

U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Reports No. 50-315/86025(DRP); 50-316/86025(DRP)

Docket Nos. 50-315; 50-316

Licenses No. DPR-58; DPR-74

Licensee: American Electric Power Service Corporation
Indiana and Michigan Electric Company
1 Riverside Plaza
Columbus, OH 43216

Facility Name: Donald C. Cook Nuclear Power Plant, Units 1 and 2

Inspection At: Donald C. Cook Site, Bridgman, MI

Inspection Conducted: June 17 through July 28, 1986

Inspectors: B. L. Jorgensen
J. K. Heller

Approved By: *R. L. Nelson for*
B. L. Burgess, Chief
Projects Section 2A

R. L. Nelson
Date
8/5/86

Inspection Summary

Inspection on June 17 through July 28, 1986 (Reports No. 50-315/86025(DRP);
No. 50-316/86025(DRP))

Areas Inspected: Routine unannounced inspection by the resident inspectors of licensee actions on previously identified items; operational safety verification; reactor trips; maintenance; surveillance; reportable events; and independent inspections.

Results: Of the seven areas inspected, no violations or deviations were identified in six areas. Two violations (Level IV - failure to establish or maintain required alarm response procedures - Paragraph 3.c; Level IV - permissible time limits for operation with a safety injection accumulator having low level were exceeded - Paragraph 3.i) were identified in the remaining area.

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DETAILS

1. Persons Contacted

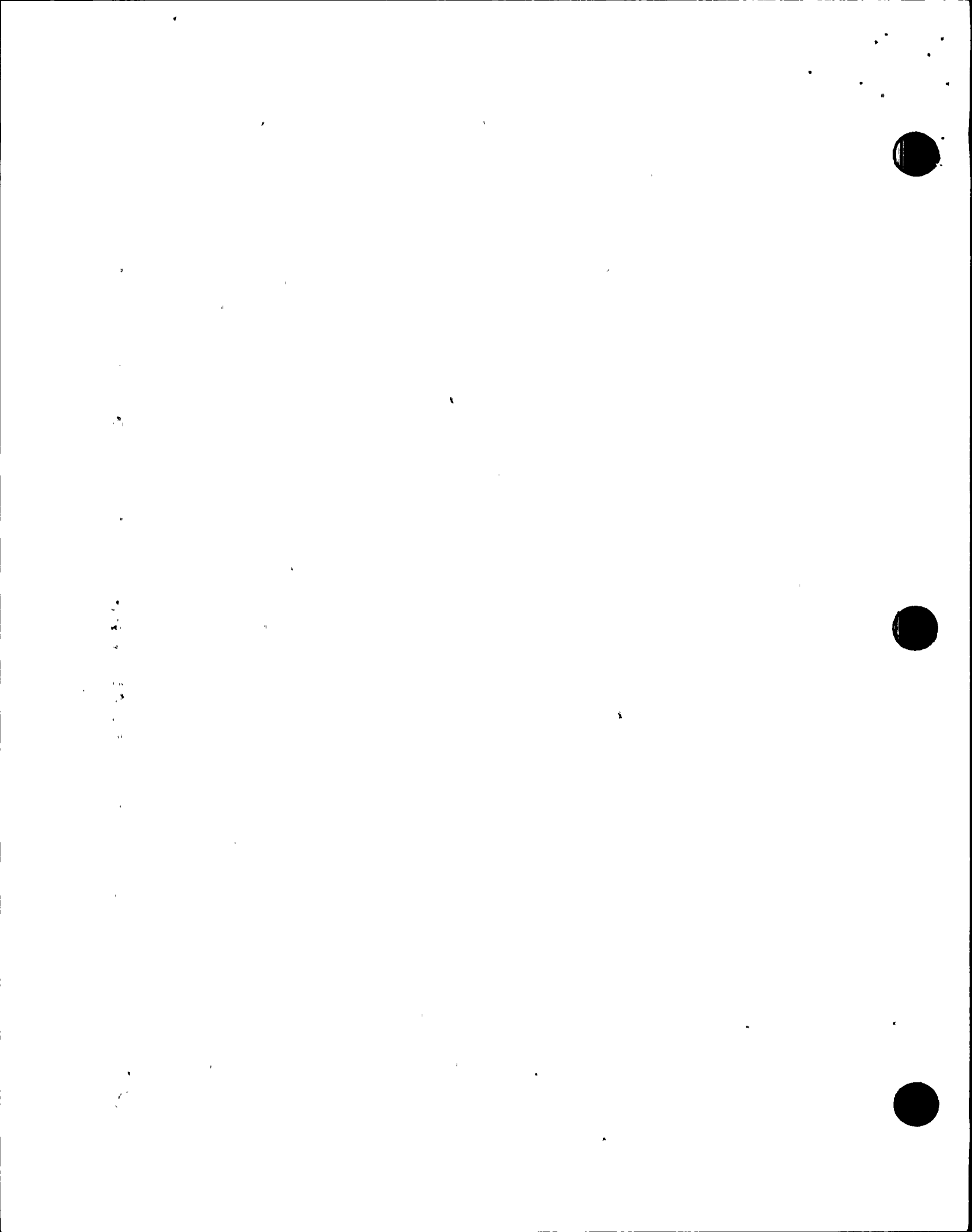
- *W. Smith, Jr., Plant Manager
- *A. Blind, Assistant Plant Manager, Administration
- *J. Rutkowski, Assistant Plant Manager, Production
- L. Gibson, Assistant Plant Manager, Technical Support
- *B. Svensson, Licensing Activity Coordinator
- T. Kriesel, Technical Superintendent, Physical Sciences
- J. Allard, Maintenance Superintendent
- K. Baker, Operations Superintendent
- J. Stietzel, Quality Control Superintendent
- T. Beilman, C&I/Planning Superintendent
- J. Sampson, Operations, Production Supervisor
- D. Wizner, Maintenance, Production Controller
- M. Horvath, Quality Assurance Supervisor
- J. Rischling, Administrative Compliance Coordinator, QC
- R. Clendenning, Radiation Protection Supervisor
- J. Fryer, Environmental Supervisor
- T. Postlewait, Technical Superintendent, Engineering
- D. Draper, Operations Procedure Coordinator
- P. Jacques, Fire Protection Coordinator
- *M. Terry, Administrative Compliance Coordinator, QC
- *D. Gallagher, Administrative Compliance Coordinator, QC
- *R. Hueter, Quality Assurance Auditor, AESPC

