APPENDIX B

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

NRC Inspection Report No. 50-285/92-03

Operating License No. DPR-40

Licensee: Omaha Public Power District (OPPD)
444 South 16th Street Mall
Mail Stop 8E/EP4
Omaha, Nebraska 68102-2247

Facility Name: Fort Calhoun Station (FCS)

Inspection At: Blair, Nebraska

Inspection Conducted: January 29-30, February 18-21, and April 20-24, 1992

Inspectors: Howard F. Bundy, Reactor Inspector, Test Programs Section Division of Reactor Safety

Dr. Dale A. Powers, Senior Reactor Inspector, Test Programs Section, Division of Reactor Safety

Approved: James E. Gagliardo, Chief, Test Programs Section, Division of Reactor Safety — 5/26/92 Date

Inspection Summary

Inspection Conducted January 29-30, February 18-21, and April 20-24, 1992 (Report 50-285/92-03)

Areas Inspected: Special, announced inspection of the licensee's in-core fuel loading and fuel storage configurations, core component performance, outage work controls and critical path scheduling, potential for fuel-related problems identified at other facilities, fuel handling procedures and practices, post-irradiation examinations of fuel assemblies, and disposition of degraded core components.

Results: The licensee's staffing to accomplish the refueling process was reasonable. The licensee had initiated the Outage Management Team concept, which had beneficially resulted in relatively little personnel traffic in the control room.

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The licensee failed to implement adequately pre-established controls designed to preclude foreign materials from entering the refueling and spent fuel pool exclusion areas. One consequence of this failure resulted in the licensee shutting down refueling operations when the inspector observed that the refueling machine load cell was not registering the weight of a fuel assembly, which was being withdrawn from the core, because of materials that were inappropriately stored under the refueling machine console had fallen and tripped the power supply switch for the load cell. Other requirements specified by the subject procedure had also not been satisfied. The licensee's corrective actions taken following their notification of this apparent violation were prompt and comprehensive.

The licensee's use of visual and audio aids to assist the refueling crew in containment was considered a programmatic weakness. There were no flood lights in the reactor vessel and the top of the core was not visible to the unaided eye. The hoist-mounted underwater light for the closed-circuit camera, used to verify fuel manipulations, was not always being used, and the television monitor's image on the refueling machine was degraded. The refueling crew did not have binoculars or other means of aiding in the viewing of the core components. The licensee had provided an audible neutron count rate system, but (1) the system's design had not been implemented properly and (2) members of a refueling crew were unaware of the location of the count rate speaker. The refueling crew did not always check the path of the refueling machine trolley and bridge to ensure that personnel and equipment were clear of the system.

The licensee had no formal guidance to preclude operators from leaving unattended irradiated fuel assemblies that were suspended from the refueling machine or the fuel handling machine. The licensee had a good core alignment procedure that was used to reduce the potential for interference between fuel assemblies and the upper guide structure.

The licensee had maintained a competent nuclear engineering staff with notable in-house computing and reload safety analysis capabilities. The licensee's nuclear engineering staff was assertive in maintaining cognizance of fuel performance and potential adverse impacts as evidenced by the requirement that all proposed changes to fuel assembly design be reviewed and approved by the licensee prior to implementation in the reload batch. The licensee's fuel examination activities and first-time application of a fuel assembly reconstitution process were successful and performed without encountering any significant problem.

Through modifications, the licensee had removed plant equipment, but had not properly reflected such deletions in existing procedures such as the control element assembly change machine (stand) and part-length control element assemblies.

One apparent violation (failure to implement pre-established controls to prevent foreign material from entering the refueling cavity and the spent fuel pool, paragraph 2.7), one unresolved item (degradation of a vital-to-vital area barrier, paragraph 3.), and one inspection followup item (erroneous alarm setpoint on the component cooling water system, paragraph 2.10) were identified.

DETAILS

1. PERSONS CONTACTED

OPPD

*J. Adams, Nuclear Design Engineer
#R. Andrews, Division Manager, Nuclear Services
J. Bobba, Supervisor, Maintenance
J. Chase, Outage Manager
R. Clemens, Supervisor, Outage Projects
*G. Cook, Supervisor, Station Licensing
+D. Eid, Engineer, Station Licensing
J. Fluehr, Senior Nuclear Design Engineer
#S. Gambhir, Division Manager, Production Engineering
+J. Gasper, Manager, Training
#W. Gates, Division Manager, Nuclear Operations
M. Guinn, Supervisor, Reactor Physics
K. Guliani, Senior Engineer - Nuclear
*A. Gurtis, Senior Quality Assurance Engineer
#K. Holthaus, Manager, Nuclear Engineering
#R. Jaworski, Manager, Station Engineering
#W. Jones, Senior Vice President
*L. Kusek, Manager, Nuclear Safety Review Group
+T. Matthews, Acting Supervisor, Nuclear Licensing
*W. Orr, Manager, Quality Assurance/Quality Control
#T. Patterson, Manager, Fort Calhoun Station
M. Puckett, ALARA Coordinator
D. Ritter, Supervisor, Security Operations
D. Rosloniec, Station Technical Advisor
+J. Sefick, Security Manager
P. Sepcenko, Senior Field Engineer - Nuclear
#R. Short, Manager, Nuclear Licensing
#C. Simmons, Station Licensing Engineer
*J. Spilker, Reactor Engineer
D. Stading, Project Engineer, Nuclear Projects
B. Weber, Supervisor, Reactor Analysis

NRC

+R. Azua, Resident Inspector
*R. Mullikin, Senior Resident Inspector

During the inspection, the inspectors also contacted other licensee personnel.

*Denotes those in attendance at the interim exit meeting on February 21, 1992.
+Denotes those in attendance at the exit meeting on April 24, 1992.
#Denotes those in attendance at both exit meetings.

2. FUEL INTEGRITY AND REACTOR SUBCRITICALITY (60705/60710/86700/92702)

The objectives of a Fuel Integrity and Reactor Subcriticality (FIRS) inspection are to review, inspect, and determine the adequacy of the licensee's activities related to the protection of reactor fuel. Attachment 1 to this inspection report is a tabulation of documents that were reviewed by the inspectors during the inspection and provided the basis for the findings documented in this report. Other licensee documents that discussed fuel-related activities and associated equipment designs and operational characteristics were made available to the inspectors and were examined in much less detail. In general, the reviews of procedures and records were not detailed in nature, but rather were broad overviews to determine that essential issues were addressed in reasonable fashion. Information on several aspects of the licensee's activities were based on interview statements taken from licensee staff members and, a sampling of those statements were verified by review of the Technical Specifications or the licensee's procedures and records. Emphasis, however, was given to reviewing the following areas:

- In-core fuel loading and fuel storage geometrical controls to preclude configurations that have not been specifically approved by NRC in safety evaluation reports and that conceivably could result in situations involving inadequate shutdown margin or inadvertent criticality;
- Operational work control practices, communications, procedures, physical systems and equipment, and training that preclude unsafe fuel movements from occurring;
- Licensee evaluations and corrective actions that were performed subsequent to any self-identified problems that were indicative of accident sequence precursors or that had the potential to lead to fuel damage; and
- The susceptibility of the licensee's operations, procedures, and equipment to fuel-related problems that have occurred at other nuclear power plants.

