U.S. NUCLEAR REGULATORY COMMISSION
REGION I

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Report No.: 50-293/89-10

Licensee: Boston Edison Company
800 Boylston Street
Boston, Massachusetts 02199

Facility: Pilgrim Nuclear Power Station

Location: Plymouth, Massachusetts

Dates: August 22 - October 1, 1989

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Approved by: for A. Randy Blough, Chief, Reactor Projects Section 3A 11/15/89 Date

Inspection Summary:

Areas Inspected: Restart staff inspection to assess licensee management controls, conduct of operations and the licensee's approach to investigation and resolution of events during the 75% power plateau of the licensee's Pilgrim Power Ascension Test Program.

Results: Management involvement in planning and support of maintenance activities was evident in efforts to clean the RBCCW heat exchanger (Section 2.3.5). The licensee's performance of the Electrical and Mechanical Pressure Regulator Surveillance at the end of the 75% power plateau was extremely precise from start to finish, performance was excellent and demonstrated professionalism and attention-to-detail (Section 3.2). The licensee's conduct of critiques to determine the root causes of events has been very thorough, timely and demonstrates management's commitment to identify and correct potential problems (Sections 5.1 and 5.2). The placing of an ALARA hold on hot shop work during Reactor Water Cleanup pump repair work demonstrated the licensee's conservative attitude and determination to improve radiological conditions at the plant (Section 5.3).

There were instances where the licensee was not sufficiently aggressive or proactive in reviewing safety concerns. In particular, following the licensee identification that the spent fuel pool cooling pump holddown bolts had been filed down to less than nominal diameter (Section 6.3), NRC prompting was required before the licensee took action to look at the generic implications of the event as it may have related to other safety-related equipment. However, after questioning by the NRC Restart Staff, the licensee performed a well-defined and thorough engineering analysis of the events. Subsequent corrective actions were thorough.

A repeat occurrence of a locked high radiation area door being left unsecured and unattended was identified by the licensee. Problems with high radiation area access control have been previously identified and cited; corrective actions taken in response to these findings have not prevented their recurrence (Section 5.2, VIO 89-10-01). Also, due to failure to comply with existing procedures and poor communication practices, a miscellaneous tank with a recorded activity level in excess of technical specification limits was discharged to the environment. These records were later found to be in error and indicated no limits were violated. Pending licensee and NRC review of corrective actions taken to prevent recurrence of this event, including the adequacy of the licensee's procedures, adequacy and determination of the need for qualification of the Chemistry computer program and overall control of station discharges, this item is unresolved (Section 5.1, UNR 89-10-02).

As evidenced by the overtorquing of the holddown bolts on a sampling of safety-related pumps and motors (Section 4.3), the discharge of a miscellaneous tank (Section 5.1), performance of maintenance on the "D" Salt Service Water pump (Section 4.2) and with respect to the unlatched locked high radiation area door (Section 5.2), an area of weakness continues to be failure to adhere to approved station procedures.

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DETAILS

1.0 Summary of Facility Activities

At the beginning of this report period, the plant was operating at approximately 75% power while the licensee was performing testing required by the Power Ascension Test Program at the 50-75% power ascension plateau.

On August 24, 1989, the licensee completed the original testing scope of the Power Ascension Test Program up to the 75% power plateau.

At 7:16 p.m. on August 30, 1989 the reactor automatically scrammed from about 75% power due to a failure of the main generator's voltage regulation circuitry. Safety-related systems responded as designed. A failed potential transformer that feeds the main generator voltage regulator caused a turbine runback and the reactor scrammed on reactor vessel high pressure.

On September 5, 1989 at 6:05 p.m. with the reactor in the cold shutdown condition, the licensee experienced an inadvertent actuation of a portion of the Residual Heat Removal (RHR) System/Low Pressure Coolant Injection loop selection logic circuitry. This resulted in an automatic start of the "A" Emergency Diesel Generator and the cycling of several RHR system valves. Details are provided in Section 2.3.2.

At 10:16 p.m. on September 6, 1989 the licensee brought the reactor critical. The turbine generator was synchronized to the grid at 10:14 a.m. on September 7, 1989. The plant reached 75% power on September 8, 1989 and the plant was at 75% power at the close of this report period. The licensee continued to experience degraded condenser vacuum due to fouling of the condenser tubes with marsh grass. Due to the degraded condenser vacuum, the licensee has periodically operated the reactor at reduced power levels in order to perform backwashing of the condenser water boxes.

On September 20, 1989 the licensee conducted a dry run emergency preparedness drill in preparation for the scheduled full scale emergency preparedness exercise on October 12-13, 1989. The dry run included site evacuation and accountability of station personnel and activation of emergency response facilities.

NRC inspection activities during this report period were conducted by the onsite Pilgrim Restart Staff led by Mr. Charles S. Marschall, Senior Resident Inspector and Restart Manager. The Pilgrim Restart Staff is composed of the Pilgrim resident inspectors, NRC regional-based and headquarters-based inspectors and an NRC contractor. On August 31 and September 1, 1989, Mr. Jon Johnson, Chief, Projects Branch No. 3 was onsite to tour the plant. On September 20, 1989 two Region I Emergency Preparedness inspectors were in the Plymouth area to observe the licensee's dry run emergency preparedness exercise.

At the beginning of this report period with the plant at 75% power, the Restart Staff was in around-the-clock shift coverage observing the licensee's power ascension testing. Around-the-clock shift coverage was discontinued at 6:30 p.m. on August 24, 1989, consistent with reduced testing activity. The Restart Staff maintained extended shift coverage throughout the remainder of this inspection period.

2.0 Operations

2.1 Sustained Control Room Observations

Control room activities were routinely observed during this inspection period to ensure control room staffing was maintained, access to the control room was controlled and operator behavior was appropriate to plant configuration and activities in progress.

Shift turnovers typically included a briefing for operations, maintenance, instrumentation and control, and radiological protection personnel. While briefings were held at each shift turnover with very few exceptions, the quality of briefings varied from one shift to another. Some shift briefings were thorough while others were perfunctory; on some occasions background conversations and control room panel alarm checks contributed to an already high background noise level, causing briefings to be inaudible. These instances were isolated, however, and in general, shift briefings were indicative of a professional control room atmosphere.

On two occasions in responding to inspector questions, the Nuclear Watch Engineers (NWE) expressed frustration with the limitations placed on their authority. In one instance regarding appropriateness of the method used to lock valves (i.e., the use of lead wire seal to attach a chain to the valve handwheel), the NWE indicated the method used had been reviewed by management and was therefore acceptable. In another instance, the NWE implied management guidance on operation of the recirculation pumps restricted his ability to interpret Technical Specifications (see discussion below). In each case, the NWE's gave further consideration to the questions raised by the NRC, and resolved the issues in an appropriate and conservative manner. When these instances were addressed with plant management, the Plant Manager conducted interviews with Senior Reactor Operator licensed personnel to ensure the responsibilities associated with an SRO license were understood and appreciated.

In one case the inspector questioned the licensee's position on Technical Specification 3.6.F.1 limits for mismatch in recirculation pump speeds. Mismatch is limited for reactor power greater than 80% to less than 10%, and for reactor power equal to or less than 80% to less than a 15% On August 30, 1989, based on questions raised by the inspector, the Station Director requested that the Nuclear Engineering Division (NED) provide an interpretation of Technical Specification 3.6.F.1. to clarify whether the mismatch referred to a percentage of full rated pump speed or actual pump speed. The response from NED and the supporting technical evaluation supplied by General Electric, concluded Technical Specification 3.6.F.1 referred to rated recirculation pump speed. This result agreed with the licensee's original understanding of the specification. The technical evaluation addressed core flow coastdown and Low Pressure Coolant Injection (LPCI) loop selection restrictions. The technical evaluation was thorough, sound, and supported the conclusion discussed above. The licensee noted, however, that the present Technical Specification basis may be incomplete, since it does not address the core flow coastdown effect which leads to the most limiting criterion for the mismatch. The licensee will consider a change to the Technical Specification basis for recirculation pump speed mismatch to include core flow coastdown.

