U.S. NUCLEAR REGULATORY COMMISSION

REGION V

Report No: 50-397/84-35

Docket No: 50-397

License No. NPF-21

Licensee: Washington Public Power Supply System
P. O. Box 968
Richland, WA 99352

Facility Name: Washington Nuclear Project No. 2 (WNP-2)

Inspection at: WNP-2 Site near Richland, Washington

Inspection Conducted: November 3-30, 1984

Inspectors: PH Johnson 1/11/85
for A. D. Toth, Senior Resident Inspector Date Signed

PH Johnson 1/11/85
for R. S. Waite, Resident Inspector Date Signed

Approved by: PH Johnson 1/11/85
P. H. Johnson, Chief Date Signed
Reactor Projects Section 3

Summary:

Inspection on November 3 - 30, 1984 (50-397/84-35)

Areas Inspected: Routine, unannounced inspection by the resident inspectors of control room operations, engineered safety feature (ESF) status, surveillance program, maintenance program, power ascension test program, licensee event reports, special inspection topics, and licensee action on previous inspection findings.

The inspection involved 203 inspector-hours onsite by two resident inspectors, including 31 hours during backshift work activities.

Results: No violations or deviations were identified.

DETAILS

1. Persons Contacted

Washington Public Power Supply System

J. Shannon, Director of Power Generation
*J. Martin, Plant Manager
R. Corcoran, Operations Manager
*J. Baker, Acting Operations Manager
K. Cowen, Technical Manager
*R. Graybeal, Health Physics and Chemistry Manager
*J. Landon, Maintenance Manager
*J. Peters, Administrative Manager
*P. Powell, Licensing Manager
*C. Powers, Assistant Plant Manager
*D. Walker, Plant Quality Assurance Manager
M. Wuesterfeld, Reactor Engineering Supervisor

*Personnel in attendance at exit meeting November 30.

The inspectors also interviewed various control room operators, shift supervisors and shift managers, engineering, quality assurance, and management personnel relative to activities in progress and records.

2. General

The Senior Resident Inspector and/or the Resident Inspector were onsite November 1-2, 5-10, 13, 15-16, 19-21, and 26-30. Backshift inspections were conducted November 5-10, 15-16, 19-20, and 27-30.

Several regional office inspectors visited the site this month for routine inspection activities. Their activities were documented in separate inspection reports.

- The regional office Project Inspector (D. Willett) was onsite November 14-16 for routine operations inspection and review of power ascension test records.

- Regional office operator examiners (R. Pate, G. Johnston and P. Gage) and consultants (I. Levey and G. Sly) were onsite November 6-9 to examine reactor operator license candidates.

- A regional office supervisor (P. Johnson) was onsite November 7-9 to meet with plant management, inspect the site, and meet with the resident inspectors.

- NRC headquarters management representatives from the office of the Executive Director for Operations (J. Sniezek and E. Blackwood) were onsite November 6 to meet with the resident inspectors and inspect the facility.

3. Plant Status

During the period of this report, the plant was conducting power ascension testing in test conditions 5 and 6. A reorganization in November realigned the Support Services Directorate, headed by D. Bouchey, to report to the Assistant Managing Director For Operations. The Technology Directorate was renamed the Engineering Directorate.

4. Operations Verifications

The resident inspectors reviewed the control room operator and shift manager log books on a daily basis for this report period. Reviews were also made of the Jumper/Lifted Lead Log and Nonconformance Report Log to verify that there were no conflicts with Technical Specifications and that the licensee was actively pursuing corrections to conditions listed in either log. Events involving unusual conditions of equipment were discussed with the control room personnel available at the time of the review and evaluated for potential safety significance. The licensee's adherence to Limiting Conditions for Operation (LCO's), particularly those dealing with ESF and ESF electrical alignment, were observed. The inspectors routinely took note of activated annunciators on the control panels and ascertained that the control room licensed personnel on duty at the time were familiar with the reason for each annunciator and its significance. The inspectors observed access control, control room manning, operability of nuclear instruments, and availability of onsite and offsite electrical power. The inspectors also made regular tours of accessible areas of the facility to assess equipment conditions, radiological controls, security, safety and adherence to regulatory requirements.