NRC Inspection Manual Procedure 60705, "Preparation for Refueling"; Inspection Procedure 60710, "Refueling Activities"; Inspection Procedure 86700, "Spent Fuel Pool Activities"; and Inspection Procedure 92702, "Followup on Corrective Actions for Violations and Deviations," provided guidance for this inspection effort.

2.1 Fuel-Related Incidents at Other Facilities

The inspectors discussed with the licensee several fuel-related events that have occurred at commercial nuclear power plants. Specifically, the incidents that were discussed are described in NRC Information Notices (INs) and Bulletins that were issued during the past decade. Attachment 2 contains a listing of those INs and Bulletins which the inspectors discussed with the licensee for this assessment

The inspectors questioned the licensee as to whether or not such publicized problems had occurred at FCS; the licensee's equipment or operations had the potential for such problems; or any corrective actions had been initiated by the licensee to preclude such problems. Those INs that had some relevancy to the licensee's operations, that the licensee had evaluated and taken significant corrective action, or that were of particular interest to the inspectors are discussed below. Other INs were viewed by the inspectors not to be directly applicable to the licensee's operations. This nonapplicability was because, for instance, the licensee's designs, practices, or procedures at the time of the IN issuances should have precluded such incidents from occurring. These nonapplicable INs are not discussed in this report. A synopsis of the relevant INs is discussed below.

2.1.1 IN 88-92

In IN 88-92 and Supplement 1, "Potential for Spent Fuel Pool Draindown," the licensee was informed of a potential for the inadvertent draindown of a spent fuel pool. The initiator for such an event could be a catastrophic failure of the seal on the spent fuel pool fuel transfer gate, especially when the fuel transfer tube was open to containment and the refueling cavity was dry and operator presence was not required in the immediate area. The inspectors also discussed the hypothetical scenarios given in NRC Inspection Report 50-482/91-28 that provided the NRC Augmented Inspection Team findings of the event involving a loss of spent fuel pool level and cooling that occurred at the Wolf Creek Generating Station on September 23, 1991.

The licensee's assessments of the potential for an inadvertent draindown scenario of the spent fuel pool were conducted as a result of their reviews of both IN 88-92 (and Supplement 1) and IN 88-65, "Inadvertent Drainages of Spent Fuel Pools." The assessments concluded that no changes were required.

The FCS spent fuel pool area contains two wet cavities: the fuel transfer canal and the spent fuel pool. The cavities are separated by a short corridor, which has a slot for the fuel transfer gate. The fuel transfer gate is a non-hinged partition that is positioned with the help of the auxiliary building crane. The licensee's practice is to have the fuel transfer gate in place separating the spent fuel pool from the transfer canal when fuel is not being moved. For storage, the fuel transfer gate is placed on the wall in the transfer canal. The fuel transfer tube is separated from the refueling cavity by a blind flange. On the auxiliary building side, a gate valve separates the tube from the transfer canal.

The FCS spent fuel pool has four resistance temperature detectors. Temperature indications are available locally in the auxiliary building, and alarm indications are available locally and in the control room. Two differential pressure level detectors are installed in the spent fuel pool, one for high level and one for low level. Level indications are available locally in the auxiliary building, and alarm indications are available locally and in the control room. There are also radiation area monitors in the vicinity of the spent fuel pool to alert operators to a decreasing spent fuel pool water level. Once every 8-hours, the auxiliary building operator was required to confirm the level of the spent fuel pool and the operation of a spent fuel pool cooling pump. A control room operator was required to confirm the spent fuel pool heat exchanger outlet temperature every 8 hours.

The inspectors questioned the licensee on the administrative controls exercised during non-refueling operations over the blind flange and the gate valve of the fuel transfer tube and the refueling cavity drains and equipment hatch access lids when activities such as maintenance on the fuel transfer trolley cart were in progress. The question was asked to determine whether the fuel transfer tube was a potential leakage path when the fuel transfer tube was dry and in communication with containment. The licensee's representative responded that drains to the containment sump and the auxiliary building radwaste system were not necessarily required to be secured during surveillance and maintenance activities on the fuel transfer system.

There are three beneficial design aspects that minimize the potential for an inadvertent draindown of the FCS spent fuel pool. First, the spent fuel pool transfer gate seal is a solid, rubber seal that does not require pressurization. Second, the top of the spent fuel transfer gate opening is about 3 feet above the top of the active spent fuel. The inspector noted to the licensee's representative that page 9.5-6 of the Updated Final Safety Analysis apparently needed revision to indicate correctly that a plate had been installed across the bottom of the "gate opening" rather than across the bottom of the gate. The plate had been added to raise the minimum spent fuel pool draindown level. Third, the licensee employed a permanently installed reactor vessel-to-cavity metal seal ring, rather than a pneumatic seal ring, which would be more prone to leakage.

The inspectors questioned the licensee on a second scenario. This scenario involved the potential for exposing an irradiated fuel assembly that might be either (1) held in a vendor's examination stand in the spent fuel pool or (2) suspended from the spent fuel pool fuel handling machine when a spent fuel pool draindown event occurred. In regard to the former circumstance, the inspector learned that the examination and reconstitution efforts during the 1992 refueling outage utilized a vendor's examination stand that sat directly upon the spent fuel racks and did not require fuel assemblies to be completely lifted from the racks. The design of this examination stand, consequently, ensured that spent fuel, which was examined, was suspended at a lower elevation than with other types of examination stands. Therefore, the susceptibility to exposing irradiated fuel in the event of a spent fuel pool draindown scenario was lessened.

In regard to the second circumstance when electrical power might be lost, the reactor engineer stated that the FCS fuel handling machine has a handcrank that could allow an operator to manually lower any fuel assembly that might be suspended from the machine at the time of the power failure. (As discussed below in paragraph 2.6, the licensee's procedures did not forbid operators from leaving unattended irradiated fuel assemblies that might be suspended from the refueling machine or the fuel handling machine.)

