fire-induced faults. The inspectors reviewed Procedures 4.12.4.1, "Fire," Revision 19, and 4.12.1.1, "Control Room Evacuation and Remote Cooldown," Revision 29, and verified that the licensee had incorporated the necessary operator actions. The corrective measures were consistent with guidance provided in Generic Letter 86-10, "Implementation of Fire Protection Requirements," dated April 24, 1986.

E8.2 Thermolag Work: The licensee completed Thermolag replacement work consistent with their commitments to the NRC.

IV. Plant Support

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 inspector 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 November 30, 1999. 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 92901:	Operations Followup
IP 92903:	Engineering Followup

ITEMS OPENED AND CLOSED

Opened

50-397/99013-01	VIO	Inadequate corrective actions to address prior RCIC unreviewed safety question (Section E1.1).
50-397/99013-02	URI	Failure to meet FSAR commitments for flood protection (Section E1.2).

Closed

50-397/98005-06	IFI	Weakness in operator requalification program (Section O8.1).
50-397/98-006-00 50-397/98-006-01	LER	Discrepancies found in low voltage bus calculations (Section E8.1).

LIST OF ACRONYMS USED

ASME	American Society of Mechanical Engineers
CFR	Code of Federal Regulations
CRD	control rod drive
EPRI	Electric Power Research Institute
FSAR	Final Safety Analysis Report
gpm	gallons per minute
IFI	inspector followup item
LER	licensee event report
NRC	U.S. Nuclear Regulatory Commission
PDR	public document room
RCIC	reactor core isolation cooling
TM	technical memo
URI	unresolved item
VIO	violation