a. Clearance Order Review

During an audit of Clearance Order 84-11-217 for the high pressure core spray diesel the inspector noted that danger tags hung on two diesel starting air valves were hung on two valves other than those specified, but which accomplished the same function (isolating starting air). The inspector verified this with an Equipment Operator. The inspector notified the Shift Manager and Control Room Supervisor and this item was immediately resolved by placing the tags on the correct valves and restoring the mistagged valves to the correct position. The licensee stated that the involved individuals would be counseled on the necessity of hanging tags as instructed in the Clearance Order.

b. Primary Containment Verification

During a walkdown of the high pressure core spray (HPCS) system the inspector noted that a vent valve, HPCS-V-85, had not been included in licensee procedure 7.4.6.1.1, "Primary Containment Verification" as requiring verification, in accordance with Technical Specification 4.6.1.1.b, of being closed. The valve was found by the inspector to be in the correct position. The licensee informed the inspector at the exit meeting that procedure 7.4.6.1.1 was corrected to include this valve. The licensee has initiated a study to ensure that all containment isolation valves are included on the appropriate valve checklist. (84-35-01)

c. Vent Valve Not Identified on Control Room Drawing

During a walkdown of Loop B of the residual heat removal (RHR) system the inspector identified a vent valve which had been ommitted from Control Room drawing M521, Sheet 2. The inspector identified this to the licensee and the licensee committed to investigate the condition. Subsequent to the exit meeting, the inspector noted that corrective actions for a previous inspection item were to have corrected such omissions of vent and drain valves from drawings. This inspection issue is reopened for consideration of the effectiveness of the licensee's prior corrective actions. (83-48-01)

No violations or deviations were identified.

5. Surveillance Program Implementation

The inspectors ascertained that surveillance of safety-related systems or components was being conducted in accordance with license requirements. In addition to observing and occasionally witnessing and verifying daily control panel instrument checks, the inspectors observed portions of several detailed surveillance tests by operators and instrument and control technicians.

No violations or deviations were identified.

6. Monthly Maintenance Observation

Portions of selected safety-related systems maintenance activities were observed. By direct observation and review of records the inspector determined whether these activities were consistent with LCOs; that the proper administrative contols and tagout procedures were followed; that equipment was properly tested before return to service; and independently verified that the equipment was returned to service. The inspector also reviewed the outstanding job orders to determine if the licensee was giving priority to safety related maintenance and verify that backlogs which might affect system performance were not developing.

No violations or deviations were identified.

7. Power Ascension Test Program

The inspectors observed and witnessed tests, examined equipment, interviewed personnel, and reviewed records and procedures relative to conduct of the power ascension program described in Chapter 14 of the Final Safety Analysis Report (FSAR).

The inspector reviewed records of apparent test results (which address level 1 and 2 criteria compliance), selected parts of procedure and data packages, and Plant Operations Committee meeting minutes for tests conducted during test condition #5. Tests were conducted as noted in the test matrix of FSAR Table 14.2-4, with the following exceptions:

- Test No. 1: The chemical and radiochemical data taken at test condition #3 encompassed the reactor power levels and core flows of test condition #5. Therefore the licensee considers that the test objectives were achieved during test condition #3, without need to repeat the data collection at condition #5. (See FSAR Figure 14.2-3). The licensee submitted an FSAR Amendment #35 (November 8, 1984) to clarify this item. The NRC technical review staff had not seen the amendment prior to the test, but advised the inspector that they concur with the equivalency determination by the licensee.

- Test No. 34: The reactor vessel internals vibration testing at test condition #4 (natural circulation) was not performed and was deemed unnecessary by the licensee. The licensee received General Electric concurrence in deletion of the natural circulation condition testing via a September 21, 1984 Field Deviation Disposition Request for test specification 22A6601 Revision 0, (FDDR No. KK1-424). This stated that: "The deleted tests at intermediate power are covered by tests under 100 percent power and vice versa. Therfore, all necessary data would be obtained. The added tests would be beneficial to reliability." The licensee submitted an FSAR Amendment #35 (November 8, 1984) to clarify this matter. The NRC technical review staff has considered this matter and concurred that it was not a major modification of the test program and did not require prior NRC approval of its omission.