The inspector also contacted a number of other licensee and contract employees and informally interviewed operations, maintenance, and technical personnel.

*Denotes personnel attending Management Interview July 30, 1986.

2. Licensee Actions on Previously Identified Items

- a. (Open) Violation (315/86022-01; 316/86022-01): Heise gauges were left installed on the Unit 1 main steam lines following calibration. Inspection Reports No. 315/86022(DRP); No. 316/86022(DRP) documented that the gauges were left in place after **1 THP 6030 IMP.076 was completed. However, during a review of completed surveillance procedures, the inspector found that **1 IMP 4030 STP.002 was performed at the same time and used the same gauges. This was discussed with the Maintenance Superintendent, Plant Manager, and at the exit interview. At the exit interview the licensee was encouraged to consider this information when responding to the violation.
- b. (Open) Violation (315/86017-01; 316/86017-01): The refueling frequency surveillance test for the Essential Service Water System (ESW) did not prove that the valves controlling the ESW flow would respond to the correct "accident" position. The licensee procedures



(**2 THP SP.125 and **2 THP SP.128) to verify correct response of the controlling valves found examples of inadequate flow; these are documented in Condition Reports No. 1-06-86-695 and No. 2-06-86-696. This item remains open pending further review.

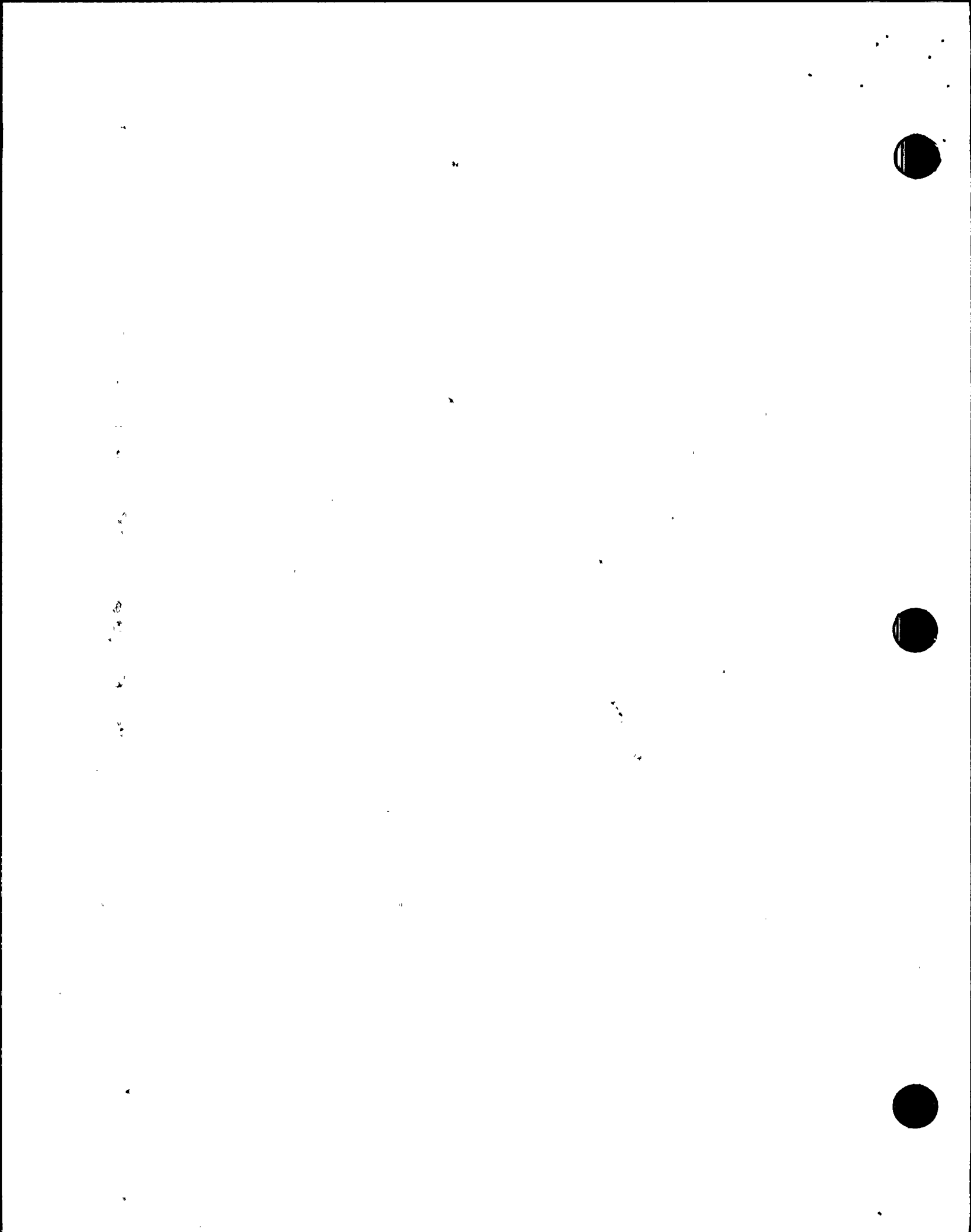
- c. (Closed Tracking Number not assigned): Inspection Reports No. 50-315/86017(DRP); No. 50-316/86017(DRP), identified that procedures **1 OHP 4030 STP.022E and STP.022W, involving the East and West Essential Service Water Systems, contained internal inconsistencies. These included valves being listed in the "Objective" section as being stroke-tested by procedures which were not so tested; and valves not being listed in the "Objective" section which were being tested. These inconsistencies were corrected and the revised procedures were in effect as of June 19, 1986.