2.2 Plant Tour Observations

The inspectors routinely conducted plant tours and noted that in general, plant cleanliness remained a licensee strength. The cleanliness of the plant illustrates that management's attention to plant conditions remains high.

2.3 Review of Plant Events

2.3.1 Plant Electrical Transient Due to a Failed Potential Transformer

Event Description

On August 30, 1989, a reactor scram occurred from about 75% reactor power. Control room personnel received numerous alarms on the electrical control panel. Immediate investigation revealed that the initiating event was a blown fuse of the potential transformer that feeds the voltage regulator. This failure resulted in demand for a higher generator excitation voltage. Accordingly, megavaras, voltage and current on the generator increased. High current caused generator current/stator cooling water flow mismatch to be greater than 15%, initiating a turbine runback. When the speed/load changer decreased, the turbine control valves closed and the turbine bypass valves opened. Although the reactor operator reduced recirculation flow in an attempt to reduce reactor power and pressure, reactor pressure increased to 1069 psig as steam flow exceeded the bypass valve capacity and the reactor scrammed on high pressure. Scram recovery was routine and all safety systems responded as designed.

Items reviewed during this inspection included the sequence of events log, plant response using EPIC traces, alarm typer printouts computer data from the electrical grid controller, and Rhode Island-Eastern Massachusetts-Vermont Energy Control (REMVEC). Several functional and operational procedures and schematic drawings were also reviewed and discussed to assess the licensee's evaluation of the event.

Evaluation of the Electrical Events

The licensee identified the specific root cause of this event to be hardware failure as well as incorrect wiring of one relay. The licensee's review indicated that a 24000/120 V potential transformer (X2) in the generator excitation/voltage regulator system failed and the primary side fuse blew, possibly due to aging. The design included a voltage balance relay (No. 260), GE Model CFVB, in the generator voltage regulator protection scheme to block other relays or protective devices that will operate incorrectly and to transfer the generator voltage regulator control from auto to manual when a potential transformer fuse blows.

This event should have resulted in the operation of the 260X2 auxiliary relay of the balance relay to switch the manual/auto voltage regulator control scheme from auto to manual mode to allow time for operator intervention. However, due to a wiring error that existed since the original manufacture of the potential transformer, relay contacts were swapped. During licensee review, the wiring scheme for the voltage balance relay and the as-built conditions were found to be inconsistent with the vendor wiring diagram. This deficiency prevented the voltage balance relay from switching the voltage regulator from auto to manual when an unbalance was sensed. The potential transformer failure resulted in a loss of signal to the voltage regulator which in turn increased the excitation voltage. This increase in excitation voltage caused the generator to be in an over-excited condition. Therefore, the generator output voltage and reactive current and power increased accordingly.

A contributing cause was lack of adequate functional testing of the generator voltage balance relay. Procedure 3.M.3-39, "Turbine/Generator Lockout Test and Associated Annunciator Verification," had been used only to verify the balance relay alarm function rather than individually verifying the function of relays 260/X1 and 260/X2. A complete functional test of these relays could have identified the wiring discrepancy. However, the functional testing of the relays is not a required test.

Overvoltage During Transient

The inspector reviewed the licensee's evaluation to verify the impact of overvoltage on electrical equipment, especially on the 4.16 KV safety bus and it's related loads. During the transient, the generator output voltage increased from 24350 V to approximately 27974 V. The duration of the transient, based on REMVEC computer data was approximately 35-40 seconds. The licensee did not have saturation curves for the unit auxiliary transformer to determine the voltage levels at the secondary side of the transformer. In addition, there are no permanent recorders or overvoltage relays installed to monitor the bus voltage conditions. Therefore, in order to find the maximum voltage increase in the lower voltage busses, the Nuclear Engineering Department (NED) utilized the fact that the Electrical Protection Assembly (EPA) breakers (EPA5 and 6) will trip on an overvoltage condition over their nominal 120 VAC supply. A review of the calibration records, prompted by the NRC, verified that the overvoltage trip set point is within the allowed maximum of 132 V. The EPA breaker assemblies were energized during the overvoltage event and did not trip. Assuming the worst case condition that none of the transformers were in saturation except for EPA transformer X20 and no voltage drop existed between the busses, the maximum voltage transient on the system was calculated to be approximately 15-16% above the normal voltage at the 4.16 kV and 480 V busses.

The licensee's evaluation shows that the unit auxiliary transformer did saturate at the primary input voltage of 122% above the nominal voltage because the EPA breakers did not trip and its transformer (X20) did not undergo saturation. The licensee stated that, based on the information received from the vendor, the transformer saturates when only 15% above the nominal voltage. The evaluation shows that the maximum voltage on a 480 V bus and 4160 V bus were about 6.25% and 9% above the normal voltage, respectively. The 4160 V bus, breaker and cables are rated for 5000 volts and 480 V bus and cables are rated for 600 V. The overvoltage level was below this rating.

In order to ensure that no damage was experienced at the 4.16 kV safety busses, the control rod drive (CRD) motor operating during the event was subjected to a polarization index test to ensure insulation integrity. The two 480 V motors that were running during the event (the turbine building component cooling water pump; and reactor building component cooling water pump) were also subjected to a polarization index test. The polarization index tests showed no evidence of degradation and the tests were completed satisfactorily.

The licensee performed random checks on instrument loops to assure that the transient did not adversely affect plant instrumentation. A check of several recorder traces showed that these traces were steady and did not shift during the transient.

The licensee concluded that due to equipment supply voltage tolerances; the relatively short time the high voltage condition existed; and the probable saturation of the unit auxiliary transformer (which provided, in effect, a protective function to the lower voltage safety buses), the overvoltage condition was not detrimental to safety equipment operability. No unacceptable conditions were identified.

The licensee took the following corrective actions prior to plant restart: (1) insulation check of the generator and exciter fields; (2) the wiring discrepancy for the voltage balance relay was corrected, both in the field and on the wiring diagram (Drawing No. E47, Revision E8); (3) replacement of the potential transformer and fuse; (4) performance of a Doble test and dissolved gas-in-oil analysis for the main transformer to verify the integrity of the transformer; (5) as stated above, the operating motor from the 4160 V safety-related bus and two motors from the 480 V safety-related bus were polarization index tested to confirm that no damage was done to the equipment; (6) functional testing of generator protective relaying; (7) functional test procedure 3.M.3-39 was revised to ensure the proper functioning of the voltage balance relay and appropriate protective devices; and, (8) dissolved gas-in-oil analysis was performed for the unit auxiliary transformer. Test results showed no degradation of the transformer. This transformer was originally factory tested at voltage levels 200% of rated voltage at 120 cycles for one minute. Therefore the transient voltage during this event was of no significant concern.

The wiring discrepancy identified as a result of this event prohibited the automatic transfer of the voltage regulator to manual upon loss of a potential transformer. The licensee stated that this condition existed since initial plant startup. They also stated that they are planning to notify other utilities to verify the wiring condition on the voltage balance relay to prevent similar occurrence. The potential for the problem to exist in other locations at the plant was investigated. The licensee determined that this type of relay is also used in two emergency diesel generator circuits. However, they only perform an alarm function and no adverse conditions exist in the wiring.