The inspectors witnessed the following power ascension tests during this period:

a. Control Rod Drive Tests

During the main steam isolation valve full closure test at 97 percent power the licensee monitored the scram times of all control rods, including the four control rods of Group A, which had been determined to be the slowest in that group during the heatup power ascension phase at rated pressure. The times of FSAR Section 14.2.12.3.5.4 were met, with the slowest rod reaching position 05 at 2.36 seconds. Additionally, the computer data for all rods showed that each individual control rod met the technical specification and FSAR criteria for the four slowest rods.

No violations or deviations were identified.

b. Main Steam Isolation Valve Tests

On November 10, the inspector witnessed the main steam isolation valve (MSIV) full isolation (all eight valves) test while at test condition #6 (92.3% power). The inspector also examined the real time computer printouts during and after the tests to ascertain compliance with the test criteria of FSAR Section 14.2.12.3.25.4., and reviewed test results with the responsible shift technical advisors. The test was performed with no operator action for the first 30 seconds, to assure determination of natural plant response. Valve stroke times met 2.5 to 5 seconds criteria, and the reactor scram occurred automatically. A maximum reactor water level of 43 inches initially occurred with a later maximum of 93 inches, controlled to prevent flooding of steam lines. The maximum reactor dome pressure of 1065 psig in 30 seconds met the 1125 level 2 and 1150 level 1 criteria. A zero positive change in heat flux met the 0% level 2 and 2% level 1 criteria; pressure relief valve acoustic monitors and tail pipe temperatures confirmed that the valves reseated, RCIC actuation occurred automatically; and recirculation pumps tripped at level L2.

This test was conducted one day earlier than planned due to increasing coolant chemistry problems arising from condenser tube leaks. Licensee management took appropriate actions to call in needed personnel to support the operations and engineering staffs. Pretest briefings were held, operations management cleared the control area of personnel not directly involved in the test, and personnel assignments were defined for monitoring instruments. The operators responded to this major transient with a good deal of communication with each other and appeared very professional and regimented in their attention to alarms and annunciators and other indications of expected and unexpected conditions. The established scram recovery procedure was avaiable to the operators and was used.

No violations or deviations were identified.

c. Core Performance Tests

The inspector reviewed the 96.9% (test condition 6) core performance printout from the process computer on November 10 (P1 printout) just prior to the main steam isolation valve full closure test. The data showed the following compliance with technical specification thermal limits (most limiting values within the reactor core):

MCPR:	1.55	
MRPD:	12.14	(MFLPD: .906)
MAPLHGR:	10.67	

On November 23 the process computer showed the 100% power values for most limiting conditions within the reactor core:

MCPR:	1.38	(limit = 1.24 min.)
MRPD:	12.61	(limit = 13.4 max.)
MAPLHGR:	11.31	(limit = 12.2 max.)

No violations or deviations were identified.

d. Remote Shutdown Panel Control of Shutdown Cooling

The remote shutdown cooling test was scheduled to be done in conjunction with the generator trip test, but was conduted earlier after an unplanned reactor scram on November 27. From the control room location the operators reduced reactor pressure to 70 psig by using the turbine bypass valves, and controlled reactor water level by manipulating controls of condensate-feedwater and reactor water cleanup systems. The main steam isolation valves (MSIVs) were not closed and the reactor was not isolated during this test. Pressure relief valves or the RCIC system was not used. From the remote shutdown panel, the operators initiated flow to prewarm the RHR piping and commence flow to the reactor vessel. Temperatures of the reactor system were reduced by at least 50F. The following discrepancies were observed by the inspectors:

(1) The test engineer did not sign the test prerequisites verification checksheet item 8.2.28.7.B.2, which prescribed initial pretest conditions above 125 psig.

(2) The test was not commenced from the prerequisite 125 psig condition, but rather the transfer of control to the remote shutdown panel was not performed until the control room operators first reduced pressure to 70 psig. (This obviated the need for the shutdown panel operators to implement procedure step 8.2.28.8.B.8, which called for demonstration that pressure could be reduced remotely from 125 psig to 70 psig while controlling pressure and water level using safety relief valves (SRVs) and the reactor core isolation cooling (RCIC) system).

