DCD

Docket No. 50-255

APR 21 1989

Consumers Power Company
ATTN: David P. Hoffman
Vice President
Nuclear Operations
1945 West Parnall Road
Jackson, MI 49201

Gentlemen:

This refers to the routine safety inspection conducted by Messrs. E. R. Swanson, J. K. Heller, and M. A. Kunowski during the period of March 14 through April 10, 1989, of activities at the Palisades Nuclear Generating Plant authorized by NRC Provisional Operating License No. DPR-20, and to the discussion of our findings with Mr. G. B. Slade and others of your staff and at the conclusion of the inspection.

The enclosed copy of our inspection report identifies areas examined during the inspection. Within these areas, the inspection consisted of a selective examination of procedures and representative records, observations, and interviews with personnel.

During this inspection, certain of your activities appeared to be in violation of NRC requirements, as specified in the enclosed Notice. A written response is required.

The responses directed by this letter and the accompanying Notice are not subject to the clearance procedures of the Office of Management and Budget as required by the Paperwork Reduction Act of 1980, PL 96-511.

In accordance with 10 CFR 2.790 of the Commission's regulations, a copy of this letter, your response and the enclosed inspection report will be placed in the NRC Public Document Room.

We will gladly discuss any questions you have concerning this inspection.

Sincerely,

W. L. Axelson, Chief
Projects Branch 2

Enclosures:
1. Notice of Violation
2. Inspection Report
No. 50-255/89009(DRP)

See Attached Distribution

IE01
1/1

RIII
Sutphin/lms

RIII
Burgess

RIII
Axelson

8905050133 XA 4p.

NOTICE OF VIOLATION

Consumers Power Company

Docket No. 50-255

As a result of the inspection conducted on March 14 through April 10, 1989, and in accordance with the General Policy and Procedures for NRC Enforcement Actions (10 CFR Part 2, Appendix C), the following violation was identified:

10 CFR Part 50, Appendix J, III.D.2.(b) (iii) states that air locks opened during periods when containment integrity is required by the plant's Technical Specifications shall be tested within 3 days after being opened. For air lock doors having testable seals, testing the seals fulfills the 3-day test requirements.

Contrary to the above, the licensee failed to do a between the seal test of the escape air lock door within 3 days after the door was used on March 2, 1989.

This is a Severity Level IV violation (Supplement I).

Pursuant to the provisions of 10 CFR 2.201, you are required to submit to this office within thirty days of the date of this Notice a written statement or explanation in reply, including for each violation: (1) corrective action taken and the results achieved; (2) corrective action to be taken to avoid further violations; and (3) the date when full compliance will be achieved. Consideration may be given to extending your response time for good cause shown.

4/21/89
Dated

W. L. Axelson, Chief
Projects Branch 2

8905050135 1p.

U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Report No. 50-255/89009(DRP)

Docket No. 50-255 License No. DPR-20

Licensee: Consumers Power Company
212 West Michigan Avenue
Jackson, MI 49201

Facility Name: Palisades Nuclear Generating Plant

Inspection At: Palisades Site, Covert, Michigan

Inspection Conducted: March 14 through April 10, 1989

Inspectors: E. R. Swanson

J. K. Heller

M. A. Kunowski

Approved By: B. L. Burgess, Chief
Reactor Projects Section 2A

4/21/89
Date

Inspection Summary

Inspection on March 14 through April 10, 1989 (Report No. 50-255/89009(DRP)
Areas Inspected: Routine unannounced inspection by the resident inspectors of: actions on previously identified items; operational safety verification; radiological controls; maintenance; surveillance; security; safety assessment/quality verification; reportable events; Engineered Safety System walkdown; and allegation followup. Several SEP open items (Safety Issues Management System (SIMS) items) were reviewed.
Results: Of the areas inspected, one violation was identified (leak testing of the containment escape lock - Paragraph 6.f). The inspection did not disclose any notable weaknesses in the licensee's programs. The inspection noted strengths in the areas of operator turnovers and the licensee's ability to maintain black board conditions during most of the period. No new Open Items and/or Unresolved Items were identified.

