BEFORE THE UNITED STATES NUCLEAR REGULATORY COMMISSION

Application of SOUTHERN CALIFORNIA EDISON COMPANY and SAN DIEGO GAS & ELECTRIC COMPANY for a Class 104(b) License to Acquire, Possess, and Use a Utilization Facility as Part of Unit No. 1 of the San Onofre Nuclear Generating Station	)	DOCKET NO. 50-206 Amendment No. 181

SOUTHERN CALIFORNIA EDISON COMPANY and SAN DIEGO GAS & ELECTRIC COMPANY, pursuant to 10 CFR 50.90, hereby submit Amendment Application No. 181.

This amendment consists of Proposed Change No. 222 to Provisional Operating License No. DPR-13. Proposed Change No. 222 is a request for approval of the thermal shield replacement support system design. Proposed change No. 222 is also a request to revise License Condition 3.M to continue the thermal shield monitoring program for Cycle 11 operation.

The proposed design of the replacement support system will correct the deficiencies in the original design, and the thermal shield monitoring program will continue to monitor the condition of the thermal shield during Cycle 11 operation to provide early detection of unexpected degradation.

In the event of conflict, the information in Amendment Application No. 181 supersedes the information previously submitted.

Based on the significant hazards analysis provided in the Description and Significant Hazards Consideration Analysis of Proposed Change No. 222, it is concluded that (1) the proposed change does not involve a significant hazards consideration as defined in 10 CFR 50.92, and (2) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed change.

Subscribed on this 20 day of April, 1990.

Respectfully submitted,

SOUTHERN CALIFORNIA EDISON COMPANY

By: Harold B. Ray
Harold B. Ray
Vice President

Subscribed and sworn to before me this
20 day of April, 1990.

Yomar V. Cleary
Notary Public in and for the
State of California

OFFICIAL SEAL
YOMAR V. CLEARY
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My Comm. Exp. May 8, 1992

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of SOUTHERN CALIFORNIA EDISON COMPANY and SAN DIEGO GAS & ELECTRIC COMPANY (San Onofre Nuclear Generating Station, Unit No. 1)	))))	Docket No. 50-206

CERTIFICATE OF SERVICE

I hereby certify that a copy of Amendment Application No. 181 was served on the following by deposit in the United States Mail, postage prepaid, on the 23 day of April, 1990.

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DESCRIPTION AND SIGNIFICANT HAZARD CONSIDERATION ANALYSIS OF PROPOSED CHANGE NO. 222 TO PROVISIONAL OPERATING LICENSE NO. DPR-13

This is a request for approval of the revised design of the thermal shield support system and to revise License Condition 3.M of the Provisional Operating License No. DPR-13 for San Onofre Nuclear Generating Station, Unit 1 (SONGS 1).

EXISTING LICENSE CONDITION

Please See Attachment 1.

PROPOSED LICENSE CONDITION

Please See Attachment 2.

DESCRIPTION OF CHANGE

Thermal Shield Support System Replacement

The thermal shield support system replacement will result in a thermal shield configuration which is similar to, but stronger than the original design. The replacement support system corrects the deficiencies in the original design, which resulted in the unexpected degradation of the support system discovered during the Cycle 10 refueling outage.

Lower Support Blocks

The six lower support blocks will be enlarged to triple the contact area, thereby, greatly increasing the joint efficiency and, therefore, the strength of the design. The number of bolts will be increased from five to seven resulting in a total cross sectional increase of more than 75%, while the total cross sectional area of the dowel pins will be increased by more than 150%. The bolts and dowel pins will be restrained either by a locking nut or a deformed head design to preclude the problems experienced at Haddam Neck.

Flexures

The thermal shield support system replacement incorporates an enhanced flexure design with lower stress concentrations and improved geometry. The flexures will be preloaded to minimize mean stresses at operating conditions. In addition all existing welded connections will be replaced with bolted/dowel pin fasteners to facilitate preloading and ease of installation. One flexure will be relocated to reduce the loading on the highest load flexure.

Limiter Keys

For increased radial support at the top of the thermal shield during abnormal events, the original four limiter keys will be replaced. The replacement design will be similar to the existing design, but will be bolted rather than welded for ease of installation. The keyway gap will be reduced for greater strength, and keyways will be reconditioned.

The design and replacement plan for the thermal shield support system are discussed in Attachment 3.

License Condition 3.M

License Condition 3.M requires Edison to implement a program to monitor the condition of the reactor vessel thermal shield throughout Cycle 10 or until the thermal shield support system is replaced. This License Condition also includes requirements to develop acceptance criteria for neutron noise/loose parts monitoring, and initiate a plant shutdown in the case of a seismic event of 0.25g or greater.