(3) Reactor level and pressure were controlled from the control room during the test. This was not consistent with procedure step 8.2.28.8.B.6 "NOTE" which called for control room operations only to protect or secure systems not related to the controlled shutdown of the reactor. Condensate pumps, feedwater control valves, reactor water cleanup discharge valves, and recirculation sytem valve 23B were manipulated from the control room.

(4) Control room instrumentation was used to ascertain compliance with the maximum 100F/hour cooldown rate, since the pressure instrumentation in the remote shutdown panel (0-1200 psig gage) was insufficiently sensitive to ascertain the temperature changes at the 60 psig range (using steam table for pressure/temperature conversion).

(5) The operators did not attempt to demonstrate the plant emergency procedure requirement to take pyrometer readings of temperatures on RHR piping, prescribed by the plant emergency procedure (PPM-4.12.1.1 step 17.1.), referenced in the test procedure as the method to place the RHR system in shutdown cooling mode of operation.

(6) The reactor level gage at the remote shutdown panel was pegged high (+60 inches) for the duration of the test, affording the operators at this station no information as to whether the level was approaching an unacceptably high level. Actual level controlled from the control room was being maintained at about +40 inches. The discepancy was attributed to the remote shutdown panel gage having been calibrated for hot water conditions associated with the plant at the initial hot standby/hot shutdown condition. (Control from the remote

shutdown panel, for depressurizing at this high temperature condition, was previously conducted in test condition #1).

(7) Failure to close the main steam isolation valves, or equivalvent measures to isolate the reactor vessel, permitted conditions which did not correspond to those that would exist during an actual evacuation of the control room.

The inspector identified the procedure deviations and technical issues to licensee management and staff. The test results had not yet been reviewed by the plant operations committee (POC), which is charged with review and acceptance of test results. The licensee stated that the inspector's observations would be considered during the overall data evaluation and presented to the POC for prescription of corrective actions where appropriate. The licensee's test results review and followup actions will be examined during a future inspection. (84-35-02)

e. Engineered Safety Feature Equipment Room Heat Load

The inspector verified that the licensee had defined and scheduled a test of the RHR room cooling system, in accordance with the commitment of FSAR Question/Response 423.031. This test was added to the schedule for test activities during the reactor cooldown after the generator trip test. The test procedure (8.4.8) calls for isolation of the normal ventilation ducts to the RHR room, running of the RHR pumps with minimum flow rate on the RHR heat exchangers to maximize heat loads on the room cooling units, and extrapolation of data to maximum standby service water temperature of 77-85 F.

The procedure did not include specific or general acceptance criteria or analysis guidance. The inspector interviewed the responsible system engineer and design engineer and ascertained that analysis methods had been defined; particularly, consideration had been given to methods of extrapolation to account for equipment performance and Loss of Coolant Accident (LOCA) heat load effects on reactor building temperature and suppression pool temperature. At the exit meeting the licensee confirmed that appropriate acceptance criteria would be added to the procedure.

No violations or deviations were identified.

8. Licensee Event Reports

The inspector reviewed selected Licensee Event Reports (LERs) and supporting information to verify that the licensee had reviewed the events, corrective action had been taken, no unreviewed safety questions were involved, and violations of regulations or Technical Specification conditions had been identified. Violations were evaluated for significance relative to 10 CFR 2, Appendix C.

The Assistant Plant Manager noted that two LER's involved failure to perform required surveillance tests and initiated management corrective action through issuance of an October 31, 1984 memorandum to the plant staff to emphasize the need to properly implement the surveillance program. Particular emphasis was placed on being sensitive to interaction between multiple Technical Specification Action Statements and compliance with the most restrictive limiting conditions for operation. This action appeared appropriate to the circumstances. The inspector confirmed that this information had reached the plant control room staff.

The resident inspectors reviewed data and interviewed personnel relative to the following event reports.

