U. S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-29/83-03

Docket No. 50-29

License No. DPR-3 Priority -- Category C

Licenseee: Yankee Atomic Electric Company

1671 Worcester Road

Framingham, Massachusetts 01701

Inspection at: Framingham, Massachusetts and Rowe, Massachusetts

Inspection conducted: February 15-18, 1983

NRC Personnel: E Kelly 3/25/83
E. Kelly, Reactor Engineer date signed

D L Haverkamp 3/25/83
D. Haverkamp, Reactor Licensing Engineer date signed

P. K. Eapen 3/25/83
P. Eapen, Reactor Engineer date signed

Approved by: Edward G Greenman /for 3/28/83
R. Gall., Chief, Reactor Projects Section No. 1A, date signed
Projects Branch No. 1,
Division of Project and Resident Programs

Inspection Summary:
Inspection on February 15-18, 1983 (Inspection Report Number 50-29/83-03)
Areas Inspected: Special safety inspection by three region-based inspectors (59 hours) of licensee actions taken to comply with requirements described in NUREG-0737, Item II.B.2, Design Review of Plant Shielding.
Results: No violations were identified.

DETAILS

1. Persons Contacted

   Yankee Atomic Electric Company

   R. Aron, Operations Engineer, Yankee
   B. Drawbridge, Technical Director, Yankee
   *S. Fournier, Senior Engineer, Systems Engineering Group, Yankee Nuclear Services Division (YNSD)
   *A. Hodgdon, Radiological Engineer, Radiological Engineering Group, YNSD
   K. Jurentkuff Jr., Assistant Operations Manager, Yankee
   A. Kadak, Project Manger, Yankee, YNSD
   *J. Kay, Senior Licensing Engineer-Yankee, YNSD
   P. Littlefield, Manager, Radiological Engineering Group, YNSD
   *J. Parillo, Radiological Engineer, Radiological Engineering Group, YNSD
   N. St. Laurent, Assistant Superintendent, Yankee

   *Denotes those present at exit interview on February 18, 1983

2. Plant Shielding Design Review

   a. Background and Scope

   As discussed in Item II.B.2 of NUREG-0737, "Clarification of TMI Action Plan Requirements," each power reactor licensee was required to perform a radiation and shielding design review of spaces around systems that may, as a result of an accident, contain highly radioactive materials. The design review was intended to identify the location of vital areas* and equipment in which personnel occupancy may be unduly limited or safety equipment may be unduly degraded by the radiation fields during post-accident operations of these systems. Additionally, each licensee was required to provide for adequate access to vital areas and protection of safety equipment by design changes, increased permanent or temporary shielding, or post-accident procedural controls. The design review was to determine which types of corrective actions were needed for vital areas throughout the facility.

   These requirements were discussed in Item 2.1.6.b of NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Requirements"; were issued by NRC letters dated September 13 and October 30, 1979 to all operating nuclear power plants; and were incorporated into NUREG-0660, "TMI-2 Action Plan." Significant changes in requirements or guidance were described in an NRC letter to all licensees of operating plants dated September 5, 1980, and were subsequently described in Item II.B.2 of NUREG-0737. Lastly,

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* Any area which will or may require occupancy to permit an operator to aid in the mitigation of or recovery from an accident is designated as a vital area.

an NRC letter to all licensees of operating power reactors dated March 17, 1982 (Generic Letter No. 82-05) requested reconfirmation of the schedule for completing Item II.B.2 of NUREG-0737.

With respect to operating power reactor licensee's, the October 30, 1979 NRC letter indicated that licensee's plant shielding design reviews were among those items for which post-implementation NRC review is acceptable. Although prior NRC approval was not required, licensees were to document their methods of implementation by the required completion date, e.g., design review by January 1, 1980 and plant modifications January 1, 1981.

With respect to documentation specified by NUREG-0737 for vital area access, operating license applicants were to provide to the NRC a summary of the shielding design review, a description of the results of this review, and a description of the modifications made or to be made to implement the results of the review. The submittals were to include:

(1) Specification of source terms used in the evaluation, including time after shutdown that was assumed for source terms in systems;

(2) Specification of systems assumed in the analysis to contain high levels of radioactivity in a post-accident situation;

(3) Specification of areas where access is considered necessary for vital system operation after an accident; and,

(4) The projected doses to individuals for necessary occupancy times in vital areas and a dose rate map for potentially occupied areas.

