U.S. NUCLEAR REGULATORY COMMISSTION
REGION I

Report No. 50-423/88-05

Docket No. 50-423

License No. NPF-49

Licensee: Northeast Nuclear Energy Company
P.O. Box 270
Hartford, CT 06101-0270

Facility Name: Millstone Nuclear Power Station, Unit 3

Inspection At: Waterford, Connecticut

Inspection Conducted: February 23 - April 4, 1988

Reporting Inspector: G. S. Barber, Resident Inspector

Inspectors: W. J. Raymond, Senior Resident Inspector
G. S. Barber, Resident Inspector

Approved by: E. C. McCabe, Jr.
E. C. McCabe, Chief, Reactor Projects Section 1B   4/22/88 Date

Inspection Summary: Inspection on February 23 - April 4, 1988

Areas Inspected: Routine onsite inspection (132 hours) of Plant Operations, previous inspection findings, unqualified flow transmitters for the C&D Recirculation Spray System (RSS) pumps, overtemperature delta-T spiking, maximum reactor power determination, improperly locked suction valve for the "A" Auxiliary Feedwater Pump, safety system operability, Plant Incident Reports, physical security, environmental qualification of Litton-Veam connectors, oil leakage from Rosemount Detector Internal Diaphragms, increase in RCS liquid and containment airborne activity, Licensee Event Reports, maintenance, and surveillance.

Results: Inspection identified no unsafe plant conditions. A violation was identified when an AFW pump suction valve was found unlocked.

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DETAILS

1.0 Persons Contacted

Inspection findings were discussed periodically with the supervisory and management personnel identified below.

S. Scace, Station Superintendent
C. Clement, Unit Superintendent, Unit 3
M. Gentry, Engineering Supervisor
R. Rothgeb, Maintenance Supervisor
K. Burton, Staff Assistant to Unit Superintendent
J. Harris, Operations Supervisor
D. McDaniel, Reactor Engineer
R. Satchatello, Health Physics Supervisor
M. Pearson, Operations Assistant

2.0 Facility Activities

The plant remained at 99% power throughout most of the inspection period. There was a power reduction to 70% on March 19 due to an apparent increase in reactor coolant system (RCS) activity (see Detail 10.0). The plant returned to 99% power the following day after an error was discovered in the RCS gross activity calculation.

An overtemperature delta-T spiking problem caused the plant to enter the inspection period at 99% power (see Detail 4.2). The licensee lowered Tavg by 2 degrees F as an interim solution, and the plant was returned to 100% power on March 23.

Hot full power operation continued for the rest of the inspection period, except for periodic 2-4% power reductions for flux mapping, protection system testing and selected surveillances.

3.0 Previous Inspection Findings

(Closed) Unresolved Item 87-17-02; Supplement for LER 87-31-00 to identify and correct root cause of June 14, 1987 Reactor Trip (90714, 92701)

The inspector reviewed LER 87-031-01 and noted that the licensee identified the root cause of the June 14, 1987 reactor trip as an oversized Turning Gear Oil Pump and a defective Bearing Header Relief Valve. Refueling outage lube oil system inspections determined that the Bearing Lube Oil Header Relief Valve valve disc was offset at a slight angle to the valve seat. This relief valve continually relieves to maintain bearing header pressure constant during lube oil system transients. It is believed the offset caused degraded valve response during the transient that resulted in the plant trip.

Testing of the Turning Gear Oil Pump (TGOP) and review of design documents determined that the TGOP was oversized by 4 to 7 psi. When energized, the TGOP raised bearing lube oil header flow and pressure. When de-energized, system pressure dropped below the turbine trip setpoint before the Bearing Header Relief Valve could adjust to the lower flow. The Bearing Header Relief Valve was repaired and the Turning Gear Oil Pump impeller was shaved to lower discharge pressure. Subsequent testing showed a less severe lube oil system pressure transient, with a 9 psi margin between the lowest lube oil header pressure experienced and the low bearing oil pressure turbine trip setpoint.