LER-84-72, Isolation of Reactor Water Cleanup

LER-84-72-01, Isolation of Reactor Water Cleanup

LER-84-87, Rod Worth Minimizer Not Declared Inoperable

LER-84-97, Reactor Water Cleanup Isolation

LER-84-99, Spurious Reactor Core Isolation Cooling Isolation

LER-84-101, Spurious Reactor Water Cleanup Isolations

LER-84-104, Reactor Scram

LER-84-106, Incorrect Instrument Installation

LER-84-108, Reactor Scram on Main Steam Isolation Valve Closure

LER-84-110, Electro-Hydraulic Valve Actuator Failure

LER-84-116, APRM Upscale Trips in Hot Shutdown Mode Set Beyond Allowable Technical Specification Values

(Closed, LER-84-72 and 72-01) - The inspector ascertained that the reactor water cleanup system isolation alarm and trip device had been modified to alarm/annunciate 45 seconds prior to trip, as opposed to the original alarm coincident with trip 45 seconds after initiating signal. The surveillance and alarm/annunciator procedures (*7.4.3.2.1.47 and *4.601.A3-3.4) were modified by Procedure Deviation Forms dated October 9 and 18 and were in place in the control room this report period. These steps allow the reactor operator to acknowledge the alarm and adjust flows to avoid inadvertant isolation for cases of short transient flow conditions.

(Open, LER 84-87) - During the inspector's review of LER 84-087, two items were identified which appear to have contributed to this event in addition to those identified in the LER:

1) Licensee procedure 3.2.1 "Normal Shutdown to Cold Shutdown" did not appear to provide adequate guidance to the operators as to when to commence surveillance procedure 7.4.1.4.1.2 during a shutdown. Procedure 3.2.1 instructs the operator to commence performance of procedure 7.4.1.4.1.2 after verifying the rod worth minimizer (RWM)

"Below Low Power Setpoint (LPSP)" light illuminates,i.e. at less than 20% power (the LPSP is the point at which the RWM automatically initiates).

2) As a result of information obtained during resolution of a Nonconformance Report (NCR) generated because of this event, the licensee has determined that the Rod Worth Minimizer (RWM) was not designed for compliance with Technical Specification 4.1.4.1.a as currently written, which requires the RWM to be operable "...in Operational Condition 1 within 8 hours prior to RWM automatic initiation when reducing THERMAL POWER, by verifying proper indication of the selection error of at least one out-of-sequence rod."

Prior to this event several shutdowns were completed but in each case examined by the inspector it appears that the RWM was declared Inoperable prior to passing thru the LPSP and two operators were stationed as required per action statement 3.1.4.1.a. Because of this, the error in the RWM was not discovered sooner. Since this event, several shutdowns have been completed and in each case examined by the inspector the RWM was declared inoperable and operators were stationed as required per Technical Specification 3.1.4.1.a.

The licensee has initiated corrective action as discussed in the LER. In addition, the licensee has committed to provide written instructions to the Shift Mangers to station two operators at the reactor control console during shutdowns prior to reaching the LPSP while in the transition zone and until RWM operability is verified in accordance with Technical Specification 3.1.4.1. The licensee has also committed to revise procedure 3.2.1 to include instructions that surveillance procedure 7.4.1.4.1.2 be commenced within 8 hours prior to RWM automatic initiation when reducing power in Operational Condition 1.

(Closed, LER-84-97) - See LER-84-72, above.

(Closed, LER-84-101) - See LER-84-72, above.

(Closed, LER-84-99) - The licensee issued Procedure Deviation Form (PDF) 84-1030 on October 19 to add precautions to place the key-lock switch into "Test" position when reading the leak detection system isolation panel meters. A caution tag was also posted on each of the two panels with this information. On November 19, the inspector observed that the panel walkdown checklist (Procedure 7.0.0) in use by the operators included the precautionary PDF.

(Closed, LER-84-104) - The inspector interviewed plant personnel and examined documents relating to the vendor representative action in the control room which resulted in the reactor scram. Control room night orders have been issued to instruct operations staff to closely control vendor activities in the control room. Additional actions to define program controls for similar plant staff and vendor troubleshooting activities have been committed by licensee management as described in NRC inspection report 84-31, paragraph 9.e.; those program controls will be reviewed during future NRC inspections relating to followup item 84-09-03.

(Closed, LER-84-106) - The inspector considered this item relative to Technical Specification 3.3.7.5, FSAR Section 7.5.2, and Regulatory Guide 1.97. He interviewed the WPPSS responsible system engineer and examined the relevant design change documents. These showed that an original equipment Barton-760 gauge was replaced with a Rosemont-1153 gauge in accordance with engineering direction by General Electric, which included incorrect connection of the reactor vessel water level reference leg to the "High" side of the new gauge. Thus one of two redundant fuel zone level indicators (MS-LT-44A for recorder MS-LR-615) was inoperable for over 10 months. The matter was identified by the licensee, reported to NRC, corrected, and did not relate to any prior violation.

