U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Reports No. 50-373/82-49(DPRP); 50-374/82-15(DPRP)

Docket Nos. 50-373; 50-374 Licenses No. NPF-11; CPPR-100

Licensee: Commonwealth Edison Company
Post Office Box 767
Chicago, IL 60690

Facility Name: LaSalle County Station, Units 1 and 2

Inspection At: LaSalle Site, Marseilles, IL

Inspection Conducted: October 1-29, 1982

Inspectors: Roger D. Walker for W. Guldemond 11-18-82

Roger D. Walker for D. Williams 11-18-82

Roger D. Walker for A. Madison 11-18-82

Approved By: R. D. Walker, Chief R. D. Walker 11-18-82
Reactor Projects Section 2A

Inspection Summary

Inspection on October 1-29, 1982 (Reports No. 50-373/82-49(DPRP); 50-374/82-15(DPRP))

Areas Inspected: Routine, unannounced resident inspection. The inspection consisted of Licensee Actions on Previous Inspection Findings; Operational Safety Verification; Monthly Surveillance Observation; Followup of Part 21 Reports; Licensee Event Report Followup; Plant Trips and Safety System Challenges; Fire Protection/Prevention Program Implementation; Followup of IE Information Notices; Pressure Isolation Leak Testing; and Independent Inspection Effort. The inspection involved a total of 190 inspector-hours onsite including 30 inspector-hours during off-shifts.

Results: In the ten areas inspected, no items of noncompliance were identified.

DETAILS

1. Persons Contacted

R. Holyoak, LaSalle Project Manager
*G. J. Diederich, Superintendent
*R. D. Bishop, Administrative and Support Services Assistant Superintendent
J. G. Marshall, Operating Engineer
*J. C. Renwick, Technical Staff Supervisor
R. Kyrouac, Quality Assurance Supervisor

The inspectors also talked with and interviewed members of the operations, maintenance, health physics, and instrument and control sections.

*Denotes personnel attending exit interviews.

2. Licensee Actions on Previous Inspection Findings

(Open) Open Item (373/82-41-04): This open item documented problems encountered in NRC Regions IV and V with personnel air locks manufactured by Chicago Bridge and Iron. The problems related to the reliable functioning of door mechanical interlocks. A review of the licensee's test methods and the construction of the interlocks initially showed this not to be a problem at LaSalle. However, a recent event involving the Unit 1 drywell airlock supports the generic concerns noted before. The licensee is reviewing the incident and the inspectors will follow their actions.

(Closed) Open Item (373/80-05-13): This open item is based on the apparent conflict between 10 CFR 55 requirements for reactivity manipulations and the use of economic generating control. It has been forwarded to the Office of Nuclear Reactor Regulation for disposition as a generic issue.

(Closed) Open Item (373/80-24-14): This open item relates to final closure of IE Circular 80-10. This circular is being separately tracked in a circular file.

(Closed) Open Item (373/81-06-11): This open item documents the susceptibility of the steam condensing mode of RHR to single component failure and raises questions concerning compliance with 10 CFR 50, Appendix A Design Criteria. In consultation with the Office of Nuclear Reactor Regulation, it was determined that the LaSalle Accident Analysis does not take credit for the steam condensing mode. Thus, General Design Criteria requirements are not applicable. The licensee is reviewing the Final Safety Analysis Report to determine if changes are appropriate.

3. Operational Safety Verification

The inspectors observed control room operations, reviewed applicable logs, and conducted discussions with plant operators during the month of October 1982. The inspectors verified the operability of selected emergency systems, reviewed tagout records, and verified proper return to service of affected components. Tours of both reactor buildings and

turbine buildings were conducted to observe plant equipment conditions, fire hazards, fluid leaks, and excessive vibrations and to verify that maintenance requests had been expeditiously initiated and resolved for equipment in need of maintenance.

At 11:00 a.m. on October 19, 1982, the licensee telephonically notified NRC Region III of an item reportable pursuant to 10 CFR Part 50.55(e). As part of an ongoing program to verify that torquing requirements for safety-related components were satisfied, it was discovered that the manufacturer of the Emergency Core Cooling System (ECCS) pumps, Ingersoll-Rand, had changed the torquing requirements for pump holddown bolts without informing the licensee. The pump Technical Manuals had originally required that the holddown bolts be torqued to 2900-3000 ft-lbs. In March 1981, the requirement was changed to 400-500 ft-lbs.