NUREG-0737 did not state that licensees of operating reactors were to submit the above documentation to the NRC. Rather, they were to have available for review the final design details of the implementation* of the Item II.B.2 position and clarification. (Information equivalent to that submitted by operating license applicants is expected to be available for review as documentation of the design review that provided the bases for final design details.) If deviations to that position and clarification were necessary, licensees were to provide detailed explanation and justification for the deviations by January 1, 1981.

The licensee's plant shielding design review and corrective actions were reviewed during this inspection. This review included licensee submittals to the NRC, a sampling verification of the shielding

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* In addition to providing clarification of requirements, NUREG-0737 revised the completion date for modifications resulting from the plant shielding design review to January 1, 1982.

design review methodology and representative calculations, a review of selected emergency procedures to determine if the vital areas where personnel must go are safely accessible, and a review of corrective actions taken or planned by the licensee (including plant modifications).

b. Licensee Submittals to the NRC and Previous Staff Evaluation

In the case of Yankee Nuclear Power Station (Yankee), the shielding design review and corrective actions were discussed by Yankee Atomic Electric Company (YAEC or the licensee) in letters to the NRC dated December 31, 1979 and April 9, 1980. The licensee's shielding study and planned actions were evaluated by the NRC staff to meet the Category "A" Lessons Learned requirements for this item (NUREG-0578 Item 2.1.6.b), as discussed in an NRC letter to the licensee dated April 18, 1980. The licensee subsequently discussed the status of corrective actions and an alternative approach to substantial shielding modifications, including a description of the alternative design details and the implementation schedule for the modifications, in letters to the NRC dated December 15, 1980, July 6, 1981, November 30, 1981, and May 14, 1982.

The above licensee and NRC letters were reviewed during this inspection to determine the licensee actions completed or to be taken, and the extent of previous staff review and evaluation, regarding the plant shielding design review for Yankee. The licensee's statements and commitments contained in these letters provided the bases, in part, for the inspector's verification that plant shielding modifications have been adequately identified and implemented as discussed in other paragraphs of this report. The information contained in these letters is described below.

J. Kay (YAEC) letter to H. Denton (NRC) dated December 31, 1979 (Subject: Lessons Learned Short-Term Requirements)

--Responded to several lessons learned short-term requirements, as requested in the NRC letter dated October 30, 1979. With respect to NUREG-0578 Item 2.1.6.b, Design Review of Plant Shielding, the licensee stated that as a result of the preliminary shielding review at Yankee, primary emphasis for shielding design changes should be directed toward shielding of the source.

--Specifically, to meet the shielding requirements with the specified source term of NUREG-0578, the licensee stated that it would be necessary to construct a concrete shield around the elevated vapor container. Based on the shielding provided by the new structure, local area shielding would be upgraded to meet access requirements. A study had been performed to estimate the schedule and manpower requirements for design and construction of a containment shield enclosure. Although YAEC's review of the scoping study had not been completed, the study indicated a project of at least three years duration with a minimum of 18 months for design engineering and licensing. In addition, there were areas of design input which depended upon the results of Systematic Evaluation Program (SEP) evaluations, including seismic design and consideration of missiles. Based on the potential impact of the SEP review schedule upon the design of this project, Yankee requested that NRC include its design review of plant shielding in the SEP program.

--The licensee had identified the systems carrying radioactive fluids under post-accident conditions and had reviewed the shielding design of the areas and equipment requiring access. The control room, portions of the primary auxiliary building, and the area adjacent to the safety injection pumps were the areas which had to be addressed.

--With respect to control room access requirements, the review indicated that the shielding provided by the rear wall of the control room, the side which is furthest away from the control room, should be increased to reduce the shine dose. The licensee committed to have the increased shielding installed prior to January 1, 1981.

--With respect to primary auxiliary building sampling sink access requirements, it would be necessary to move or rebuild the sampling station.