(Closed) Unresolved Item 88-02-02; Control of Vendor Notification System for Field Change Notices - NI Gain Adjustment (92701)

The licensee requested, in writing, that the vendor explain why there was no requirement for a Power Range (PR) Nuclear Instrument (NI) operability evaluation after the low leakage core load for Cycle 2. It was specifically requested that the evaluation consider whether the PR NIs could be adjusted to match the secondary heat balance without hardware modifications. In addition, the licensee requested that the vendor explain what has been done to prevent recurrence.

The vendor reply, by Memorandum 88NE*-G-0032 dated March 2, 1988, explained how the NI problem occurred and what has been done to prevent future recurrence. The memorandum stated that any changes in "as constructed" or "operating" characteristics that could affect neutron leakage from core to the detector could also affect NI sensitivity.

The second part of the memorandum dealt with actions to prevent recurrence. It addressed two concerns related to the lack of sensitivity of the PR NIs. First, protection setpoints may need to be adjusted based on lower detector currents. Second, establishment of a minimum acceptable PR NI current below which a field modification would be necessary should be evaluated by the licensee. The vendor specified, as a part of his design review process, that the desirability of aligning the intermediate range (IR) and PR NIs prior to reload startup should be considered by evaluating the effects of the predicted radial power distribution on the NIs. Vendor internal procedures have been changed to require forwarding the radial power distribution to licensees with their reload design. Additionally, the attachment to the memorandum also addressed the advisability of hardware modifications to ensure minimum PR NI sensitivity requirements are met. The inspector reviewed the vendor's actions to and concluded that, if adequately implemented, the internal procedures established should prevent recurrence.

There still exists a generic concern that lessons learned by the vendor in the field are not being fed back into the design review process. In this case, the lack of PR NI sensitivity was first identified by the vendor in 1983. Plants with low leakage core loads were sent information letters describing the potential effects of the low leakage core loads on the NIs. The NI sensitivity problem for future low leakage core loads at other plants was not considered. This failure of the vendor to incorporate lessons learned in the field into the design review process is an inadequacy. Since the actions taken by the vendor in this specific case are adequate, this unresolved item is closed. However, the vendor's factoring of lessons learned in the field into the design review process is being forwarded to the NRC Vendor Inspector Branch for consideration.

4.0 Equipment Problems

4.1 Unqualified EQ Flow Transmitters for the C&D RSS Pumps

The licensee determined at 1:05 p.m., March 8, that the Environmental Qualification (EQ) for C&D Recirculation Spray System (RSS) Flow Transmitters (FTs) 3RSS-FT40C&D was inadequate. This was discovered during a routine licensee audit of the EQ components listed in the Production Maintenance Management System (PMMS). The auditor noted that the PMMS listed no EQ requirements for the FTs while the Master EQ list required them to be Category 2 (Cat 2). Licensee field verification of the FTs proved that they were, in fact, different than that specified on the master EQ list. The licensee reviewed the reporting requirements of 10 CFR 50.72 and 10 CFR 50.73 to determine if the non-conformance was reportable. In addition, 10 CFR 50.49 and Generic Letter (GL) 86-15 were reviewed to supplement the licensee's reporting determination. The licensee determined that this non-conformance was not reportable since it did not cause the system to become inoperable. The inspector reviewed the licensee's reportability determination and noted no inadequacies.

Although GL 86-15 does not specify reporting requirements, it states that, when a licensee discovers a potential deficiency in the environmental qualification of equipment, the licensee shall make a prompt determination of operability, shall take immediate steps to establish a plan with a reasonable schedule to correct the deficiency, and shall have written justification for continued operation. This justification does not require NRC review and approval. The licensee reviewed the operability of the C&D RSS pumps and determined that they were operable based on the following Justification for Continued Operation:

> The affected flow transmitters provide flow indication for Containment Recirculation Pumps C&D. Containment Recirculation Pumps provide two functions: one is the spray down of Containment during an accident (TS 3.6.2.2), the other provides recirculation flow capability for long term injection (TS 3.5.2). The long term injection capability requires one pump and heat exchanger per train. The principal pumps for this function are the A&B pumps in each train, based on their design to provide automatic recirculation of sump water to the safety injection pump suctions. RSS pumps A&B have qualified transmitters.
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> When required by their respective safety signals, the C&D pumps will start automatically and spray down the containment regardless of the status of their flow indicators. The C&D pumps are thus consi-

dered operable to perform the required spray function. The inspector concurs with the licensee's operability determination since alternate indications are provided to ensure their proper functioning.