The licensee checked several bolt torques on a Unit 2 low pressure core spray pump and a residual heat removal pump on October 19, 1982. The measured values fell in the range of 2000-2500 ft-lbs, a value consistent with an initial torque of 2900-3000 ft-lbs. This was confirmed by a review of pump installation documentation. It was presumed that the same situation existed for Unit 1 ECCS pumps.

Based on their initial evaluation of this item, the licensee determined that an applied torque of 2900-3000 ft-lbs could result in a bolt stress at or slightly in excess of the specified bolting material yield stress. However, the fact that the measured torques were consistent with the originally applied torques with allowance for normal relaxation, lead the licensee to conclude that little, if any, yielding had occurred and that the bolting configuration was acceptable.

A conference call was held at 2:00 p.m. on October 19, 1982, between members of the NRC Region III Division of Project and Resident Programs and Division of Engineering and Technical Programs, members of the licensee's Station Nuclear Engineering Department, the LaSalle Technical Support Group and Station Construction, and representatives of Sargent and Lundy. The purpose of the conference call was to discuss the details of the licensee's analysis that lead to the conclusion that the existing bolting configuration was acceptable and to specify any additional actions which may be necessary to support that conclusion.

The following information was discussed during the conference call. The vendor's original torquing requirements of 2900-3000 ft-lbs would produce a bolt stress of approximately 17,500 psi. The specified bolt material, high strength steel, has a yield stress of approximately 36,000 psi. The specified bolt diameter was 1 3/4 inches. The bolts actually installed are 1 7/8 inch diameter high strength steel. Calculations of actual bolt stress for a torque of 3000 ft-lbs yield 35,000 psi if a friction coefficient of .4 is used and 70,000 psi if a friction coefficient of .2 is used. Based on this information, it was concluded that the pump holddown bolts may have been torqued beyond their minimum yield stress. However, since measured torque values had not changed unexpectedly from initially installed values, it was concluded that if any yielding had occurred, it was minor and did not represent a challenge to bolt strength or integrity.

The NRC representatives accepted the licensee's conclusions on the information provided. However, the licensee was requested to provide details of their analysis in their forthcoming 10 CFR 50.55(e) written report, documenting the results of torque checks on all of the accessible holddown bolts on affected Unit 1 and Unit 2 ECCS pumps, and justifying for leaving the torques at the measured values as opposed to the vendor recommended values. The licensee agreed to provide this information. This item remains open (373/82-49-01).

On October 26 and again on October 28, Unit 1 experienced an automatic downshift of recirculation pumps from fast to slow speed. The downshift was initiated when feedwater flow fell below 30% due to a malfunction in the "A" turbine driven reactor feedwater pump control system. In both cases, the "B" recirculation pump failed to start on slow speed. This was attributed to a lockout circuit which prevents pump start when flow mismatch in the two recirculation loops exceeds a present value.

Following both events, the inspector monitored the plant recovery to pre-downshift conditions. On October 26, it was noted that the "B" recirculation flow control valve could not be repositioned following transfer of the "B" recirculation pump from slow to fast speed unless the valve was first moved slightly off its minimum position stop. It was surmised that the differential pressure across the valve when it is in the minimum stop position with the pump in fast speed produced a hydraulic lock which prevented valve movement.

The operating procedure for shifting recirculation pumps from slow speed to fast speed, LOP-RR-05, requires that the recirculation flow control valves be placed at their minimum position prior to the transfer. However, the procedure provides no guidance as to what constitutes minimum position. Given the problems encountered in the October 26 transfer, such guidance is warranted. This concern was relayed to the Unit 1 Operating Engineer who agreed to review the procedure and make appropriate changes.

On October 28, the shift from slow to fast recirculation pump speed was made with the "B" recirculation flow control valve slightly off its minimum position stop. The inspector noted that the pump transfer procedure had not been clarified. It was determined that written directions had been provided to the operators on positioning the "B" valve prior to transfer. The inspector again expressed concern over the lack of guidance in the normal operating procedure. The Unit 1 Operating Engineer stated that final procedural guidance was still being determined, but that the procedure would be changed to reflect the problems encountered. This item remains open (373/82-49-02).