Proposed Change No. 222 revises the License Condition 3.M to implement a program to monitor the thermal shield during Cycle 11 operation after the thermal shield support system has been replaced. The proposed License Condition also includes requirements to develop neutron noise/loose parts monitoring acceptance criteria for Cycle 11 operation.

BACKGROUND

Thermal Shield Design

The reactor vessel thermal shield at Unit 1 is designed to attenuate both neutron and gamma radiation from the core before it impinges on the reactor vessel "beltline" area. This, in turn, minimizes the radiation-induced embrittlement of the reactor vessel and helps maintain the material strength and ductility of the reactor vessel.

The thermal shield is fabricated from stainless steel and is approximately 2-1/2 inches thick, 134 inches in outer diameter, and 163 inches high, weighs approximately 48,000 pounds, and is supported at the bottom by six stainless steel blocks located at 60° intervals around the circumference. The support blocks attach the thermal shield to the reactor core barrel. At the top of the thermal shield, six flexures attach the shield to the core barrel, while accommodating differential vertical expansion between the thermal shield and the core barrel. Additionally, four

"limiter" keys installed 11 inches below the top of the shield allow vertical movement while limiting radial motion between the thermal shield and core barrel.

Cycle 10 Inspection

In late 1987, Westinghouse Electric Co. informed SCE that the reactor vessel thermal shield at the Haddam Neck plant had been found to have degraded supports. As a result, the SONGS 1 thermal shield was visually inspected on January 1, 2, and 3, 1989, during the Cycle 10 refueling outage . The inspection found:

- Five of the six flexures were broken (four were known to be broken from prior inspections).
- Three of thirty support block bolts appeared to be degraded.
- One of the dowel pins in a lower support block had a broken tack weld.

No other damage was found.

The inspection results were evaluated and it was demonstrated that safe operation of the reactor was not impacted by the existing condition of the thermal shield.

Cycle 10 Thermal Shield Monitoring

The plant was returned to service on July 31, 1989. However, as a condition of continued operation, SCE implemented a program to monitor the condition of the thermal shield during Cycle 10 operation, instituting a weekly monitoring program.

During Cycle 10 operation, two independent systems are being used to monitor the condition of the thermal shield - - neutron noise and loose parts monitoring. The neutron noise monitoring system employs electrical current signals from the power range neutron detectors. The loose parts monitoring system uses four accelerometers mounted on the reactor vessel flange. The two monitoring systems provide early detection of potential degradation in the thermal shield support system.

DISCUSSION

Thermal Shield Support System Replacement

In an October 6, 1989 letter, SCE committed to correct the thermal shield degradation during a combined Cycle 11 refueling and thermal shield outage scheduled to begin no later than

June 30, 1990.

DESIGN

The design of the replacement thermal shield support system includes the following improvements:

- The design of the bottom support blocks will be revised to eliminate gaps between the thermal shield and the support blocks, and between the support blocks and the core barrel. New support blocks will be wider and will have additional fasteners.

- The existing flexures will be replaced with new flexures to be made of improved material. In addition, the geometry of the flexure region will be improved to reduce stresses due to radial and tangential loading, and the design will include a preload on the web portion of the flexures to minimize mean stresses at operating conditions.

- The limiter keys will be replaced with new keys of similar design. The new limiter keys will be bolted rather than welded, and the clearance gaps will be reduced. Prior to the installation of the keys, the keyways will be reconditioned.

The proposed design of the support system is a significant improvement over the original design. Enhancements are made in the stiffness and strength of the lower support blocks, and the strength of the flexures. Lessons learned from the Haddam Neck experience have also contributed to the design of the SONGS 1 thermal shield support system. The dowel pins and bolts will be restrained either by locking nuts or deformed heads to prevent fasteners from backing out as was the case at Haddam Neck. These design improvements will restore the thermal shield to a condition stronger than the original design, and will correct the deficiencies in the original design.

DESIGN QUALIFICATION

The degradation of the thermal shield support system at SONGS 1 has been attributed to flow induced vibrational (FIV) loads resulting in high cycle fatigue failure of the lower support block bolts and flexures. This conclusion is based on analytical work performed during the Cycle 10 outage, and Westinghouse evaluations performed in early 1970's for another Westinghouse plant of similar design.

The revised design of the support system, consisting of six lower support blocks, six flexures, and four limiter keys, is based on

a flow induced vibration parametric study. The design will be qualified to the ASME Code whenever possible. However, if a component can not be qualified to the ASME Code, justification for an alternative design limit or qualification method will be provided. The loads identified for the design qualification are dead weight, thermal load, pressure load, flow induced vibrational load, seismic load, and bolt preload.