The inspector determined that the FCS plant designs and procedures minimized the consequences and potential of an inadvertent spent fuel pool draindown event.

2.1.2 IN 89-51

In IN 89-51, "Potential Loss of Required Shutdown Margin During Refueling Operations," the licensee was alerted to another licensee's violation of required shutdown margin upon the placement of highly reactive reload fuel assemblies into intermediate locations during core alterations. In response to the IN, the licensee's review determined that future cycles of FCS operation would likely involve increased fuel enrichments as a result of actions to be taken to mitigate pressurized thermal shock effects upon the reactor vessel wall. Consequently, the licensee speculated that the refueling crews might need flexibility to utilize intermediate fuel assembly parking and, therefore, might be requesting changes to the refueling procedure sequence and fuel assembly placements. Consequently, the licensee revised Operating Procedure OP-11, "Reactor Core Refueling Procedure," to require the reactor engineer to conduct a reactivity verification check for required shutdown margin prior to positioning fuel assemblies in intermediate positions. This action appeared to be a prudent response to the IN.

2.2 Shutdown Margin and Premature Criticality

In the spent fuel pool, the licensee was storing fuel assemblies into one of two separate regions. The fuel racks in Region 1 contained Boraflex, a neutron poison material. The fuel racks in Region 2 did not contain a poison and were, therefore, restricted to fuel assemblies that did not exceed pre-established reactivity criteria. Specifically, Technical Specification 2.8 (11) requires that prior to moving a fuel assembly to Region 2 that the burnup and enrichment must fall within certain parameters. To perform this analysis, the licensee used Special Procedure SP-BURNUP-1. During refueling operations, the licensee limited travel of the spent fuel pool fuel handling machine to Region 1 by mechanical interlocks.

As discussed above in paragraph 2.1, the licensee was required to perform a reactivity analysis for any desired deviation to the fuel movement plan that would permit intermediate positioning of fuel assemblies.

The licensee's representatives stated that there had been no instances at FCS when the Technical Specification 2.10 required shutdown margins had been violated.

2.3 Service Information on Fuel-Handling Equipment

The inspector questioned the licensee's representative about the process for processing vendor-supplied service information on their fuel-handling equipment. Of particular interest was the thoroughness of the licensee's process to ensure that adequate evaluations were conducted of safety-related, post-installation information that could result in the determination of deficient equipment. Post-installation service information for fuel-handling equipment was processed in accordance with Standing Order SO-G-62. The licensee's organization that maintained vendor manuals tracked completion of technical evaluations for significant changes to vendor manuals. No recent service information on fuel-handling equipment had been received by the licensee.

2.4 Commitments Related to Fuel-Handling Activities

The licensee was committed to Regulatory Guide 1.33, Revision 0, "Quality Assurance Program Requirements (Operation)" in Technical Specification 5.8.1. The guide endorses ANS 3.2 -1972, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants." This standard requires written procedures for core alteration, accountability of fuel, and partial or complete refueling operations. Specific procedures (listed in Appendix A of Regulatory Guide 1.33) are also required for each refueling outage and for receipt and shipment of fuel. Review by the inspectors of those procedures listed in Attachment 1 indicated that this commitment had been implemented.

2.5 Outage Work Controls, Responsibilities, Delegations, and Critical Path Scheduling

The inspectors reviewed the licensee's procedural delegations of responsibilities for refueling operations. The inspectors specifically examined documented specifications related to select managers, licensed operators, and other key outage personnel. The inspectors gave particular attention to whether clear lines of authority and provisions for internal coordination had been pre-established for outage activities.

During the 1992 refueling outage, the licensee employed for the first time, the Outage Management Control Center concept. This new process had resulted in a substantial number of work activities being pre-processed or completely processed outside of the control room. When the inspectors toured the control room, they observed that the process had beneficially resulted in a quiet control room with relatively little personnel traffic. Fuel handling was in process during the control room tours. The inspectors noted that a designated control room person was in constant communication with personnel in the containment. The inspectors noted that the fuel position status board and the master copy of the procedure governing fuel movements were being maintained.

The licensee had delegated responsibilities for refueling activities in Operating Procedure OP-11. Clear lines of authority had been specified. Standing Order Procedure SO-O-1 required each shift crew during periods of core alterations to include a senior reactor operator with no concurrent operational duties to supervise directly core alterations. The shift supervisor retained overall responsibility for refueling activities. Standing Order G-52 limited the working hours to a reasonable maximum for key operations and maintenance personnel. An extra refueling crew had been assigned to each shift to ensure adequate break time. Operating Procedure OP-11 required establishment of adequate communications when moving fuel and cessation of work if those communications were lost. Also, Operating Procedure OP-11 required stopping fuel movements for a number of reasons including: decreasing refueling water level, loss of control room ventilation, and unexpected sustained increasing count rate on any operating wide-range logarithmic channel of a neutron monitor.

The inspectors discussed with the outage manager and other licensee personnel the process whereby critical path schedules were developed. The process for revising the critical path schedule to correct problems and include emergent work was also discussed. In addition, the inspectors reviewed the Summary Working Schedule dated January 28, 1992. The schedule logic appeared reasonable.

The outage responsibilities were covered by the FCS 1992 Refueling/Maintenance Outage Responsibility Charter. It specified the outage control center as the focal point for all outage activities. The following were listed as key safety functions:

- Decay heat removal capabilities,
- Reactor coolant system inventory control,
- AC/DC power availability,
- Reactivity control, and
- Containment closure capability.

Only scheduled activities were supposed to be performed and all changes were supposed to be analyzed for their impact on safety. The nuclear safety review group (NSRG) was supposed to review any scope or schedule changes which potentially impacted safety-related systems or components. Detailed outage planning, scheduling, and execution were performed in accordance with Standing Order SO-M-104, which appeared to be comprehensive. A "1992 Fort Calhoun Refueling Outage Handbook" was widely distributed and appeared to be very useful for temporary outage personnel. It covered various subjects including plant arrangement, policies, organizations, telephone numbers, major work scope, radiation work permit information, schedules, safety, and security.

A meeting agenda titled, "OPPD Shutdown Plant Issues Meeting," dated January 28, 1992, was reviewed by the inspectors. It covered a large number of outage safety, training, and planning issues. It was noted that an extra shift supervisor, as well as a dedicated switchyard coordinator, would be assigned during the outage.

The inspector reviewed an "FCS Plan of the Day," dated February 19, 1992. It covered relevant outage safety issues including critical system safety status, ALARA concerns, Technical Specifications Limiting Conditions for Operations, fire barrier impairments, major equipment out-of-service, and late surveillances. The inspectors were informed that the NSRG would have daily contact with the outage planning personnel.