(Closed, LER-84-108) - The inspector verified that the plant scram resulted from the reactor operator prematurely placing the mode switch into "Run". This was a departure from the sequence described in startup procedure 3.12, which required in an earlier step that reactor pressure be increased to 920 psig. The sequence departure was performed without obtaining the prescribed approval of the Shift Supervisor. Revision 8 of the startup procedure includes a "CAUTION" following step 62, to ensure that low pressure annunciators are cleared prior to moving the mode switch to "RUN". Use of this procedure was noted during the November 20 startup; it provides clearly formatted instructions to avoid a recurrence of the reported event.

(Closed, LER-84-110) - This LER reported problems on October 3 with the control room emergency ventilation system similar to the mid-October situation reviewed by the inspectors, as described in inspection report 84-31, paragraph 7.c. Both events reflected failure to recognize entry into the technical specification action statement. That action statement prohibited handling of irradiated fuel in the secondary containment, core alterations and operations with a potential for draining the reactor vessel. None of these operations was underway or planned. Normal plant operations of reactor water cleanup blowdown and shutdown heat removal line preheating/blowdown were not considered to have potential for draining the core, since automatic isolation of the flowpaths and operability of automatic ECCS makeup systems existed in accordance with technical specification operability requirements. Failure to identify entry into action statements has been addressed by the licensee, with emphasis in revised procedures to assure proper recording of such actions in the operations logs.

(Closed, LER-84-116) - The resident inspectors reviewed circumstances of incorrect setpoint of the Average Power Range Monitors (APRM) immediately subsequent to the discovery of the event. Details of this review were discussed in NRC inspection report 84-31, Paragraph 8.f.).

No violations or deviations were identified.

9. Licensee Actions on Previous NRC Inspection Findings

The inspectors reviewed records, interviewed personnel, and inspected plant conditions relative to licensee actions on previously identified inspection findings:

a. (Closed) Followup item (84-18-02) - Flexible conduit waterproof jackets were found damaged. The inspector examined records of 1983 installation of BISCO LOCA-Seal plugging material at the entry of the conduit to the junction boxes of important connections, to prevent entry of water into the equipment. He also examined the scope of work in progress in the drywell for inclusion of taping or replacement of damaged flexible conduit. Procedure 10.25.57 revision 2 was in use for evaluation of damage and performance of repairs by Bechtel field engineer and craft personnel. Valve operators for valves RWCU-V-1 and RHR-V-50B were included in the scope of this procedure. These reviews showed the followup item to have been resolved.

b. Report 50-397/84-26 Paragraph 4: the licensee had committed to submittal of a technical specification clarification regarding chlorine monitors and actuation of isolation valves. At the exit meeting this period, the licensee stated that a plant modification request had been issued (PMR-84-1570) to prescribe modifications to discontinue chlorine use at the site, and install a sodium chloride system for water treatment. Once this is installed, a related technical specification change would be submitted. Since this action is expected to be complete within a few months, an interim technical specification related submittal to NRC would not be issued. The inspector acknowledged the licensee's plans.

10. Licensee Action On 10 CFR 50.55(e) Construction Deficiencies

Various construction deficiency reports were issued by the licensee during the construction phase of the project. Those reports of conditions and corrective actions taken or planned were reviewed by NRC regional staff at the time of submission. Fulfillment of reporting requirements, report completeness, corrective actions, generic aspects of the items, and need for onsite followup were evaluated. Many of these reports were further examined during followup inspections of records and hardware at the site.

During the current period, the inspector performed a review of the WNP-2 plant files of several such reports. The purpose of this inspection was to ascertain that the plant records verify that the described corrective actions had been implemented and/or incorporated into controlled corrective action programs subject to tracking and prioritization evaluations. The following items are considered closed, as noted:

(80-11-J) No. 120 - Procurement of Safety equipment to less than IEEE-323 requirements: NRC in-office review found the licensee reports of this matter and planned corrective actions acceptable.

(80-12-A) No. 127 - Deficiency in Limitorque Motor Operators: corrective actions taken were reviewed in conjunction with report No. 226 (83-01-C).

