UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
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November 17, 2003

Mr. Harold B. Ray, Executive Vice President
Southern California Edison Co.
San Onofre Nuclear Generating Station
P.O. Box 128
San Clemente, California 92674-0128

SUBJECT: NRC INSPECTION REPORT 050-00206/03-013

Dear Mr. Ray:

An NRC inspection was conducted on October 27-31, 2003, at your San Onofre Nuclear Generating Station, Unit 1 facility. This inspection was an examination of activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspection included an examination of selected procedures and representative records, observations of activities, and interviews with personnel. The enclosed report presents the results of that inspection. Overall, the inspection determined that you are conducting decommissioning activities in compliance with regulatory and license requirements.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be made available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Should you have any questions concerning this inspection, please contact the undersigned at (817) 860-8191 or Mr. Robert J. Evans, Senior Health Physicist, at (817) 860-8234.

Sincerely,

/RA EECollins for/

D. Blair Spitzberg, Ph.D., Chief
Fuel Cycle and Decommissioning Branch

Docket No.: 50-206
License No.: DPR-13

Enclosure:
NRC Inspection Report
050-00206/03-013

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ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket No:	50-206
License No:	DPR-13
Report No:	050-00206/03-013
Licensee:	Southern California Edison Co. P.O. Box 128 San Clemente, California 92674
Facility:	San Onofre Nuclear Generating Station, Unit 1
Location:	San Clemente, California
Dates:	October 27-31, 2003
Inspectors:	Robert J. Evans, P.E., C.H.P., Senior Health Physicist Fuel Cycle & Decommissioning Branch William C. Huffman, Project Manager Division of Waste Management, Decommissioning Branch Office of Nuclear Material Safety & Safeguards
Accompanied By:	Scott P. Atwater, Inspector-In-Training Fuel Cycle & Decommissioning Branch
Approved By:	D. Blair Spitzberg, Ph.D., Chief Fuel Cycle & Decommissioning Branch
Attachment:	Supplemental Inspection Information
ADAMS Entry:	IR05000206-03-013 on 10/27/2003 - 10/31/2003; Southern California Edison Co., San Onofre Nuclear Generating Station; Unit 1. Decommissioning Report. No Violations.

EXECUTIVE SUMMARY

San Onofre Nuclear Generating Station, Unit 1
NRC Inspection Report 050-00206/03-013

This inspection was a routine, announced inspection of decommissioning activities being conducted at San Onofre Nuclear Generating Station, Unit 1. The inspection included a review of safety reviews, design changes and modifications; spent fuel pool safety; decommissioning performance and status review; and solid radioactive waste management and transportation of radioactive materials. Overall, the licensee was conducting decommissioning in accordance with regulatory and procedural requirements.

Safety Reviews, Design Changes, and Modifications at Permanently Shutdown Reactors

- The licensee's safety review and design change program was in compliance with 10 CFR 50.59 requirements. The licensing change control process was effectively screening and evaluating the impacts of facility and procedural changes. The licensee had developed and implemented a training program for 10 CFR 50.59 screeners and evaluators. All evaluations reviewed were accomplished by trained and qualified individuals. Audits of the safety review program had been completed in accordance with quality assurance requirements (Section 1.2.a).

- The inspectors conducted a detailed review of the licensee's implementation of design changes to the Unit 1 turbine building gantry crane. The modifications were being implemented in accordance with the proposed change and design drawings. At the conclusion of the onsite inspection, the license amendment request had not been approved by the NRC; therefore, the licensee was conducting the work at risk. The crane was not being used for any safety related function; therefore, the implementation of the modifications had no impact on the safe storage of fuel (Section 1.2.b).

Spent Fuel Pool Safety at Permanently Shutdown Reactors

- The licensee was maintaining the spent fuel pool within permanently defueled technical specifications and procedural requirements (Section 2.2).

Decommissioning Performance and Status Review at Permanently Shutdown Reactors

- Site tours indicated that radiation protection controls were in place to support decommissioning activities. The licensee was cutting up the sphere enclosure building and safety precautions were being taken to prevent the dropping of concrete blocks (Section 3.2.a).

- The licensee recently experienced an industrial accident that resulted in an injured worker in the radiologically restricted area. The licensee issued an Action Request to review the incident and to consider corrective actions necessary to upgrade their ability to evacuate injured individuals via the overhead crane (Section 3.2.a).