In summary, the licensee performed a thorough engineering analysis of the cause and subsequent effects on safety-related components due to the overvoltage transient. However, there was a disconnect between the licensee's engineering department and onsite personnel, in that, onsite personnel were unaware of the scope and depth of the engineering analysis performed by the Nuclear Engineering Department. This appeared to be caused by a lack of communications. Additionally, NRC prompting was required for the licensee to look at surveillance records on the EPA breakers. Corrective actions were well defined and thorough. The overvoltage condition that existed during the event was not detrimental to the equipment operability. Control room operators responded appropriately and the plant technical support staff responded effectively to identify and correct adverse conditions prior to returning the plant to operation.

2.3.2 Inadvertent Actuation of Low Pressure Coolant Injection Loop Selection Logic

On September 5, 1989 at 6:05 p.m. with the reactor in the cold shutdown condition, the licensee experienced an inadvertent actuation of a portion of the Residual Heat Removal System (RHRS)/Low Pressure Coolant Injection (LPCI) loop selection logic circuitry due to a personnel error during surveillance testing. The actuation resulted in an automatic start of the "A" Emergency Diesel Generator (EDG), and automatic closing of the Recirculation System Loop "B" pump discharge valve (MO-202-5B) and the RHRS/LPCI Loop "A" injection valve (MO-1001-29A). In addition, the licensee experienced automatic opening of the RHRS/LPCI Loop "B" injection valve (MO-1001-29B) and the RHRS Loop "A" heat exchanger bypass valve (MO-1001-16A). No injection into the reactor vessel occurred.

Pending further investigation, the licensee suspended conduct of the surveillance and restored the circuitry to normal. The licensee determined that prior to the actuation, Instrumentation and Control (I&C) technicians were performing surveillance test (TP) 88-78, "General Electric CR 2820 Time Delay Relays for Automatic Depressurization System (ADS), Core Spray (CS), Residual Heat Removal (RHR) and Reactor Protection Systems (RPS)." Step 2.6 of Attachment 2 requires the technician to install the jumper across contacts 1 and 2 of relay 14A-K10A (Panel 932). The jumper is installed in order to record the time delay for Core Spray System time delay relay 14A-K14A. Instead, the jumper was inadvertently placed across contacts 1 and 2 of relay 10A-K10A in Panel 932. The root cause of the placement of the jumpers across the wrong relay was attributed to the I&C technician noting the K10A part of the relay number on the relay but not observing the first half of the relay number. A contributing cause may have been the internal lighting within the panels. While the lighting outside of the panels (general area) was adequate and the technicians had flashlights, the light fixtures (inside at the top of the panels) were not working at the time the surveillance was performed.

As corrective action, the licensee revised the test procedure to include double verification for those procedure steps that involved the installation of a jumper or insulating boot. At the time, the procedure included double verification for the removal of jumpers. The licensee's corrective actions were appropriate.

2.3.3 <u>Inoperable High Pressure Coolant Injection System</u>

On September 7, 1989, the High Pressure Coolant Injection (HPCI) system was declared inoperable and a seven-day Technical Specification (3.5.C.2) limiting condition for operation (LCO) was entered at 6:55 p.m. The system was declared inoperable due to a mechanical overspeed trip which occurred during a surveillance test.

The cause for the mechanical overspeed trip was failure of the ramp generator signal converter (RGSC) module that is part of the HPCI turbine speed control system. The failed RGSC module was removed and sent to the manufacturer for additional testing to determine the cause of the failure. A new module was installed and the turbine speed control system was calibrated. The HPCI system was tested for operability with satisfactory results and the LCO was cleared on September 9, 1989 at 8:23 p.m. While the HPCI system was inoperable, testing of applicable systems required by Technical Specification, including the Reactor Core Isolation Cooling System, Automatic Depressurization System, Low Pressure Coolant Injection System and Core Spray System was conducted with satisfactory results. The licensee's thorough analysis in determining the root cause led to appropriate corrective action.

2.3.4 Breach of Secondary Containment

On September 7, 1989 the licensee identified that both reactor building airlock doors were inadvertently opened simultaneously for two to three seconds, resulting in a momentary breach of secondary containment integrity. The normal electrical interlock, preventing both doors from being opened at the same time, was not in service and security guards were posted as a compensatory measure at both the inner and outer doors. However, the guards failed to prevent the coincident ingress/egress of two persons leaving and entering the airlock. The NRC was notified at 9:25 p.m. via ENS.

This event occurred during the changing of posts by the security guards posted as compensatory measures to maintain secondary containment. The licensee identified the root cause of this event to be the failure of the Security Post order to adequately address the method that the security guards change their post, which occurs every half hour. The Post order addressed personnel egress/ingress but failed to address specifically the changeout of security guards. The licensee has revised the compensatory procedures (post order) to address, in detail, the change of post by those guards serving as a compensatory measure for the airlock doors. Although revising the post order to detail the method in which the guards are to change posts may alleviate future personnel error in this area, repair to the airlock doors to eliminate the need for compensatory measures is planned for the October mini-outage. Interim corrective actions of modifying the post orders have been observed to be satisfactory.

2.3.5 Reactor Building Component Cooling Water (RBCCW) Heat Exchanger Surveillance

On September 26, 1989, the licensee entered a Technical Specification limiting condition for operation (LCO) when the "A" RBCCW heat exchanger failed to meet surveillance acceptance criteria for service water flow, and the "D" Residual Heat Removal (RHR) pump failed to meet requirements for discharge head and flow. The simultaneous surveillance failures rendered both trains of containment cooling inoperable. The LCO required one train of containment cooling be restored to an operable status or the plant be placed in cold shutdown within 24 hours. The licensee simultaneously pursued manual cleaning of the heat exchanger and re-calibration of the flow transmitter used to perform the RHR surveillance.

Although mechanical maintenance personnel removed one head of the RBCCW heat exchanger and extracted a large number of mussels in less than 12 hours, the subsequent flow test did not prove successful. Subsequently, successful results for the RHR surveillance were obtained using redundant flow instrumentation in the RHR system.

As a result of the successful RHR surveillance, the licensee continued their actions under the 7-day LCO for the RBCCW heat exchanger. The licensee re-opened the heat exchanger, cleaned the tubes and again performed the RBCCW flow surveillance with negative results. Trouble shooting activities by control room personnel indicated the flow indicator used for the RBCCW surveillance had been fouled by biological growth. Subsequent measurements of RBCCW differential pressure supported by engineering analysis were used to demonstrate adequate flow existed through the heat exchanger, and the RBCCW system was declared operable. The engineering analysis used for the surveillance will be reviewed during a future inspection.

Maintenance efforts to clean the RBCCW heat exchanger were controlled and coordinated well and were indicative of management involvement in planning and support of maintenance activities. However, control room personnel backwashed the RBCCW heat exchanger using a normal operating procedure, whereas the intent was troubleshooting to identify the cause of the failed RBCCW surveillance. Licensee management committed to provide clear guidance to control room personnel on identification and review of trouble shooting activities.

2.4 ESF System Walkdown

The inspector performed a walkdown of the accessible portions of the Reactor Core Isolation Cooling (RCIC) System, Procedure 2.2.22, "Reactor Core Isolation Cooling System," Revision 32, to verify its operability. The inspector reviewed the licensee's current completed valve lineup for the system, then did an independent walkdown and verification of selected valves. This independent verification was performed with the aid of a nuclear plant operator who physically manipulated valves during the walkdown.

