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. 184

SOUTHERN CALIFORNIA EDISON COMPANY and SAN DIEGO GAS & ELECTRIC COMPANY, pursuant to 10 CFR 50.90, hereby submit Amendment Application No. 184.

This amendment consists of Proposed Change No. 226 to Provisional Operating License No. DPR-13. Proposed Change No. 226 is a request to revise the Basis section of Technical Specification 3.1.4, "Leakage and Leakage Detection Systems." The proposed change clarifies the basis for establishing the maximum acceptable leakage rate for the reactor coolant system. Part of the existing basis is no longer consistent with plant commitments and capabilities. PCN-226 will remove this part of the Basis, which refers to a loss of alternating current power, and revise the remaining part of the Basis to clarify that the reactor coolant leakage limitations are based on the equipment limitations currently discussed in the Basis of the technical specification.

In the event of conflict, the information in Amendment Application No. 184 supersedes the information previously submitted.

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Based on the significant hazards analysis provided in the Description and Significant Hazards Consideration Analysis of Proposed Change No. 226, 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 3rd day of October, 1990.

Respectfully submitted,

SOUTHERN CALIFORNIA EDISON COMPANY

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

Subscribed and sworn to before me this
3rd day of October, 1990.

Holly D Roset
Notary Public in and for the
State of California

NOTARY PUBLIC
STATE OF CALIFORNIA
County of Orange
HOLLY D. ROSET
My Appointment Exp. 7-18-94

James A. Beoletto
Attorney for Southern
California Edison Company

By: James A. Beoletto
James A. Beoletto

Enclosure 1

DESCRIPTION AND SIGNIFICANT HAZARD CONSIDERATION ANALYSIS OF PROPOSED CHANGE NO. 226 TO PROVISIONAL OPERATING LICENSE NO. DPR-13

The following is a request to revise the basis of Section 3.1.4, "LEAKAGE AND LEAKAGE DETECTION SYSTEMS" of the Appendix A Technical Specifications for the San Onofre Nuclear Generating Station, Unit 1 (SONGS 1).

DESCRIPTION OF CHANGE

Technical Specification 3.1.4 limits reactor coolant leakage to six gallons per minute (gpm). The existing basis states that this limitation is based on the capability to remove decay heat for a period in excess of 12 hours during a loss of power event concurrent with a failure of onsite generation. The existing basis also states that the six gpm value is based on equipment limitations of the radwaste system. This proposed change will remove the discussion of a 12 hour loss of power event from the Basis and will clarify that the correct basis for the six gpm value is the radwaste system limitations.

The assumption of a 12 hour loss of offsite power event, or station blackout, is not consistent with our current analysis which assumes a two hour event. Additionally, in response to 10 CFR 50.63, we have developed a new station blackout analysis which postulates a duration of four hours. This new analysis is currently under review with the NRC. When the new analysis is approved, new or revised technical specifications specifically addressing station blackout considerations may be required. Currently however, it is not appropriate to address station blackout concerns in the existing technical specification because it was designed to control reactor coolant leakage during normal operating conditions. For these reasons, the discussion of a loss of power event in the Basis of Technical Specification 3.1.4 will be removed altogether.

EXISTING TECHNICAL SPECIFICATIONS

See Attachment 1

PROPOSED TECHNICAL SPECIFICATIONS

See Attachment 2

DISCUSSION

Summary of Changes

PCN-226 revises the Basis of Technical Specification 3.1.4 to clearly state that the basis of the six gallons per minute (gpm) leakage limitation is to avoid exceeding the long term flow capabilities of the radwaste system. Technical Specification 3.1.4 provides limits for the leakage rates from the

reactor coolant system during normal operation. PCN-226 removes the paragraph stating that decay heat can be removed with leakage rates of six gpm during a loss of power event lasting up to 12 hours. PCN-226 also adds a paragraph discussing flow provided during normal operation to the letdown system and the reactor coolant pump seals.

Licensee Event Report

Licensee Event Report (LER) 1-90-004, Revision 1, dated April 25, 1990, identified a potential for Reactor Coolant System (RCS) leakage rates which exceed the six gallons per minute (gpm) maximum allowed by Technical Specification 3.1.4.

The LER was written when a letdown orifice isolation valve (CV-203) was discovered to be leaking in excess of the six gpm allowed by Technical Specification 3.1.4. CV-203 can be open or closed during normal operation, depending on letdown system requirements, but is required to isolate the reactor coolant system during a loss of power event. If the valve leaks upon isolation initiated by a loss of power event, reactor coolant inventory can be lost and the possibility of violating the current Basis of Technical Specification 3.1.4 is created.