- The licensee was conducting temperature monitoring of the spent fuel Advanced Horizontal Storage Module. The concrete temperature of the one loaded module was well below the high temperature procedural limits (Section 3.2.b).

Solid Radioactive Waste Management and Transportation of Radioactive Materials

- The licensee had developed and implemented a comprehensive program for handling, sorting and disposing of radioactive wastes. The Unit 1 staff was conducting the program in accordance with approved site procedures. The use of written travelers was noted to be a positive management tool to help ensure compliance with shipping and disposal regulations (Section 4.2.a).

- The inspectors reviewed the shipment of a reactor coolant pump to an out of state disposal site and concluded that the licensee shipped the pump in accordance with NRC and U.S. Department of Transportation requirements (Section 4.2.b).

Report Details

Summary of Plant Status

San Onofre Nuclear Generating Station, Unit 1 was permanently shut down during November 1992 and was permanently defueled by March 1993. The unit remained in SAFSTOR until June 1999, when decommissioning was initiated. At the time of this inspection, the licensee was conducting decommissioning activities under the DECON option as stated in its Post Shutdown Decommissioning Activities Report dated December 15, 1998.

Work completed since the previous inspection included removal of the residual heat removal heat exchangers from containment, shipment of reactor coolant Pumps A and B to an offsite waste disposal facility, removal and packaging of the reactor vessel insulation, and removal of the condenser water boxes and tubes. The third reactor coolant pump was shipped offsite during the onsite inspection. The licensee began cutting the sphere enclosure building wall into blocks during July 2003. This work is expected to continue until February 2004. This work was tightly controlled by the licensee, in part, because of the safety hazards involved. At the time of the inspection, the licensee had cut and removed about 100 of 337 blocks.

The reactor pressure vessel was still onsite. The vessel was being stored in its shipping container adjacent to the sphere enclosure building. The reactor pressure vessel package is currently scheduled to be shipped to a disposal site in South Carolina prior to the end of calender year 2003.

Work in progress during the inspection included upgrading the Unit 1 turbine building gantry crane to support future Unit 1 fuel movement activities and construction of spent fuel canisters and storage modules. One loaded fuel canister was installed in the onsite Independent Spent Fuel Storage Installation, and the second canister of Unit 1 fuel was being dried in Unit 3.

The first canister underwent a lengthy vacuum drying period, in part, because of trapped water in the spacers below the fuel elements. As a corrective action, the licensee modified the spacers in the second canister to provide a drain path for the trapped water. At the conclusion of the onsite inspection, the licensee was approximately 24 hours into the vacuum drying phase on the second canister. A vacuum of 15 Torr had been achieved. It was too early to determine whether the spacer modification was effective in reducing vacuum drying time. The NRC will continue to review the licensee's handling of spent fuel during future inspections.

1 Safety Reviews, Design Changes, and Modifications at Permanently Shutdown Reactors (37801)

1.1 Inspection Scope

The purpose of this portion of the inspection was to ascertain whether design changes, tests, experiments, and modifications were effectively reviewed, conducted, managed, and controlled during plant decommissioning.

1.2 Observations and Findings

a. 10 CFR 50.59 Program Review

10 CFR 50.59 is a regulation that addresses the change control process and is used to determine if a proposed change to the facility, procedures, tests, or experiments is subject to a license amendment and NRC approval. The process is implemented by site procedure SO123-XV-44, "10 CFR 50.59 and 72.48 Program," Revision 6. This procedure provides instructions for both initial screening and subsequent evaluation (if necessary) of facility or procedure changes to confirm that the licensee can complete the changes without NRC approval.

The inspectors reviewed Procedure SO123-XV-44 to determined if it met the intent of 10 CFR 50.59. The inspectors ascertained that the procedure's screening criteria incorporated industry guidance found in Nuclear Energy Institute's guidelines for 10 CFR 50.59 implementation. The inspectors also confirmed that the 10 CFR 50.59 evaluation criteria precisely matched the regulatory requirements. The inspectors noted that the licensee had incorporated Nuclear Energy Institute's guidance into a resource manual to provide a comprehensive reference of the 10 CFR 50.59 experience and practices from an industry perspective.