The inspector verified correct breaker positions for the RCIC system valves, and compared the procedure to the plant process and instrument drawings (M245 and M246) to confirm that the system procedure matches the drawings and the as-built configurations.

Licensee administrative controls were adequate to ensure operability of the RCIC system.

3.0 Surveillances

3.1 Routine Surveillance Tests

The following surveillance tests were witnessed or reviewed during this inspection period.

-- Procedure 8.5.3.2, "Salt Service Water Capability and Operability Tests," observed September 14, 1989

-- Procedure 8.E.10, "LPCI System Instrument Calibration," reviewed September 26, 1989.

Administrative controls of surveillance activities to ensure compliance with Technical Specifications were considered adequate.

3.2 Electric and Mechanical Pressure Regulator Surveillance (EPR and MPR)

On August 24, 1989 the licensee performed surveillances on the EPR and MPR and turbine thrust bearing clearances. The test was preceded by a detailed pre-evolution briefing conducted by the Nuclear Watch Engineer (NWE). The NWE covered the contents of the procedure, the special characteristics of all controls and indications, and assignments to operators of emergency actions should any problems be encountered. The control room was cleared of unnecessary personnel and those remaining were ordered to remain quiet and clear of the operation. The Nuclear Operations Supervisor (NOS) read each step of the procedure and directed the actions of the control board reactor operator (reader-worker method). The reactor operator's manipulation of the sensitive turbine pressure regulators throughout the 20-minute test was extremely precise. Data was properly recorded and all administrative requirements of the procedure completed without error.

The licensee's performance of these surveillances from start to finish, was excellent and demonstrated professionalism and attention-to-detail.

4.0 Maintenance and Modifications

4.1 Recurring Feed Pump Auxiliary Oil Pump Cycling Problems

During observation of control room activities and shift turnover briefings it was noted that there have been recurring problems with the reactor feed pump auxiliary oil pumps. Examples of problems which have occurred in the last six months include: (1) the auxiliary oil pump had cycled on and off after the feed pump was tripped, (2) the auxiliary oil pump continued to run after the feed pump was started, then the feed pump tripped on low oil pressure after operators manually secured the auxiliary oil pump, and (3) auxiliary oil pumps cycling on and off with the feed pump running. The auxiliary oil pump is designed to start following the stopping of a feed pump as the lubricating oil pressure decreases due to the coast down of the attached shaft driven oil pump. The feed pumps and associated equipment are nonsafety-related; however, they are "important-to-safety" in that faulty operation can lead to plant transients and trips.

Erratic operation of the auxiliary oil pumps has been a long-standing problem and several auxiliary oil pump motors had been replaced due to failure, apparently related to abnormal operation of the system, over the past several years. In addition, the auxiliary oil pumps had been frequently run in manual to reduce cycling and the possibility of feed pump trips. Review of the maintenance history revealed that fourteen maintenance work requests (MWRs) had been written on auxiliary oil pumps since February 1989.

System engineers learned while researching the inspector's concerns that the installed pressure switches controlling the auxiliary oil pumps were not the correct model. Due to problems with the RFP oil system since construction, a site initiated Plant Design Change (PDC) had been installed in early 1980. The plant design change added an additional flow control relief valve and changed the auxiliary oil pump pressure switch to one with a greater dead-band between the set pressure and reset pressure. However, the pressure switch, plant drawings, and Instrumentation and Control calibration cards were apparently not changed as required by the PDC. Post modification testing that would have identified the error was apparently not conducted.

System engineering personnel have responded to the above concerns with extensive research. Corrective actions include the following: (1) duplicative maintenance work requests have been cancelled, leaving only three active MWRs on the auxiliary oil pumps, (2) a new pressure switch vendor manual has been ordered, and (3) calibration cards have been reviewed and I&C technicians interviewed to determine that the switches are set with the widest possible dead-bands (about 4 psid). The feed pump oil system switches have been inspected to determine the exact model of pressure switches installed. Based on that inspection, the licensee found four of the pressure switches on the RFP lube oil skids have incorrect cover plates and therefore, there is no assurance the correct switches are installed. This will require the switches to be replaced. The licensee has written an Engineering Service Request (ESR) to engineering to review the application of the switches and make a recommendation as appropriate.

The probable cause of the auxiliary oil pump cycling was inadequate implementation of the old PDC and inadequate post modification testing that would have identified the pressure switch problem. This may have been attributed to the fact that at the time, review of balance-of-plant equipment may have been less that that for safety-related equipment. Modification program changes and management controls instituted since the modification should preclude recurrence of similar problems.

Based on the number and age of maintenance requests on the RFP auxiliary oil pumps, the inspector concluded that management oversight and supervisory review of this long term equipment problem in solving balance of plant equipment, could be improved by more aggressive action on the part of the licensee. The problems, although not directly safety significant, do relate to reliability and unit operability. The results of the licensee's corrective actions will be reviewed during a future inspection.

4.2 Maintenance on "D" Salt Service Water (SSW) Pump

On September 12, 1989 the inspector observed two licensee attempts to measure vibration levels at the upper motor bearing of the "D" Salt Service Water (SSW) pump after extensive repairs. The vibration test results would be used to determine if the pump was ready for post-work testing.

The first of two vibration tests was performed using procedure 3.M.1-15, "Vibration Monitoring for Preventative Maintenance and Balancing," at low tide with net positive suction head (NPSH) at a minimum. The test failed, however, when one of the measured horizontal values read 0.40 in/sec which exceeded the "Alert" value of 0.314 in/sec. When the licensee repeated the test at high tide later that day, the same horizontal value was measured at 0.50 in/sec. Until questioned by the inspector, maintenance personnel apparently intended to declare the test successful if it had met the acceptance criteria during the second test even though the measured vibration exceeded the alert level at low tide. However, Quality Control personnel indicated to the inspector that they would not have considered the passing of the second test to be acceptable because of the first test failure. The inspector noted that the procedure does not provide for reperformance of the vibration test at high tide if it failed to meet the acceptance criteria at low tide. Further, this practice does not appear to have a sound technical basis.

Since the test also failed at high tide, maintenance personnel recognized the requirement for additional maintenance. The pump motor was rebalanced and the vibration test was repeated prior to maintenance work package completion.

In addition, the inspector noted that the procedure form required that similar vibration data be obtained from three other locations along the upper (motor) end of the pump shaft, but this information was not obtained during either test. The licensee stated that the normal practice was to obtain data at only one location, although the additional data locations were specifically foot-noted for trending purposes.

Because the inspector noted the inconsistencies in procedural adherence while the test was in progress, the inspector was unable to determine whether licensee management controls, such as Quality Control reviews and supervisory oversight, would have resulted in appropriate rejection of the test results. However, it was clear that in this instance, the licensee's goals for strict adherence to procedures were not met. The licensee has subsequently reemphasized this concept through handouts to station personnel. Continued management attention to strict adherence to procedures is warranted.

4.3 Pump/Motor Holddown Bolt Overtorquing

During the performance of maintenance on the "B" Spent Fuel Pool (SFP) cooling pump, examination by the licensee of a broken SFP pump holddown bolt revealed that its shank had been filed, at some unknown time in the past, to less than nominal diameter. In response to NRC concerns over whether holddown bolts had also been filed down on safety-related equipment, the licensee performed an inspection of holddown bolts on a sampling of safety-related pumps and motors. During this inspection, one maintenance team re-torqued the bolts to the previously recorded break-away torque rather than in accordance with the referenced vendor's manual. This caused several of the holddown bolts to be overtorqued.