The licensee reviewed 10 CFR 50.49 and noted that certain sections specify that qualified instrumentation not mislead the operator during accident circumstances. In that respect, the licensee took the following compensatory actions to cope with the unavailability of accurate C&D RSS flow indication:

-- The operators have been provided correlation between pressure, flow, and RSS pump motor current. The motor current monitors will not be affected by temperature and radiation levels at the location of the monitors for flow.

-- The indicators have been caution tagged to alert the operators of the potential for incorrect readings under accident conditions.

-- The information has been disseminated to the shifts during their shift briefs.

-- The Operations Supervisor has issued a Night Order indicating that RSS pumps C&D are not to be used to satisfy TS 3.5.2.

The inspector reviewed the licensee's compensatory actions and identified no inadequacies. Additionally, the licensee took prompt action to order qualified Category 2 replacement FTs and expects to receive the replacements within 6 to 8 weeks. The licensee will maintain the stated compensatory measures until the qualified FTs are received, installed and satisfactorily tested. The inspector has no further questions on the operability of the C&D RSS pumps. An EQ team inspection, during the week of March 14-18, reviewed other technical aspects of this issue. Their findings are contained in Region I Inspection Report 50-423/88-04.

4.2 OverTemperature Delta-T Spiking

The licensee identified that numerous spikes have occurred in the overtemperature delta-T (OTdT) circuitry. This circuitry, used for both control and protection functions, provides runbacks and reactor trips for DNB (departure from Nucleate Boiling) protection. The spiking first occurred on February 19 and was initially believed to be due to setpoint drift. Further licensee investigation has shown it to be due, in part, to the replacement of the RTD (resistance temperature detector) bypass manifold with well-mounted RTDs. These RTDs are mounted at 120 degree increments around the circumference of the T-hot (Th) and T-cold (Tc) piping.

The spiking appears to be caused by a combination of factors. First, in order to meet time response requirements, the RTD elements had to be mounted as close to the bottom of the well as possible. The well protrudes into the flow stream to sense actual variations in Tc and Th. The closeness of the RTD elements to the bottom of their wells makes them very sensitive to small changes in temperature. As Th spikes upward, it does two things to the control and protection circuitry. First, it causes an actual increase in core delta-T because Th has increased with a constant Tc. Second, it causes a penalty to be added to the OTdT setpoint because of the apparent increase in Tave (average core coolant temperature). This has caused either the OTdT runback or trip circuitry to actuate.

Another factor is the failure to achieve complete mixing of two different in-core fluid streams at the core exit. The first stream is from the coolant being heated as it passes the fuel rods. The second stream is from bypass flow that does not contact the heated surfaces of the fuel rods. This bypass flow is part of the NSSS design and accounts for approximately 6% of total flow. Since this flow does not get heated, it enters the outlet plenum 60 degrees F lower than the rest of the flow. It mixes freely in the outlet plenum, but temperature variations still exist in coolant entering the Th piping. This temperature distribution may shift across the Th piping cross section, compounding the spiking problem.

The spikes are visible on the computer display, with routine variations of up to 0.5 degrees F. Larger variations are possible and can cause bistable trips. This has occurred on all 4 channels at different times and causes trips or runbacks while at 100% power. By reducing power to 99%, sufficient margin exists to prevent actuation of protection and control bistables. The licensee measured a 3 to 4 degrees F temperature gradient in the cross-section of the Th piping. The vendor predicted a temperature gradient of about 7 degress F. The licensee maintained power at 99% until an In-Service Test (IST) could be run.

The IST was run during the week of March 14 and confirmed the earlier observations. Half-degree spikes occurred frequently and constituted a background noise level. Spikes lasting up to 20 seconds and large enough to cause an OTdT trip or runback on a given channel also were detected. The licensee also contacted two other plants (Salem and Catawba) to see if they have experienced similar problems since they have removed their RTD bypass manifolds. These other plants have not seen the same problems; their OTdT calculated setpoints and time response acceptance criteria are such that they don't impose the same penalties and increase measured core delta-T enough to cause trips.