The inspectors reviewed randomly selected 10 CFR 50.59 screens of various facility changes and found that all screens were completed in accordance with procedural requirements. The 10 CFR 50.59 information pertinent to any facility or procedural change was maintained in the licensee's Action Request (AR) system. The AR system can be readily searched and viewed in a paperless database called MOSAIC. The inspectors observed that both the 10 CFR 50.59 screens and evaluations were generally completed online in the AR system without hard copy paper forms. Electronic forms have been prepared with the procedural screening criteria embedded in the form template. Both the screens and evaluations are independently reviewed by a qualified person other than the preparer. The inspectors found that in all cases the screens provided a sufficient summary to understand what was being changed and why. Furthermore, the responses to the screening criteria were thorough and appeared to adequately address the criteria questions from a technical perspective.

Full 10 CFR 50.59 evaluations are completed infrequently by the licensee. Consequently, the inspectors' review included evaluations that were several years old. The inspectors found that the evaluations addressed the correct regulatory criteria in all cases. The responses indicated that the criteria had been thoroughly examined by the licensee in terms of technical detail and overall safety considerations. The inspectors had some observations related to the implementing procedure for 10 CFR 50.59 evaluations. The observations discussed with the licensee included:

- Per Procedure SO123-XV-44, after a 10 CFR 50.59 evaluation has been completed by the preparer and the reviewer, the licensee's Nuclear Safety Group (NSG) shall review "selected" evaluations and document the results of the review. It was not clear to the inspectors which evaluations were selected for NSG review. In practice, it appeared that all 10 CFR 50.59 evaluations were

reviewed by the NSG. The licensee provided the inspectors with an example of a quarterly report by the NSG that confirmed reviews of 50.59 evaluations. The inspectors observed that documentation of the NSG review on the electronic 10 CFR 50.59 evaluation form was simply an electronic acknowledgment that the review had been accomplished. No documentation of what considerations were involved in the independent NSG review of the evaluations were discussed in any of the evaluations examined.

- Per Procedure SO123-XV-44, the licensee's Onsite Review Committee (OSRC) shall review "requested" 10 CFR 50.59 evaluations. It was not obvious to the inspectors which evaluations had been reviewed by the OSRC. The licensee provided the inspectors with an OSRC report that contained an example of a 10 CFR 50.59 evaluation that was considered by the committee.

- Evaluation AR#030400078-99 assessed the dry cask loading and transfer of Unit 1 fuel from the Unit 3 spent fuel pool to the Independent Spent Fuel Storage Installation. The designated category of this 10 CFR 50.59 evaluation was "Working" (i.e., open) even though the implementation of the proposed activity was already underway. Although the evaluation preparer and reviewer had completed their requirements relative to this evaluation, the evaluation had still not been reviewed by NSG, the OSRC, or Nuclear Regulatory Affairs. The inspectors questioned how the fuel transfer activity could be underway with an incomplete 10 CFR 50.59 evaluation. The licensee stated that the intent of the supplemental reviews was of a quality assurance nature and that these reviews could be performed after the activity affected by the evaluation was implemented so long as the reviewer and preparer activities had been completed. However, the licensee acknowledged that Procedure SO123-XV-44 did not discuss this situation. The licensee issued an Action Request to address this potential procedural weakness. Since the 10 CFR 50.59 evaluation had been completed by both the reviewer and preparer, the inspectors did not consider this issue to be a compliance problem with 10 CFR 50.59.

The inspectors reviewed the licensee's 10 CFR 50.59 training program for both screeners and evaluators. The training material was instructive, comprehensive and reflective of industry experience. In addition to completing the required training courses and associated tests, screeners and evaluators must meet a personnel qualification standard that requires demonstration of practical capability in performing 10 CFR 50.59 related activities. Furthermore, the licensee required requalification every 2 years. The inspectors judged that the training program was effective and should provide assurance that 10 CFR 50.59 evaluations were being performed properly by licensee's staff.

The inspectors also verified that the preparer and reviewer of every 10 CFR 50.59 screen and evaluation reviewed during the inspection was qualified by the licensee's training program at the time the 10 CFR 50.59 screen or evaluation was performed. The inspectors were able to accomplish this task using the licensee's online training database referred to as EQIS.