(80-12-C) No. 132 - Inadequate piping layup after hydrotest: the inspector determined that engineering directives had been issued for excavations to examine the service water piping, a study had been done to identify systems which had been hydrotested, that these had been examined, and specifications had been changed to provide post-hydrotest layup instructions.

(81-06-B) No. 134 - Procedure for coating application did not include ANSI requirements: this was determined to be not reportable and found acceptable during regional in-office review of the licensee letter of September 1, 1981.

(82-07-B) No. 201 - Overstress of instrument lines by thermal displacement: the inspector ascertained that corrective actions were documented as complete and verified by a project quality assurance surveillance record.

(82-09-A) No. 210 - Diesel generator corroded relay contacts: regional inspectors identified a violation regarding corrective action for this matter, as described in NRC Inspection Report 83-37 (tracking item 83-37-01); this matter was resolved as described in NRC Inspection Report 84-18.

(82-10-B) No. 216 - Relief valve vents: the inspector ascertained that the FSAR described the deletion of the steam condensing mode of the residual heat removal system, and observed the absence of the relief valves in the plant, disconnect of electrical supplies to valves, and chain locking of valves during routine plant tours. The licensee has defined a phase two program for removal of additional equipment and extraneous instruments previously associated with the steam condensing mode.

(83-01-E) No. 223 - Non-1E equipment connected to 1E electrical bus without isolation device: the inspector ascertained that the required corrective engineering directive had been issued and the plant tracking log showed that this was no longer an open item.

(83-01-C) No. 226 - Inadvertent installation of nonconforming valve motor operator: the inspector ascertained that nonconformance report documents had been issued to assure replacement of the valve operator. The plant tracking log showed that this was no longer an open item. Also, the file showed that the engineers had reviewed all DC and AC valve motor operators to assure matching of the voltage ratings and wiring terminations with the installed plant wiring.

(83-04-A) No. 251 - Unqualified motor operators on valves. The inspector ascertained that startup deficiency reports had been issued and were on the current plant tracking log for assuring that the valve operator motors are replaced prior to acceptance of the planned fuel pool cooling system installation. The tracking log shows required completion dates as prior to first plant refueling (which would require use of the fuel pool).

(83-07-C) No. 270 - Indeterminate grade nuts and incorrect grade bolts on ECCS pumps: this matter was identified during an NRC construction assessment team (CAT) inspection, as documented in NRC Inspection Reports 83-29 and 83-38 (NRC item 83-38-04). Corrective actions were reviewed and found acceptable by NRC inspectors, as documented in NRC Inspection Report 83-49, with exception of an outstanding question regarding bolting of pump head discharge plates, to be addressed in the licensee's final report on this matter. The final report was received (dated October 13, 1983) and found acceptable during NRC regional office review.

(83-07-F) No. 262 - Lack of redundant means of detecting RWCU leakage: the inspector ascertained that the plant tracking log no longer showed the corrective action documents as open items.

(83-08-B) No. 277 - Incorrect motor starting resistor settings: NRC in-office review found the licensee reports of this matter and planned corrective actions acceptable. The inspector ascertained that at least one of the licensee referenced startup deficiency reports (SDR) had been on the master completion list prior to fuel load and has subsequently been cleared from the plant tracking log (SDR 11296).

(83-10-B) No. 288 - Removal of sand adjacent to containment vessel had possible effect on stress analysis: NRC in-office review found the licensee reports of this matter and planned corrective actions acceptable. The resident inspector had also ascertained that the analysis had been conducted as stated, during the construction phase of the project, in conjunction with inspection of licensee corrective actions to assess and avoid corrosion of the containment shell in the area of the occasionally wet sand.

(83-12-A) No. 303 - Containment isolation valves installed improperly: NRC in-office review found the licensee report of this matter and planned/completed corrective actions acceptable. The resident inspector also witnessed the completed corrective action and interviewed plant engineering personnel regarding the change.

11. Management Meeting

On November 30 the inspectors met with the plant manager and his staff to discuss a summary of the inspection findings for this period. Attendees at this meeting are identified in paragraph 1 (*). Additionally, the inspector met with the Plant Manager weekly to review status of inspection findings, and as required with department managers to define data and information needs relevant to the inspections in progress.