The documentation and interviews indicated a proactive approach by the licensee to outage safety. However, an event occurred which involved an apparent violation of a number of licensee policies. This event involved loss of shutdown cooling flow control and flow indication on April 12, 1992. It was described in NSRG Investigation Report IR-920273, which was approved April 23, 1992. The report noted that the surveillance testing which resulted in the event had not been approved by the outage control personnel as required and was performed earlier than scheduled. An unusual power supply configuration was one of the causes. As a part of the corrective action, the inspectors anticipate that the licensee will analyze the breakdown in implementation of the policies discussed above.

2.6 Fuel-Handling Controls

The licensee had provided refueling crews with guidance on special fuel-handling techniques in Operating Procedure OP-11, "Reactor Core Refueling." The guidance addressed methods to aid in the insertion or withdrawal of fuel assemblies that were hanging up, the removal of fuel assemblies stuck to the core support plate, the rotation of the refueling machine mast or cable shaking to reduce fuel assembly interactions, and the use of the fuel assembly guide to assist in seating bowed fuel assemblies. Operating Procedure OP-11 required post-core loading verification of fuel assembly locations. This detailed core alignment process was used by the licensee to minimize the potential for interference between fuel assemblies and the upper guide structure. Core alignment problems, like those discussed in IN 90-77, Supplement 1, are somewhat mitigated in the Combustion Engineering nuclear steam supply system because, as at FCS, the locating pins on the upper core alignment plate are larger in diameter and less susceptible to bending deformation than that in some other vendor designs. This alignment verification process was considered by the inspectors to be a programmatic strength.

Other guidance on fuel handling was provided in Operating Instruction OI-FH-1, "Fuel Handling Equipment Operation." This procedure addressed the new fuel elevator, spent fuel handling machine, refueling machine, and the tilt and transfer machine. Numerous matters were addressed in this procedure such as bridge and trolley positions that correlated to spent fuel pool fuel storage locations, various core component weights, and operational techniques to be used by operators.

The inspectors and the senior resident inspector observed fuel-handling operations underway in the refueling cavity and the spent fuel pool. The clarity of the water for operator visibility was observed to be fair at the time of core off load. The inspectors were told that the clarity of the refueling water had been improved as compared to previous outages. Presumably, the use of improved submersible filter units was, at least, partially responsible for the increased visibility. The senior resident inspector witnessed fuel-handling operations during core reload operations and observed the clarity of the water to be murky.

The inspectors observed that lighting in the reactor vessel was poor. At the time of the inspectors' tour of the containment, there was no underwater lighting in the reactor vessel except for Cherenkov lighting in the areas where fuel assemblies had been removed. In general, the top of the core was dark and no core components were visible to the unaided eye. The inspectors noted that the refueling crew did not have binoculars or a periscope with which to view the refueling operations. The inspectors observed that the licensee was utilizing an underwater television camera, which was mounted to the refueling mast hoist box. The monitor's image on the refueling machine was poor, and an inspector made that comment to the operators. The senior reactor operator stated that the camera's light was not working. At that time, the inspectors' tour guide offered that the console's potentiometer circuitry for the camera's light was rate sensitive, and that some times when the potentiometer was turned too rapidly that the breaker would trip. (The inspector noted later that Operating Instruction OI-FH-1, "Fuel Handling Equipment Operation," provided a precaution about the sensitivity of the light circuitry.) The operator checked and confirmed that the breaker was tripped. After resetting the breaker, the monitor's image was good. The camera's light was for local illumination only and did not illuminate a significant portion of the core. Later, the inspectors inquired as to why the licensee had not employed more lighting in the refueling cavity. The licensee's representatives indicated that previous attempts to use a contractor's underwater lighting system for core vacuuming had resulted in overloading the available circuits and had caused excess breaker trips in containment. The licensee's representative indicated that to install additional power for underwater lighting would necessitate a plant modification and that such consideration had previously been abandoned. The inspectors concluded that the licensee's lack of core lighting for aiding the refueling crews was a program weakness. The impact of the lack of lighting was, however, partially mitigated by the licensee's provisions for the underwater camera and light on the refueling mast hoist box.

The inspectors observed that during movements of the refueling machine bridge and trolley both operators were seated facing the same direction at the console. It was not apparent to the inspectors that the crew was checking the path of the trolley or refueling machine to ensure that personnel and equipment, which were moving around the operating deck of the refueling cavity, were clear of the system. The inspectors discussed with the licensee representatives that a third member of the refueling crew (such as the equipment operator who ran the fuel assembly upender) might be useful as a spotter and could also observe if fuel assemblies were fully withdrawn into the mast, if the hoist box was at the up limit, and if any people or cables were in the path of the refueling machine trolley or bridge.

While on the refueling machine, the inspector could not hear the audio neutron count rate, which is derived from the nuclear instrument power signals. The inspector questioned the licensee's representative on the matter. The senior reactor operator placed his ear near to a speaker on the console and stated that he could not hear the count rate. (During a later tour of the control room, the inspector noted that the neutron count rate was audible in the proximity of the speaker.) In response to the inspector's observation, the licensee issued a Nonconformance Report on March 11, 1992. The licensee subsequently determined that the audible count rate speaker was not the speaker located on the refueling machine, but a speaker mounted on a wall near the refueling cavity. The licensee determined that the volume of the count rate speaker was limited because the speaker's impedance of 13 ohms was not in conformance with the circuit design, which specified 45 ohms. Consequently, the licensee prepared Modification Request 92-017 to resolve the configuration nonconformance. For the FCS, there is no regulatory requirement for an audible neutron count rate in containment during fuel-handling operations. The inspectors could not find reference to this operator aid in the licensee's training plans. The inspector observed, however, that the licensee had provided for such operator aid, but (1) the system's design had not been implemented properly and (2) members of the subject refueling crew were unaware of the lack of an audible count rate and were unaware of the location of the count rate speaker.

The licensee performed timely preventive maintenance, calibration, and checkout of fuel-handling and transfer equipment by completion of the following procedures:

- OP-ST-FH-0001, completed March 28, 1992,
- OP-ST-FH-0005, completed March 26, 1992,
- OP-ST-FH-0002, completed March 25, 1992,
- SE-ST-FH-0006, completed January 10, 1992,
- SE-ST-FH-0007, completed March 9, 1992,
- IC-RR-FH-0800, completed March 24, 1992, and
- MM-RI-FH-0700, completed February 14, 1992.