The inspectors also reviewed independent audits of the 10 CFR 50.59 program by the licensee's quality assurance organization. The independent audits were well structured and thorough. No significant problems were identified by the audits. Some minor findings identified during the audits were confirmed to have been subsequently corrected or resolved.

b. Modifications to the Turbine Gantry Crane and Turbine Building North Extension Area

By letter dated July 25, 2003, the licensee submitted a license amendment request to the NRC. Proposed Change No. 271 requested a revision to sections of the Defueled Safety Analysis Report changing the design of the turbine gantry crane, turbine gantry crane capacity, and the structural descriptions of the turbine building. During the inspection, the licensee was implementing these design modifications although the NRC had not formally approved the changes. The licensee chose to begin implementing the modifications, in part, to support the Unit 1 fuel loading schedule.

The NRC inspectors conducted an in-depth review of the modifications to ensure that the licensee was implementing the changes in compliance with the license amendment request. The structural enhancements to the gantry crane and the turbine building north extension area provided a means for transferring design basis seismic stresses to earth while still carrying a 105-ton load. The modified turbine gantry crane, along with the new Ederer X-SAM trolley, will provide a single failure proof crane in accordance with NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants." The modifications being implemented included:

- Stiffening the turbine gantry crane legs and bridge girders with steel plates to increase the load capacity from 100 tons to 105 tons, as required for the new Ederer X-SAM trolley,

- Strengthening the turbine building north extension structure by adding two vertical columns with associated cross bracing,

- Installing seismic bumpers on the turbine gantry bridge legs to limit upward motion during a seismic event,

- Replacing the turbine gantry crane's existing trolley with a new trolley,

- Constructing a new cask decontamination station below the existing station in the turbine building north extension structure, and

- Installing a "LEGO" block footer on the west side of the transfer trailer runway for support and for a retaining wall.

At the end of the onsite inspection, the overall implementation of these design changes was approximately 30 percent. The inspectors concluded that the modifications were being implemented in accordance with Proposed Change No. 271 and the design drawings. As stated earlier, the change had not been approved by the NRC, and the field work was being completed at risk by the licensee. The crane was not being used

for any safety related function; therefore, the implementation of the crane and turbine building modifications had no impact on the safe storage of fuel. The licensee was aware that it could not use the modified crane for fuel handling activities until the NRC approved the license amendment request.

1.3 Conclusions

The licensee's safety review and design change program was in compliance with 10 CFR 50.59 requirements. The licensing change control process was effectively screening and evaluating the impacts of facility and procedural changes. The licensee had developed and implemented a training program for 10 CFR 50.59 screeners and evaluators. All evaluations reviewed were accomplished by trained and qualified individuals. Audits of the safety review program had been completed in accordance with quality assurance requirements.

The inspectors conducted a detailed review of the licensee's implementation of design changes to the Unit 1 turbine building gantry crane. The modifications were being implemented in accordance with the proposed change and design drawings. At the conclusion of the onsite inspection, the license amendment request had not been approved by the NRC; therefore, the licensee was conducting the work at risk. The crane was not being used for any safety related function; therefore, the implementation of the modifications had no impact on the safe storage of fuel.

2 Spent Fuel Pool Safety at Permanently Shutdown Reactors (60801)

2.1 Inspection Scope

The inspectors reviewed the safe storage of spent fuel including pool siphon and drain protection; pool instrumentation, alarms, and leakage detection; pool chemistry and cleanliness control; criticality controls; pool support equipment operation; and power supplies.

2.2 Observations and Findings

The permanently defueled technical specifications (PDTS) provide the safety limits, limiting conditions of operation, and surveillance requirements for the spent fuel pool (SFP). Plant tours, record reviews, and interviews with plant operators were conducted to verify if the Unit 1 SFP was being maintained in accordance with PDTS and procedural requirements.

Safety Limit D2.1 specifies that the water level in the SFP shall be maintained above plant elevation 16 feet whenever fuel assemblies are stored in the SFP. Further, a minimum water level limit of 40-foot, 3 inches was specified in plant procedures. The inspectors reviewed the operations logs for the period of August-October 2003 and noted that the pool had been maintained between 40-foot, 10.5 inches and 40-foot, 7 inches during that time frame. The pool level was 40-feet, 8 inches during the inspection.