DETAILS

1. Persons Contacted

Consumers Power Company

G. B. Slade, Plant General Manager
+J. G. Lewis, Technical Director
R. D. Orosz, Engineering and Maintenance Manager
R. M. Rice, Operations Manager
+W. L. Beckman, Radiological Services Manager
+R. E. McCaleb, Planning and Administrative Director
+R. A. Fenech, Operations Superintendent
+K. E. Osborne, Projects Superintendent
+K. M. Hass, Reactor Engineering Superintendent
+R. P. Margol, QA Administrator
R. M. Brzezinski, I&C Superintendent
L. J. Kenaga, Staff HP
+C. S. Kozup, Licensing Engineer
+J. R. Brunet, Licensing Analyst
D. J. Malone, Licensing Analyst
R. J. Frigo, Operations Staff Support Supervisor

U.S. Nuclear Regulatory Commission (USNRC)

+E. R. Swanson, Senior Resident Inspector
+J. K. Heller, Resident Inspector

+Denotes those present at the Management Interview on April 12, 1989.

Others members of the plant staff, and several members of the Contract Security Force, were also contacted during the inspection period.

2. Actions on Previously Identified Items (92701, 92702)

a. (Closed) Open Item 255/04S09-00(DRP): Effects of charging line and letdown system pipe breaks. The NRC documented acceptance of the licensee's analysis, submitted May 31, 1985, which demonstrated that the subject pipe breaks need not be postulated in a Safety Evaluation dated February 4, 1987.

b. (Open) Open Item 255/04S10-00(DRP): Completion of evaluation of specific components for seismic design. Consumers Power responded to the NRC's questions on April 30, 1987. The NRC's review of this issue is tracked under SEP Topic III-6 and Unresolved Safety Issue A-46.

c. (Open) Open Item 255/04S12-00(DRP): Evaluation of specific structural loads referenced in SEP Topic III-7.B and IPSAR 4.12. The NRC is reviewing the licensee submittals.

d. (Open) Open Item 255/04S15-02(DRP): Consumers Power committed to submit revised Technical Specifications for primary coolant leakage detection systems inside containment as documented in the IPSAR 4.15.2 (NUREG-0820). As committed in their January 24, 1989 letter, the licensee will submit revised Technical Specifications as part of their restructured Technical Specification effort.

e. (Closed) Open Item 255/04S28-02(DRP): Install forced cooling for inverter and charging cabinets and AFW junction boxes. The inspector verified that fans had been installed in the inverter cabinets. The NRC is preparing a Safety Evaluation of this topic (SEP IX-5, IPSAR 4.28.2).

f. (Closed) Open Item 255/86035-08(DRP): Develop a long term solution to the freezing of the stack gas flow recorder sensing lines. A monthly periodic activity control (PAC) has been generated to periodically blow down the sensing lines. PAC RIA-002 has been performed on a monthly basis (+/- 25 percent) since May 1988. In a letter to File PR12/01/86A-258 the system engineer stated that this is an acceptable alternative to modifying the sensing lines or revising the piping.

g. (Closed) Open Item 255/86035-10(DRP): The licensee will evaluate the flow characteristic of Auxiliary Feedwater (AFW) flow control Valves FCV-0736A, FCV-0737A, FCV-0727 and FCV-0749 to determine the methodology to improve flow control below 100 gpm. The licensee chose to install per FC-789 a 1 and 1/2" bypass flow control valve for the "C" AFW pump. The inspector reviewed the control room prints and confirmed that the prints reflected the Facility Change. Also, the inspector discussed the modification with control room operators and confirmed that they had been trained on the modifications. An Engineering review team has selected FC-789 for additional reviews. Items identified by that review are discussed in Inspection Report No. 255/89007.

h. (Open) Open Item 255/86035-12(DRP): Replace AFW Valve CV-0521 because internal valve leakage is causing a slow rotation of the Turbine Driven Auxiliary Feedwater (TDAFW) pump. The valve was repaired by the vendor per Work Order 24706677. Internal Correspondence DAB 89*002 stated the valve was repaired satisfactory and is leak tight. A tour of the AFW pump room on March 22, revealed that the TDAFW still has a slow rotation. The inspector discussed this with control room operators and found that at the start of operations subsequent to the 88 refueling outage that the pump was not rotating. The inspector has asked the licensee to revisit this item and determine if additional actions are required.

i. (Closed) Open Items 255/86035-18, -19 and -20(DRP): Resolve the component cooling water pumps mechanical seal leakage problems. During the 88 refueling, the "C" pump was modified at the vendor facilities per SC 86-269 to accept a different mechanical seal.