The inspectors found the data used in and derived from the performance of the above procedures was satisfactory and supportive of fuel handling.

Operating Procedure OP-11 required the minimum reactor coolant system boron concentration to be the refueling boron concentration, which was based on reactor subcriticality considerations. Starting one shift prior to fuel reload boron analyses were required each shift. The shift supervisor was required to verify that the boron concentration was greater than the refueling boron concentration.

The inspectors found no guidance in the licensee's documents on fuel-handling controls that precluded operators from leaving unattended irradiated fuel assemblies that were suspended from the refueling machine or the fuel handling machine. This type of situation is most conceivable for non-critical path activities in the spent fuel pool when examinations might be temporarily discontinued for breaks, shift turnovers, etc. The licensee's representative confirmed that no formal guidance on this matter existed. Although there is no regulatory requirement barring such a practice, the inspectors noted that fuel handling practice should prohibit such a circumstance. The licensee's representatives said they would consider the generation of such a procedural restriction.

The reactor engineer informed an inspector that the core load verification had identified that a source assembly was not fully seated, but there were no loading or alignment errors found in the Cycle 14 core. The source assembly was repositioned and fully seated.

2.7 Flow Blockage

The licensee had established Standing Order SO-M-10, "Foreign Material Exclusion," to control loose parts from being inadvertently introduced into the refueling cavity and the spent fuel pool. The spent fuel pool was the only area defined as a continuous foreign material exclusion (FME) area. The standing order established provisions for defining exclusion areas wherein foreign materials were to be controlled and accounted for or entirely prohibited (i.e., materials susceptible to degradation). The standing order identified the positions of FME coordinators who were responsible for observing activities and ensuring that the requirements of the procedure were implemented by: controlling access and egress of materials and personnel including the requisite recording on accountability logs; reviewing work control requirements; granting permission for the entrance of critical material needs that were discouraged by the procedure; maintaining stop work authority; and conducting work completion inspections. The responsibility for assigning the FME coordinators was delegated to supervisor, crew leader, and lead personnel. Operating Procedure OP-11, "Reactor Core Refueling," required in Section 4.0, "Initial Conditions," that an FME coordinator was to be assigned and exclusion procedures placed in effect prior to fuel movement.

On February 19 and 20, the inspectors toured the containment and the auxiliary buildings, respectively. During the tours, operations personnel were in the process of off-loading fuel assemblies from the core. The inspectors noted that the refueling cavity and the spent fuel pool operating decks were being maintained as FME areas via tape and rope barriers. In containment, the inspector audited the material accountability log. Some items logged into the FME area, such as a camera, could not be located by the licensee's staff. Other items, such as boxes of latex gloves, were in the area, but were not indicated on the log. In addition, the log sheets that were being used were not those specified by the standing order.

In regard to the boxes of latex gloves, the inspectors concluded that their storage in the FME area was not in concert with the intent of the standing order. Moreover, their storage in the FME area resulted in a suspension of fuel-handling operations. Specifically, when an inspector mounted the refueling machine to observe fuel-handling operations, the inspector observed that the load cell was not indicating the weight of a fuel assembly that was being withdrawn from the core. The inspector brought this concern to the attention of the operators who then suspended lifting operations in order to determine why the load cell was failing to indicate load. Subsequently, the NSRG Chairman came to assist the refueling crew, and determined that there were boxes of latex gloves stored under the operating console of the refueling machine and that one of the boxes had fallen and tripped a toggle switch, which supplied power to the load cell.

The inspectors noted that various materials stored on the periphery of the operating deck, which were just outside of the taped FME area, were materials discouraged by the standing order, such as clear plastic. Other items within the FME area were not attached to lanyards as the procedure specified. During the tour of the spent fuel pool, the inspectors noted that there was no FME coordinator and no accountability log available.

The inspectors informed the licensee that work activities in the refueling cavity and the spent fuel pool were in apparent violation (285/9203-01) of Standing

Order SO-M-10. Subsequently, the licensee issued Corrective Action Report No. 92-019 on February 22, 1992. Pursuant to the Corrective Action Report, quality control (QC) personnel conducted surveillances of the two subject FME areas. The surveillances began on February 21 and ended on February 24, 1992. The first three QC surveillances found similar findings noted by the inspectors. The fourth QC surveillance determined that a state of compliance had been established, that extraneous material had been removed, and that accountability logs had been reestablished. Other corrective actions that the licensee took included the training of a dozen maintenance FME coordinators. The training was completed on February 23, 1992. In addition, work activities, except radiation protection and operations activities, and tours were restricted in the FME areas until the FME coordinator training was completed. Also during February through April, the licensee conducted a quality assurance (QA) surveillance to compare Standing Order SO-M-10 against the requirements of INPO Good Practice MA-315, "Foreign Material Exclusion." The QA report was issued on April 15, 1992. For long-term corrective actions, the licensee planned to revise Standing Order SO-M-10 and establish a formal training program for FME coordinators.

During the refueling outage, the licensee identified some loose parts in FME areas. These loose parts included a flashlight end cap, a nail, and rags. The licensee also conducted an effort to inspect the refueling cavity for debris. This effort resulted in miscellaneous items being removed from the refueling cavity.

The inspector noted another matter for the licensee's consideration. This matter concerned the placement of a personnel protective clothing changeout station on the spent fuel pool operating deck. In particular, the placement of the changeout station was on a narrow walkway where personnel would be removing protective boots, etc., while maintaining their balance on the edge of the spent fuel pool, which did not have a protective railing to keep personnel or items of their clothing from falling into the water. The inspectors questioned the licensee representative as to why the changeout station was not located at an area outside of the immediate administrative security door. The licensee representative indicated that they would reconsider the appropriateness of the location for the changeout station.

2.8 <u>Fuel and Core Component Performance</u>

The licensee presented the inspectors with an overview of its fuel and core component performance. During the discussion, the inspectors also questioned the licensee's representatives regarding the occurrence of any fuel-handling problems such as undue mechanical interference between spacer grids during fuel loading or unloading, physical damage incurred to irradiated core components, etc. The licensee's representatives indicated that there had been no known fuel-handling damage incurred at FCS. The licensee's representatives stated that some fuel failures had occurred during Cycles 6, 7, 8, and 10 operation. These failures were attributed to weld defects, over-ramping during startup, fretting because of debris, densification-related pellet cladding interaction, and some unknown mechanisms. The licensee's representative stated that they had no evidence of any control element assembly (CEA) that had perforated rodlets. The licensee's representative stated that there had been no fuel assembly guide tube sleeving performed at FCS.