[NOTE: Personnel access considerations and modifications for post-accident sampling were not included in the scope of this inspection. NRC evaluation and inspection of this item is being performed separately, with respect to NUREG-0578 Item 2.1.8.a and NUREG-0737 Item II.B.3.]

--With respect to the area adjacent to the safety injection pumps, the licensee stated that access is not required for short-term operation. However, if long-term operation is required, there is the possibility that access may be required for maintenance purposes. In this event, it would probably be necessary to shield the three trains from one another and it may be necessary to reroute the cross-connecting piping. YAEC believed that any rerouting of safety injection piping, if required, should be deferred until the SEP review is completed in order that all contingencies (seismic, missiles, etc.) may be considered.

J. Kay (YAEC) letter to D. Ziemann (NRC) dated April 9, 1980 (Subject: Resolution of TMI "Category A" Implementation Audit Outstanding Items)

--Responded to several areas that required additional information, as identified during the NRC staff's "Category A" implementation audit on April 2, 1980.

--With respect to NUREG-0578 Item 2.1.6.b, the licensee (1) provided more details of the design review for plant shielding, (2) included the identification of vital areas and operator actions which would be necessary following an accident, and (3) included a discussion of the extent of potential plant modifications and a justification for deferring them to the SEP program.

--The licensee stated that primary emphasis of the shielding review was given to the direct radiation from the unshielded containment structure. The results of the dose analysis demonstrated that most of the buildings on-site would not be habitable for several days following an event of the magnitude assumed in the NUREG-0578. Calculations showed that a concrete shield, approximately 3 feet thick surrounding the containment, would be required to reduce the radiation levels to acceptable levels in the early stages of the accident.

--Licensee review of plant emergency procedures defined five areas in the plant that may require access within a short time following an accident. Those areas included the control room/onsite technical support center, hydrogen monitoring/containment air sampling station, auxiliary feedwater station, liquid sampling station, and containment isolation system (CIS) relay station. These areas were either (1) adequately shielded, (2) being provided with additional shielding, or (3) being provided with appropriate design changes to permit post-accident operation of needed systems and equipment.

--With respect to the extent of potential plant modifications and a justification for deferring them to the SEP program, the licensee stated:

"As a result of the shielding review discussed above, it appears that consideration be given to shielding the containment or the justifying a more realistic source term based on a plant systems anaylsis. Yankee (YAEC) believes these considerations should be deferred to the SEP review.

The intent of the SEP review is to determine the need for plant modifications and to defer potential major modifications of this nature until an integrated assessment can be performed. This integrated assessment would give consideration to all SEP topic reviews and determine the impact of the modification on existing plant systems to obtain the optimum design. Therefore, the need for a shield enclosure should be reviewed under SEP because, as noted in Reference (b)*, construction of a shield enclosure is a major project of at least three years duration involving many design interfaces which depend on the results of SEP topic evaluation."

*J. Kay letter to H. Denton dated December 31, 1979

D. Zieman (NRC) letter to J. Kay (YAEC) dated April 18, 1980 (Subject: Evaluation of Category "A" Lessons Learned Implementation)

--Provided the staff's evaluation of the implementation of Category "A" Lesson learned requirements (excluding 2.1.7.a) at Yankee.

--With respect to NUREG-0578 Item 2.1.6.b, the staff evaluation was based on the licensee's submittals dated December 31, 1979 and April 9, 1980 (discussed above). The evaluation discussed the licensee's preliminary shielding review assumptions, analysis results, and calculations, which showed that a concrete shield around the containment vessel would be required to reduce post-accident radiation to acceptable levels. The evaluation also discussed the licensee's proposal that further actions (such as containment enclosure or performing a more realistic analysis) be deferred to the SEP review.

--The staff concluded that the licensee had met the Category "A" requirements for this item. However, the evaluation did not indicate, specifically, either acceptance or denial of the licensee's proposal to defer further actions to the SEP review.

L. Heider (YAEC) letter to D. Eisenhut (NRC) dated December 15, 1980 (Subject: Post-TMI Requirements - Implementation Date Commitments)

--Provided commitments to the implementation schedule for various items listed in NUREG-0737. With respect to Item II.B.2, the licensee stated that the plant shielding design review had been deferred to the SEP program per the NRC letter to YAEC dated April 18, 1980. [As noted above, that NRC letter did not specifically defer this item to the SEP review. However, the letter did discuss the licensee's proposal and did not reject the deferral of major plant shielding modifications to the SEP assessment.]