The licensee is considering four possible solutions to the OTdT spiking problem. The first two involve changes to the setpoint calculation. The first setpoint change would be to increase the K1 factor from 1.08 to 1.10 and the second setpoint change would be to increase the Tau-one factor from 8 to 12 seconds. Both of these solutions would necessitate re-verification of all FSAR Chapter 15 analyses for which OTdT trip provides protection. The third solution involves imposing a 1 second signal filter in the Th circuitry. This would require a plant modification and could only be used subject to satisfactory time response testing. The last solution lowers Tave 2 degrees F to 585 degrees F. From a protection standpoint, the inspector noted that this will move the core further away from DNB for all analyzed accidents. The licensee chose this option, for the interim, until a further evaluation can be completed. The inspector reviewed the licensee's operation at 585 degrees F and 100% power and noted no inadequacies. The inspector will evaluate the licensee's final solution in future inspections.

5.0 Facility Tours (71707, 71710)

5.1 Maximum Reactor Power Determination

While conducting a routine walkdown of safety system status in the control room, the inspector observed indicated Nuclear Instrument (NI) Power at 101% on the Nuclear Steam Supply System (NSSS) computer display. The NSSS computer display also showed that core thermal output for the last hour was 3411 megawatts thermal (MWt) which is at or below the Technical Specification (TS) Rated Thermal Power (RTP). The inspector questioned the operators on the ramifications of operating above 100% indicated NI power. They stated that licensed power level is 3411 MWt and that operations at or below that level ensures compliance with TS. The inspector questioned the actions they would take if the computer was lost. They stated that they would reduce power until indicated power by NIS was less than or equal to 100% power and remain there until the computer was restored.

The inspector reviewed the FSAR safety analysis to ensure that operation above an indicated 100% power by NIs did not adversely impact the accident analysis assumptions. FSAR Chapter 15 Section 15.0.3.1 evaluates accidents at two separate power levels. They are:

a. The guaranteed NSSS thermal power output. This power output includes the thermal power generated by the reactor coolant pumps.

b. The engineered safety features design rating. The NSSS supplied engineered safety features are designed for thermal power higher than the guaranteed value. This higher thermal power value is designated as the engineered safety features design rating. This power output includes the thermal power generated by the reactor coolant pumps.

Where initial power operating conditions are assumed in accident analyses, the guaranteed NSSS thermal power output, plus allowance for errors in steady state power determination, is assumed. Where demonstration of adequacy of the containment and engineered safety features are concerned, the engineered safety features design rating, plus allowance for error, is assumed. The thermal power values used for each transient analyzed are given in FSAR Table 15.0-2. In all cases where the 3,579 megawatt thermal (MWt) rating is used in an analysis, the resulting transients and consequences are conservative compared to using the 3,425 MWt rating.

The inspector noted that the only other accident that could be potentially affected would be a rod ejection at full power since the rate of change of NI power would be less for an equivalent positive reactivity insertion at succeedingly higher indicated power levels. Even though the actual rate of change of power is less than indicated, this is not significant because the high flux trip provides the necessary protective action. This trip would actually occur sooner since indicated NI power is higher than actual. The inspector had no further questions regarding operation above 100% indicated NI power with the available NSSS computer indicating less than or equal to licensed thermal power.

5.2 Improperly Locked Suction Valve for the "A" Auxiliary Feedwater Pump

A routine safety inspection of equipment configuration in the Engineered Safety Features (ESF) Building at 3:30 p.m., March 4 found the demineralized water storage tank (DWST) Suction Valve for the "A" Motor-Driven Auxiliary Feedwater Pump (MDAFW) (3FWA-V2) open but not properly locked. The inspector noted the discrepancy while pulling on the lock. It was not properly secured. The lock had been closed down on one link at one end of the chain but not at the other end. The chain could be pulled free of the handwheel and the valve could have been operated without proper authorization. This is a Violation of the administrative controls used to control locked valves (VIO 88-05-01).