Table D3.1.3-1 of TS requires that both chloride and fluoride be maintained less than or equal to 0.15 parts per million. The pool was sampled monthly by the licensee. The sample results for July-October 2003 were reviewed. The sample results document that chloride and fluoride concentrations were less than 0.05 parts per million during this time frame.

The SFP was routinely sampled for other chemical constituents such as boron concentration, pH, conductivity, sulfate, tritium content, and gamma ray activity. The inspectors reviewed the chemistry results for July-October 2003. All parameters were within the ranges specified in the applicable chemistry procedure.

The procedural limit for SFP temperature was 150 degrees Fahrenheit with a high temperature alarm setpoint of 125 degrees. The pool temperature varied between 68 and 100 degrees during August-October 2003 indicating that pool temperature remained below the procedural limit during that time frame. The pool temperature was 73 degrees during the inspection.

A leak chase system monitors for leakage from the SFP liner. Procedures require pumping of the leak detection well whenever the water level in the well exceeds plant elevation of 2.5 feet. The inspectors reviewed the SFP liner leak detection records to ascertain whether leakage was increasing or decreasing. According to the licensee's records, the liner leak rate was 0.32-0.44 gallons per day since June 2003. The liner leak rate included condensation and groundwater in-leakage into the collection system. This leak rate has been relative constant in recent years.

2.3 Conclusions

The licensee was maintaining the SFP within PDTS and procedural requirements.

3 Decommissioning Performance and Status Review at Permanently Shutdown Reactors (71801)

3.1 Inspection Scope

The inspectors evaluated whether the licensee and its contracted workforce were conducting decommissioning activities in accordance with license and regulatory requirements.

3.2 Observations and Findings

a. Site Tours/Control of Decommissioning Activities

The inspectors conducted tours of the Unit 1 facility to observe decommissioning work in progress. The inspectors observed radiological area postings, boundaries, and housekeeping. Access to the restricted and contaminated areas was controlled by radiation caution signs, barricades, boundary lines, locked doors, and gates. Radiological boundaries were well defined and posted in all areas.

The inspectors conducted radiological ambient gamma radiation surveys of the Unit 1 restricted areas using a Ludlum Model 2401-EC2 survey meter (NRC No. 016295G), in part, to confirm the accuracy of the restricted area postings. The radiation areas and high radiation areas were properly posted with warning signs and barriers as appropriate.

During site tours, the inspectors observed the licensee cutting portions of the concrete sphere enclosure building wall. The licensee planned to cut the wall into 337 blocks. The maximum design weight of each block is 20,000 pounds. The first block was cut, lifted, and lowered to the ground on July 30, 2003. At the end of the onsite inspection, the licensee had cut almost 100 blocks. The licensee was using extra rigging to ensure that blocks would not fall during the cutting and lifting process.

Since the previous inspection, the licensee experienced two incidents inside of the Unit 1 radiologically restricted area. The first incident involved a dropped radio, while the second incident involved a personnel injury. The first incident occurred on September 27, 2003. Workers on a platform dropped a radio. The radio fell about 35 feet. As the radio fell, it broke into several pieces. One or more pieces hit two employees. No one was injured and an AR was issued to investigate the incident. Corrective actions included changes in how radios are attached to workers.

The second incident occurred on October 14, 2003, when an employee slipped off a ladder while climbing down from the top of the Unit 1 containment building. At the time of the injury, the individual was inside of the radiologically restricted area but outside of a contaminated area. The injured individual was transported out of the Unit 1 radiologically controlled area using an overhead crane with a man-basket. The individual was radiologically surveyed but was not contaminated. The individual was then transported to a local hospital for medical attention. The incident revealed potential weaknesses in man-basket access during emergency situations. An AR was issued to investigate the incident and to consider potential corrective actions.

b. Advanced Horizontal Storage Module Thermal Monitoring Program

Section 5.2.5 of the NUHOMS Certificate of Compliance No. 1029 (Technical Specifications) discusses the temperature measurement requirements for each loaded Advanced Horizontal Storage Module (AHSM). According to this requirement, the concrete temperature and ambient air temperature are to be monitored daily. If any temperature indicator for a loaded AHSM indicates a temperature rise of greater than or equal to 80 degrees Fahrenheit in a 24-hour period, or a maximum temperature of 225 degrees, action must be taken to try to avoid exceeding the concrete and cladding temperature limits.