J. Kay (YAEC) letter to D. Eisenhut (NRC) Dated July 6, 1981 (Subject: Shielding Design Review)

--Discussed the previous shielding review results, that identified the potential need for plant modifications, and the previous requests to include the shielding evaluation with the SEP integrated assessment. Also described the licensee's continued detailed shielding review independent of the SEP schedule.

--Detailed reviews were conducted of areas containing vital equipment, focusing on required operator actions in response to an emergency situation. The reviews were conducted with the objective of evaluating alternative approaches to substantial shielding modifications by identifying the need for additional local shielding and/or remote operability of required equipment. Based on these evalautions the licensee concluded that a viable alternative existed, (e.g. remote operation of required equipment).

--Continuous occupancy of the Control Room and Technical Support Center was determined possible due to sufficient shielding. Provisions had been made to ensure logistical supplies (i.e. food supplies, sanitary facilities, sleeping facilities) would be available for personnel who would be working in the plant, post-accident. Additionally, plant staff emergency training included the plant access arrangements that would be used for high radiation conditions.

--Detailed designs for the modifications were expected to be completed within two months.

J. Kay (YAEC) letter to D. Eisenhut (NRC) dated November 30, 1981 (Subject: Shielding)

--Described the licensee's alternative design to substantial shielding modifications, which the licensee designated as the Post-Incident Cooling (PIC) System (See Note below). This system was designed to permit shutdown, cooldown and maintenance of cold shutdown conditions from within certain shielded areas of the plant. Those areas would allow continuous or infrequent personnel occupancy in the event a large source term is generated within the vapor container which restricts general access to the plant site. The shielded areas included: (1) the main control room and technical support center located on the operating floor level of the turbine building, (2) the switchgear room located on the mezzanine floor of the turbine building below the control room, and (3) the pump room area located on the ground floor of the turbine building. The PIC system was designed to provide the required monitoring and control equipment to safely bring the plant to a cold shutdown condition and maintain it in that condition for as long as 2 or 3 months or until general access to the site is reestablished.

[NOTE: The PIC system is not an additional separate "system", but actually is a combination of existing systems and equipment, and component modifications that would accomplish the licensee's design objectives.]

--Described the PIC system design basis assumptions, functional requirements, existing functional components, and scheduled modifications. There were nine functions required to shutdown, cooldown and maintain cold shutdown conditions that would be performed by the PIC system. These functions and necessary modifications are summarized below:

(1) Function: Decay Heat Removal and Cooldown-Remove heat from the Main Coolant System (MCS) by steaming the steam generators and providing makeup to the steam generators.

Modifications Needed: Add atmospheric steam dump valves remotely operable from the main control room.

(2) Function: Cooldown to Cold Shutdown-Lower MCS temperature to less than 200°F with the shutdown cooling system, and remove this heat through the component cooling and service water systems to Sherman Pond.

Modifications Needed: Relocate the controls, for the four shutdown cooling system isolation valves in the lines to main coolant loop 4, from the primary auxiliary building to the main control room. Add motor operators to the following valves and locate their controls in the main control room:

- shutdown cooling pump discharge valve
- component cooling pumps discharge valves
- shutdown cooling and low pressure surge tank cooling heat exchangers' component cooling discharge valves and combined discharge valve
- one service water pump discharge valve
- combined service water discharge valve from the component cooling heat exchangers

Add remote indication in the main control room for shutdown cooling heat exchanger discharge flow and temperature, and component cooling water discharge temperature.

(3) Function: Main Coolant Inventory, Pressure and Boration Control-Control MCS pressure above saturation temperature and regulate system pressure as temperature is lowered with pressurizer heaters, charging pumps and the pressurizer relief valve. Also provide the capability to emergency borate the system, if required, with the charging pumps.

Modifications Needed: None*

(4) Function: Emergency Core Cooling-Maintain MCS inventory and supply core cooling in the event of a LOCA.