The inspectors concluded that the licensee had maintained a competent nuclear engineering staff with notable in-house computing and licensing reload safety analysis capabilities. The inspectors found the licensee's nuclear engineering staff to be assertive in maintaining cognizance of fuel performance and potential adverse impacts. For instance, this "heads-up" attitude was reflected in that the licensee had established fuel contracts with Combustion Engineering and Westinghouse which required all proposed changes to fuel assembly design to be reviewed and approved by the licensee prior to implementation in the reload batch. The licensee's representatives stated that there had been no FCS fuel design changes made under the provisions of 10 CFR Part 50.59.

During the reactor shutdown at the end of Cycle 13 operations, the licensee became aware of coolant iodine activity levels that indicated fuel cladding perforation. The increased iodine activity level was a transient spike and did not result in a Technical Specification limit violation. The licensee anticipated from the iodine activity that a tight leaker(s) was present in the core. Identification and disposition of the suspected leaker is discussed in the following paragraphs.

2.9 Fuel Assembly Post-Irradiation Examination and Reconstitution

During the outage, the licensee examined several fuel assemblies from Cycle 13 and prior cycles of operation. The examination process involved 44 Batch N, 3 Batch M, and 3 Batch P fuel assemblies. The examinations were performed in the spent fuel pool and utilized video taping and through transmission ultrasonics (to determine the presence of water in fuel rods). These examinations were performed by personnel supplied by ABB Combustion Engineering. The licensee's representative stated that all fuel handling was performed by the licensee's personnel. The licensee's representative stated that two procedures were written to endorse the contractor's procedures to examine and reconstitute fuel assemblies. The inspectors did not review these contractor and licensee procedures.

One of the fuel assemblies, which was designated as N008, was intended for use in the Cycle 14 core and was found to contain a perforated fuel rod. The assembly, which was a reconstitutable assembly, was manufactured by Combustion Engineering, and, at the time of core discharge, had a burnup of 24,547 MWD/MTU. The licensee subsequently reconstituted Assembly N008 in the spent fuel pool. Nonconformance Report 92-029 described this effort. The replacement rod that was used was a stainless steel dummy rod. The licensee stored the perforated rod in an enclosed canister, which was placed in a spent fuel assembly's guide tube. The licensee did not determine the mechanism responsible for the failure of the fuel rod. However, the licensee suspected a manufacturing defect as the likely causal agent, because the failure appeared not to be attributable to debris fretting or handling damage. No significant mishaps occurred during the licensee's reconstitution process. During the outage, the licensee discussed the use of the reconstituted fuel assembly with personnel from the Office of Nuclear Reactor

Regulation. The impact of the replacement stainless steel rod on the peaking factors was small for the Cycle 14 core. The licensee performed a 10 CFR Part 50.59 safety evaluation of the reconstitution and determined that the reconstitution did not involve an unreviewed safety question. Specifically, the maximum integrated radial peaking factor (Fr) increased by 0.14 percent, which the inspector understood was bounded by the radial peaking factor upper tolerance uncertainty of 4.00 percent in the original Cycle 14 reload safety analysis. Other safety implications of the use of the replacement rod, such as seismic analysis, were determined to be negligible.

The licensee provided a written statement that, prior to this reconstitution, there had not been any field changes to fuel assemblies or CEAs at FCS. The inspector concluded that the licensee's fuel examination and reconstitution processes were successful and performed without encountering any significant problem.

2.10 Documentation of Configuration Control

During the review of licensee documents, the inspectors noted that plant equipment, which had been removed by plant modifications, had not always been properly reflected in existing procedures. The examples of this problem were reference to the CEA change machine (stand) and part-length CEAs. These historical components were discussed in Operating Instruction OI-FH-1, Operating Procedure OP-11, and Training Lesson Plan 4-4-10. The inspectors found that the CEA change machine was removed prior to the refueling outage in the fall of 1991 and that the discontinuation of part-length CEAs occurred several years ago. The inspectors informed the licensee representative of this observation. The licensee representatives said they would examine their practices to ensure that plant documents properly reflect existing plant design.

A separate configuration control problem the inspectors identified involved the spent fuel pool instrumentation. The inspectors discussed with the licensee's representative the associated alarms available to control room operators. In the discussion, Annunciator Response Procedure CB-1, 2, 3/A1 was examined. This procedure discussed the alarm setpoint temperatures for both sides of the spent fuel pool heat exchanger. The inspectors noted that the alarm setpoint temperature that was specified for the component cooling water side of the heat exchanger was obviously in error. The given temperature was 200°F, whereas the system operated below 100°F and the spent fuel temperature alarm was specified in the same procedure as 110°F.

The licensee's representative agreed that the alarm setpoint was erroneous and checked the calibration procedure for the applicable initiating Device TIC-479. It was determined that implementing Calibration Procedure CP-479 also referred to the same 200°F alarm setpoint. However, Calibration Procedure CP-479 was entitled "Aux. Cooling Water from Letdown Heat Exchanger Temperature." The licensee determined that the adjacent Calibration Procedure CP-477 appeared to be identical to Procedure CP-479, except for three places in each procedure where device numbers were given.

The inspectors were informed that the subject annunciator response procedure, a document labelled as "SAFETY RELATED," had been reviewed in the Project 1991 upgrade effort, but the erroneous alarm setpoint had not been identified. In further investigation, the licensee found that the "Component Cooling Water," Revision 0, March 1989 design basis document had listed in Attachment 8, the TIC-479 200°F setpoint as a nonsafety-related function without a design-basis calculation. The generation of a calculation for this setpoint was characterized as Category 5 (in a Category 1-to-5 priority rating with Category 5 being the least important). Furthermore, the May 8, 1985, revision of Procedure CP-479 also did not identify the alarm setpoint problem. Consequently, the setpoint had been in error throughout plant operation.

The safety significance of the incorrect alarm setpoint is somewhat mitigated by the existence of an equivalent backup alarm (i.e., 110°F spent fuel pool temperature). Consequently, in the event of a spent fuel pool heatup accident, operators should have had indication of the problem from another annunciator alarm, which is derived from a nonsafety-related device.

At the exit meeting, the inspectors identified the alarm setpoint problem as an apparent violation. The licensee's management responded that (1) the alarm setpoint was in error; (2) in the 1989 review, the reviewer noted the setpoint did not have a backup design basis calculation, but also questioned the value of the setpoint; (3) although the annunciator response procedures were safety related, this particular setpoint was not considered safety related; (4) the alarm annunciator was not considered inoperable because it was not required by Technical Specifications, but was used for information only; (5) the Technical Specifications required the monitoring of other spent fuel pool instrumentation that would alert operators of insufficient spent fuel cooling; and (6) the review of the subject setpoint would be expedited. As a result of this information and regional management's review, the inspector informed the licensee's representative on April 27, 1992, that the issue would not be considered a violation, but would be addressed as an inspection followup item (285/9203-02).

