UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA ST., N.W.
ATLANTA, GEORGIA 30303

Report No.: 50-302/83-24

Licensee: Florida Power Corporation
3201 34th Street, South
St. Petersburg, FL 33733

Docket No.: 50-302

License No.: DPR-72

Facility Name: Crystal River 3

Inspection at the Crystal River site near Crystal River, Florida

Inspector: J B Kahle 9/7/83
J. B. Kahle Date Signed

Approved by: K P Barr 9/12/83
K. P. Barr, Section Chief Date Signed
Operational Programs Branch
Division of Engineering and Operational Programs

SUMMARY

Inspection on August 8-12, 1983

Areas Inspected

This routine, unannounced inspection involved thirty-five inspector-hours on site in the areas of solid radioactive waste management, packaging and transportation of radioactive materials, followup on selected NUREG 0737 items, followup on inspector identified items, QA audits of health physics programs, and followup on licensee identified items.

Results

Of the six areas inspected, no violations or deviations were identified in five areas; one apparent violation was found in one area (transportation).

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REPORT DETAILS

1. Persons Contacted

   Licensee Employees

   *E. M. Howard, Site Nuclear Operations Director
   *G. L. Boldt, Nuclear Plant Operations Manager
   *B. Hickle, Chem-Rad Superintendent
   *S. Mansfield, Nuclear Compliance Specialist
   S. Lashbrook, Acting Plant Health Physicist
   D. Wilder, Chem-Waste Supervisor
   G. Clymer, Acting Nuclear Waste Manager
   L. Bayer, QC Supervisor
   B. Chastain, Plant Engineer
   W. Johnson, Training Instructor
   B. Crane, Training Manager
   K. Wilson, Site Nuclear Licensing Supervisor
   L. Giles, Assistant Shift Supervisor
   J. Barrett, Engineer
   S. Primo, Engineer
   G. Halnon, Enginner
   E. Welch, Engineer

   Other licensee employees contacted included two technicians, two operators, and two office personnel.

   NRC Resident Inspector

   T. Stetka

   *Attended exit interview

2. Exit Interview

   The inspection scope and findings were summarized on August 12, 1983, with those persons indicated in paragraph 1 above. Management representatives acknowledged the item of noncompliance but emphasized that the safety of transporting radioactive material had not been compromised.

3. Licensee Action on Previous Enforcement Matters

   Not inspected.

4. Unresolved Items

   Unresolved items were not identified during this inspection.

5. Followup on Inspector Identified Items

   a. (Closed) IFI - 83-15-01, Accidental exposure of TLD badges. The licensee was unable to identify the source of radiation to which the February batch of TLD badges were exposed. The investigation showed that no radioactive sources were received or shipped during the time the badges were received or shipped. The licensee assumed that the badges were accidently exposed during the shipment of the badges between Eberline and the licensee's site. The inspector had no further questions.

   b. (Closed) IFI - 83-15-02, Procedures for receipt of radioactive material. The inspector verified that procedures were maintained for picking up and receiving packages of radioactive material. Procedures were maintained for surveillance of packages for radiation and contamination pursuant to 10 CFR 20.205. The inspector had no further questions.

6. Solid Wastes

The inspector discussed the procedural requirements for dewatering primary and radwaste demineralizer resins and the concreting method for solidifying evaporator bottoms. This work is accomplished by a contractor under the auspices of the licensee. Burial site dewatering limits and other requirements were discussed with licensee representatives.

The inspector reviewed the licensee's procedures for segregating radioactive wastes which are conducive to compaction and wastes which are not compacted. Compacted wastes are compacted in 55-gallon drums while the noncompacted wastes (wood, metal, large objects, etc.) are placed in large metal boxes, approximately 4'x 4'x 6' (B-25's). The inspector discussed methods used to assure that boxes or drums contain no liquids when shipped to the burial site. Procedural controls and visual inspections are the primary methods. Measurement of the quantity of radioactivity in the drums and boxes is determined by the technique described in the Health Physics Journal by taking contact readings along the side of the container and using the graphs provided for converting to the radioactivity quantity.