The inspector noted that valve 3FWA-V2 was in its proper position (open) along with its redundant suction (3FWA-66) from the Condensate Storage Tank (CST). The emergency suction supply was also available since the inspector observed the spool piece in its proper storage location. It could have been installed by procedure and the system could have been aligned to the emergency suction, if needed.

The inspector called the control room and reported the improper locking arrangement to the Shift Supervisor (SS). He dispatched a Senior Control Operator (SCO) on relief shift to the ESF building. The SCO arrived and properly secured the lock under observation of the inspector. The SCO agreed to accompany the inspector as he finished his walkdown of other accessible locked major flowpath valves for the AFW and Safety Injection (SI) systems. No other inadequacies were identified. The inspector returned to the control room to discuss the locked valve and valve lineup requirements with the SS. The inspector questioned whether there might be other valves without their locks properly attached. The licensee agreed to check all locked valves outside of containment to verify their proper configuration. When performing the review of equipment configuration, the licensee found valve RHR*V30 was not locked as required by the Piping & Instrumentation Diagram (P&ID). However, the valve lineup (A-3) from OP-3310 did not require the valve to be locked. The licensee locked the valve closed and submitted a change to correct the oversight (PIR 3-52-88).

The inspector will review the licensee's root cause determination and corrective actions in future inspections.

6.0 Plant Operational Status Reviews (71707)

The inspector reviewed plant operations from the control room and reviewed the operational status of plant safety systems. Actions taken to meet technical specification requirements when equipment was inoperable were reviewed to verify the limiting conditions for operations were met. Plant logs and control room indicators were reviewed to identify changes in plant operational status since the last review and to verify that changes in the status of plant equipment was properly communicated in the logs and records. Control room instruments were observed for correlation between channels, proper functioning and conformance with technical specifications. Alarm conditions in effect were reviewed with control room operators to verify proper response to off-normal conditions and to verify operators were knowledgeable of plant status. Operators were found to be cognizant of control room indications and plant status. Control room manning and shift staffing were reviewed and compared to technical specification requirements. No inadequacies were identified. The following specific activities were also addressed.

6.1 Safety System Operability Review (70710)

The high pressure safety injection, quench spray, auxiliary feedwater, recirculation spray, charging, residual heat removal, safety injection accumulators, and the emergency diesel generator systems were reviewed to verify the systems were operable in the standby mode. The review included consideration of: proper positioning of major flow path valves; operable normal and emergency power supplies; indicators and controls functioning properly; and a visual inspection of major components for leakage, cooling water supply, lubrication and general condition. Except for the item described in Detail 5.2, no inadequacies were identified.

6.2 Review of Plant Incident Reports (90714)

The plant incident reports (PIRs) listed below were reviewed during the inspection period to (i) determine the significance of the events; (ii) review the licensee's evaluation of the events; (iii) verify the licensee's response and corrective actions were proper; and, (iv) verify that the licensee reported the events in accordance with applicable requirements, if required. The PIRs reviewed were: 10-88 dated 1/22, 12-88 dated 1/24, 13-88 dated 1/24, 28-88 dated 2/9, 31-88 dated 2/12, 35-88 dated 2/16, 36-88 dated 2/18, 38-88 dated 2/19, 44-88 dated 2/29, 45-88 dated 3/1, 46-88 dated 3/1, 47-88 dated 3/1, 48-88 dated 3/3, 50-88 dated

3/3, 52-88 dated 3/4, 53-88 dated 3/7, 55-88 dated 3/11, 56-88 dated 3/15, 57-88 dated 3/16, 58-88 dated 3/18, 59-88 dated 3/20. The following items warranted inspector followup:

-- PIR 51-88 dated 3/4, Administrative Controls for Locked Valves (see Detail 5.2).

-- PIR 60-88 dated 3/21, Various Limit and Pressure Switches may not be environmentally qualified (see Detail 8.0).

-- PIR 61-88 dated 3/21, EEQ Instruments with Litton-Veam Connectors (see Detail 8.0).

-- PIR 62-88 dated 3/19, High RCS Activity (See Detail 10.0).