Operating Instruction SO1-12.1-9, “Inspection of Advanced Horizontal Storage Modules (AHSMs)”, Revision 0, provided instructions for inspection of the loaded modules and collection of temperatures. The inspectors observed the operation of the AHSM temperature monitoring system and discussed system operation with the computer engineer to determine system capabilities and redundancies. The inspectors also interviewed the on duty shift manager to determine his knowledge of system operation

and data collection requirements. Finally, the inspectors reviewed the results of recent AHSM inspections.

At the time of the inspection, one storage module had been loaded. The most recent temperature readings taken from the computer indicated that the concrete temperature had reached an equilibrium temperature of approximately 100 degrees, indicating that no program temperature limit was being exceeded.

The inspectors concluded that the temperature monitoring system was capable of performing its intended function. The shift manager appeared knowledgeable of the data collection requirements, and the operators were performing their surveillances in accordance with procedural instructions.

3.3 Conclusions

Site tours indicated that radiation protection controls were in place to support decommissioning activities. The licensee was cutting up the sphere enclosure building and safety precautions were being taken to prevent the dropping of concrete blocks. The licensee recently experienced an industrial accident that resulted in an injured worker in the radiologically restricted area. The licensee issued an AR to review the incident and to consider corrective actions necessary to upgrade their ability to evacuate injured individuals via the overhead crane. The licensee was conducting temperature monitoring of the spent fuel AHSM. The concrete temperature of the one loaded module was well below the high temperature procedural limits.

4 Solid Radioactive Waste Management and Transportation of Radioactive Materials (86750)

4.1 Inspection Scope

The purposes of this portion of the inspection effort were to determine whether the licensee properly processed, packaged, stored, and shipped radioactive materials, and to determine whether transportation activities were being conducted in compliance with applicable NRC and U.S. Department of Transportation regulations.

4.2 Observations and Findings

a. Radioactive Waste Management

The licensee was conducting decommissioning activities under the DECON option as stated in its Post Shutdown Decommissioning Activities Report dated December 15, 1998. DECON is defined as the immediate removal and disposal of all radioactivity in excess of levels which would permit release of the facility for unrestricted use. Details of how the licensee handles solid radioactive waste (radwaste) during decommissioning is described in Section 5.5 of the Unit 1 Defueled Safety Analysis Report.

The major sources of dry active wastes were debris generated by radioactive equipment and material removal and building demolition. During decommissioning, the waste material was being sorted based on whether it was contaminated, potentially contaminated, or not contaminated. Wastes located in containment were generally considered to be contaminated and were being routinely disposed. Wastes located in potentially non-contaminated areas such as the Turbine Building were being scanned for radioactivity to ascertain whether the item was contaminated. If the item was not contaminated, then it was free released. If the item was contaminated or if the surveyor cannot ascertain whether the item was contaminated, then the item was routinely disposed as radioactive waste.

The radioactive wastes were classified as Class A, B, or C wastes based on exposure rates. The predominate waste stream was Class A wastes. Class A wastes were typically packaged in intermodal boxes and shipped to a disposal facility in Utah. After emptying, the boxes were returned to Unit 1 for reloading. Cargo containers are Type A containers that were used to ship compactable wastes such as bagged trash to a waste broker in Tennessee for volume reduction. The cargo containers were also returned to the site for reuse after emptying. Mixed wastes were typically shipped in “B-25" boxes to the Utah site for macroencapsulation and disposal.

The inspectors noted that the licensee had developed “travelers” to assist workers during loading of boxes with radwastes. These travelers were condensed instructions obtained from procedures and provided guidance of what material could be placed in the different types of containers. The inspectors noted that travelers were attached to all intermodals, B-25 boxes, and cargo containers. The travelers reminded workers of the restrictions for placement of material into these boxes, including liquid restrictions and dose rate limitations. Travelers, in conjunction with logs of the material being placed into the respective containers, helped ensure continued compliance with Department of Transportation shipping regulations and waste disposal classifications.

The inspectors observed the licensee free releasing equipment from the site. The inspectors noted that each individual component was being checked at least once, but was usually checked more than once. Small items were surveyed using a small articles monitor. Very small items such as bolts were scanned by hand using calibrated survey meters. Larger items that would not fit into the small articles monitor or had a self-shielding design were also scanned by hand. The items were then sorted and stored in either non-radioactive or radioactive material skiffs. The material in the radioactive skiffs were disposed, while the material in the non-radioactive skiffs were free-released.