No violations, deviations, unresolved items or open items were identified.

3. Operational Safety Verification

The inspector observed control room operation including manning, shift turnover, approved procedures and Limiting Condition for Operation (LCO) adherence, and reviewed applicable logs and conducted discussions with control room operators during the inspection period. Observations of the control room monitors, indicators, and recorders were made to verify the operability of emergency systems, radiation monitoring systems, and nuclear and reactor protection systems, as applicable. Reviews of surveillance, equipment condition, and tagout logs were conducted. Proper return to service of selected components was verified. Tours of the auxiliary building, turbine building, Unit 1 containment building, and screenhouse were made to observe accessible equipment conditions, including fluid leaks, potential fire hazards, and control of activities in progress. In addition, routine facility tours with the Plant Manager were conducted.

- a. Unit 1 began the inspection period in a maintenance and testing outage which commenced following a Unit trip on May 28, 1986. Activities completed during the outage included replacement of the turbine backup overspeed trip circuit logic cards (considered the cause of the trip); repairs to the reactor coolant pumps (see Paragraph 5.f below); ice weighing; and Limitorque valve inspections (see Paragraph 8.c below). The Unit achieved criticality at 6:42 a.m. on July 7, 1986 and subsequently escalated to normal full power operation, except for a generator off-line period on July 10 and 11, to perform selected time response testing. A Unit trip (see Paragraph 4) on July 22, 1986 led into shutdown to Mode 5 for selected tests and repairs.
- b. Unit 2 began the inspection period in a scheduled refueling, maintenance, modification and testing outage which began February 28, 1986. While in preparation for critical approach on July 7, 1986 with most of the shutdown bank rods withdrawn, a reactor trip signal



(see Paragraph 4 below) resulted in tripping of the withdrawn rods. Subsequently, the Unit achieved criticality on July 7, 1986 at 6:12 p.m., and commenced low power physics testing. The Unit was again out of service following a reactor trip from about 69 percent power on July 18, 1986 (see Paragraph 4) until July 20, when criticality and power escalation were re-established.

c. Unit 1 Operating Procedures for the Component Cooling Water (CCW) system were selectively reviewed, as follows:

- 1-OHP 4021.016.001 "Filling and Venting the Component Cooling Water system (CCW)", Revision 5, May 15, 1984.
- **1-OHP 4021.016.002 "Interchanging of the Spare Component Cooling Water Pump with the East or West CCW Pump", Revision 3, October 9, 1984.
- 1-OHP 4021.016.003 "Operation of the Component Cooling Water System During Reactor Startup and Normal Operation", Revision 5, March 20, 1986.

A few minor internal inconsistencies were noted in the review. For example, procedure **1-OHP 4021.016.002 above, in both its References and Initial Conditions sections, referred to a procedure (...016.004) which no longer exists. Another procedure was listed in the References section with an error in the title. Also, a couple of possible procedure enhancements were identified. For example, two steps in procedure I-OHP 4021.016.001 refer to a valve be operated only by name, not by valve number, as in the remaining instructions. These and other matters were identified to the Procedures Coordinator in the Operations Department for consideration in future revisions. No significant problems were noted.

On July 2, 1986 the inspector reviewed the controlled copies in each main control room of the following procedures:

Unit 1

- 1-OHP 4024.104 "Annunciator No. 4 Response - Essential Service and Component Cooling."

Unit 2

- 2-OHP 4024.204 "Annunciator No. 4 Response - Essential Service and Component Cooling."

Several Discrepancies were noted in each procedure. The Unit 2 procedure (2-OHP 4024.204) contained two different instruction pages for response to "drop" (panel window) number 077, one of which was an outdated version reflecting that "drop" as a blank, unused window. Drop 077 currently annunciates essential service

water 2E pump start on low header pressure, which is correctly reflected on the newer page. Also, some seventeen different procedure instructions, for "drops" 031 through 049 excluding 040 and 042, remain in the procedure despite the fact the panel has been modified and the subject "drops" are now disconnected spares.

The Unit 1 procedure (1-OHP 4024.104) contained a more serious discrepancy. "Drops" 75 through 80 inclusive, annunciate offnormal automatic safety-related pump starts as follows:

Drop 75:	ESW pump 1E	Unit 2 SI Start
Drop 76:	ESW pump 1W	Unit 2 SI Start
Drop 77:	ESW pump 1E	Unit 1 low pressure start
Drop 78:	ESW pump 1W	Unit 1 low pressure start
Drop 79:	CCW pump 1E	Unit 1 low pressure start
Drop 80:	CCW pump 1W	Unit 1 low pressure start

The subject procedure did not provide instructions for responding to or correcting these conditions but treated them as spares. Procedures for correcting abnormal, offnormal or alarm conditions are among those specified in Appendix "A" of Regulatory Guide 1.33, and, as such, are procedures which are required by Technical Specification 6.8.1.a to be established, implemented and maintained. Failure to establish or to maintain procedures for the alarms involving automatic pump starts as described above is considered a violation of the subject Technical Specification (Violation 315/86025-01).