In addition, modified sleeves and nuts were installed on the pump. Internal Correspondence GJS 89*002 stated that the repairs appear successful in eliminating the seal leakage problem. The licensee plans to modify the "A" and "B" pump during an upcoming refueling outage.

j. (Closed) Open Item 255/88020-05(DRP): The thermal overload on one phase of the Emergency Diesel Generator starting air compressor motor was tripping. It was found that the overloads were recently replaced with a new style and one phase required a minor calibration. No similar problems have recurred.

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

3. Operational Safety Verification (71707, 71710, 42700)

Routine facility operating activities were observed as conducted in the plant and from the main control rooms. Plant startup, steady power operation, plant shutdown, and system(s) lineup and operation were observed as applicable.

The performance of licensed Reactor Operators and Senior Reactor Operators, of Shift Technical Advisors, and of auxiliary equipment operators was observed and evaluated including procedure use and adherence, records and logs, communications, shift/duty turnover, and the degree of professionalism of control room activities.

Evaluation, corrective action, and response for off normal conditions or events, if any, were examined. This included compliance to any reporting requirements. Two events (D-PAL-89-012 and 036) identified and implemented corrective action for late, and failure to implement, fire watch requirements.

Observations of the control room monitors, indicators, and recorders were made to verify the operability of emergency systems, radiation monitoring systems and nuclear reactor protection systems, as applicable. A strength noted was the "black board" conditions maintained during most of the period except during testing and repair activities. Operators were well informed as to the reasons for any alarms which indicates that good turnovers are being conducted. Reviews of surveillance, equipment condition, and tagout logs were conducted. Proper return to service of selected components was verified. Activities reviewed are discussed below.

a. Steam Generator Tube Leakage

The unit began the reporting period at a reduced power level of approximately 60 percent to maintain primary to secondary leakage at an acceptable level, while plans for a steam generator tube inspection outage were formalized. On April 3, during a licensee initiated conference call, the plant manager informed NRC personnel consisting of the resident inspection staff, the NRR project manager,

section chiefs from Region III Divisions of Reactor Safety and Reactor Projects, and NRR steam generator specialists, of six options that were being considered by senior Consumers Power corporate managers pertaining to operation of Palisades until the upcoming outage. The option that was implemented on April 4, consisted of a power increase in three distinct steps to 90 percent power with a specified holding period between steps. Limitations pertaining to the leakrate and off gas were imposed and contingencies established if the limits were exceeded. A 90 percent power limit was a previously established administrative limit to resolve NRC concerns pertaining to tube leakage. Prior the power increase, the resident inspection staff was informed of the option selected. At the close of this inspection period, the power level had been increased to 80 percent power with no changes in primary to secondary leakrate.

b. Post Accident Operation of the Component Cooling Water System

On March 27, the licensee became aware of a potentially significant design deficiency in the Component Cooling Water (CCW) system relating to the lack of a qualified (safety grade) air system to support the operation of the CCW containment isolation valves. The CCW piping inside containment is not seismically qualified and has not been reviewed for high energy line break (HELB) concerns. Containment isolation valve CV-0910 does not have an air accumulator and would fail open on loss of air. Therefore, a HELB could result in a complete loss of CCW and a containment integrity violation. Postulating a single failure of the 1-2 diesel generator, all containment cooling would be lost. This concern should have been evaluated under the NRC's Systematic Evaluation Program, but apparently was not because the system portion inside containment is not considered safety related. The issue was identified under the licensee's Configuration Control Project design basis reconstitution effort, a part of a continuing corrective action. It is expected that a LER will be submitted discussing this discovery.

c. Off Gas Monitoring

(1) The licensee has installed an additional off gas flow meter that is calibrated in the 0-3 cfm range. The other flow meter was not calibrated in this range and would not provide repeatable readings in this range. Primary to secondary leakrate calculation results dropped by a factor of 10, as result of the more accurate readings from the new meter.

(2) The inspector observed an auxiliary operator take an off gas flow reading and implement the procedural requirements of Paragraph 7.8.1 to SOP 13, "Air Ejector, Gland Steam Condenser and Condenser Vacuum pump."

(3) During a Turbine Building tour on April 5, the inspector observed the needle of the .5 to 3 cfm off gas flow meter rapidly bouncing from off scale low to approximately .75 CFM. The inspector found that the instrument root isolation valves had not been closed subsequent to the last off gas measurement. An Auxiliary Operator closed the root isolation valve and the reading returned to zero. This was discussed with the Shift Supervisor and Operation Superintendent.