7.0 Observations of Physical Security (81064)

Selected aspects of site security were verified to be proper during inspection tours, including site access controls, personnel and vehicle searches, personnel monitoring, placement of physical barriers, compensatory measures, guard force staffing, and response to alarms and degraded conditions. The following item was also addressed:

7.1 Physical Barriers and Detection Aids

The inspector conducted a walkdown of the protected and vital area barriers and reviewed detection aid adequacy on March 29 with NRC Security inspectors on-site. Protected Area (PA) barriers were observed to ensure that they channeled personnel, vehicles, and materials to protected area entry control points, and that they also delayed unauthorized penetration attempts sufficiently to permit detection, assessment, and response by site security personnel. Selected Vital Areas (VAs) were inspected to ensure that physical barriers existed to prevent unauthorized penetration of vital areas from inside the protected area. Isolation zones were checked to be free from obstructions that could conceal an individual or object and to ensure that they were of sufficient width to observe an unauthorized entry into the PA. The inspector noted that a portion of a barrier might be susceptible to forced entry under the certain circumstances. The licensee took appropriate actions to upgrade the barrier and the inspector had no further questions on this item.

Intrusion detection was inspected for ability to detect attempted penetrations or attempts to gain unauthorized access. Alarm systems were checked to ensure they provided adequate coverage. No inadequacies were identified. Inspection Report 50-423/88-06 documents additional findings of this security inspection.

8.0 Environmental Qualification of Litton-Veam Connectors (92701)

Litton-Veam (L-V) connectors are electrical connectors used to connect conduit coming from detectors to the signal processing and conditioning equipment. These particular connectors have quick disconnect capability and are used quite frequently with Rosemount detectors. The connectors are required to remain operable during the worst case analyzed transient for their location. Testing was performed on Litton-Veam connectors to ensure that their leakage currents would be minimal when subject to the deleterious environment generated during a Loss of Coolant Accident (LOCA) or a High Energy Line Break (HELB). The tests were run since excessive moisture or chemical intrusion into the connector could cause grounds and excessive leakage currents.

An EQ (environmental qualification) team inspection (see Inspection Report 50-423/88-04 for further details) was conducted during the week of March 14-18. The team questioned the qualification of the Litton-Veam connectors used for the Rosemount transmitters. Unaged detectors successfully completed both LOCA (loss of coolant accident) and HELB (high energy line break) tests. However, aged L-V connector test results were questionable for the LOCA test. The leakage currents (IRs) were below spec on one phase to phase combination. The licensee felt that this was a result of a recording error since the low leakage currents should have shown up on at least two phase combinations if there was moisture intrusion into one phase. (Each phase is tested together with each of the other two phases and should show a low reading with both.)

At the EQ inspection exit meeting on March 18, the NRC staff questioned the operability of the L-V connectors for the near term. The licensee concluded that they were operable from an EQ standpoint because the unaged connectors passed the LOCA and HELB tests and because the plant is very early in its design life. A shift briefing was conducted with operators prior to assuming the shift on swing shift (3:30 p.m.-11:30 p.m.) March 18. The briefing identified instruments with Rosemount transmitters and L-V connectors, and provided compensatory measures to accomplish EOP actions during LOCA or faulted SG conditions. All other shifts were similarly briefed prior to assuming watchstanding duties. Also, suspect instruments now have bright red stickers attached to indicate possible degradation during adverse environmental conditions. The inspector observed the first shift briefing, and confirmed that all shifts were briefed by noting SS log entries. The inspector reviewed the EOPs related to faulted SGs (steam generators) and LOCAs and determined that the licensee's measures should adequately compensate for inoperable indications during accident conditions. The inspector operability questions for the near term have been satisfactorily addressed. Further licensee actions to resolve NRC concerns will be addressed during NRC follow-up of IR 50-423/88-04.

9.0 10 CFR 21 Report - Oil Leakage from Rosemount Detector Internal Diaphragms (36100)

The licensee submitted a 10 CFR 21 report based on the frequency and similarity of failures of Rosemount transmitters used for RCS flow indication. There were 5 similar failures of these DP transmitters between March and November of 1987. The failures all involved oil leakage from between two internal diaphragms of the detector, causing RCS flow indication to fail in the non-conservative direction (high). RCS flow is used in the reactor trip circuitry for DNB protection. The failures all occurred in RCS flow instrumentation (5 of the 12 such flow transmitters). Rosemount detectors are also used for pressurizer level indication; these have not failed at Millstone 3.