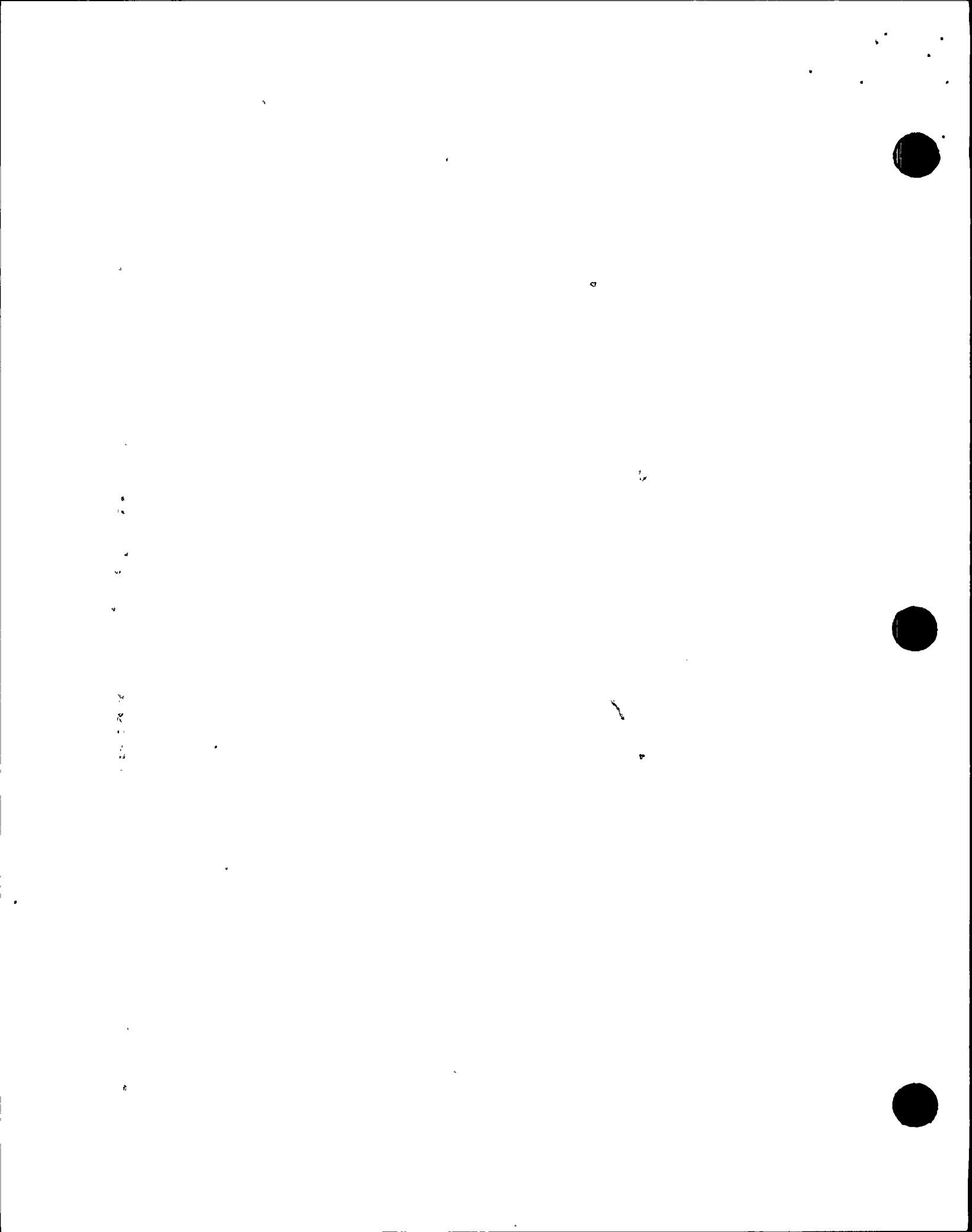
Each of the items associated with annunciator response procedure reviews noted above was identified to and discussed with the Operations Procedure Coordinator on the date of discovery. The circumstances were similar to those discussed in a previous report (reference: I.E. Reports No. 50-315/86022(DRP); No. 50-316/86022(DRP)) but this time involved important information about safety-related equipment, for which some operator response action (as reflected in their Unit 2 procedure analogues) is required.

- d. During a tour of the auxiliary building, the inspector performed an independent survey of the spent fuel pit area and found some boxed equipment with direct exposure readings up to about 50 mR/hr stored on a passageway across the northeast end of the area. The area is accessed either through a gate off the 650 foot elevation from the West, or by climbing a stair from the trackway area on the 609 foot elevation. The entrance gate at 650 foot was posted identifying the area as a Radiation Area, and a survey map identified general radiological conditions. Neither a posting nor a survey map was provided for entry up the stair. This was discussed with the Radiation Protection Manager who pointed out the 609 foot elevation trackway area is itself a posted Radiation Area, such that the stairway need not be separately posted. He agreed, however, that



a posting and a survey map were desirable and indicated both would be provided. During a later building tour, the inspector noted both a Radiation Area posting and a survey map had been affixed to the stairway gate.

