April 26, 1999

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

SUBJECT: SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 2 - ISSUANCE OF AMENDMENT RE: SHUTDOWN COOLING CHECK VALVES REPAIR (TAC NO. MA5298)

Dear Mr. Ray:

The Commission has issued the enclosed Amendment No. 152 to Facility Operating License No. NPF-10 for San Onofre Nuclear Generating Station, Unit No. 2. The amendment is in response to your application dated April 24, 1999, to permit repair of certain check valves in the shutdown cooling (SDC) system.

This one-time temporary amendment allows the facility to be outside the licensing basis regarding remote shutdown capability of the SDC system as described in the Updated Safety Analysis Report, Section 5.4.7.1.2, during the period of the repair. The amendment is effective for 7 days from the date of issuance or until the repair of the check valves is completed, whichever occurs first. This amendment does not involve any changes to the plant's technical specifications.

A copy of our related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,
ORIG. SIGNED BY
L. Raghavan, Senior Project Manager
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-361
Enclosure: 1. Amendment No. 152 to License No. NPF-10
2. Safety Evaluation

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* See previous concurrence

OFC	PDIV-2/PM	PDIV-2/LA	SRXB*	OGC* NLO	PDIV-2/SC
NAME	LRaghavan	CJamerson	JWermiel	APH	SDembek
DATE	4/26/99	4/26/99	4/26/99	4/26/99	4/26/99

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-w/comments as noted

OFC	PDIV-2/PM	PDIV-2/LA	SRXB	OGC	PDIV-2/SC	PERB	SPSB
NAME	LRaghavan	CJamerson	J. WERMIEL		SDembek		
DATE	4/ /99	4/ /99	4/ 26/99	4/ 99	4/ /99	4/ /99	4/24/99

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

Mr. Harold B. Ray
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NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

April 26, 1999

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L. Raghavan

L. Raghavan, Senior Project Manager
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-361

Enclosure: 1. Amendment No. 152 to License No. NPF-10
2. Safety Evaluation

cc w/encls: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SOUTHERN CALIFORNIA EDISON COMPANY

SAN DIEGO GAS AND ELECTRIC COMPANY

THE CITY OF RIVERSIDE, CALIFORNIA

THE CITY OF ANAHEIM, CALIFORNIA

DOCKET NO. 50-361

SAN ONOFRE NUCLEAR GENERATING STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 152
License No. NPF-10

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Southern California Edison Company, et al. (SCE or the licensee) dated April 24, 1999, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, License No. NPF-10 is amended to approve changes to the Updated Final Safety Analysis Report Section 5.4.7.1.2 as indicated in the attachment to this license amendment.

3. This license amendment is effective as of the date of its issuance and is applicable for 7 days from the date of issuance or until the check valve repair is completed, whichever occurs first.

FOR THE NUCLEAR REGULATORY COMMISSION

Stephen Dembek, Chief, Section 2
Project Directorate IV and Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the UFSAR Section 5.4.7.1.2

Date of Issuance: April 26, 1999

periods from normal operating temperature to the refueling temperature. The initial phase of the cooldown is accomplished by heat rejection from the steam generators to the condenser or atmosphere. After the reactor coolant temperature and pressure have been reduced to approximately 350°F and 376 lb/in.^{2}a, the SCS is put into operation.

In their shutdown cooling function, the LPSI pumps take suction from one of the two RCS hot legs. Heat is removed by circulating this flow through the shutdown cooling heat exchangers (SCHXs). The cooled flow returns to the RCS through four LPSI headers connected to the cold legs. Plant cool-down rate is controlled by flow control valves which permit proportioning the amount of shutdown cooling flow passing through the heat exchangers and heat exchanger bypass line. The SCS reduces reactor coolant temperature to refueling temperature and maintains this temperature during refueling operations.

The SCHXs are also used during the recirculation mode following a loss-of-coolant incident for containment spray purposes, as discussed in subsections 6.2.2 and 6.5.2.

The SCS is used in conjunction with steam generator atmospheric dump and emergency feedwater to cool down and depressurize the RCS following a small break LOCA (see section 6.3).

No components of the SCS for Unit 2 are shared by Unit 3.

5.4.7.1.2 Design Criteria

In addition to the functional requirements of paragraph 5.4.7.1.1, the following design requirements form the design basis for the SCS:

A. The functional requirements defined in paragraph 5.4.7.1.1 must be met assuming the failure of a single active component.