During the course of the above events, the inspector noted a second problem. In order to prevent cavitation in the recirculation system there is an interlock which prevents placing recirculation pumps on fast speed with feedwater flow less than 30%. The procedural requirements for transferring recirculation pumps from slow to fast speed contained in LOP-R05 place the plant in a condition where, in preparation for the transfer, feed flow falls below 30%. Thus, in order to transfer the pumps from slow to fast speed it is necessary to bypass this interlock.

It is further necessary to bypass the function which signals the recirculation pumps to shift from fast to slow speed at less than 30% feed flow as feed flow does not instantaneously increase when the pumps are shifted to fast speed. The licensee is evaluating this situation in order to eliminate the conflict. A temporary procedure change has been processed to specify bypassing the interlocks. This item remains open (373/82-49-03).

The inspector, by observation and direct interview, verified that the physical security plan was being implemented in accordance with the station security plan, and that radiation protection controls were being implemented.

On October 14, the inspector monitored access control activities in the contractor access facility at the beginning of dayshift. No discrepancies were noted.

During the inspection period, the inspector noted that the vehicle access gates adjacent to the main access facility were open for an excessive percentage of the period. In all instances, proper control over the area was established. However, discussions with the Security Supervisor revealed that the subject gates had an excessive failure rate and that various modifications were being considered. Given this poor operating history, the inspector will track this as an open item (373/82-49-04).

No items of noncompliance or deviations were identified in this area.

4. Monthly Surveillance Observation

The inspector observed technical specification required surveillance testing on the Unit 1 Standby Gas Treatment System and the Reactor Vessel Low-Low Level MSIV Closure System and verified that testing was performed in accordance with adequate procedures, that test instrumentation was calibrated, that limiting conditions for operation were met, that removal and restoration of the affected components were accomplished, that test results conformed with Technical Specifications and procedure requirements and were reviewed by personnel other than the individual directing the test, and that any deficiencies identified during the testing were properly reviewed and resolved by appropriate management personnel.

The inspector noted that the performance of the Reactor Vessel Low-Low Level MSIV Closure Calibration and Functional Test was interrupted for approximately one hour. During this period, the subject instruments were left unattended and lockwires were not re-installed on the instrument valves. The inspector expressed a concern to the licensee that an interruption of a surveillance with the subject instrument test incomplete for an hour was too long. The licensee acknowledged this concern.

No items of noncompliance or deviations were identified in this area.

5. Followup on Part 21 Reports

(Closed) 82-01-PP-P: ITT Controls notified American Warming and Ventilating Inc. on February 5, 1980, that AH90 and NH90 series Hydromotor Electrohydraulic Actuators may not meet the specified safety factor with respect to thrust delivered to the damper assembly. An evaluation of dampers affected and corrective actions required was performed by Sargent and Lundy and transmitted to the licensee on November 26, 1980. M.C.C. Powers, the original installers of the dampers, performed the necessary rework and notified the licensee of completion in a letter dated October 20, 1981.

(Closed) Comsip, Inc. reported to the NRC on July 31, 1980 that a drive shaft on a Reliance IE motor driving a Dia-Vac diaphragm pump had failed. After a review by the licensee, it was determined that four motors at LaSalle, supplied by Comsip, Inc. contained the type of drive shaft that had failed.

New pump motors with improved shafts were installed by Morrison, Inc. in September 1981.

On October 21, 1982, the inspector verified that the pumps had been replaced by visually inspecting the affected panels. Visual verification was required because documentation of the replacement could not be located. The inspector expressed his concern for lack of documentation to the licensee.

(Closed) A 10 CFR Part 21 Report concerning GMC, model 645E4 series diesel engines was submitted by POWER Systems which stated that the original estimate of heat generated by the diesel was lower than actual heat generated. While LaSalle does use this model diesel for emergency power supplies, a heat balance was performed in November 1981 which showed the engine room air flow rates were adequate.

No items of noncompliance or deviations were identified in this area.

6. Licensee Event Reports Followup

Through direct observations, discussions with licensee personnel, and review of records, the following event reports were reviewed to determine that reportability requirements were fulfilled, immediate corrective action was accomplished, and corrective action to prevent recurrence had been accomplished in accordance with Technical Specifications.