Prior to the examination on the sampling of the pump and motor holddown bolts, the licensee had conducted a meeting with the maintenance supervisors in attendance to discuss the general conduct of the inspection, stressing how to perform this work without affecting pump/motor alignment. Step 3A of the maintenance work plan (MWP) for the maintenance request (MR) states "Measure and record breakaway torque (torque wrench calibrated in the CCW direction). Remove and inspect each holddown fastener (one at a time) on pump and motor and then reinstall and torque. Stake studs and restore any locking devices. Ref: 3.M.4-92." Procedure 3.M.4-92 is titled, "Bolting and Torquing Guidelines."

One of the maintenance teams removed each holddown bolt one-at-a-time, recorded the breakaway torque, inspected the bolt and re-torqued to the breakaway torque. Re-torquing the holddown bolts to the breakaway torque rather than the torque values specified in the referenced vendor's manual occurred on three pump/motor combinations: the High Pressure Coolant Injection (HPCI) pump and motor, the "A" Standby Liquid Control (SBLC) pump and motor, and the emergency diesel generator (EDG) fuel oil transfer pump. In several cases, the breakaway torque was greater than the specified torque in the vendor's manual.

The licensee's critique of the event identified the root cause to be that the maintenance supervisor left the meeting with a mind-set that the major concern was how to perform the work without affecting motor/pump alignment and an understanding that the fasteners should be restored to the "as found" condition. The MWP was not specific enough to remove that mind-set. The quality control personnel that observed this evolution had the mind-set that the purpose of the evolution was to inspect the fasteners and did not question the re-torquing of the fasteners to break-away torque values. The licensee identified contributing root causes to be (1) MWP step 3 had four distinct actions which could cause confusion in the implementation of this activity; (2) the MWP did not specify what reference to use for the specified torque value (i.e., torque per Vendor Manual); and (3) the MWP was confusing in that 3.M.4-92 (torque procedure) cannot be used if the torque value is specified in the Vendor Manual, as was the case.

Corrective actions by the licensee included: (1) performing an engineering analysis of the over-torqued bolts and either accepting them as-is or untorquing the bolts and re-torquing to the specified torque value; (2) for those bolts with less than the required applied torque, torque them to the specified torque value; (3) provide training to the maintenance and quality control personnel; (4) itemizing the steps of the MWP by individual actions to make the steps more comprehensible and specific and (5) requiring that the MWP is clear as to what referenced document must be used and, unless necessary, only one document should be referenced per step.

Although the licensee considers the root cause of this event to be the mind-set of the maintenance team and quality control personnel that were involved in this job, the inspector considers a root cause of the overtorquing of the bolts to be failure to follow the MWP. Although step 3 of the MWP has several actions to be performed and also inproperly references procedure 3.M.4-92, the procedure instruction and revision column on the MWP for step 3 clear identifies the vendor manual and revision to be used. The work package contained a copy of the applicable pages of the vendor manual with the specified torque values clearly identified. Although procedure 3.M.4-92 was inappropriately referenced in the the MWP, this procedure would clearly not apply to this evolution since it states its purpose is limited to cases where torque values are not specified elsewhere. In addition, the quality control personnel on the job site also failed to observe that the MWP was not being followed. However, despite the difference between the licensee's and the inspector's root cause determination, the licensee's corrective actions for this event were appropriate.

With respect to the original reason for inspecting the pump and motor holddown bolts on a sampling of safety-related equipment, the licensee was not sufficiently aggressive in pursuing possible generic implications associated with the SFP cooling pump holddown bolts which were found filed down. Repeated promptings by the NRC had been necessary before the licensee completed the necessary inspection to ensure no question of safety-related equipment operability existed. Once performed, the additional licensee inspections provided reasonable assurance that additional bolts had not been filed down.

5.0 Radiological Controls/Chemistry

5.1 Radwaste Miscellaneous Tank Discharge

On August 30, 1989, the licensee conducted a routine waste water discharge of approximately 500 gallons to the ocean. This waste water had been collected from floor drains that normally collect water that contains quantities of soap and dirt, but only small amounts of radioactivity. At the time of the release, however, licensee records indicated that waste water radioactivity levels were above the Radiological Environmental Technical Specification and 10 CFR 50 Appendix I limits. This was later found to be erroneous, and the water was actually well within release limits.

The inspector performed a detailed review of this event. The inspector reviewed various logs and record and interviewed licensee personnel involved in assessment of the event.

The release was conducted in accordance with procedure 7.9.2, "Liquid Radioactive Waste Discharge." Tank samples were taken, analyzed and processed by a Chemistry computer code to obtain the total quantity of radioactivity released during the previous thirty-one days, including the radioactivity contained in the tank to be released. This computer code had not been qualified according to procedures for safety-related computer codes because the liquid radwaste discharge system is not safety-related.

The computer code radioactivity data, printed on Form CH-12.C, indicated that isotopic concentrations were four to eight orders of magnitude higher than expected.

Both the Chemistry Division and the Radiological Technical Support Division were involved in calculating, evaluation and processing approval of the release. However, personnel involved failed to halt the authorization process and notify supervision that the calculated release exceeded limits. Per procedure, the Watch Engineer is not required to see the entire Chemistry/Radiological package; he only reviews the discharge permit which does not include the calculated dose values nor the concomitant limits.

The unusual values on the discharge form were noted and the licensee recalculated the curie values manually. This discharge had occurred with the formal record indicating that the activity was above the allowable safe limit. The recalculation of radioactivity in the waste water sample and the concomitant dose rates revealed that, in reality, the dose values were well within their normal ranges and that the actual radioactive discharge to the ocean was benign. No Technical Specification limits were violated.

The licensee determined that the root causes of the event were:

(1) an unexplained error in a non-qualified computer program;
(2) failure of licensee personnel to comply with existing procedures;
(3) poor communication practices; and (4) human factor deficiencies within the procedure.

Immediate licensee corrective actions include: (1) the non-qualified computer code was administratively disabled and the values it produced were immediately recalculated manually to determine if Technical Specifications were violated (they were not); (2) plant management was notified and decided to conduct a formal critique; (3) all waste water discharges were stopped until procedure 7.9.2 was revised (this action is now completed); and (4) all previous calculations accomplished by the non-qualified computer code were verified by manual calculations to ensure that no Technical Specification limits had been exceeded since reliance on the computer began in mid-July (no significant errors were found).

The inspector concurred with the licensee's assessments of the root cause. This item will remain unresolved pending completion of corrective actions and an NRC specialist review of corrective actions taken to prevent recurrence of this event. This review will include the adequacy of the licensee's procedures in this area, the adequacy and determination of the need for qualification of the Chemistry computer program in accordance with station procedures and the overall administrative control of radioactive liquid discharges. (UNR 89-10-02)

5.2 Unsecured Locked High Radiation Area Door

On September 14, 1989, the licensee discovered an access door to the condenser bay, an area designated to be a "locked high radiation area" (LHRA), unsecured and unguarded. The door had been unsecured and unguarded for approximately one hour following an earlier exit by several personnel. In addition, two individuals accessed the LHRA who had not been authorized entry. Further, it was determined by the licensee that not all doors from the condenser bay LHRA were verified locked by the radiological protection (RP) technician as required by licensee procedures.