- e. On July 8, 1986 the inspector performed independent radiological surveys on a sole-use LSA radioactive waste shipment awaiting transport to a licensed waste burial site. The shipment, consisting of two "liners" (large cylindrical metal drums) and a metal box, was found to be well secured. All radiological measurements made by the inspector were in good agreement with those made by licensee personnel, considering differences in survey meters used, and were well within regulatory limits for measurements made at two meters from the truck. Contact readings were also typically far below limits, though a single spot on the box was found to be up to about 25 percent of the limit. The inspector and licensee surveys both included measurements below the truck bed and at the vehicle cab. No problems existed in either of these areas.
- f. Additional items identified on facility tours included: superseded "local control" procedures left behind atop 4 KV switchgear (Unit 1); temporary gauges installed on turbine driven auxiliary feedwater pump steam supply lines (Unit 1); and test transducers temporarily installed on main steam headers (Unit 2). These items were discussed with appropriate licensee representatives and the inspector's questions satisfactorily addressed.
- g. During a tour of the turbine building the inspector was able to open a vital door without the use of the card reader. Apparently an alarm was generated because a security guard responded in a timely fashion. The door was made operable within four hours by replacing the door strike. This is an unresolved item pending a review by a Region III security specialist. (Unresolved Item 315/86025-02; 316/86025-01).
- h. During a tour of the protected area the inspector was able to open a door below the Unit 2 Steam Generator No.'s 2 and 4 main steam isolation valves without using the card reader. This door was an access to the auxiliary building (Radiation Area). The inspector could have entered the Radiation Area without signing-in (written or by card reader) on the "REM" computer. This was discussed with the Radiation Protection Manager and the Plant Manager, and will be subject to further review during a future inspection. (Open Item 315/86025-03; 316/86025-02).
- i. Subsequent to Unit 2 startup from a refueling outage on July 7, 1986 the licensee was monitoring accumulator level and pressure as required. Accumulator No. 22 level and pressure appeared to indicate possible reactor coolant system inleakage in that both seemed to increase slightly over about the following week. During this time, low power physics testing was completed and the plant commenced power escalation, entering MODE 1 at 12:38 a.m. on July 11. By July 15,



the licensee had decided to investigate the level indication by independently determining level with an ultrasonic device. A containment entry on that date appeared to validate the indicated level; i.e. the level (volume) of contained borated water calculated from the ultrasonic result agreed within about one-half percent with that indicated in the control room. This was, however, in error. The ultrasonic device establishes level by measurement (in inches) from the lower instrument tap centerline. When this measurement was made on July 15, probably in part because involved personnel were wearing full-face respiratory protection, the measured value of one-oh-five and three-quarters inches was recorded and conveyed for calculation as one-one-five and three-quarters. The coincidence of the errant measure report with the indicated (instrument) volume camouflaged the mis-communication.

Following an unrelated reactor trip on July 18, and following the draining of some borated water from the No. 22 accumulator (as indicated volume continued to increase slowly) additional ultrasonic verification measurements were made. These established that the contained volume was lower than the specified minimum of 929 cubic feet (as low as about 900 cubic feet), and had apparently been low for greater than the 13 hour (cumulative) time requirement of Technical Specifications 3.5.1, Action a. for achieving HOT SHUTDOWN with low volume. This is considered a violation of the referenced Technical Specification (Violation 316/85025-03).

Two violations, one unresolved item, one open item, and no deviations were identified.

4. Reactor Trips/Safety Feature Actuations

- a. Unit 2 received a reactor trip signal while in Mode 3 at 1:57 a.m. on July 7, 1986, from a "spike" on Nuclear Instrument NI-32 which caused an RPS "high flux" trip signal. The licensee was preparing for a critical approach at the end of a refueling, modification, maintenance and testing outage which began February 28, 1986. Two banks of shutdown rods had been withdrawn, and an investigation to repair an indicated failure on rod L-13 in the third shutdown bank was in progress, when NI-32 appeared to fail low. During troubleshooting of this failure using the vendor recommended procedure, the spike occurred, causing the trip signal. All withdrawn rods responded as designed.

Rod L-13 was repaired, as was NI-32. The Unit achieved initial criticality subsequent to the refueling at 6:12 p.m. on July 7, 1986.

- b. Unit 2 tripped from about 69 percent power at 9:52 a.m. on July 18, 1986 when a feed flow instrument (FFC-221) failed low. The faulty low reading for No. 2 steam generator feed flow caused the level control system to overfeed, resulting in turbine trip/reactor trip on steam generator No. 2 high level. All systems appeared to respond as designed.

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The licensee replaced the faulty instrument, completed other minor repairs involving an accumulator narrow-range level instrument and a steam generator No. 4 level transmitter, and the plant was again critical at 8:47 a.m. on July 20, 1986.

- c. Unit 1 tripped from about 90 percent power at 1:07 p.m. on July 22, 1986. Reactor trip was precipitated by a generator trip/turbine trip due to indicated main generator stator cooling water low pressure. A calibration activity was in progress which involved the stator cooling water pressure switches, and it appears an error was made in handling of the requisite lead lifting processes to perform the calibration.

All safety systems responded as required. One source-range nuclear instrument failed to automatically re-energize as expected. This was corrected by removing and reinserting the control power fuse. Root cause is under investigation.

The inspector verified system responses and reviewed and discussed licensee evaluations relating to cause and to corrective and preventive actions. Further review and evaluation of each of these matters is anticipated in follow up on the Licensee Event Reports associated with the events.

No violations, deviations, unresolved items or open items were identified.

5. Maintenance

Station maintenance activities of safety related systems and components listed below were reviewed to ascertain that they were conducted in accordance with approved procedures, regulatory guides and industry codes or standards and in conformance with Technical Specifications.

The following items were considered during this review: the Limiting Conditions for Operation were met while components or systems were removed from service; approvals were obtained prior to initiating the work; activities were accomplished using approved procedures; and post maintenance testing was performed as applicable.

The following maintenance activities were reviewed or observed.