As discussed in a July 25, 1989 NRC letter, regarding postulated pipe breaks in primary coolant loop piping, SONGS 1 satisfies the conditions specified in Generic Letter 84-04 "Safety Evaluation of Westinghouse Topical Reports Dealing with the Elimination of Postulated Pipe Breaks in PWR Primary Main Loops." Therefore, consideration of dynamic effects associated with postulated pipe ruptures of the primary coolant loop piping is not required for SONGS 1.

Cycle 11 Thermal Shield Monitoring Program

Although the replacement thermal shield support system is not expected to degrade, as an additional precaution, and to monitor the condition of the thermal shield, the monitoring program will be continued for Cycle 11 operation. This monitoring program will basically be an extension of the existing program.

SIGNIFICANT HAZARD CONSIDERATION ANALYSIS

As required by 10 CFR 50.91(a)(1), this analysis is provided to demonstrate that a proposed license amendment to revise the License Condition 3.M for monitoring the reactor vessel thermal shield and to revise the design and replace the thermal shield support system does not represent a significant hazard consideration. As discussed below, in accordance with the three factor test of 10 CFR 50.92 (c), implementation of the proposed license amendment was analyzed using the following standards and was found not to: 1) involve a significant increase in the probability or consequences for an accident previously evaluated; or 2) create the possibility of a new or different kind of accident from any accident previously evaluated; or 3) involve a significant reduction in a margin of safety.

Background

In late 1987, Westinghouse informed SCE that the reactor vessel thermal shield at the Haddam Neck plant had been found to have degraded supports. Because of the similarity between the Haddam Neck and SONGS 1 thermal shield designs, the SONGS 1 thermal shield was visually inspected in January of 1989, during the Cycle 10 refueling outage. The inspection found five of the six flexures were broken (four were known to be broken from prior inspections), three of

thirty support block bolts appeared to be degraded, and one of the dowel pins in a lower support block was also found to have a broken tack weld. No other damage was found.

The inspection results were evaluated and it was determined that safe operation of the plant was not impacted by the existing condition of the thermal shield.

On July 31, 1989, the plant was returned to service. However, as a condition to continue operation, SCE implemented a program to monitor the thermal shield during Cycle 10 operation. SCE instituted weekly monitoring using signals from the Nuclear Instrumentation System and from a set of four accelerometers mounted on the vessel flange. In addition, SCE committed to replace the thermal shield support system during a combined thermal shield and Cycle 11 refueling outage scheduled to begin no later than June 30, 1990.

1. Will operation of the facility in accordance with this proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

RESPONSE: No

Thermal Shield Monitoring

The replaced thermal shield support system is not expected to degrade. However, as a precaution, the thermal shield monitoring program will be continued for Cycle 11 operation. The monitoring methods proposed consist of neutron noise and loose parts monitoring. The equipment and procedures to be used are similar to those used in the Cycle 10 monitoring program.

The thermal shield monitoring acceptance criteria will be established by performing a baseline evaluation for 60 days at $\geq$85% power following return to service for Cycle 11 operation. An interim monitoring acceptance criteria is not required because the thermal shield support system will not degrade significantly during the short period of the baseline evaluation.

Based on Cycle 10 experience, 60 days at $\geq$85% power will provide sufficient time to establish a baseline and develop the acceptance criteria for Cycle 11 thermal shield monitoring.

The current monitoring program (License Condition 3.M) requires the plant be shut down to inspect the thermal shield in case of a seismic event of 0.25g or greater.

This requirement was imposed by the NRC due to the degraded condition of the thermal shield supports, and will not be needed for Cycle 11 operation because the new thermal shield support system will be designed to withstand a seismic event of 0.67g, the design basis earthquake for SONGS 1.

This proposed change establishes the thermal shield monitoring program for Cycle 11 operation. The implementation of the monitoring program does not impact any of the plant's safety features including the condition of the thermal shield.

Thermal Shield Support System Replacement

The thermal shield support system replacement corrects a deficiency in the original design. Although degradation of the replacement thermal shield support system is not expected, analysis (presented to the NRC in March/April 1989) was performed to show that complete failure of all support blocks and flexures would not result in the thermal shield tilting or dropping out of position. Nevertheless, the postulated case of the thermal shield dropping or tilting was considered as an incredible "worst case" scenario, and it was shown that if the thermal shield were to drop, it would rest on the lower radial supports and that the normal core flow would not be significantly affected.

SCE has followed the Haddam Neck thermal shield event closely, and the lessons learned from the Haddam Neck experience have also been factored into the proposed design of SONGS 1 replacement thermal shield support system. Specifically, the dowel pins and bolts will be restrained either by locking nuts or deformed heads to prevent fasteners from backing out as was the case at Haddam Neck.