The condenser bay is controlled as a locked high radiation area with the highest whole body dose rate measured, at 75% power, of 1.6 rem/hour; general area dose rates are in the range of 5-450 mrem/hour. Following completion of work in the condenser bay at about 11:20 a.m. on September 14, 1989, two I&C technicians exited the condenser bay via the southeast door. Upon exiting the LHRA door, the I&C technicians failed to check that the door was locked and secured in accordance with the Radiation Work Permit and licensee procedure 6.1-012, "Access Control to High Radiation Areas." After the I&C technicians exited the condenser bay southeast door, the RP technician checked the door by pulling the door from inside the condenser bay. It did not open. It was later determined that although the door latching mechanism was locked, the latch did not extend into the door jamb. The RP technician then exited the condenser bay through the northeast door and verified it locked. He did not, however, verify the other condenser bay doors locked. Only the doors that were accessed were checked whereas the procedure requires a check of all doors. This failure to check all accessible doors upon exiting a LHRA occurred at three separate times and is contrary to procedure 6.1-012, Precaution 6.0[5] which states the "the RP technician shall check all accessible doors to a LHRA upon leaving the area and verify them locked." This step had been added to this procedure as corrective action per Licensee Event Report (LER) 89-05, for the previous failure to maintain a LHRA access door secured on February 3, 1989.

The licensee conducted a critique and determined the root cause to be personnel failure to follow procedural requirements for LHRA door verification upon exit and personnel failure to follow procedural requirements for entry into a LHRA. Contributing causes were determined by the licensee to be (1) General Employee Training did not include sufficient detail on Pilgrim high radiation area (HRA) access requirements and (2) malfunctioning of the door, which had been determined to be sticking against the door jamb and would not self-close. The two individuals who entered the LHRA without proper authorization were determined to have received 5 mrem and 15 mrem of exposure.

Immediate corrective actions by the licensee included: (1) re-familiarization was provided to all radiological operations personnel on the HRA Control procedure; (2) the southeast condenser bay door was latched and verified locked; (3) the other condenser bay doors and all other LHRA doors on site were verified to be shut and locked; (4) the licensee inventoried all LHRA keys and found them all to be present and properly controlled; and (5) a guard was posted at the southeast condenser bay door until the door's operability had been verified.

Permanent corrective actions taken by the licensee include: (1) a revision was approved to procedure 6.1-012 to include a job aid identifying the accessible doors for each LHRA and containing a signature block to be signed for each door that is verified shut and locked; (2) General Employee Training has been revised to emphasize and clarify the requirements for HRA entry and exit; (3) request further evaluation of HRA procedures and practices by quality assurance auditing; (4) a "For Your Information" notice was issued to re-empahasize and clarify HRA entry requirements for station personnel; (5) appropriate maintenance personnel received special training in HRA controls; and (6) a maintenance request was written to repair the southeast door.

The licensee's response to the unsecured LHRA was prompt and their investigation was thorough. The critique identified several areas of weakness and the licensee's corrective actions, both immediate and permanent in response to this event are appropriate and aggressive.

10 CFR 20.203 (c)(2) requires, in part, that each entrance or access point to a high radiation area shall be maintained locked and with positive control over individual entry. In addition, Technical Specification 6.13.2 requires that each high radiation area in which the intensity of radiation is greater than 1000 mrem/hour shall have locked doors to prevent unauthorized entry into such areas. Finally, procedure 6.1-012 requires, in part, that areas controlled under these procedures remain locked or guarded at all times and that positive control is maintained of each individual entry by authorized personnel. Inadequate control of locked radiation areas had been an area of previous NRC concern. Notices of Violation (NOV) had been issued in the past during inspections 50-293/87-03, 50-293/87-11, 50-293/87-19 and 50-293/87-57. In addition, during inspection 50-293/88-37, a licensee-identified violation for which no NOV was issued, was identified with respect to failure to properly control a LHRA access door. Corrective actions taken in response to the above Notices of Violation have not been effective in precluding recurrence. Although this event was promptly identified by the licensee, and prompt, aggressive corrective actions were taken, failure to maintain the area locked is a violation (VIO 50-293/89-10-01). The

response to the Notice of Violation should include discussions of why previous corrective actions have not remained effective and actions to be taken to assure durability of measures to prevent recurrence.

5.3 ALARA Hold on RWCU Pump Repair Work

During maintenance on the "B" Reactor Water Cleanup (RWCU) pump on September 8, 1989, the "As Low As Reasonably Achievable" (ALARA) engineer placed an ALARA "hold" on the maintenance work in progress after contamination was found on an individual's shoe upon exiting the process buildings. This individual had not been involved in the RWCU pump repair nor in the hot shop where repairs were being performed, but had walked past the outside of the hot shop.

The level of activity on the individual's shoe was determined to be about 450,000 disintegrations per minute (dpm). Surveys of the hot shop, just inside the door, revealed 15,000 dpm. The licensee hypothesized that when the "A" RWCU pump repair had been completed and removed from the glove bag, loose fines may have been on the bag the RWCU pump was in, allowing contamination onto the hot shop floor. Work on the "A" RWCU pump, including removal of the pump from the glove bag, had been done under RP supervision.

Upon determining that the RWCU pump repair work had potentially caused loose surface contamination, the licensee placed an ALARA hold on all hot shop work and the hot shop was cleaned. To prevent recurrence of possible contamination spread in the hot shop, the licensee; (1) removed unnecessary items which were being stored in the hot shop; (2) cleaned the glove bag used to perform repairs on the RWCU pumps, including removal of unnecessary tools and vacuuming; and, (3) the contamination levels in the decontamination trough in the hot shop were reduced, as well as the installation of a polyvinyl chloride (PVC) curtain around the trough to improve negative ventilation.

Licensee actions with respect to identifying the source of the contamination, placing of the ALARA hold on hot shop work and identification of methods to reduce possible future contamination problems were prompt and conservative. The placing of the ALARA hold demonstrates the licensee's conservative attitude and determination to improve radiological conditions at the plant.

6.0 Management Self-Assessment

During the course of inspection, licensee self-assessment activities were observed during Management Oversight and Assessment Team (MO&AT) meetings and routine plant activities. On certain occasions, apparently incomplete or outdated information was presented to senior plant management. This information was not always questioned by senior management. The inspector considered this to indicate a need for more plant specific knowledge among

(MO&AT) members, as well as a more questioning attitude. When the instances were brought to the attention of senior managers by the NRC, action was taken to prevent recurrence. The action included elevation of the Regulatory Affairs Department in the organizational structure and full time representation of the department manager on the MO&AT. In addition, the licensee intends to improve the level of plant-specific knowledge among senior managers by channeling some of the senior managers through the six-month long SRO certification program.

The long-term effectiveness of these actions will be monitored during the course of routine inspection activities. Immediate corrective action in the form of better knowledge of the basis for information supplied has also been observed occasionally; however, this approach has not been uniformly applied.

7.0 Management Meetings

An NRC Restart Assessment Panel meeting was held on August 28, 1989 at the NRC Region I Office in King of Prussia, Pennsylvania. The Restart Assessment Panel was briefed by the licensee on their plans for the remainder of the 75% plateau and the remainder of the Power Ascension Test Program. The licensee's handout for the presentation is included as Attachment II to this report.

An NRC Restart Assessment Panel meeting was held on September 18, 1989 at the NRC Region I Office in King of Prussia, Pennsylvania. The NRC resident inspectors participated via teleconference. The Restart Assessment Panel received a presentation from the licensee on their assessment of the results of the 50% - 75% Power Ascension Program. The licensee's handout for the presentation is included as Attachment III to this report.