- a. Repair of Unit 1 and Unit 2 Turbine Driven Auxiliary Feedwater Pump (TDAFP) steam supply line insulation.
- b. Installation of seismic support for conduit/cable for 1CD Emergency Diesel Generator (EDG) bearing temperature monitoring system per Request For Change (RFC) No. 01-2698.
- c. Door frame adjustments of Turbine Driven Auxiliary Feedwater Pump room doors.

- d. Installation of emergency lights per RFC No. 12-2698.
- e. Change out of 2AB Emergency Diesel Generator lube oil and repair of lube oil cooler.
- f. Job Orders:

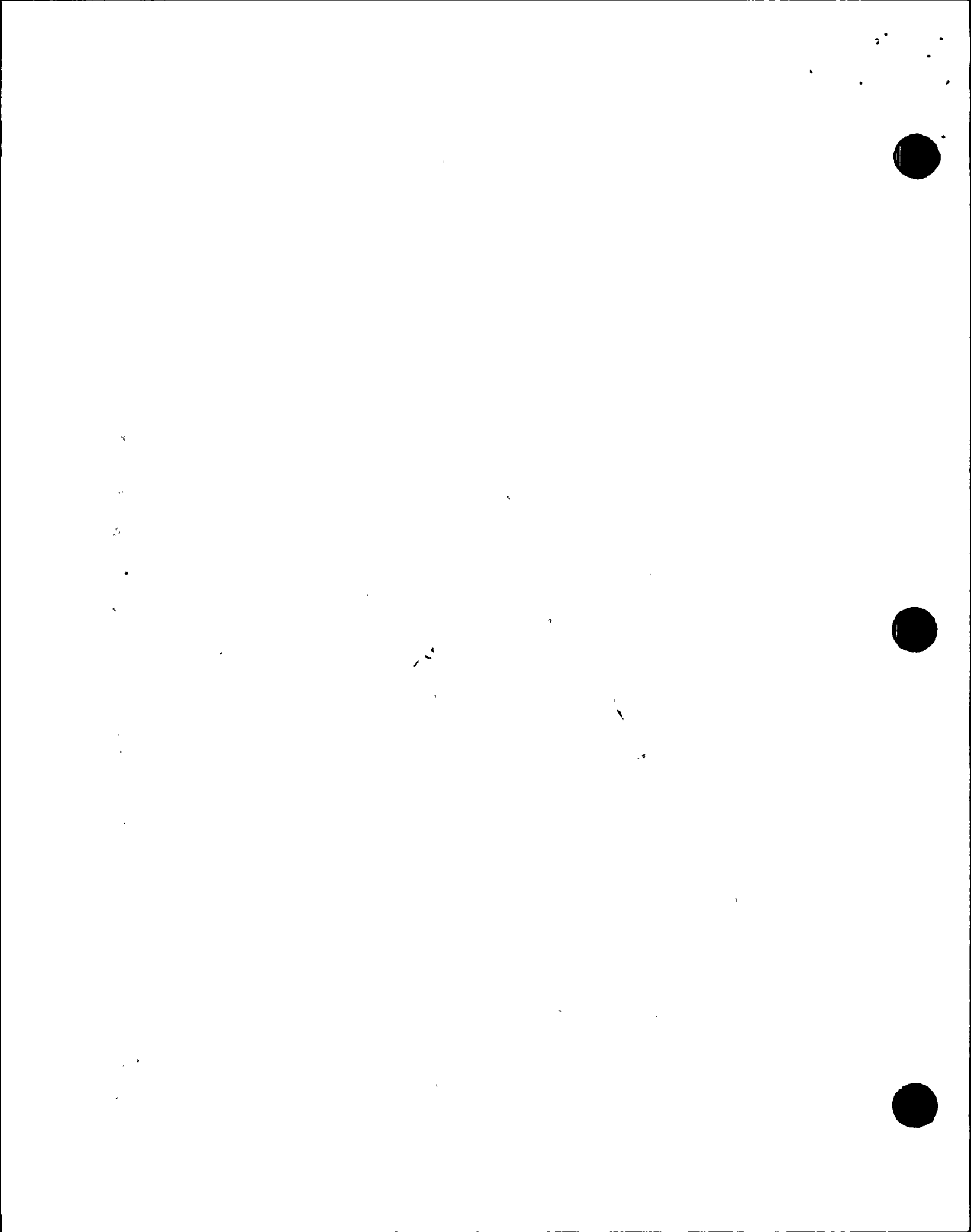
<u>Number</u>	<u>File</u>	<u>Description</u>
No. 95893	ME-PP-PP-45	Replacement of RCP-11 "0" ring
No. 95894	ME-PP-PP-45	Replacement of RCP-12 "0" ring
No. 95895	ME-PP-PP-45	Replacement of RCP-13 "0" ring
No. 95896	ME-PP-PP-45	Replacement of RCP-14 "0" ring

- g. Job Orders:

<u>Number</u>	<u>File</u>	<u>Description</u>
No. 97786	ME-VCM-CCM	Remove/install plugs for CCM-430, 431, 432 and 433.
No. 94567	ME-VCM-CCM	Tighten packing on CCM-453.
No. 92832	PS-35	Install blank flanges to facilitate hydro.
No. 92841	PS-35	Repair inlet and outlet flange for CCW to "E" RHR Heat Exchanger.
No. 97850	ME-VCM-CCM	Repair leaking valve CCM-451.