Volume reduction is accomplished through special and routine annual training of licensee and contractor personnel. Signs have been posted at the entrance to the plant and in work areas which address the methods of volume reduction. Decontamination is accomplished whenever possible. Examination and inspection of contaminated wastes are performed frequently to identify objects and materials which should not be in the radiation control area (RCA) and objects and materials which can be decontaminated.

Through discussions with licensee representatives, it was determined that the licensee is familiar with 10 CFR 61 requirements and is presently working these requirements into the radioactive waste procedures.

The inspector examined containers of radioactive waste and the associated storage area on the berm outside the auxiliary building. It appeared that the containers were properly closed, marked and labelled. A radiation survey showed no unacceptable radiation intensities.

The inspector had no further questions.

7. Packaging and Transportation of Radioactive Materials

A licensee representative stated that the revised DOT regulations have been used since July 1, 1983. The licensee's procedures for shipping radioactive materials are being revised to be consistent with the revised regulations and are in final revision form. A licensee representative showed the inspector a copy of the Federal Register, Vol. 48, No. 48, Thursday, March 10, 1983, which provides the new DOT regulations. It was apparent that the licensee representative was familiar with the new changes and the impact on their operation.

The inspector selectively examined several records of radioactive material shipments. Details pertaining to the shipments were discussed with the licensee representative. It was concluded from the discussions that the quantity of radioactive material was not appropriately determined for a shipment of a contaminated plenum stand made to Allied Nuclear, Inc., License No. WN-10171-1, on February 28, 1983. The quantity was determined by multiplying the radioactivity on the highest smear for removable contamination and the total area of the stand. Licensee representatives were informed that the technique showed only what contamination or radioactivity was removed by smearing and not the total radioactivity involved. The plenum stand was sent to Allied Nuclear for decontamination and release as nonradioactive material, consequently the quantity of radioactive material will never be known or if the quantity met the low specific activity (LSA) limit as defined in 49 CFR 173.389(c). The licensee was informed that failure to appropriately measure and determine the radioactivity on the plenum stand was a violation of 10 CFR 71.5, which requires that licensee deliver packages of radioactive material for transport in accordance with 49 CFR regulations. 49 CFR 172.203(d)(iii) requires that the description for a shipment of radioactive material must include the activity contained in each package of the shipment in curies, millicuries, or microcuries. (83-24-01)

For making the shipment as radioactive material LSA, the licensee determined the radioactivity concentration by dividing the radioactivity on the plenum stand by the weight of the stand. The inspector stated that 49 CFR 173.389(c)(5) definition for LSA was more appropriate because the plenum stand would be nonradioactive material externally contaminated with radioactive material. The licensee felt that both definitions of LSA were applicable because of the added information following the 49 CFR 173.389-(c)(4) definition, "NOTE: This includes, but is not limited to, materials of low radioactivity concentration such as residues or solutions from chemical processing; wastes such as building rubble, metal, wood, and fabric scrap, glassware, paper, and cardboard; solids or liquid plant waste, sludges, and ashes." The licensee had determined that the plenum stand met the metal waste category. The licensee stated that he had no problem in using the surface contamination instead of the concentration definition. After much discussion it was concluded that the shipment would have been made in the same manner had the total radioactivity been appropriately determined and the surface contamination instead of the concentration definition been used.

8. Health Physics Audits

The inspector examined a QA audit report of an audit conducted from August 23, 1982, through September 8, 1982, of health physics activities at the plant. The documentation showed that the audit was well planned and approved prior to the auditing activities and covered all areas of radiation protection. Findings were acknowledged, reviewed and investigated. Causes for discrepancies were determined and schedules established for corrective actions and actions to prevent recurrences. The inspector had no further questions.

9. NUREG 0737

a. II.B.2.2, Shielding Design Review

In response to NUREG 0737 Item II.B.2.2, "Plant Shielding Modifications for Vital Area Access", a design review of the Crystal River 3 plant shielding was performed. In accordance with the requirements, radiation source terms were specified, systems assumed to contain high levels of radioactivity as a result of a postulated accident were determined, vital areas requiring access were identified, and dose rates in various plant areas and vital areas were calculated.