2.11 Movement of Reactor Vessel Upper Internals Package

The inspector reviewed the following procedures, which involved removing the reactor vessel upper internals package: MM-RR-RC-0305, MM-RR-RC-0306, MM-RR-RC-0307A, and MM-RR-RC-308A. They required appropriate monitoring of the load cell. A safe transfer path was procedurally outlined. Appropriate instructions were given to ensure secure attachment of the reactor vessel internals lift rig. The senior reactor operator's presence and tool accountability was required from the start of work. During reactor vessel head removal, spotters and monitoring nuclear instrumentation were required to preclude inadvertent removal of a control element assembly. Instructions were included to watch for interferences. Radiation monitor coverage was required at all times. Crane speed was restricted to 6 inches per minute during movement of the upper guide structure.

The inspector could find no requirement to establish communications with the control room in the above-specified procedures. Communication with the control room is generally warranted during any core alterations. This observation was provided to the licensee representatives at the exit for its consideration. The licensee's representatives responded that they would take action to implement such communications.

2.12 Mid-Loop Operations

An inspector walked down instrumentation associated with reactor coolant system mid-loop operations. Installation configuration within the containment appeared acceptable. Control room instrumentation appeared useful and user friendly. The overall adequacy of equipment and instrumentation used during mid-loop operations will be assessed during a future inspection pursuant to Temporary Instruction 2515/103, "Loss of Decay Heat Removal (Generic Letter 88-17), 10 CFR Part 50.54(f), Programmed Enhancements (Long Term) Review."

2.13 Containment and Fuel Building Activities

Operating Procedure OP-11 required completion of a containment integrity checklist prior to core alterations. Also, containment and spent fuel storage area radiation monitors (RMs) were required to be in operation and calibrated. Verification of the isolation logic was required for containment RMs. Portable RMs were required to be on or near the refueling machine (FH-1) and spent fuel handling machine (FH-12). During movement of irradiated fuel in the spent fuel areas, Operating Instruction OI-FH-1 required diversion of the spent fuel area ventilation through the charcoal filter. Appropriate positioning of dampers and doors to obtain a confined system was required. All auxiliary building controlled area exterior doors were either required to be closed or suitably sealed to prevent disruption of the spent fuel area ventilation. A differential pressure gauge for this area was located on Panel AI-44 in the control room. The inspector noted that its indication was negative with respect to atmospheric pressure, as would be expected, during core off-loading activities.

Operating Instruction GM-OI-HE-2 required a plant review committee (PRC) approved procedure for transport of any load (including an empty hook) over the spent fuel pool with the auxiliary building crane (HE-2). Proximity interlocks were installed to prevent a collision of FH-12 and HE-2. There were caution statements in each of the procedures for handling reactor components not to handle any heavy load above irradiated fuel without the control room filter system in operation and in the filtered make-up mode. Operating Instruction GM-OI-HE-1 contained adequate restrictions for using the containment polar crane. Specifically, with the reactor vessel head removed, transport of any load over the core without a PRC-approved procedure was prohibited.

2.14 Fuel-Handling Qualification and Training Program

The licensee had a documented training program for both licensed and non-licensed operators. Operation of the refueling machine required a licensed operator. Other fuel-handling equipment could be operated by non-licensed personnel. An inspector reviewed the lesson plans for fuel-handling activities and found them to be comprehensive. The inspector then selected one refueling crew and reviewed their training records for fuel-handling activities. All crew members had received training for the activities to which they were assigned.

The agenda for "OPPD Shutdown Plant Issues Meeting," dated January 28, 1992, listed training requirements for both licensed and non-licensed operators. Licensed operators were required training in shutdown risk management, loss of shutdown cooling simulator exercise, raw water malfunctions, and reactor control. Non-licensed operators were required training in loss of shutdown cooling, nozzle dam mockup, emergency diesel generator local operations, reactivity control, and containment integrity. The scope of operator training appeared appropriate.

2.15 Summary

One violation (paragraph 2.7), one inspection followup item (paragraph 2.10), and no deviations were identified in the review of this program area.

3. VITAL-TO-VITAL AREA BARRIER

During the inspection on February 18, an inspector requested confirmation that the licensee was posting a security officer at a specific vital-to-vital area barrier when certain surveillance and maintenance activities were performed on the fuel transfer system. The nature of the surveillance and maintenance activities typically resulted in the degradation of that vital-to-vital area barrier. The licensee's security manager was unaware of the issue. The licensee subsequently determined that an officer had not been posted at the barrier when such surveillance and maintenance activities had been previously performed. The inspector estimated that there had probably been dozens of instances in which the subject surveillance and maintenance activities were periodically performed.

The inspector reviewed the licensee's fuel-handling documents and was unable to locate a requisite precaution to alert the licensee's operations staff of the need to post the barrier prior to conducting the surveillance and maintenance activities. Prerequisite comments, however, did address the radiological implications of the surveillance and maintenance activities. The inspector inquired as to whether the licensee had become aware of this issue from an NRC inspection at another utility, and learned that some of the licensee's staff had read the inspection report, were aware of the issue, and had interpreted a comment in the OPPD Physical Security Plan that negated the need for the security officer posting, but had not discussed the issue with the security personnel. The integrity of the unposted barrier was designated as an unresolved item (285/9203-03) requiring further review in a subsequent NRC inspection.

3.1 Summary

One unresolved item and no violations or deviations were identified in the review of this program area.

4. EXIT MEETING

On February 21, and April 24, 1992, the inspectors met with members of the licensee's organization denoted in paragraph 1 and summarized the scope and findings of this inspection. As discussed in paragraph 2.10, an issue involving an erroneous alarm setpoint on the component cooling water heat exchanger temperature was originally characterized as an apparent violation at the April 24, 1992, exit meeting. The licensee's management provided additional information on the issue during the exit meeting. On April 27, 1992, following regional management review, an inspector informed the licensee's representative that the issue would be characterized as an inspection followup item.

The licensee did not identify as proprietary any of the materials provided to, or reviewed by, the inspectors during this inspection.