Modifications Needed: None*

(5) Function: Emergency Power-Supply the electrical power requirements of safe shutdown and cooldown equipment in the event of a loss of the normal off-site power supply system.

*All of the controls and indications required are available in the main control room or the switchgear room.

Modifications Needed: None*

(6) Function: Post-Accident Hydrogen Control-Vent any non-condensible gases from the MCS and monitor the hydrogen buildup in containment.

Modifications Needed: Install a redundant hydrogen analyzer inside containment and a control and indication panel in the main control room.

(7) Function: Demineralized Water Replenishment-Replenish the onsite demineralized water supply.

Modifications Needed: Install cross-connect piping and manual valves between the service water and water treatment systems, in the pump room area.

(8) Function: Spent Fuel Pit Makeup-Replenish the spent fuel pit water losses due to evaporation and/or boiling.

Modifications Needed: Install a make-up line and fill valve, remotely operable from the main control room. Change existing high/low alarm windows for spent fuel pit level to individual high and low level alarm windows.

(9) Function: Personnel and Logistical Support-Insure that operating personnel have adequate drinking water, food supplies, and other necessities available within shielded areas.

Modifications Needed: Implement emergency response personnel staffing plans and have the required logistical support items available within shielded areas.

D. Moody (YAEC) letter to D. Crutchfield (NRC) dated May 14, 1982 (Subject: Status of TMI Action Plan Item Implementation)

--Provided the status of various TMI Action Plan items, as requested by the NRC letter dated March 17, 1982 (Generic Letter No. 82-05).

--With respect to NUREG-0737 Item II.B.2 modifications, the licensee stated that the installation of modifications related to the PIC system (as described in the YAEC letter dated November 30, 1981) would be accomplished during the 1982 refueling outage.

c. Shielding Design Calculations and Dose Estimates

The inspector reviewed the details of the licensee's shielding calculations with various licensee representatives. These details included the mathematical models, assumptions, source terms, dose rates and doses to personnel during post accident access to vital areas. The licensee employed three in-house computer codes to calculate doses: DIDOS-III, SKYSHINE AND RASCAL.

DIDOS-III is a three dimensional point-kernel shielding code used to determine direct gamma radiation doses from a cylindrical geometrical source. This code was used to calculate doses from radioactive fluid-carrying pipes.

SKYSHINE estimates indirect (scattered) radiation from atmospheric reflections. The mathematical model is a modified single-scatter approach that includes absorption by, and buildup within, the medium. The code was used to calculate the "skyshine" (air-scattered) doses at the facility.

The RASCAL computer program calculates gaseous source term activity and estimates radiation dose rate levels and integrated doses (cloud immersion) in various areas of the reactor facility. This program was used to estimate the immersion dose rates and integrated doses to personnel in the control room and other vital areas.

The inspector reviewed the licensee's benchmark documentation which compared the results obtained using the above computer codes with results from standard accepted industry codes or practices. These documents indicated that the licensee's in-house computer codes employed mathematical models that were consistent with state-of-the-art methods for shielding calculations.

The inspector also reviewed the licensee's documents which demonstrated how doses to personnel during post-accident access to vital areas were maintained within the guidelines of GDC-19 and NUREG-0737 Item II.B.2. One concern was identified regarding the assumptions that supported the time-motion study. The inspector noted that one of the licensee's cummulative dose estimates was based on the assumption that a key person would need to come onsite to perform post-accident mitigating functions or otherwise augment on-shift emergency response personnel. This individual would receive a major portion of the GDC-19 dose limit (5 REM whole body, or its equivalent to any part of the body for the duration of the accident) during transit to the turbine building. The licensee's calculations assumed a vehicle transit speed of 40 miles per hour. The inspector noted that during adverse weather conditions (snow and ice) it may not be possible to maintain this speed, and, therefore, this assumption appears nonconservative. However, using a lower speed for vehicle transit in certain cases (times early after the accident) would cause cummulative doses to exceed the GDC-19 dose limit.