The licensee increased the RCS flow surveillance to monthly and contacted Rosemount to obtain more information. Rosemount has not promulgated any information on problems with these instruments (Rosemount Model 1153HO5PC). The serial numbers which failed were 408187, 408189, 408191, and 408195. Rosemount reportedly indicated that the failures were random and that there is no generic problem.

After the initial licensee contact with Rosemount, the inspector received information from the Senior Resident Inspector at James A. Fitzpatrick Nuclear Power Plant (JAFNPP) that indicates that this may, in fact, be a generic problem. During 1985 and 1986, JAFNPP experienced six Rosemount detector failures: four were caused by oil leakage from the internal diaphragms. Two models were affected; Rosemount Model Nos. 1153DB5 and 1153GB5 with a partial listing of these serial numbers; 410541, 410548 and 410574. JAFNPP evaluated the failures at the time and determined they were not reportable under 10 CFR 21. This issue will be reviewed in future inspections.

10.0 <u>Increase in RCS Liquid and Containment Airborne Activity (92701)</u>

The licensee reported that RCS gross activity as measured at 6:38 p.m., March 19, showed a gross activity of 0.58 microcuries per gram (uCi/g) against a limit of 0.96 uCi/g. This was an apparent increase relative to the limit. The shift supervisor questioned the 0.96 uCi/g limit since that value had previously been 104 uCi/g. The chemistry technician stated that 0.96 uCi/g was accurate. A power reduction was began at 3% per hour. During the power reduction, chemistry supervisor review noted that an error had been made in calculating the 0.96 uCi/g limit. This limit is calculated as 100/E-bar (uCi/g). E-bar is the average of the sum of the average beta and gamma energies per disintegration of radionuclides in the sample. During Cycle 1, the 100/E-bar limit had been calculated to be 104 uCi/g (i.e., E-bar ≈0.96). The chemistry technician had mistakenly divided the 104 uCi/g limit into 100 to calculate an incorrect limit of 0.96 uCi/g. When this error was identified by the chemistry supervisor, the control room was informed. Power was returned to 99% at 2% per hour on March 20.

Although the 0.58 uCi/g was well within the 104 uCi/g limit, licensee trending of gross activity and Dose Equivalent Iodine 131 (I-131) showed an increase to about 1.3 and 0.15 microcuries per cubic centimeter (uCi/cc), respectively. This indicates a small defect in the fuel. The licensee contacted the vendor, who stated that the defect cannot be located until equilibrium is reached in the I-131/I-133 ratio and the gross activity.

Health Physics personnel have been trending containment airborne activity to determine the effects of the fuel pin defect. Gaseous activity inside containment increased from of 5.2 E-07 uCi/cc early during power ascension (2/12/88) to approximately 2.3 E-04 uCi/cc (3/25/88). Particulate activity increased from 2.0 E-09 uCi/cc (2/12/88) to 2.5 E-08 uCi/cc (3/25/88). These levels are comparable to containment gaseous (1.9 E-09 uCi/cc on 10/3/87) and particulate (1.7 E-08 uCi/cc on 10/3/87) seen at the end of Cycle 1 and 100% power.

The inspector reviewed and compared both RCS liquid and containment airborne activity levels and noted that both activities are asymptotically approaching equilibrium. Further significant increases in activity levels are not expected. The inspector has reviewed the licensee's trending of various activities and has no further questions at this time. This issue will be reviewed in future inspections.

11.0 Licensee Event Reports (LERS) (90714)

Licensee Event Reports (LERs) submitted during the report period were reviewed to assess LER accuracy, the adequacy of corrective actions, compliance with 10 CFR 50.73 reporting requirements, and to determine if there were generic implications or if further information was required. Selected corrective actions were reviewed for implementation and thoroughness. The LERs reviewed were:

LER 87-047-00 Core eration Performed without Proper Communications or SRO Coverage due to Procedural Error.