A review of SOP 13 reveals that this configuration was contrary to the restoration steps of Paragraph 7.8.1. This was discussed at the management interview.

d. During a tour of the Auxiliary Building on April 10, the inspector identified leakage (a small spray) from a main feedwater vent line near the containment penetration. The leakage was at the cap on the line. The inspector verified that the line and manual Valve FW-746 were not considered part of the containment integrity boundary and that a work request was subsequently initiated.

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

4. Radiological Controls (71707)

During routine tours of radiologically controlled plant facilities or areas, the inspector observed occupational radiation safety practices by the radiation protection staff and other workers.

Effluent releases were routinely checked, including examination of on-line recorder traces and proper operation of automatic monitoring equipment.

Independent surveys were performed in various radiologically controlled areas. The inspector witnessed the conduct of a survey being performed in the West Safeguards Room.

5. Maintenance (62703, 42700)

Maintenance activities in the plant were routinely inspected, including both corrective maintenance (repairs) and preventive maintenance. Mechanical, electrical, and instrument and control group maintenance activities were included as available.

The focus of the inspection was to assure the maintenance activities reviewed 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 activities were inspected:

a. Repair of Sigma Indicator TIC-0111H (W.O. 24901095)
b. Rebuild of Relief Valve RV-0527 (W.O. 24901031)
c. Auxiliary Feedwater Pump Governor packing adjustment (W.O. 24901446)
d. Blowdown flow meter cleaning and repair (W.O. 24901304)
e. Pump (P-60) alignment and coupling (W.O. 24901341, RWP P890240)
f. Remove, calibrate and reinstall the 0-3 cfm flow meter (W.O. 24901663)
g. Install mechanical tube plugs in "B" steam generator (W.O. 24900478). The inspector confirmed the plugs were receipt inspected by review of Purchase Order CP11-7012Q.

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

6. Surveillance (61720, 61726, 42700)

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 activities were inspected:

a. QO-21 Auxiliary Feedwater System Valve Test (CV-0521)

This test demonstrated the operability of the steam admission valve and also the turbine control after adjustment of the governor packing. The test was successful but also revealed that the steam pressure control was still not optimum in that the relief valve lifted several times before steady state operation was achieved.

The inspector verified that long range plans exist to resolve this issue which include the installation of new air supplies to the steam valves which will modulate their operation and, if that is not successful, system design changes.

b. DWO-1 Daily Control Room Surveillance.
c. SHO-1 Operators Shift Surveillance.
d. SO-4b Escape Air Lock Penetration Leak Test.
e. DWO-13 Local Leak Rate Test for Inner and Outer Personnel Air Lock Door Seals"
f. SO-4a Personnel Air Lock Penetration Leak Test

The inspector performed a technical review of SO-4b and DWO-13 using 10 CFR 50, Appendix J, Standard Technical Specifications and the vendor manuals (Licensee Files C-53) as references and referred the following items to the site licensing engineer.

(1) The inspector found that a between the seal test of the escape lock door was not performed after the escape air lock penetration test was completed on March 2, 1989. Interviews with personnel involved with the test indicated that the air lock doors were opened a number of times to restore the air lock to service following the test. Restoration includes removal of the strong backs and fluffing of the seals to remove the seal set induced by the strong backs.

10 CFR 50, Appendix J, III.D.2.(b) (iii) requires that air locks opened during periods when containment integrity is required shall be tested within three days after being opened. For air lock doors that have testable seals, testing the seals fulfills the three-day test requirements. In the event that the testing for the three-day interval cannot be done at Pa, the test pressure shall be stated in the Technical Specification.

The current Technical Specifications do not address 10 CFR 50, Appendix J, III.D.2.(b) (iii) but the licensee does have a Technical Specification change request that is pending with NRR. The proposed Technical Specification does require a reduced pressure between the seal test after use of air locks. The licensee is implementing this requirement after each use of the personnel air lock but not for the escape air lock. The escape air lock vendor's manual does not recommend a between the seals test be performed at Pa but does state the seal should be capable of a reduced test pressure.

Failure to do a between the seals test of the escape air lock after each use of the air lock is a violation of 10 CFR 50, Appendix J, III.D.2.(b) (iii) (255/89009-01(DRP)).