At periodic intervals during the inspection period, meetings were held with senior facility management to discuss the inspection, scope and preliminary findings of the resident inspectors. A final exit interview was conducted on October 27, 1989. No written material was given to the licensee that was not previously available to the public.

ATTACHMENT I

INSPECTION REPORT 50-293/89-10

Persons Contacted

R. Bird, Senior Vice President - Nuclear
K. Highfill, Vice President, Nuclear Operations
R. Anderson, Plant Manager
D. Eng, Outage and Planning Manager
E. Kraft, Deputy Plant Manager
R. Fairbanks, Nuclear Engineering Department Manager
D. Long, Plant Support Department Manager
L. Olivier, Operations Section Manager
N. DiMascio, Radiological Section Manager
J. Seery, Technical Section Manager
G. Stubbs, Maintenance Section Manager
L. Olivier, Operations Section Manager
T. Sullivan, Chief Operating Engineer
J. Neal, Security Division Manager
W. Clancy, Systems Engineering Division Manager
B. Sullivan, Fire Protection Division Manager

DRAFT
8/28/89
DMM

FINAL ASSESSMENT REPORT PRELIMINARY OUTLINE

Purpose of the Report

1. To document completion and demonstrate effectiveness of plans, programs, actions to ensure safe and reliable restart and continued operation of Pilgrim undertaken pursuant to the restart plan, PAP, RRSA and other Company initiatives.

2. To document the basis for closure and request dissolution of CAL-86-10 and the CAL Supplement.

DRAFT
8/28/89
DMM

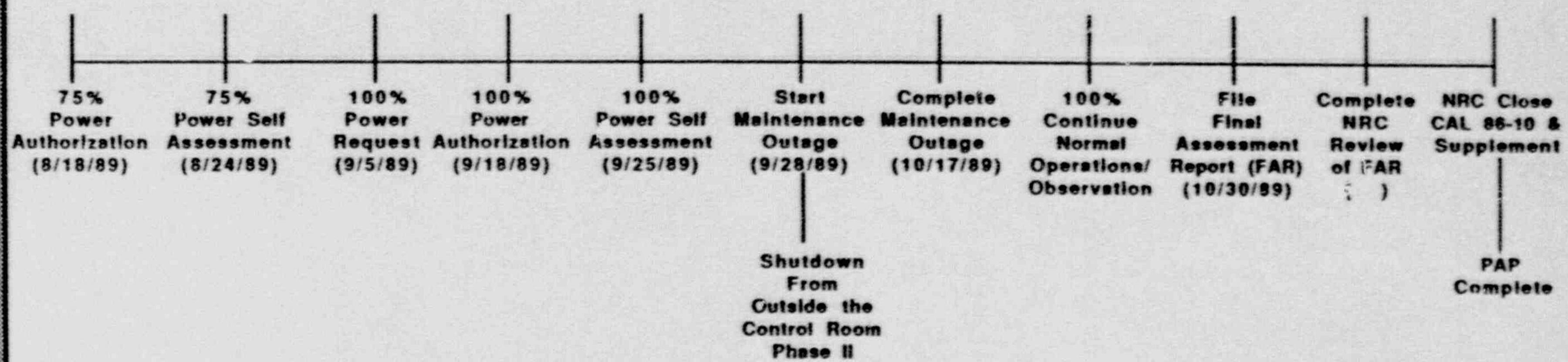
FINAL ASSESSMENT REPORT PRELIMINARY OUTLINE

EXECUTIVE SUMMARY

- Purpose of Report
- Resolution of PAP Hold Point Self Assessments
- Resolution of Major Issues from RRSA
- Closure of Restart Plan, Volume 2 Items
- Processes for Continuing Improvements
- Documentation of Completion of PAP
- Organization Changes and Succession Plan
- Conclusions

DRAFT
8/28/89
DMM

Preliminary Timeline for Submittal of Final Assessment Report and Completion of PAP



SLIDE #1

100 % POWER PRESENTATION

SCOPE OF SELF ASSESSMENT

RESULTS OF MANAGEMENT ASSESSMENTS

PROCEDURES: LESSONS LEARNED

OVERSIGHT AT 75% POWER

FINAL ASSESSMENT REPORT

ACTIVITIES PRIOR TO CLOSING CAL 86-10

SLIDE #2

OVERSIGHT MANHOURS

8,525 MANHOURS OF DIRECT OVERSIGHT SINCE BEGINNING POWER ASCENSION PLAN

630 MANHOURS ABOVE 50%

LINE MANAGEMENT

2600 MANHOURS
200 MANHOURS ABOVE 50% POWER

PEER EVALUATORS

3500 MANHOURS
240 MANHOURS ABOVE 50% POWER

QA/QC

2425 MANHOURS
190 MANHOURS ABOVE 50% POWER

SLIDE - #3

KEY QUESTIONS

IS THERE ANYTHING THAT WARRANTS FURTHER INVESTIGATION?

ARE WE MAKING PROGRESS ON THE ISSUES WHICH WE HAVE PREVIOUSLY IDENTIFIED AS NEEDING ADDITIONAL MANAGEMENT ATTENTION?

SLIDE #4

SELF ASSESSMENT

QUESTIONS FOR MANAGERS

ARE WE READY FOR FULL POWER OPERATIONS?

WHERE AND WHEN ARE FURTHER IMPROVEMENTS NEEDED?

WHEN POWER ASCENSION IS COMPLETE, CAN IMPROVEMENTS IN PERFORMANCE BE SUSTAINED?

SLIDE #5

PLANT PERFORMANCE

PLANT CONDITION SOUND

SURVEILLANCES CURRENT

CONDENSER VACUUM PERFORMANCE

MAIN GENERATOR POTENTIAL
TRANSFORMER FAILURE CORRECTED

SLIDE #6

MANAGEMENT AND ORGANIZATION EFFECTIVENESS

MAINTENANCE MANAGEMENT STRENGTHENED

MAINTENANCE EFFECTIVE

RADWASTE PROGRAMS IMPROVED

RFI PROCESS A GOOD PRACTICE

PROCEDURE UPGRADE PROGRAM

DIRECTED BY NEW VP (ADMINISTRATION)