During the exit interview on February 18, 1983, the licensee's representatives stated that the need to bring an individual onsite would be assessed, based on emergency plan staffing requirements for response personnel. During a subsequent telephone discussion on March 16, 1983, a licensee repre.entative stated that two individuals, a chemist and an emergency communicator (NRC telephone), were needed to come onsite during off-normal hours to meet the NRC's emergency response staffing guidelines. However, it may not be necessary, or appropriate, to bring these individuals onsite if very high radiation levels exist outside the vapor container. In that event, the inspector noted that the licensee's emergency plan and implementing procedures should address alternative emergency staffing provisions, consistent with the guidelines discussed in NUREG-0737 Supplement 1. Since this concern, (although related to post-accident vital areas access requirements discussed in NUREG-0737 Item II.B.2) is germaine to emergency response staffing plans, the item will be reviewed during a subsequent NRC emergency preparedness followup assessment (029/83-03-01).

d. Vital Area Accessibility-Procedure Review

The inspector reviewed three emergency procedures that would be implemented by the licensee in the event of a loss of coolant accident. The review included (1) a plant walkdown of the procedures to determine the ability to perform the procedural steps and the accessibility of manual valves or breakers that may require local operation, and (2) an assessment of potential exposures to plant personnel based on the results of the licensee's shielding design review. The procedures reviewed included: Operations Procedure OP-3051, "Loss of Main Coolant Pressure and/or Safety Injection Initiation," Revision 6 dated 01/83; OP-3000, "Emergency Shutdown from Power," Revision 18 dated 11/82; and, OP-3106, "Loss of Main Coolant," Revision 24 dated 01/83.

Based on this review, the inspector determined that (1) the procedures could be performed from the vital areas identified by the licensee's shielding design review, (2) the procedures contained appropriate provisions to assure controlled access to vital areas for post-accident operations, and (3) post-accident doses to plant personnel would be within the guidelines of NUREG-0737.

The inspector had no further questions regarding the accessibility of vital areas associated with these procedures.

e. Corrective Actions

The inspector reviewed the licensee's assessment of vital areas, as documented in a licensee memorandum, "Vital Areas Requiring Access in a Post Accident Environment at Yankee-Rowe," SEG 366/80 (W. Jones to P. Littlefield), dated October 7, 1980. The memorandum described the plant engineering review of emergency procedures and an evaluation to determine the need to occupy various areas in response to an accident. The memorandum also listed 14 areas that may require continuous occupancy or infrequent access. Based on the results of this review, the licensee determined that the calculated doses would preclude post-accident access to most plant areas and that certain operational actions could not be performed without appropriate corrective actions.

The licensee's initial plans (vapor container shielding) and revised approach (PIC system) for plant modifications are discussed in paragraph 2.b. The inspector reviewed the licensee's evaluation of existing post-accident system capabilities and system modifications (collectively designated as the PIC system), which were needed to meet the requirements described in NUREG-0737 Item II.B.2. The evaluation was documented in a licensee memorandum "Design Details of Yankee-Rowe Shielding Review," SEG 327/81 (S. Fournier to A. Kadak) dated September 9, 1981. The memorandum listed the 22 systems and supporting systems that were reviewed by the licensee, the specific modifications that were required, and the 4 remaining vital areas. The plant modifications identified in this memorandum were described in the licensee's submittal dated November 30, 1981.

The inspector verified that the PIC system modifications have been satisfactorily completed, as discussed in paragraph 3. In addition, the inspector verified that the licensee had incorporated appropriate procedural changes, as a result of plant modifications, based on a sampling review of the procedures described in paragraph 2.d and the following additional procedures:

--OP-2100 "Plant Startup from Cold Shutdown", Revision 28, dated 12/82
--OP-2105 "Plant Cooldown from Hot Standby", Revision 19, dated 12/82
--OP-2162 "Operation of the Shutdown Cooling System," Revision 10, dated 12/82
--OP-3053 "Inadequate Core Cooling", Revision 5, dated 12/82
--OP-3107 "Steam Generator Tube Rupture", Revision 12, dated 3/83
--OP-3203 "Loss of Feedwater", Revision 12, dated 3/83

The inspector also verified that the logistical supplies, as discussed in the licensee's submittals to NRC dated July 6, 1981 and November 30, 1981, were in place. The inspector had no further questions regarding the licensee's shielding design review corrective actions.