The licensee's responses were reviewed. The assumptions and methodology employed by the licensee in the shielding design review were found to be consistent with the requirements. Source terms were based on source term requirements contained in NUREG 0737. The systems identified as potentially containing high concentrations of radioactivity following an accident were found to be consistent with system functions.

Licensee responses to this item were dated January 11, 1980, December 15, 1980, October 2, 1981, January 6, 1982, April 14, 1982, June 4, 1982, June 18, 1982 (two letters), June 30, 1982, September 8, 1982, September 14, 1982, September 17, 1982 and October 12, 1982. The licensee identified areas which would require access or occupancy in order to mitigate the consequences of the postulated accident. Each area was evaluated in the plant shielding design review to ensure that these areas would be accessible without exposing an individual to radiation in excess of GDC 19 criteria. The licensee identified the control room, diesel generator room, nuclear sample room, hydrogen purge equipment, containment air monitor RM-A6, radioactive waste disposal control board, radiochemistry laboratory and count room as vital areas requiring access during post-accident conditions. A comprehensive review by Florida Power Corporation of the waste disposal control board showed that access or relocation of the waste disposal panel did not increase the capabilities of operators to control and mitigate the consequences of an accident.

Due to the findings of the shielding design study, the licensee determined that no shielding modifications are required. The inspector noted that Florida Power Corporation had not completed installation of their permanent Post-Accident Sampling System (PASS) which will be located in a vital area in the auxiliary building. Evaluation of shielding requirements for post-accident sampling will be conducted in conjunction with NUREG 0737, Item II.B.3, "Post-Accident". As a result of the shielding study, certain modifications were accomplished to make it unnecessary to have post-accident access to certain areas. These modification changed decay heat valves (DHV) 7, 8, 39 and 40 to motor operated valves and added motor operated bypass valves for the Makeup and Purification System filter. These were MUV-100, 194, and 452. Modification of the waste gas system such that venting of the holdup tank can be performed entirely form a remote, low radiation area. Verification was made by the inspector that modifications have been completed and the equipment and systems are operable. Several emergency procedures were reviewed to determine the adequacy of worker protection for re-entry to control and mitigate the consequences of a postulated accident. Entry into the reactor building would not be necessary to mitigate the consequences of an accident. A plant walk-down of the procedure to collect a reactor building atmosphere air sample under accident conditions was conducted. It was determined that this vital area was accessible and the stay times to collect the samples were within the times used in the design review shielding analysis.

Based on the NRC Region II review of the Florida Power Corporation Crystal River 3 shielding design review, inspection of the plant modifications made as a result of the shielding study and the performance of an independent assessment of the vital area accessibility and personnel doses in a post-accident condition, the requirements of NUREG 0737, Item II.B.2.2 have been met and are acceptable.

b. III D.3.3 Improved Iodine Sampling

The inspector verified that the licensee had adequate instrumentation and sampling media for collecting samples to determine airborne iodine concentrations in areas within the facility where plant personnel maybe present during an accident. Verification was made that the technicians have been trained and procedures are maintained for determining the airborne iodine concentrations. The inspector had no further questions.

10. Radiation Dose Update

The licensee provided the inspector with the latest data of radiation dose to personnel on a monthly basis. The monthly data for 1982 and 1983 are as follows:

	1982 (mrem)	1983 (mrem)
January	5433	6894
February	37795	81055
March	3881	28545
April	2847	101548
May	4121	106462
June	1583	61060
July	1109	
August	4491	
September	3060	
October	19322	
November	5364	
December	74385	