(2) The inspector noted that the licensee Technical Specification change request is for a reduced pressure between the seals test. However, the proposed Technical Specification does not provide an acceptance criteria. A review of Standard Technical Specification reveals that a general acceptance criteria for reduced pressure testing appears to be .01 La. The acceptance criteria implemented in DWO-13 is approximately .025 La. The inspector asked the licensing engineer to review the information and assure the appropriate acceptance criteria is used. In addition, the inspector discussed this item with the NRR Project Manager on March 16 for consideration during review of the Technical Specification change request.

g. MO-7A Emergency Diesel Generator Test.

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

7. Security (71707)

Routine facility security measures, including control of access for vehicles, packages and personnel, were observed. Performance of dedicated physical security equipment was verified during inspections in various plant areas. The activities of the professional security force in maintaining facility security protection were occasionally examined or reviewed, and interviews were occasionally conducted with security force members. Tours of the central and secondary alarm station were routinely conducted.

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

8. Safety Assessment/Quality Verification (35502, 40500)

The effectiveness of management controls, verification and oversight activities, in the conduct of jobs observed during this inspection, was evaluated.

The inspector frequently attended management and supervisory meetings involving plant status and plans and focusing on proper coordination among Departments.

The results of licensee auditing and corrective action programs were routinely monitored by attendance at Corrective Action Review Board (CARB) meetings and by review of Deviation Reports, Event Reports, Radiological Incident Reports, and security incident reports. As applicable, corrective action program documents were forwarded to NRC Region III technical specialists for information and possible followup evaluation.

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

9. Reportable Events (92700, 92720)

The inspector reviewed the following Licensee Event Reports (LERs) by means of direct observation, discussions with licensee personnel, and review of records. The review addressed compliance to reporting requirements and, as applicable, that immediate corrective action and appropriate action to prevent recurrence had been accomplished.

(Closed) LER 255/84001: Loss of offsite power and communications. Appropriate modifications, training, and procedure changes were made, which resolved the myriad of issues involved in the event. Licensee Reports D-PAL-84-003, D-PAL-84-007, D-PAL-87-004, NAPO Report P85-36, and PRC minutes 85-37 were reviewed.

(Closed) LER 255/840022-02: Containment temperature exceeded the analyzed value. The licensee revised their surveillance procedures to require a plant shutdown if the containment temperature exceeds the currently analyzed 145 degrees Fahrenheit. A Technical Specification limit is planned to be submitted as part of the Restructured Technical Specifications. The licensee is continuing to evaluate methodology for determining air temperature and the adequacy of installed instrumentation.

(Closed) LER 255/86004-02: Main Steam Relief valves failed to meet the "as found" acceptance criteria. A Technical Specification change was submitted and approved relaxing the tolerance requirements from +/- 1 percent to +/- 3 percent, which bounded the tolerances identified during this testing.

(Closed) LER 255/86017-01: Primary Coolant System (PCS) leakage was greater than one gpm requiring a plant shutdown. Two sources were found. Relief Valve RV-2006 which protects the letdown system piping had not reseated and was discharging the PCS to the quench tank. Disassembly of the valve determined that the bellows were distorted. Evaluation of the valve determined that it was suitable for its service and it was repaired and reinstalled. The secondary cause of the leakrate, was the failure of three valves in the reactor head vent system to reseat. These valves, PRV-1067, PRV-1068, PRV-1072, were found to contain metal shavings which were left in the system during fabrication. This conclusion was reached after the shavings were analyzed by Battelle (Columbus) to determine their source. The valves were replaced, the head vent system was flushed and corrective action was taken to improve cleanliness controls during modification work (refer to D-PAL-86-154).

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

10. Engineered Safeguards System Walkdown (71710)

The inspector performed a walkdown of the Chemical Spray Additive System and verified that each accessible valve in the flow path was in its required position and operable, that power was aligned for components that actuate on an initiation signal, that essential instrumentation was operable; and that no conditions existed which could adversely affect system operation. The licensee was provided the identification of a valve which was missing a label, and questions about changes to the valve lineup for their resolution.

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

11. Allegation Followup (99024)

(Closed) Allegation (AMS No. RIII-88-A-0163): The following discussion relates to an allegation concerning the radiation protection program, which was evaluated during this inspection. The evaluation consisted of a review of licensee records and NRC Inspection Reports, and discussions with licensee personnel.

Concern No. 1: A dose of 1800 mrem assigned to an individual for steam generator work in February 1978 was too low.