The items to be free released were scanned primarily in a maintenance area located south of the plant, but several other exit points were used to free release items after scanning. Some smaller items such as tools were routinely released following survey at the radiologically restricted area access control point. Concrete blocks from the sphere enclosure building were being scanned and released in the yard area adjacent to the Unit 1 containment structure. The inspectors observed the instrumentation being used to survey the equipment and material and noted that the instrumentation appeared fully functional at all locations.

The NRC inspectors reviewed material release logs to ensure that all equipment being free released has been surveyed and the survey results documented. The licensee did not always document the free release of each individual component. In many cases, the items were bulk released. Each skiff being released was certified by an individual the individual components have been directly surveyed or an evaluation conducted to ensure that the item was free of residual contamination. As an example, electrical conduit was released by combination of survey and evaluation. The exterior of the conduit was radiologically surveyed but an evaluation typically concluded that the interior of the conduit was most likely not contaminated.

According to procedure SO123-VII-20.9.2, “Material Release Surveys,” Revision 4, the release criterion was no detectable licensed radioactivity above background. The inspectors observed items being surveyed at several locations. The background values varied according to location. By procedure, the background cannot exceed 150 counts per minute for beta-gamma friskers, the most common hand-held instrument. During the inspection, no location being used to survey equipment exceeded the background limit of 150 counts per minute.

The area with the highest background was the north side of Unit 1 where the concrete blocks were being surveyed and free released. The background readings were elevated because of the temporary storage of the reactor pressure vessel shipping canister. Surveys were allowed to continue in this area because the background count rate did not exceed the procedural limit. A review of the radiological survey records indicated that background varied from 60 to 100 cpm. In all cases, the surveyor certified that no detectable counts above background were identified during the scans. The licensee expects the background levels to decrease significantly in this area when the canister is eventually shipped offsite.

The inspectors conducted a review of all Unit 1 radioactive material in storage. The licensee had two barrels of spent filters in storage in a locked cage in the Unit 1 radiologically controlled area. The licensee planned to dispose of this material at a later date when additional filters are accumulated. Some spent resin was being stored in the spent resin storage tank. The resin material will be processed and disposed at a later date.

The licensee also processed and disposed of wet solid wastes. The current waste stream included slurry from concrete cutting operations. This material was being stored in drums. The licensee dried the drums using one of four electric drying heaters. Once dry, the material was disposed as radioactive waste. Other potential wet solid wastes included hoses and mop heads. The licensee cut up coiled hoses to remove trapped water and mop heads were allowed to air dry prior to disposal as dry active wastes.

The licensee had Unit 1 material in temporary storage at the Multipurpose Handling Facility. The facility is an in-process staging area for the accumulation of solid radwaste until it was released from the site for shipment. In particular, the facility was used to store higher activity wastes that required special handling. At the time of the inspection, Unit 1 material in storage included one high integrity container of filters collected from the reactor vessel internals cutting project. These filters will be shipped as Class C

wastes for disposal at a licensed waste disposal facility located in South Carolina. The licensee is currently waiting for permission from the disposal site before shipping the filter material to the disposal site.

Three carbon steel liner containers of reactor mirror insulation material were also in storage. Recently, the licensee completed the cutting of the former reactor vessel insulation inside of containment. The licensee segmented the insulation using remote torch cutting techniques because the insulation was a significant radiological hazard. This material is scheduled for disposal at a state-licensed facility in Utah. The licensee is awaiting approval by the disposal facility to ship and dispose of Class A containerized wastes at the facility.

The licensee's solid radwaste program was compared to the guidance provided in several implementing procedures, including the Material Release Surveys, Solid Radioactive Waste Packaging for Class A Unstable Wastes, and Surveys for Release of Liquids, Sludges, Slurries, and Sands procedures. The inspectors concluded that the licensee's staff was implementing the solid radwaste program in accordance with site procedures.

b. Shipment of Reactor Coolant Pump

The inspectors reviewed the licensee's packaging, shipment, and disposal of reactor coolant Pump C. The pump was the third and final pump to be disposed. The inspectors reviewed the waste classification plans and transportation documents and confirmed that the waste and transportation classifications were in agreement with regulatory requirements.