3. Implementation of TMI TAP (NUREG-0737) Requirements - Item II.B.2 Plant Shielding Modifications

The inspector reviewed the licensee's implementation of modifications that resulted from the plant shielding design review discussed in paragraph 2. The inspector reviewed the engineering design documents and respective system/equipment modifications listed below:

-- Engineering Design Change Request (EDCR) 79-34, "Control Room Shield Wall", approved July 18, 1980

-Added 12-inch thick concrete shield between the turbine hall and the control room

-- EDCR 81-33, "Post Incident Cooling Instrumentation," Change No. 1, approved June 23, 1982

- Provided pump bearing vibration monitoring in the control room for three high pressure safety injection pumps, three low pressure safety injection pumps and two emergency feedwater pumps
- Provided shutdown cooling system flow and temperature monitoring at the shutdown valve panel in the control room
- Provided component cooling temperature (from the shutdown cooling heat exchanger) monitoring at the shutdown valve panel
- Provided north and south spent fuel pit level indication at the gravity drain tank transfer pump panel in the control room, and provided a common annunciator panel alarm

-- EDCR 81-35A "Atmospheric Steam Dump", approved July 16, 1982 (with minor Change Nos. 1-3)

- Replaced the four manual atmospheric steam dump valves with motor operated valves capable of operation from the control room

- Because these valves are located outdoors, provided each valve with internal space heaters for the valve motor and limit switch compartment

-- EDCR 81-35B "Spent Fuel Pit Fill Line", Change No. 1, approved August 8, 1982

- Provided a tap-off from the condensate system (component cooling surge tank supply) to the spent fuel pit, including 3/4-inch stainless steel tubing and a solenoid-operated flow isolation valve, controlled from the gravity drain tank transfer pump panel

-- EDCR 81-35C, "Service Water to Demineralized Water Cross Tie" Change No. 1, approved June 30, 1982

- Provided piping, valves and a spool piece to permit transfer of water from the service water header to either the demineralized water tank or the primary water storage tank, from a shielded location

-- EDCR 81-37, "Emergency Power Load Revisions", approved June 14, 1982

- Provided motor operators and associated controls for eight previously manual valves, including component cooling valves CC-605, 608, 631, 633, and 635; shutdown cooling valves SC-611 and service water valves SW-404 and 609

- Provided emergency power sources for safety injection valves SI-MOV-48, 514, 515, 516, 517 and 518

- Installed dual contactors and associated controls for existing safety injection tank recirculation valve CS-MOV-532 and safety injection valve SI-MOV-4 to prevent spurious operation

- Modified the control of safety injection valves SI-MOV-514 and 515 to permit throttling

- Modified and transferred the control of shutdown cooling valves SC-MOV-551, 552, 553 and 554 from the primary auxiliary building to the control room

- Modified the 480 volt emergency switchgear buses 1, 2 and 3 to accommodate new emergency motor control centers 3 and 4, and new alternate feeds to emergency motor control centers 1 and 2

- Provided control switches (OFF only) and indication in the control room for gravity drain tank transfer pumps P-25-1 and P-25-2

-- Plant Alteration (PA) 82-126, "Auxiliary Boiler Controls Modification," approved August 5, 1982

- Installed dual controls for the auxiliary boilers on the south wall of the turbine hall (ground floor) to allow an operator to remotely switch the boilers from low-fire to high-fire, and vice versa, from a shielded location

With respect to the plant modifications approved by the above engineering design documents, the inspector verified that design changes had been properly reviewed, approved and controlled in accordance with licensee procedures; that procedures and drawings had been changed as necessary (sampling review); and that personnel had been provided appropriate training reference material. The inspector did not, specifically, review modification installation documents or records of completed training, as these items are periodically reviewed during routine NRC inspections. However, the inspector observed the modified systems and equipment (piping, valves, motor operators, motor control centers, emergency switchgear, control panels, parameter indications, shielding, etc.) during a plant walk-through on February 17, 1983. Based on the inspector's reviews and observations described above, the shielding design review plant modifications were determined to be satisfactorily completed.

4. Exit Interview

The inspectors met with licensee representatives (denoted in paragraph 1) at the conclusion of the inspection on February 18, 1983, to discuss the inspection scope and findings, as described in this report.