LER 1-90-004 notes that this possibility can only occur during an extended loss of power event or station blackout. The existing Basis of Technical Specification 3.1.4 mentions that with a leakage rate of up to six gpm, decay heat removal can be accomplished for a period of up to 12 hours in the event of a sustained loss of all offsite power concurrent with the failure of on-site generation. The corrective actions for LER 1-90-004 required the basis of Technical Specification 3.1.4 to be revised to be consistent with current plant licensing provisions regarding loss of power events. The LER also requires the basis to identify consistent letdown system isolation requirements. PCN-226 was developed to satisfy these corrective actions in the LER.

Licensing Background

A review of the past changes to Technical Specification 3.1.4 determined that the capability to remove decay heat during a loss of power event lasting up to 12 hours was part of the original basis for the allowable leakage rate in the late 1960's. The allowable leakage rate at that time was 10 gpm.

On January 13, 1972 the Atomic Energy Commission issued Change Number 7 to the technical specifications. This change lowered the leakage limit in Technical Specification 3.1.4 from 10 gpm to 6 gpm. The leakage limitation was reduced to avoid exceeding the long term processing capabilities of the gas handling part of the radwaste system. This new value was also established as the "lowest practical value" in accordance with the requirements of the "Interim Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Power Reactors", which was adopted by the Atomic Energy Commission on June 19, 1971.

Change Number 7 also revised the paragraph in the Basis of Technical Specification 3.1.4 addressing the capability to remove decay heat during a loss of power event lasting up to 12 hours. However, only the leakage value was changed - from 10 gpm to 6 gpm - and no wording changes were made. While this change was conservative because the leakage rate was being decreased, it made the Basis of Technical Specification 3.1.4 misleading because the capability to remove decay heat with a leakage rate of 6 gpm during a loss of power event lasting up to 12 hours is now stated as the basis of the 6 gpm value. The actual basis for the 6 gpm value is the limitations of the radwaste system. The Basis should have been revised to reflect this.

The erroneous statement in the Basis of Technical Specification 3.1.4 connects station blackout survival to the six gpm leakage limitation. As a result, LER 1-90-004 was generated when CV-203 was discovered to be leaking in excess of six gpm.

Technical Background

The leaking letdown orifice isolation valve, CV-203, is one of three valves used in parallel (CV-202, CV-203, and CV-204) to control letdown system flow. Each of the three valves controls flow during normal operation to a calibrated orifice. By opening different valves or different combinations of these valves, the flow supplied to the letdown system can be controlled.

The containment isolation valves for the letdown system are located downstream of the letdown system pressure isolation valves and downstream of a relief valve, RV-206. The relief valve setpoint is approximately 485 psig. During normal operation, reactor coolant flowing into the letdown system is reduced from the high pressure of the reactor coolant system to the letdown system pressure by flowing through the orifices controlled by the three letdown system orifice isolation valves and through a downstream back pressure control valve (PCV-1105) set to maintain 350 psig.

The orifice isolation valves, CV-202, CV-203, and CV-204 are designed to fail closed on a loss of power. The letdown system containment isolation valves, CV-525 and CV-526, are also designed to fail closed, however these valves do not close automatically during a loss of power event since they are powered from the station batteries and will remain energized until closed by operator action.

If one or more of the letdown system orifice isolation valves leaks and CV-525 and CV-526 remain open, then reactor coolant will flow into the letdown system. During a loss of power event the capability to return this flow to the reactor coolant inventory is lost. If CV-525 and/or CV-526 are closed, as was the case postulated in LER 1-90-004, then reactor coolant will build up behind the letdown system containment isolation valve. When enough coolant has leaked to build the pressure up to the relief setting of RV-206, coolant will flow out of the reactor coolant system through the relief valve. This flow is then lost from the reactor coolant inventory. It is through this sequence of events that a letdown system orifice isolation valve which is

leaking in excess of six gpm causes the current Basis of Technical Specification 3.1.4 to be violated.

Station Blackout Background

Since the late 1960's, when the current discussion of station blackout in the Basis of Technical Specification 3.1.4 was developed, there have been several changes to the Station Blackout program at San Onofre Unit 1. Generic Letter 81-04, "Emergency Procedures and Training for Station Blackout Events," issued on February 25, 1981 requires licensees to consider station blackout consequences. This generic letter states that, qualitatively, there appears to be sufficient time available following a station blackout event to restore AC power, however, licensees may need to improve operator preparation and training to act accordingly during a station blackout event.