Job Order No. 97850 was written when valve CCM-451 "Containment Isolation Valve for Component Cooling Water from Reactor Coolant Pump Oil Coolers", failed its "Type ~~R~~ Q" Leak Rate Test. The repair included replacement of the previously installed eight inch Center-line butterfly valve (company no longer supplies spare parts for the valve) with an eight inch Pratt butterfly valve. The replacement was performed on May 19, 1985. The inspector questioned if a Request For Change (RFC) had been processed since the replacement may have changed a parameter used in the system safety evaluation. For example, the valve weight may vary between manufacturers and this may require a new seismic evaluation. The inspector discussed this with the Maintenance Superintendent and was informed that when this valve was replaced the valve designator did not distinguish between valve manufacturers, which means the valve could be replaced via a Job Order provided the same valve designator was used. The Maintenance Superintendent did state that approximately six months ago the valve designator was expanded to include the valve manufacturer. This means that future valve replacement would require an RFC if different manufacturers were used.

This Job Order was discussed with a Region III Quality Assurance Specialist and will be classified as an open item pending review during a future inspection. (Open Item 315/86025-04; 316/86025-04).



The inspector also discussed this replacement with the inservice inspection personnel to determine if new valve stroke baseline was established after the valve was replaced. The data sheet still identifies the valve as a Centerline. The new baseline is provided by Corporate personnel when the plant notifies the Corporate Office that the valve is changed out. The baseline for the Pratt and Centerline butterfly valve is essentially the same.

The inspector found that the requirement to notify the Corporate Office when the valve is replaced (via a Job Order), is a Memo from Corporate Office to the Plant. The inspector notes that this requirement may be more appropriate when documented as a Plant Administrative procedure.

<u>h.</u>	<u>Job Order</u>	<u>Description</u>
	No. 48600	N-32 Source Range NI Failed Low
	No. 48566	N-32 Source Range NI Failed Low
	No. 52023	Replace Intermediate Range N-36
	No. 52018	Repair Feed through penetration 215
	No. 52014	Replace defective connector on N-42
	No. 14671	Timer for N-32 and N-32 not working
	No. 54516	Inspect cable connections and rearrange to assure free movement of drawers
	No. 48275	Replace power plug on N-41B

One open item and no violations, unresolved items, or deviations were identified.

6. Surveillance

The inspector reviewed Technical Specifications required surveillance testing as described below 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 properly 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 deficiencies identified during the testing were properly reviewed and resolved by appropriate management personnel.

The following were observed or reviewed:

- a. 12 MHP 4030 STP.002 " Maintenance Procedure for Main Steam safety Valve Setpoint Verification (Secondary System Safety valve Settings)", performed November 1985, for Unit 1 and June 1986 for Unit 2. The inspector observed the setup of the test equipment for Unit 2 and reviewed the completed surveillance tests for Unit 1 and Unit 2. STP.002 contains instructions for performance of the test and instructions for adjusting the relief valve setpoints if any are found out-of-specification. However, the procedure does not contain a requirement to initiate a Condition Report if the relief valve

setpoints are found out-of-specification. In this case, the Condition Report would notify plant management of equipment that may not be functioning as intended and assure that reportability requirements were not missed. The inspector reviewed the Unit 2 data approximately a week after the test was completed, and found that six of the twenty relief valves tested were found with settings below the Technical Specification setpoint. STP.002 properly documents the required adjustments and retests showing the as-left settings were satisfactory, however, the test does not document that a Condition Report was written. The inspector inquired if a Condition Report had been or would be written; one was written the next day.

The inspector reviewed the Unit 1 data and found two of twenty relief valves with as-found settings below the Technical Specification setpoint, and six of twenty relief valves with as-found settings above the Technical Specification setpoints. As with the Unit 2 test, the Unit 1 test documents the required adjustments and retest showing the as-left settings were satisfactory, however, the test does not document that a Condition Report was written. This was discussed with the Plant Manager and Maintenance Superintendent and at the exit interview. This is an unresolved item pending a review to determine if the Unit 1 setpoints should be reported on a Licensee Event Report (LER) per 10 CFR 50.73(a)(2)(v), "Events That Could Have Prevented Fulfillment Of A Safety Function." (Unresolved Item 315/86025-05; 316/86025-05).

- b. **1 OHP 4030 STP.020E "East Component Cooling Water Surveillance Test", performed January through July 1986.

**1 OHP 4030 STP.020W "West Component Cooling Water Surveillance Test", performed January through May 1986.

The inspector reviewed the technical content of STP.020E and STP.020W (both were Revision "0", with change sheets 1 and 2 incorporated) and found that the procedures appear to demonstrate the operability of the Component Cooling Water System in accordance with the appropriate monthly Technical Specification surveillance requirements. The ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWP, "Operability Requirements", are contained in Section 15 of the Technical Data Book, which the inspector reviewed. The 1W, 2W, 1E, and "spare pump" differential pressure "Alert" and "Required Action" limits are correct when recalculated. However, the "Required Action - Low" limit for differential pressure on the 2E pump appeared to be non-conservative. This was discussed with the cognizant engineer who issued a Condition Report and changed the setpoint. In addition, the inspector and engineer reviewed the trending data for the last two years and verified that the pump had not exceeded its corrected low action limit for pump differential pressure.

- c. Observed: **1 OHP 4030 STP.017 TV, "Turbine Driven Auxiliary Feed Pump Trip and Throttle Valve Operability Test"
- **2 THP 4030 STP.239, " Reactor Coolant Pump Fire Protection Water System Test"

One unresolved item and no open items, violations or deviations were identified in this area.