The replacement of the support system will restore the thermal shield to a condition stronger than the original and will correct an initial design deficiency, and the on-line monitoring will detect changes in the support system prior to substantial degradation. In addition, analysis of complete support system failure, which is considered to be incredible, has shown that normal core flow would not be significantly affected. Therefore, the operation of the facility in accordance with this proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Will operation of the facility in accordance with this change create the possibility of a new or different kind of accident from any accident previously evaluated?

RESPONSE: No

This proposed change provides for the monitoring of the condition of the thermal shield during Cycle 11 operation and provides the design and replacement plan for the thermal shield support system. As stated above, the implementation of the thermal shield monitoring system does not have any impact on plant safety systems and the operation of the thermal shield. Also, as stated above, the replacement of the thermal shield support system will correct an initial design deficiency, and the monitoring program will ensure the thermal shield will continue to function as intended. Therefore, operation of the facility in accordance with this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Will operation of the facility in accordance with this proposed change involve a significant reduction in a margin of safety?

RESPONSE: No

As discussed in part 1 above, the thermal shield monitoring system does not have any effect on the operation of the thermal shield or any other plant safety system. The thermal shield is not expected to degrade following the replacement of the support system. However, the monitoring program , proposed for Cycle 11 operation, will ensure that the thermal shield will function as designed by providing early detection of degradation. Therefore, the implementation of the Cycle 11 thermal shield monitoring program and the replacement of the thermal shield support system does not impact a margin of safety.

SAFETY AND SIGNIFICANT HAZARD CONSIDERATION DETERMINATION

Based on the preceding analysis, it is concluded: (1) Proposed change No. 222 does not involve a significant hazard consideration as defined by 10 CFR 50.92; and (2) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed change.

Attachments:

1 - Existing License Condition
2 - Proposed License Condition
3 - Design and Replacement Plan for Thermal Shield Support System

ATTACHMENT 1

EXISTING LICENSE CONDITION 3.M

M. Cycle X Thermal Shield Monitoring Program

The neutron noise/loose-parts detection system shall be used to monitor the condition of the reactor vessel thermal shield throughout Cycle X or until repair. Periodic monitoring of both neutron noise and loose-parts vibrations confirms that no long term unacceptable trend of degradation is occurring. The details of this program are described below.

(1) The unit will be shut down no later than June 30, 1990 to inspect the condition of the thermal shield.

(2) During the first 7 days of $\geq$ 85% power, interim acceptance criteria for neutron noise/loose-parts monitoring will be developed. These interim criteria will be utilized until the final acceptance criteria is developed.

Final acceptance criteria for neutron noise/loose-parts monitoring will be established by performing baseline evaluations for 45 calendar days at $\geq$ 85% power following return to service for Cycle X operation. The base line data will be established by recording a minimum of 16 segments of data information, each of 20 minute duration at $\geq$ 85% power. Adjustments to the acceptance criteria will be made for cycle burnup and boron concentration changes throughout the cycle.

(3) The neutron noise/loose-parts monitoring system shall be OPERABLE in MODE 1 with:

a) At least two horizontal loose-parts detectors monitored for at least five (5) minutes 2 times per day; and,

b) at least three (3) neutron noise inputs monitored for at least twenty (20) minutes once a week, and be analyzed for cross power spectral density, including phase and coherence.

(4) The data provided by the loose-parts/neutron noise monitor shall be analyzed once per week and compared with the established criteria. If the data exceeds the acceptance criteria:

a) Within 1 day the NRC will be informed of the exceedance.

b) Within 14 days the conditions will be evaluated and a report provided to the NRC documenting future plans and actions.

c) The plant will be shutdown should the remaining flexure be demonstrated failed.

(5) Each channel of the loose-part detection system shall be demonstrated OPERABLE in MODE 1 by performance of a:

a) CHANNEL CHECK at least once per 24 hours

b) CHANNEL TEST at least once per 31 days

The surveillance requirements for neutron noise monitor are covered by the Appendix A Technical Specification 4.1.1 for the Power Range Neutron Flux.

(6) With the neutron noise/loose-parts detection instrumentation inoperable for more than 7 days, licensee shall submit a Special Report to the Commission pursuant to Appendix A Technical Specification 6.9.2 within the next 3 days outlining the cause of the malfunction and the plans for restoring the system to operable status.

(7) In the case of a seismic event of 0.25g or greater as indicated on site sensors, a controlled shut down shall be initiated. Before operations are resumed, it will be demonstrated that no thermal shield damage has occurred due to the seismic event.

(8) The provisions of Appendix A Technical Specification 3.0.4 are not applicable to this license condition.