11. Auxiliary Building Ventilation System Review

As a result of an allegation concerning the improper design of plant ventilated systems at the Crystal River Nuclear Plant, the inspector reviewed the design and operation of the Auxiliary Building (AB) ventilation system with cognizant engineers. From these discussions and an examination of the AB ventilation drawings, verification was made that the system was installed in accordance with the description provided in the FSAR. The system is designed and operated continuously to maintain a negative internal building pressure with the air movement from lesser potential to higher potential contaminated areas. A calculation showed that the number of air changes per hour exceed 10 not considering the volume of walls, equipment, tanks, etc. In the event of an alarm indicating high radiation in the auxiliary building exhaust vent (RM-A2), the supply fans automatically stop, but the exhaust fans continue operation. This further increases the negative internal pressure in the AB, thus assuring no uncontrolled leakage to the outside. During an alarm condition the AB is evacuated of personnel and re-entry is made under conditions where precautionary measures are taken by personnel. When the supply fans automatically stop, a "bypass" or "make-up" damper opens allowing for sufficient air flow from the outside to the exhaust fans to prevent cavitation. The alleger claimed that this damper was intended to provide a constant velocity of the plant exhaust air flow to meet a requirement of the radiation measurement device for the system.

Experience over the past several years has shown that, except for isolated controlled conditions, high concentrations of airborne radioactivity materials have not been detected in areas where personnel work within the AB. Radioactive contamination has not been spread to uncontrolled areas because of improper ventilation. Whole body count results show no internal deposition of radioactive material in workers as a result of exposure to excessive concentrations airborne radioactive material.

Based upon the above review, the allegation that the ventilation system was of an improper design was not substantiated.

12. IE Notice No. 83-05: Obtaining Approval For Disposing Of Very Low-Level Radioactive Waste - 10 CFR Section 20.302

Licensee representatives stated that they had received the notice and consideration was being given to making a proposal to dispose of contaminated oil pursuant to 10 CFR 20.302(a). The inspector had no further questions.

13. LER 83-001

A licensee representative stated that a study is still in progress to initiate engineering changes to correct the problem, consequently the interim corrective actions of personnel verifying that RM-A3, RM-A4 and RM-A5 on an hourly basis are still in effect. The inspector stated that followup on this item would be performed during a subsequent inspection.

14. NCOR's

a. NCOR 83-207

A review of the licensee's nonconforming operating report (NCOR) 83-207 showed that two individuals had exceeded the licensee's administrative exposure limit of 300 mrem. The investigation showed that the limit was exceeded because of an incorrect reading obtained with a Teletector. The 0-2R/hr scale was off (low) by a factor of 3. The other ranges appeared satisfactory when the instrument was recalibrated. A check of the calibration records showed the Teletector to perform satisfactorily during the previous calibration. No reason could be determined to explain why the Teletector failed to function properly on the 0-2R/hr scale. A licensee representative stated that Teletectors are not rugged instruments and precautions are taken to preclude jarring the instruments or bumping them with other objects. Usually, if an instrument is out of calibration from rugged treatment, all three scales are out. The instruments are source checked on the lower scale, 0-25 mr/hr.

TLD results showed the individuals received 492 and 444 mrem with the high range pocket dosimeters showing 475 and 430 mrem, respectively. No regulatory limits were exceeded. The inspector had no further questions.

b. NCOR 83-218

An examination of this report showed that an individual exceeded the licensee's administrative exposure limit of 300 mrem. Apparently the individual took longer than planned to perform a job. He was not wearing a high range dosimeter and his 0-200 mrem dosimeter was off scale. The TLD result showed 445 mrem.

The report showed that there was improper planning with respect to the RWP and inadequate communications regarding the wearing of high range pocket dosimeters. As a result of this administrative limit being exceeded, written instructions were issued to technicians and waste handling personnel to thoroughly discuss all work involving the handling of high level waste and clarifying when high range pocket dosimeters will be worn.

No regulatory limits were exceeded and the inspector had no further questions.

15. Training

The inspector took the abbreviated training course in lieu of the annual retraining required to obtain a yellow security badge for unescorted access to the plant. The inspector discussed ways of reducing the length of time to receive the retraining and still assure that individuals were knowledgeable of site specific requirements. Licensee representatives stated that considerations were being given to this concern and to other groups of individuals who had general radiation protection knowledge and only needed site specific retraining.