Southern California Edison responded to Generic Letter 81-04 by letter dated July 21, 1981, stating that procedures to cope with a station blackout would be developed. Consequently, procedure #SO1-1.0-60 entitled "Loss of Power" was developed. One assumption in the development of this procedure was that a two hour period would be available to restore AC power. This assumption is validated by calculation #DC-1063, "Component Cooling Water Thermal Capacity," demonstrating that two hours is a conservative assumption. This calculation addresses the time available in a station blackout condition before the inlet water temperature to the component cooling water heat exchanger reaches excessively high levels where the probability of reactor coolant pump seal degradation increases.

Following issuance of 10 CFR 50.63, Southern California Edison developed a new station blackout evaluation which was submitted on April 17, 1989 and supplemented on May 1, 1990. This evaluation is currently under review by the Nuclear Regulatory Commission (NRC). Upon NRC approval of our current station blackout evaluation it is possible that Technical Specifications specifically addressing Station Blackout situations will need to be developed. Until that time it is appropriate to remove the station blackout considerations from Technical Specification 3.1.4. We will continue to operate the plant in compliance with our station blackout evaluations and procedures, but station blackout considerations will not be addressed in this technical specification basis. Therefore, PCN-226 proposes to revise the Basis of Technical Specification 3.1.4 such that this specification will address normal operation in Modes 1, 2, 3, and 4 only and will not address station blackout considerations by deleting the discussion of a loss of power from the Basis.

Letdown System and Reactor Coolant Pump Seal Flow

PCN-226 also proposes to add a paragraph to the Basis of Technical Specification 3.1.4 clarifying that normal flow from both the letdown system and the #1 seals on the reactor coolant pumps is not included in the 6 gpm coolant leakage limitation. These flows are directed to closed systems and returned to the reactor coolant inventory during normal operation. Therefore, these flows are not lost from the reactor coolant inventory and are not considered as leakage during normal operation. The 6 gpm leakage limitation

addresses leakage flows which are lost from the reactor coolant inventory and must be replaced via the makeup system.

SIGNIFICANT HAZARD CONSIDERATION ANALYSIS

As required by 10 CFR 50.91(a)(1), this analysis is provided to demonstrate that Proposed Change No. 226, which revises Technical Specification 3.1.4, 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 revision to the Basis of Technical Specification 3.1.4 was analyzed using the following standards and 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.

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

   The proposed change does not change either the operation of the plant or the design basis of the plant. Only the Basis of Technical Specification 3.1.4 is changed. The basis for the 6 gpm leakage limitation is the long term capacity of certain parts of the radwaste system. If the leakage is greater than 6 gpm for an extended period of time, it would be possible to exceed the capacity of certain parts of the radwaste system. This proposed change revises the existing basis to state this more clearly. The current basis states that a 6 gpm leakage value is based on the capability to remove decay heat with a 6 gpm leakage during a loss of power event lasting up to 12 hours. This statement was the basis for the reactor coolant leakage rate of 10 gpm in the late 1960's before it was reduced to 6 gpm in 1972 in order not to exceed the limitations of the radwaste system. The discussion of the ability to remove decay heat is no longer current with plant commitments and has been removed from the Basis by this proposed change. Because this change updates only the basis section of Technical Specification 3.1.4, operation of the facility in accordance with this change will 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 proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   This proposed change does not change the operation of the plant or the design basis of the plant. This change only updates the basis of Technical Specification 3.1.4 to clarify the basis for the 6 gpm leakage limitation as discussed above in Question 1. Therefore, operation of the facility in accordance with this proposed 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 margin of safety?

   Response: No

   This change only updates the Basis of Technical Specification 3.1.4 and does not affect the operation of the facility or involve a reduction in margin of safety. The basis for the 6 gpm reactor coolant leakage limitation is clarified to be based on equipment limitations. The discussion of the ability to remove decay heat with a leakage rate of 6 gpm during a loss of power event lasting up to 12 hours is no longer current and has been removed. Because the operation of the facility is not altered by this proposed change, there will be no impact on a margin of saftey.

SAFETY AND SIGNIFICANT HAZARD CONSIDERATION DETERMINATION

Based on the preceding analysis, it is concluded: (1) Proposed Change No. 226 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.

PCN-226.MTG2