LER 87-048-00 Failure to Monitor Inoperable Fire Assemblies

LER 88-002-00 Insufficient Seismic Support of RCP Oil Collection System.

Within the scope of this review, no inadequacies were noted. The following LERs required inspector followup and are documented where shown:

LER 86-052-02 Area Temperature Monitoring (MS-01). See Detail 5 of Inspection Report 50-423/86-28.

LER 87-010-01 Loose Parts Detection System Channel Inoperable due to Unknown Reasons. See Detail 3 of Inspection Report 50-423/87-08.

LER 87-031-01 Reactor Trip due to Turbine Trip on Low Lube Oil Pressure. See Detail 10.0 of Inspection Report 50-423/87-17.

LER 87-037-00 Feedwater Isolation due to High Steam Generator Level Caused by Operator Error. See Detail 4.2 of Inspection Report 50-423/87-21.

LER 87-041-00	Inadequate Testing of Containment Penetration Circuit Breakers. See Detail 3.4 of Inspection Report 50-423/87-24.
LER 87-049-00	Missed Engineering Evaluation Due to Misinterpretation of TS. See Detail 4.0 of Inspection Report 50-423/87-33.
LER 88-001-00	Inadvertent Safety Injection due to Sensitive Equipment. See Detail 3.3 of Inspection Report 50-423/87-33.
LER 88-003-00	Diesel Sequenced Start due to Spurious Relay Actuation. See Detail 4.2 of Inspection Report 50-423/88-02.
LER 88-004-00	Control Building Isolation due to Chlorine Detector Failure. See Detail 4.1 of Inspection Report 50-423/88-02.
LER 88-006-00	Violation of Tech Specs. Mode Change without Required ECCS Equipment. See Detail 8.0 of Inspection Report 50-423/88-02.
LER 88-008-00	Manual Reactor Trip due to Inoperable Digital Rod Position Indication. See Detail 10.0 of Inspection Report 50-423/88-02.
LER 88-009-00	Reactor Trip and Feedwater Isolation due to Steam Generator Level Transient. See Detail 11.0 of Inspection Report 50-423/88-02.
LER 88-010-00	Improper Nuclear Instrument Calibration due to Low Leakage Core. See Detail 12.0 of Inspection Report 50-423/88-02.

12.0 Committee Activities (40700)

The inspector reviewed the minutes for Plant Operations Review Committee (PORC) meetings 3-87-156 dated 11/5/87, 3-87-174 dated 11/19/87, 3-87-179 dated 11/21/87, 3-87-180 dated 11/21/87, 3-87-181 dated 11/22/87, 3-87-182 dated 11/23/87, 3-87-184 dated 11/24/87, 3-87-185 dated 11/24/87, 3-87-186 dated 11/24/87, 3-87-205 dated 12/14/87, 3-87-211 dated 12/28/87, and 3-88-2 dated 1/2/88. The inspector noted from the written records that committee administrative requirements were met for the meetings and that the committees discharged their functions in accordance with regulatory requirements. No inadequacies were identified.

13.0 Maintenance (62703)

The inspector observed and reviewed selected portions of preventive and corrective maintenance to verify compliance with regulations, use of administrative and maintenance procedures, compliance with codes and standards, proper QA/QC involvement, use of bypass jumpers and safety tags, personnel protection, and equipment alignment and retest. The following activities were included:

-- "C" Charging Pump Volute Installation.

-- "B" Emergency Diesel Air Dryer Filter Inspection.
-- Overtemperature delta-T Inservice Test.

No inadequacies were identified.

14.0 Surveillance (61726)

The inspector observed portions of surveillance tests to assess performance in accordance with approved procedures and Limiting Conditions of Operation, removal and restoration of equipment, and deficiency review and resolution. The following tests were reviewed:

-- RCS Leakrate Calculation.
-- In-core Flux Mapping.
-- SIH Pump Operability Test.

No inadequacies were noted.

15.0 Management Meeting (30703)

Periodic meetings were held with station management to discuss inspection findings during the inspection period. A summary of findings was also discussed at the conclusion of the inspection. No proprietary information was covered within the scope of the inspection. No written material was given to the licensee during the inspection period.