Each of the three reactor coolant pumps were analyzed for potential radioactive material content. Using the sum of fractions rule specified in 10 CFR 61.55, the pump was determined to be Class A waste material. Reactor coolant Pump C was calculated to contain about 6.5 percent of the Class A waste material limit. The most limiting radionuclides were strontium-90, nickel-63, and the transuranics. The pump was shipped by exclusive use truck to a state-licensed disposal facility located near Clive, Utah, during the inspection.

The licensee determined that the reactor coolant pumps met the definition of a surface contaminated object (SCO), Group II. A SCO means a solid object which is not itself radioactive but which has radioactive material distributed on any of its surfaces. U.S. Department of Transportation (DOT) Exemption DOT-E 13212 authorized the transportation of the three reactor coolant pumps as "non-specification packages." This exemption was issued on August 25, 2003. The licensee was exempted from the requirements of 49 CFR 173.427(b)(1) which stipulate that SCO-II material be transported in authorized packaging.

In accordance with the DOT exemption, the licensee prepared the pump for shipment by welding, bolting and enclosing shut all nozzles, penetrations and openings. The licensee also treated the exterior surfaces with a special polymeric barrier coating to seal any loose contamination. Just prior to shipment, the licensee conducted a

radiological survey of the pump to ensure compliance with DOT requirements. The maximum contact dose rate was 150 millirems per hour with an exclusive use shipment limit of 200 millirems per hour. The 2-meter dose rate was 1.5 millirems per hour with a 10 millirems per hour limit. Fifteen smear samples were taken, and no removable contamination was identified.

The inspectors reviewed the shipping papers. The documents included a bill of lading, waste disposal shipping manifest, emergency instructions, driver instructions, and radiological survey. The inspectors concluded that the shipping papers met the requirements of 10 CFR 71.5, Transportation of Licensed Material.

4.3 Conclusions

The licensee had developed and implemented a comprehensive program for handling, sorting, and disposing of radioactive wastes. The Unit 1 staff was conducting the program in accordance with approved site procedures. The use of written travelers was noted to be a positive management tool to help ensure compliance with shipping and disposal regulations. The inspectors reviewed the shipment of a reactor coolant pump to an out of state disposal site and concluded that the licensee shipped the pump in accordance with NRC and DOT requirements.

5 Exit Meeting Summary

The inspectors presented the inspection results to members of licensee management at the exit meeting on October 30, 2003. The licensee did not identify as proprietary any information provided to, or reviewed by, the inspectors.

ATTACHMENT
PARTIAL LIST OF PERSONS CONTACTED

Licensee

D. Axline, Licensing Engineer, Nuclear Regulatory Affairs
R. Clark, Manager, Nuclear Safety Group
M. Love, Manager, Maintenance
J. Madigan, Manager, Health Physics
C. McAndrews, Manager, Nuclear Oversight
M. McBrearty, Engineer, Nuclear Regulatory Affairs
D. Nunn, Vice President, Engineering & Technical Services
R. Osborne, Manager, 50.59 Program
A. Scherer, Manager, Nuclear Regulatory Affairs
J. Sills, Project Manager, Unit 1 Health Physics
R. Waldo, Station Manager
J. Wambold, Vice President, Business & Financial Services

INSPECTION PROCEDURES USED

37801	Safety Reviews, Design Changes, and Mods at Permanently Shutdown Reactors
60801	Spent Fuel Pool Safety at Permanently Shutdown Reactors
71801	Decommissioning Performance and Status Review at Permanently Shutdown Reactors
86750	Solid Radioactive Waste Management and Transportation of Radioactive Materials

ITEMS OPENED AND CLOSED

Opened

None.

Closed

None.

Discussed

None.

LIST OF ACRONYMS USED

AHSM	Advanced Horizontal Storage Module
ALARA	As Low As Reasonably Achievable
AR	Action Request
CFR	Code of Federal Regulations
NSG	Nuclear Safety Group
OSRC	Onsite Review Committee
PDTS	Permanently Defueled Technical Specifications

SCO	Surface Contaminated Object
SFP	Spent Fuel Pool
DOT	U.S. Department of Transportation