50-373/82-108	Failure to Perform ADS Accumulator Backup N Surveillance
50-373/82-106	Failure of Primary Sample Containment Isolation Valve to Close
50-373/82-105	Failure of Primary Sample Containment Isolation Valve to Close in Required Time
50-373/82-110	Out of Specification Standby Liquid Control Tank Level
50-373/82-82	Out of Specification Setpoint of RHR Differential Pressure Switch
50-373/82-57	Inoperable RHR Service Water Process Radiation Monitor Pump

50-373/82-21	Failed Outside Air Radiation Monitor
50-373/82-22	Inoperable Control Room HVAC Emergency Make-Up Train
50-373/82-41	Failure to Sample With the Noble Gas Monitor Out-of-Service
50-373/82-121	Improper Ventilation System Hanger Installation
50-373/82-89	Overspeed Trip of the RCIC Turbine
50-373/82-25	Autostart of the Drywell Equipment Drain Sump Pump During ILRT
50-373/82-42	Reactor Scram Due to Improper Valve Lineup
50-373/82-99	Service Water Radiation Monitor Drift
50-373/82-91	Out-of-Tolerance RCIC Steam Flow Isolation Instrumentation
50-373/82-109	Out-of-Tolerance RCIC Pressure Isolation Instrumentation
50-373/82-113	Failed Fuel Pool Vent Exhaust Radiation Monitor*
50-373/82-104	Failed Off Gas System Hydrogen Analyzers
50-373/82-103	Failed RHR Service Water Radiation Monitor
50-373/82-93	RCIC Turbine Sightglass Oil Leak
50-373/82-85	Lubricating Oil in the Cylinders of 2A Diesel Generator
50-373/82-102	Out-of-Tolerance Reactor Vessel Level Scram Setpoint
50-373/82-114	Ventilation System Hanger Discrepancy
50-373/82-111	Failed Weld on Radwaste Boiler
50-373/82-77	Failure of RCIC Testable Check Valve to Close
50-373/82-78	Improper Snubber Installation
50-373/82-97	Failure of RCIC Testable Check Valve to Close
50-373/82-98	Improper Snubber Installation
50-373/82-100	Failure of RCIC Equalizing Valve to Close

*This report was submitted late. The initial determination was that the event was not reportable. This was changed during subsequent review.

No items of noncompliance or deviations were identified in this area.

7. Plant Trips/Safety System Challenges

At 11:50 a.m. on October 6, 1982, Unit 1 was manually scrammed from 22% power. The "B" turbine driven feedwater pump, which was supplying feedwater prior to the scram, tripped for no apparent reason. The motor driven feedwater pump failed to start automatically as designed.

Two unsuccessful attempts were made to start the motor driven feedwater pump and the reactor was manually scrammed prior to reaching the low level automatic scram setpoint. Reactor vessel level was restored and maintained by manual initiation and control of the reactor core isolation cooling system.

Subsequent investigations revealed no apparent cause for the trip of the turbine driven feedwater pump or the failure of the motor driven feedwater pump to start. At the time of the event, an instrument mechanic was performing fuse checks in the cabinet housing the breaker for the motor driven feedwater pump. However, it could not be established that these fuse checks precipitated the events leading to the scram.

During the course of the investigation, the licensee discovered that power to the low suction pressure trips for all three reactor feedwater pumps comes from a single source. With such a configuration, a failure

of the power supply could supply trip signals to all three pumps. The licensee is considering a modification to this circuit to eliminate this possibility (373/82-49-05).

Unit 1 remained shutdown until October 9 for maintenance activities and to allow the independent HVAC review group access to the Unit 1 drywell for inspections. The unit was taken critical at 10:15 p.m. on October 9.

No items of noncompliance or deviations were identified in this area.

8. Fire Protection/Prevention Program Implementation

The inspector examined the licensee's installed fire detection and suppression systems, manual fire fighting equipment, fire brigade training and administrative controls over combustible materials and ignition sources with respect to the requirements in the facility Technical Specifications and the Fire Protection/Prevention Program Implementing Procedures.

In the area of housekeeping and control of combustible materials, the following procedures were reviewed:

LAP-900-14, Revision 7, 3/12/82, Fire Protection Program
LAP-900-10, Revision 8, 5/07/82, Fire Prevention Procedure for Welding and Cutting
LAP-900-15, Revision 7, 3/18/82, Housekeeping Practices
LAP-900-16, Revision 3, 1/20/82, Fire Protection Impairment and Required Fire Watches

The periodic Fire Inspection Reports for August 8, 13, 19, 27, 1982 and September 3, 8, 16, 20, 23, 24, 1982 and the Monthly Housekeeping Reports for June 8, 1982 and August 5, 1982 were also reviewed.