ATTACHMENT 1

DOCUMENTS REVIEWED

"Reactor Core Refueling," Operating Procedure OP-11, Revision 2, December 9, 1991

"Conduct of Operations," Standing Order Procedure SO-O-1, December 19, 1990

"Plant Staff Working Hours," Standing Order No. G-52, July 25, 1988

"Fort Calhoun Station 1992 Outage Schedule," January 28, 1992

"OPPD Shutdown Plant Issues Meeting," agenda dated January 28, 1992

"Selected Topics," January 28, 1992

"Foreign Material Exclusion," Standing Order SO-M-10, Revision 10, July 10, 1991

"Spent Fuel Pool Inventory Control," Standing Order SO-O-47, July 12, 1991

"Fuel Handling Equipment Operation," Operating Instruction Procedure OI-FH-1, January 23, 1992

"Calibration of FH-12 Storage Pool Platform Bridge and Hoist Load Cell," Maintenance Procedure IC-RR-FH-0800, November 8, 1991

"Refueling Machine Preoperational Inspection and Maintenance," Maintenance Procedure MM-RI-FH-0700, August 21, 1991

"Refueling System Fuel Handling Machine (FH-1) Interlocks Test," Surveillance Test Procedure SE-ST-FH-0001, June 10, 1990

"Ref[illegible] System Spent Fuel Handling Machine Refueling Interlocks Test," Sur[illegible]e Test Procedure SE-ST-FH-0005, December 10, 1991

"Refue[illegible] System Fuel Transfer System Interlocks Test," Surveillance Test Procedure SE-ST-FH-0002, Revision 0, June 10, 1990

"Refueling System New Fuel Elevator Test," Surveillance Test [illegible]ocedure SE-ST-FH-0004, June 10, 1990

"Burnup Determination for Storage of Spent Fuel," Surveillance Test Procedure RE-ST-RX-0007, February 25, 1991

"Auxiliary Building Crane Normal Operation," Operating Instruction Procedure GM-OI-HE-2, July 25, 1991

"Removal of Reactor Vessel Closure Head," Maintenance Procedure MM-RR-RC-0305, July 25, 1990

"Removal of Reactor Internals Hold Down Ring," Maintenance Procedure MM-RR-RC-0306, July 25, 1990

"Removal of Upper Guide Structure and Raising of In-Core Instrumentation Plate," Maintenance Procedure MM-RR-RC-0307A, July 24, 1990

"Removal of Core Support Barrel," Maintenance Procedure MM-RR-RC-0308A, December 2, 1991

"Ultrasonic Examination of Fuel," Special Procedure SP-FE-11, April 4, 1987

"Irradiated Fuel Assemblies Visual Inspection and Retrieval of Foreign Objects," Maintenance Procedure RE-RI-FE-0700, April 25, 1991

"Removal of Spent Fuel Pool Gate," Maintenance Procedure MM-RR-FH-0500, February 20, 1990

"Design Basis Document: Spent Fuel Storage and Fuel Pool Cooling," Document Number SDBD-AC-SFP-102, Revision 2, September 1991

"QA Surveillance Report No. Z8-92-2," April 15, 1992

Corrective Action Report 92-019, February 22, 1992

"Non-Licensed Operator Fuel Handling Qualification Manual," Revision 3, August 1991

"Fuel Handling," Instructor Handbook, Student Handbook, and Transparency Index, Lesson Plan 4-4-10, Revision 3, December 19, 1991

"Fuel Handling Machine," Instructor Handbook, Student Handbook, and Transparency Index, Lesson Plan 7-11-13, Revision 3, April 9, 1990

"SP-BURNUP-1," memorandum from J. Spilker to PRC chairman, February 14, 1992

Special Procedure SP-BURNUP-1, "Burnup Determination for Storage of Spent Fuel," February 25, 1991

Nonconformance Report Number 92-029, March 23, 1992

Nonconformance Report Number 92-018, March 11, 1992

Annunciator Response Procedure ARP-CB-1, 2, 3/A1, "Annunciator Response Procedure A1 Control Room Annunciator A1," Revision 4, April 9, 1992

Calibration Procedure CP-479, "Aux. Cooling Water From Letdown Heat Exchanger Temperature," Revision 1, May 8, 1985

"FCS 1992 Refueling/Maintenance Outage Responsibility Charter"

"1992 Fort Calhoun Refueling Outage Handbook"

"Summary Working Schedule," dated January 28, 1992

"FCS Plan of the Day," dated February 19, 1992

"Refueling System Spent Fuel Handling Machine Interlocks Test for New Fuel Receipt," Procedure SE-ST-FH-0006, Revision 4, dated June 10, 1990

"Refueling System Spent Fuel Handling Machine Interlocks Test for Spent Fuel Shuffle," Procedure SE-ST-FH-0007, Revision 4

"Refueling System Fuel Handling Machine (FH-1) Interlocks Test," Procedure OP-ST-FH-0001, Revision 3

"Refueling System Spent Fuel Handling Machine Refueling Interlocks Test," Procedure OP-ST-FH-0005, Revision 0

"Refueling System Fuel Transfer System Interlocks Test," Procedure OP-ST-FH-0002, Revision 0

"Loss of Shutdown Cooling Flow Control and Flow Indication," Investigation Report SRG-92-287, approved April 23, 1992

"Maintenance Procedures," FCS, Unit 1 Updated Safety Analysis Report (USAR), Section 12.3.5, Appendix A

"Fuel Handling Incident," Abnormal Operating Procedure (AOP)-08, Revision 1, dated July 31, 1990

ATTACHMENT 2

FUEL-RELATED INFORMATION NOTICES (INS) DISCUSSED

IN 81-23, "Fuel Assembly Damaged Due to Improper Positioning of Handling Equipment"

IN 84-93, "Potential for Loss of Water From the Refueling Cavity" (see also Bulletin 84-03)

IN 85-12, "Recent Fuel Handling Events"

IN 86-06, "Failure of Lifting Rig Attachment While Lifting the Upper Guide Structure at St. Lucie, Unit 1"

IN 86-58, "Dropped Fuel Assembly"

IN 87-19, "Perforation and Cracking of Rod Cluster Control Assemblies"

IN 88-65, "Inadvertent Drainages of Spent Fuel Pools"

IN 88-92, "Potential for Spent Fuel Pool Draindown"

IN 89-31, "Swelling and Cracking of Hafnium Control Rods"

IN 89-51, "Potential Loss of Required Shutdown Margin During Refueling Operations" (see also Bulletin 89-03)

IN 90-77 and Supplement 1, "Inadvertent Removal of Fuel Assemblies from the Reactor Core"

IN 91-26, "Potential Nonconservative Errors in the Working Format Hansen-Roach Cross-Section Set Provided with the Keno and Scale Codes" Docket No. STN 50-482/91-32