Discussion: The alleged stated that during each of three separate entries ("jumps") into a steam generators, his dosimeter went off scale. He stated that he was assigned a dose of 1800 mrem total for the three jumps, but he contends that this assigned dose was too low. No documentation was provided by the alleger to support his contention that his dose should have been higher. A review of licensee's dosimetry records maintained for the individual indicated that the individual was provided with three separate dosimetry devices to monitor his radiation dose. According to the records, on January 23, 1978, the individual's self-reading dosimeter (SRD) went off scale and his TLD badge read 670 mrem (the inspectors could not establish the range of the dosimeter worn by the alleger), on January 24, 1978, the dosimeter again went offscale and the TLD badge read 999 mrem; and on January 26, 1978, the dosimeter again went offscale and the TLD badge read 413 mrem, for a total exposure by TLD badge of 2082 mrem. The record also indicated that on January 30, 1978, the dosimeter read 0 and the TLD badge read 4 mrem, and on January 31, 1978, the dosimeter read 0 and the TLD badge read 2 mrem, for additional total exposure by TLD of 6 mrem. In total, for the month of January 1978, the record lists the individual's final exposure according to his TLD badges as 2088, while his monthly film badge reading was 1800 mrem. (There was no indication that the individual received any radiation dose in February 1978.) At the time, Palisade's practice was to use the film badge as the official record of worker exposure, to satisfy the requirements of 10 CFR 20, and use the less accurate TLD badge and the SRD as secondary or back-up dosimeters to quickly provide exposure information. The TLDs and SRDs were read onsite after each entry, whereas the film badges had to be sent to an independent contractor offsite for a monthly readout. The 14% difference between the TLD dose and the film badge dose is consistent with the accuracy of the two methods and is not a health and safety concern nor a regulatory concern.

An inspection conducted at Palisades by NRC radiation specialists in January 1978, reviewed the licensee's control over outage activities, specifically steam generator work (Inspection Report No. 50-255/78-02). The inspectors noted that during the work, the licensee experienced a shortage of low-range (0-200 mrem) and high-range (0-1 rem and 0-5 rem) SRDs; however, the inspectors verified that workers were sent into areas only after surveys had been performed, wore film and TLD badges, and were limited on their stay times so that an over-exposure was unlikely. From a health and safety perspective, the limitation on stay time imposed by the licensee was an acceptable substitute for the lack of SRDs.

Findings: The allegation is not substantiated. Although the alleger's SRD went off scale during three steam generator jumps, the individual's dose was adequately monitored by a film badge and a backup TLD dosimeter. The 1800-mrem dose assigned to the alleger was based on the film badge, which was the licensee's officially designated dosimeter at the time. The allegation is closed.

Concern No. 2: The licensee did not evaluate an uptake received by the alleger when his airline became disconnected during a steam generator jump.

Discussion: The inspectors were not able to retrieve any licensee records that specifically dealt with the alleged airline disconnect. However, the inspectors did review records that indicated that the alleger received a whole body count on February 1, 1978, shortly after his steam generator entries. Whole body counting is an acceptable method for evaluating possible uptakes of radioactive material, including that found in steam generator channel heads during maintenance. The results of this whole body count indicated the presence of approximately four nanocuries of Cs-137 in the individual, a value which indicates exposure to airborne radioactivity of no more than about 0.1% of the NRC limit during the individual's steam generator jumps. NRC review in 1978 of the licensee's whole body counting system and results of many counts conducted during the outage determined that the licensee whole body counting system was adequate and no exposures to airborne radioactive material in excess of regulatory limits occurred during the outage (Inspection Reports No. 50-255/78-02; 50-255/78-11).

Findings: The allegation is not substantiated. The licensee conducted a whole body count of the alleger shortly after the steam generator work. Although this whole body count may not have been specifically given to evaluate any uptake resulting from the alleged airline disconnect, it served that purpose. The amount of radioactive material detected by the whole body count indicated that the individual was not exposed to airborne radioactivity in excess of regulatory limits during his steam generator work. NRC review during the period of concern determined that the licensee was providing adequate internal exposure control and assessment. The allegation is closed.

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

12. Management Interview (30703)

The inspectors met with licensee representatives (denoted in Paragraph 1) on April 12, 1989, to discuss the scope and findings of the inspection. In addition, the inspector also discussed the likely informational content of the inspection report with regard to documents or processes reviewed by the inspector during the inspection. The licensee did not identify any such documents/processes as proprietary.