OPERATIONS PROCEDURES COMPLETE: JULY 1990

MAINTENANCE PROCEDURES COMPLETE: DECEMBER 1991

INCORPORATES HUMAN FACTORS

UNAMBIGUOUS PROCEDURES

SLIDE #8

PROCEDURES: LESSONS LEARNED

CONDENSATE PIPING OVERPRESSURIZATION

PUMP AND MOTOR BOLTING

RADIOACTIVE LIQUID DISCHARGE PERMIT

UNLOCKED HIGH-RADIATION AREA DOOR

SLIDE #9

LINE MANAGEMENT OBSERVATIONS

STRENGTHS

CONTROL OF EVOLUTIONS AND SURVEILLANCES BY CONTROL ROOM PERSONNEL

OVERALL USE OF PROCEDURES

PRECISION AND FORMALITY OF ORAL COMMUNICATION IN THE CONTROL ROOM

WATCH TURNOVER AND PRE-SHIFT BRIEFINGS

CONSERVATIVE APPROACH BY OPERATIONS SUPERVISORY PERSONNEL TO EQUIPMENT PROBLEMS

RADIOLOGICAL CLEANLINESS

PERSONNEL RADIATION EXPOSURE

SLIDE 10

LINE MANAGEMENT OBSERVATIONS

IMPROVEMENTS

CONTROL OF CHEMICALS

OPERATOR ATTENTIVENESS TO DETAILS

USE OF SAFETY EQUIPMENT

SLIDE #11

LINE MANAGEMENT OBSERVATIONS

ADDITIONAL MANAGEMENT ATTENTION

STRICT ADHERENCE TO PROCEDURES

MAINTENANCE EFFICIENCY

HOUSEKEEPING IN A FEW PLANT AREAS

PROCUREMENT EFFICIENCY

SLIDE #12

PEER EVALUATOR OBSERVATIONS

STRENGTHS

PRECISION AND FORMALITY OF ORAL COMMUNICATIONS

WATCH TURNOVER AND PRE-SHIFT BRIEFINGS

PERSONNEL RESPONSE TO EQUIPMENT PROBLEMS

CONTROL OF EVOLUTIONS AND SURVEILLANCES

STRONG CONTROL BY CONTROL ROOM SUPERVISORS

FORMAL COMMUNICATIONS USED BY I&C and RADWASTE PERSONNEL WHEN INTERFACING WITH CONTROL ROOM

SLIDE #13

PEER EVALUATOR OBSERVATIONS

CONSISTENT IMPROVEMENTS NOTED

SECURITY PERFORMANCE AND PROCEDURES

LOGKEEPING IN CONTROL ROOM

EFFECTIVE PRE-EVOLUTION BRIEFINGS

CONTROL OF CHEMICALS

CONTROL OF ACCESS TO THE CONTROL ROOM

PEER EVALUATOR OBSERVATIONS IMPROVEMENTS SHOWN, CONTINUING ATTENTION NEEDED

ADHERENCE TO PROCEDURES

OPERATOR ATTENTIVENESS TO PANEL INDICATIONS

HOUSEKEEPING IN A FEW PLANT AREAS

PERSONAL PROTECTIVE EQUIPMENT

TEAMWORK AMONG WORK GROUPS

SLIDE #15

PEER EVALUATOR OBSERVATIONS

ADDITIONAL MANAGEMENT ATTENTION

MAINTENANCE WORK PLANNING

ATTENTION BY OPERATIONS TO ADMINISTRATIVE DETAILS

COMMUNICATION BETWEEN MAINTENANCE AND OTHER SUPPORT GROUPS

RP POSTINGS AND SURVEY DOCUMENTATION

SLIDE #16

QUALITY ASSURANCE OBSERVATIONS

STRENGTHS

CONDUCT OF SAFETY-RELATED SURVEILLANCES

MANAGEMENT SUPPORT TO RESOLVE DEFICIENCY REPORTS

RADIOLOGICAL CLEANLINESS

CONDUCT OF OPERATIONS DURING OFF-NORMAL EVENTS

SLIDE #17

QUALITY ASSURANCE OBSERVATIONS

ADDITIONAL MANAGEMENT ATTENTION

CONFIGURATION CONTROL EFFICIENCY

PROCUREMENT EFFICIENCY

WORK CONTROL EFFICIENCY

EXPANSION OF CLEARINGHOUSE PROCESS

SLIDE #18

FINAL ASSESSMENT REPORT
PURPOSE

DOCUMENT COMPLETION AND EFFECTIVENESS OF ACTIONS

DESCRIBE HOW WE WILL SUSTAIN IMPROVED PERFORMANCE

SUMMARY OF PRIMARY LESSONS LEARNED.

DOCUMENT BASES FOR CLOSURE OF CAL 86-10.

SLIDE 19

FINAL ASSESSMENT REPORT
PRELIMINARY OUTLINE

RESOLUTION OF MAJOR RRSA ACTION ITEMS

ADDITIONAL IMPROVEMENTS DURING THE PAP

CLOSURE OF RESTART PLAN, VOLUME 2 ITEMS

PROCESSES FOR SUSTAINING CONTINUING IMPROVEMENTS

DOCUMENTATION OF COMPLETION OF THE PAP

ORGANIZATIONAL CHANGES AND THE SUCCESSION PLANNING PROCESS

LESSONS LEARNED

CONCLUSIONS

REMAINING MILESTONES TO CLOSURE OF CAL 86-10

100% POWER REQUEST (TODAY)

NRC APPROVAL OF 100% POWER

100% POWER TESTING COMPLETE

MO&AT ASSESSMENT AT 100% POWER

MO&AT NORMAL OPERATION/OBSERVATION ASSESSMENT

SHUTDOWN FROM OUTSIDE THE CONTROL ROOM PHASE II

BEGIN MAINTENANCE OUTAGE

FINAL ASSESSMENT REPORT

COMPLETE MAINTENANCE OUTAGE

COMPLETE NRC REVIEW OF FINAL ASSESSMENT REPORT

PAP COMPLETE/CAL CLOSED

BACKUP

SOME AREAS DISCUSSED BY LINE MANAGERS

PLANT PHYSICAL READINESS FOR FULL POWER

STAFFING LEVELS FOR OPERATION

EXISTING ORGANIZATION, INCLUDING INTERNAL COMMUNICATIONS AND INTERFACES

ADEQUACY OF BUDGET

MANAGEMENT INVOLVEMENT AND CONTROL

IDENTIFICATION AND RESOLUTION OF TECHNICAL ISSUES

RESPONSIVENESS TO NRC

OPERATIONAL EVENTS (INCLUDING RESPONSE TO, ANALYSIS OF, REPORTING OF, AND CORRECTIVE ACTIONS FOR)

TRAINING AND QUALIFICATION

PROCEDURES

TECHNICAL DOCUMENTATION

PROCEDURAL COMPLIANCE

PERSONNEL PERFORMANCE IN OPERATION AND MAINTENANCE OF PLANT SYSTEMS

PERFORMANCE INDICATORS

BACKUP

MANAGERS ASSESSMENT OF READINESS FOR FULL POWER

MANAGERS CONDUCTING SELF-ASSESSMENTS:

STATION DIRECTOR

PLANT MANAGER

QUALITY ASSURANCE DEPARTMENT MANAGER

PLANT SUPPORT DEPARTMENT MANAGER

PLANNING AND OUTAGE DEPARTMENT MANAGER

NUCLEAR ENGINEERING DEPARTMENT MANAGER

EMERGENCY PREPAREDNESS DEPARTMENT MANAGER

NUCLEAR TRAINING DEPARTMENT MANAGER

PLANT OPERATIONS SECTION MANAGER

DESIGN SECTION MANAGER

REGULATORY SECTION MANAGER

RADIOLOGICAL SECTION MANAGER

SECURITY SECTION MANAGER

TECHNICAL SECTION MANAGER

MAINTENANCE SECTION MANAGER

MATERIALS MANAGEMENT SECTION MANAGER

RADWASTE & CHEMISTRY SECTION MANAGER

SYSTEMS ENGINEERING DIVISION MANAGER

FIRE PROTECTION DIVISION MANAGER

INDUSTRIAL SAFETY DIVISION MANAGER

BACKUP

WEEKLY PERFORMANCE INDICATORS

OPEN FIRE PROTECTION & SECURITY MRs

OPEN DEFICIENCY REPORTS

RADIOLOGICAL OCCURRENCE REPORTS (RORs)

PERCENT TIME REACTOR WATER CHEMISTRY OUT OF SPEC

TOTAL OPEN MRs & OPEN POWER BLOCK RELATED MRs

POTENTIAL CONDITIONS ADVERSE TO QUALITY (PCAQ)

ALARA TRACKING

CONDUCTIVITY

OPERATING EXPERIENCE REVIEW PROGRAM (OERP)

MANAGEMENT CORRECTIVE ACTION REPORTS (MCAR)

FAILURE & MALFUNCTION REPORTS (F&MR)

LIQUID RADIOACTIVE EFFLUENTS