7. Reportable Event

(Closed) LER 315/85069-00 and 315/85069-01: operation with inoperable intermediate range neutron flux detector, caused by instrument drift. The revised report dated July 7, 1986 corrects the cause determination from improper estimation of core leakage, to instrument drift presumed to be the result of a faulty detector tube, and reports on replacement of the suspect detector. The detector canister was found to have water in it. A previous review of this item, documented in I.E. Inspection Reports No. 50-315/86005(DRS); No. 50-316/86005(DRS) had concluded the original LER was deficient with respect to root cause determination and, as a consequence, proper corrective/preventive action. The previous report identified an Unresolved Item (315/86005-08) to be used to track and further evaluate these matters. As such, the LER is now redundant to the Unresolved Item and is being closed simply to eliminate duplicate tracking.

No violations, deviations, open items or unresolved items were identified.

8. Independent Inspection Areas

- a. The licensee announced a significant reorganization effective July 1, 1986. The revised organization provides for a third Assistant Plant Manager and the new position of Coordinator of Licensing Activities. Reporting realignments were made among selected organizational elements. The major divisions, each headed by an Assistant Plant Manager, and their respective supporting elements, are as follows:
- i) Production - consists of Operations, Maintenance and C&I/ Planning Departments. The Control and Instrument Section is newly assigned from the Technical Engineering Department to the C&I/Planning Department.
 - ii) Technical Support - consists of Computer Sciences, Technical-Engineering, and Technical-Physical Sciences Departments. The inservice inspection (ISI) and Nuclear Plant Reliability Data System (NPRDS) functions are newly assigned to Technical-Engineering from Quality Control.

iii) Administration - consists of Quality Control, Accounting, and Shift Technical Advisor groups; along with the Administrative Department which includes Personnel, Security and Training.

The Technical Specifications (both Units) at Section 6.3, require facility staff to meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions. The inspector reviewed the qualifications of personnel whose assignments were affected by the reorganization, against the criteria of ANSI N18.1-1971, for the following: Assistant Plant Manager-Production; Assistant Plant Manager-Technical Support; Planning Superintendent; Technical Engineering Superintendent; and Computer Sciences Superintendent. No discrepancies were identified.

- b. On June 27, 1986 the inspector attended a meeting of the Plant Nuclear Safety Review Committee (PNSRC) held to review the safety evaluation associated with a pending request for relief from Technical Specification requirements on maximum out-of-service time for an emergency diesel generator in Unit 2. The 2AB diesel was inoperable due to lubricating oil contamination by service water from a leaking heat exchanger tube. The Unit was in Mode 3, so one emergency diesel is permitted out-of-service for 72 hours. The licensee was requesting the time limit be extended to 120 hours so that cycling the plant to cold shutdown and then reheating later could be avoided. No Mode change was contemplated. The PNSRC approved the safety evaluation as showing the contemplated action did not involve an unreviewed safety question as defined in 10CFR 50.59 and the extension request was made. Other events (a pressurizer spray valve leak) subsequently caused the licensee to elect to place the plant in cold shutdown, such that no Technical Specification relief was required on the emergency diesel. In a letter to the licensee dated July 16, 1986, the NRC Region III office documented the above circumstances and indicated an extension would have been granted for a period required to issue an emergency Technical Specification change.
- c. Pursuant to a request from the NRC Region III Office, conveyed via memorandum from C. E. Norelius, Director of the Division of Reactor Projects, the inspector gathered specified information from the licensee concerning licensee inspection of Limitorque motor valve operator wiring. The additional information supplied as a result of the current inspection is considered to close the issue as to resident inspector review. NRC Region III plans further inspection in the area.
- d. This inspection included a review of licensee actions pursuant to NUREG-0737 Items I.D.2.2 and I.D.2.3 involving installation and full implementation, respectively, of a plant safety-parameter display system/console for each Unit. Such a system has been installed, tested, and declared operable, as discussed below. The review performed during this inspection excluded considerations involving overall acceptability of the licensee's onsite Technical Support

Center (TSC), of which the subject safety-parameter display system is only one feature. Overall TSC adequacy remains subject to further review by NRC specialists in emergency response planning.

The licensee implemented Items I.D.2.2 and I.D.2.3 via a facility design change processed under Request For Change (RFC) No. 12-2457. A portion of this RFC involved installation of a computerized hardware system capable of acquiring data on a wide variety of plant system parameters, along with software for data treatment (highlighting, trending, etc.) and display. As indicated in a previous inspection report (No. 50-315/86022(DRP); No. 50-316/86022(DRP)) the achievement of operability for the Unit 2 system, and review of administrative support functions such as procedures and operator training, were the only remaining areas of review on this issue. These activities have now been completed and the NUREG-0737 Items I.D.2.2 and I.D.2.3 are, insofar as resident inspector reviews are concerned, considered closed.

9. Unresolved Items

Unresolved items are matters about which more information is required in order to ascertain whether they are acceptable items, violations, or deviations. Unresolved Items disclosed during the inspection are discussed in Paragraphs 3.g and 6.g.

10. Open Items

Open items are matters which have been discussed with the licensee, which will be reviewed further by the inspector, and which involve some action on the part of the NRC or licensee or both. Open Item disclosed during the inspection are discussed in Paragraphs 3.h and 5.g.

11. Management Interview

The inspectors met with the licensee representatives (denoted in Paragraph 1) on July 30, 1986 to discuss the scope and findings of the inspection as discussed below.

- a. The violation for not having alarm response procedure - Paragraph 3.c
- b. The unresolved item for a malfunctioning door - Paragraph 3.g
- c. The open item for uncontrolled access to the auxiliary building - Paragraph 3.h
- d. The violation for inadequate accumulator level - Paragraph 3.i
- e. The reactor trips - Paragraph 4
- f. The unresolved item for out-of-service Main Steam Relief Valves and the need for an LER - Paragraph 6

g. The new plant organization - paragraph 8.a

The inspector asked those in attendance whether they considered any of the items discussed to contain information exempt from disclosure. No items were identified.