The inspector examined combustible and ignition source controls during tours of Unit 1 and 2 reactor buildings, the auxiliary building, the radwaste building, the turbine buildings, the river screenhouse building, the plant yard, and lake screen house.

The inspector also observed maintenance and work request activities to verify proper implementation of fire protection/prevention controls. During a tour of Unit 1 reactor building some combustible material was found stored next to a safety-related motor control center (MCC 133-2, equipment number 1AP62E) at the 740' level. This condition was discussed with the Fire Marshall, who had the material immediately moved to a safer storage site.

During a tour of the lake screen house on October 6, 1982, a substantial amount of oil absorbent material and paper toweling was discovered in the "A" and "B" diesel fire pump rooms. Further investigation revealed that an internal licensee letter was written on September 28, 1982, asking for timely removal of the material. Subsequent discussions with the Station Fire Marshall resolved this issue. The absorbent was laid down to soak up diesel fuel which had run onto the floor as a result of

a small fuel line leak and a storage tank overflow. The material was left on the floor in an effort to absorb as much of the fuel as possible from surface pores in the concrete. The material was expeditiously removed when it was discovered by station fire protection personnel.

The inspector reviewed the fire brigade training program as referenced in Procedures LAP-900-14, Revision 7, March 12, 1982, Fire Protection Program and LAP-900-17, Revision 3, March 7, 1982, Fire Drills. Fire brigade training attendance rosters for February 19, 26, 1982, March 5, 12, 19, 26, 1982; May 14, 21, 28, 1982; June 4, 11, 1982; August 6, 13, 20, 27, 1982; and September 3, 10, 1982 were reviewed. In addition, fire drill critique and attendance sheets for July 20, 1982, August 30, 1982, and September 23, 1982 were reviewed.

The inspector observed an unannounced fire drill on October 1, 1982. The response was timely and the individual fire brigade members demonstrated excellent training in their duties. The drill was immediately critiqued by the Fire Marshall. No deficiencies were noted.

The inspector reviewed the status of manual fire fighting equipment with respect to implementation of the following procedures:

LMS-FP-9, Revision 0, January 18, 1982, Annual Maintenance of AFFF Fire Extinguishers
LMP-FP-10, Revision 2, January 20, 1982, Monthly Fire Inspection
LMS-FP-3, Revision 0, March 1, 1979, Monthly Inspection of the Yard Loop Fire Hose Stations
LMS-FP-5, Revision 1, January 20, 1982, Annual Inspection, Maintenance, and Weight Check of Portable Fire Extinguishers
LMS-FP-06, Revision 1, January 18, 1982, Fire Hose Station Valve Operability and Flow Verification
LMS-FP-08, Revision 2, August 2, 1982, Fire Hose Station Annual Hose Inspection, Rerack, and Gasket Inspection
LMS-FP-12, Revision C, Diesel Fire Pump Surveillance

The only discrepancy noted was that welding and cutting cart portable fire extinguishers do not receive an annual inspection. These extinguishers are hand pump water extinguishers. The Station Fire Marshall stated that this type of extinguisher does not require an annual inspection.

The inspector examined manual fire fighting equipment during tours of the Unit 1 and Unit 2 reactor buildings, the auxiliary building, the radwaste building, the turbine building, the lake and river screen house buildings, and the plant yard.

For Fire Detection/Suppression Systems, the inspector reviewed the implementation of the following procedures:

LOS-FP-A2, Revision 2, February 5, 1982, Fire Protection Functional Test
LOS-FP-A3, Revision 2, July 9, 1982, Fire Protection Sprinkler and Deluge System Drain Flow and Cycling Test
LOS-CO-M1, Revision 2, March 25, 1982, Combined CO Fire Protection Flow Path Valve Position Check Sign-Off Sheet

LOS-FP-M1, Revision 2, February 4, 1982, Fuel Transfer Test to "A" Diesel Fire Pump Day Tank and Fuel Storage and Day Tank Level Check
LOS-FP-M3, Revision 4, June 17, 1982, Unit 1 and Common Fire Protection Flow Path Valve Position Check Sign-Off Sheet
LOS-FP-M4, Revision 5, April 5, 1982, Fire Protection Sprinkler and Deluge System Valve Lineup and Alarm Check
LOS-FP-W2, Revision 5, August 4, 1982, Diesel Fire Pump "A" OFP01KA Weekly Operational Check
LOS-FP-W2, Revision 5, August 5, 1982, Diesel Fire Pump "B" OFP01KB Weekly Operational Check
LMS-FP-1, Revision 2, February 20, 1981, Central File Halon 1301 Fire Protection System, Semi-Annual Inspection
LSF-172(A), Revision 3, August 27, 1982, Daily Fire Door Check
LSF-172(B), Revision 2, August 27, 1982, Weekly Fire Door Check

The inspector examined Fire Detection/Suppression Systems during tours of Unit 1 and Unit 2 reactor buildings, the auxiliary building, the radwaste building, the turbine building, the river and lake screen houses, and the plant yard.

No items of noncompliance or deviations were identified in this area.

9. Followup of IE Information Notices

For the IE Information Notices listed below, the inspector verified that the Information Notice was received by the licensee, that a review for applicability was performed, and that if the Information Notice was applicable to the facility, appropriate corrective actions were taken or were scheduled to be taken.

82-37 Cracking in the upper shell to transition cone girth weld of a steam generator at an operating pressurized water reactor

82-38 Change in format and distribution system for IE Bulletins, Circulars, and Information Notices

82-41 Failure of safety/relief valves to open at a BWR

80-34 Defective welds in safety-related panels

No items of noncompliance or deviations were identified in this area.

10. Pressure Isolation Leak Testing

In April 1981, the Office of Nuclear Reactor Regulation issued license modification orders to operating plants requiring that certain Reactor Coolant System pressure isolation valves be tested to verify their leak tightness. Of particular concern were system configurations where a check valve(s) provided isolation for a system connected to the Reactor Coolant System where the system contained low pressure piping.

At the time the order was issued, LaSalle County Station did not have an operating license. In an effort to determine that the concerns expressed in the order were adequately addressed at LaSalle, the inspector reviewed the licensee's Reactor Coolant System Pressure Isolation Valve Leakage Detection Program. The following observations were made. The requirements promulgated in the April 1981 order are incorporated in LaSalle Unit 1 Technical Specification 3.4.3.2. The licensee has approved procedures for testing the values of concern. These procedures are being appropriately implemented based on a review of completed test procedures. This item is considered closed.

No items of noncompliance or deviations were identified in this area.

11. Independent Inspection Effort

During a plant tour on October 1, 1982, the inspector observed two contractor personnel (Foley) using a fire hose station to fill a container for core drilling. When questioned, the men admitted they had not been told they could use that hose station for that purpose. The site Fire Marshall was informed. He, in turn, contacted the construction Fire Marshall who counseled both the individuals involved and their supervisor on the purposes of the fire hose station.

License Conditions 2.C.(30).(0).(i) and (ii) to NRC Operating License No. NPF-11 requires that the licensee's Shutdown Parameter Display System (SPDS) and Emergency Operations Facility (EOF) be fully functional no later than October 1, 1982. On September 21, 1982, the licensee submitted a request to the Office of Nuclear Reactor Regulation to defer the required completion dates to December 31, 1982. The basis for this request was additional time was required for adjustments to and training on the SPDS and training in the use of the EOF.

On October 1, 1982, the Office of Nuclear Reactor Regulation informed the inspector that while the request for deferral of the completion dates was acceptable, it would be impossible to admimistratively process the deferral before close of business and that this would leave the licensee technically in violation of license Conditions 2.C.(30).(0).(i) and (ii). The inspector discussed this situation with NRC Region III management and, with the consent of the Office of Nuclear Reactor Regulation, reached an agreement that enforcement action was inappropriate. This agreement was predicated on the fact that the licensee's submittal and justification were acceptable and that formal relief had not been granted on or before October 1, 1982 simply because the request could not be administratively processed in time.

On October 1, 1982, the Office of Nuclear Reactor Regulation Project Manager for LaSalle County Station informed the inspector that the submittals required by Conditions 2.C.(16) and 2.C.(30).(1).(a) of NRC Operating License NPF-11 had been received and based on this, those license conditions could be considered satisfied. This closes open item 81-00-116.

No items of noncompliance or deviations were identified in this area.

12. Exit Interview

The inspector met with licensee representatives (denoted in Paragraph 1) throughout the month and at the conclusion of the inspection period summarized the scope and findings of the inspection activities. The licensee acknowledged these findings.