B. No single active failure will allow overpressurization of the SCS. Positive isolation from the RCS is provided whenever the RCS is above the shutdown cooling initiation pressure of 376 lb/in.^{2}a (pressurizer). Isolation valves with appropriate interlocks are provided on the SCS suction line for this purpose. The valves and interlocks are discussed in paragraph 5.4.7.2.2.

Overpressure protection from the safety injection tanks is discussed in paragraph 6.3.2.2.1.

The SCS is provided with appropriate relief valves for overpressure protection. Design basis for pressure relief capacity is discussed in paragraph 5.4.7.2.2.

~~C. No single active failure prevents at least one train of the SCS from being aligned and operated from the control room either during a normal plant cooldown or following an accident. A failure modes and effects analysis of the SCS is provided in table 5.4-7.~~ *

* NOTE: APPLICABLE FOR 7 DAYS FROM THE DATE OF ISSUANCE OF THE AMENDMENT OR UNTIL THE CHECK VALVES REPAIR IS COMPLETED, WHICHEVER OCCURS FIRST.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 152 TO FACILITY OPERATING LICENSE NO. NPF-10

SOUTHERN CALIFORNIA EDISON COMPANY

SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 2

1.0 INTRODUCTION

By letter dated April 24, 1999, Southern California Edison Company, (SCE or the licensee), requested an emergency amendment to the Updated Final Safety Analysis Report (UFSAR) "Design Criteria" for the San Onofre Nuclear Generating Station (SONGS), Unit 2, relating to the Shutdown Cooling (SDC) system. The requested change would facilitate repair of certain check valves in the SDC system and allow operation of Unit 2 without the ability for achieving remote shutdown capability from the control room during the period of the repair. This one time and temporary amendment is needed until the check valves repair is completed. The licensee expects to complete the necessary repair by April 30, 1999.

2.0 BACKGROUND

The SDC system is a subsystem of the Low Pressure Safety Injection (LPSI) system and is used to remove heat from the reactor coolant system (RCS) during post-shutdown periods. The RCS heat is rejected in two steps. During the initial phase of normal cooldown, the heat is rejected from the steam generators to the condenser or atmosphere. After the reactor coolant temperature has been reduced to approximately 350°F, the SDC is put into operation. In the second step of the shutdown cooling function, the LPSI pumps take suction from one of the two RCS hot legs. Heat is removed by circulating this water through the shutdown cooling heat exchangers (SCHXs). The cooled water returns to the RCS through four LPSI headers connected to the cold legs. During normal operation, the SDC is aligned for emergency core cooling system (ECCS) and containment cooling system functions.

The SDC suction line connects the RCS hot leg to the two LPSI pumps. There are two manual isolation valves (MU015 and MU018; one for each train) between the SDC system suction header and each LPSI pump. Originally, these isolation valves remained normally closed to preclude the possibility of both LPSI pumps drawing suction from one source for certain single failures and resulting in both LPSI pumps inoperable due to net positive suction head (NPSH) problems. This design also require an operator to manually open the valves to initiate SDC.

In the early 1980s, in response to the Branch Technical Position RSB 5-1, "Design Requirements for the Residual Heat Removal System," the licensee modified the SDC system to include two swing check valves MU200 and MU202 (one check valve upstream of each isolation valve). The check valves provided the isolation function such that the manual isolation valves, MU015 and MU018, can remain open and allow iinitiation of SDC remotely from the control room. During normal operation, the check valves are normally closed. Their safety function is to remain closed during the injection and recirculation phases of ECCS operation, and to open to allow remote initiation of shutdown cooling. This design change was made to comply with Branch Technical Position RSB 5-1.

During the current Unit 3 inspection of these check valves, the licensee discovered that the disc nut was missing but the nut staking pin was in place. As a result, they radiographed the Unit 2 check valves and discovered the valves were similarly degraded. The licensee performed an operability assessment, and determined the Unit 2 check valves to be operable but degraded. The licensee plans to restore these valves to a condition equivalent to the original design as soon as possible. The licensee has evaluated various repair options, including the insights provided by the plant's probabilistic safety assessment, and concluded repairing these valves in Mode 1 operation is the most prudent course of action. The licensee estimates repairs will take between 30 and 40 hours.

3.0 EVALUATION

The licensee has evaluated different alternatives in determining the safest course of action, including repairing the valves in hot shutdown, Mode 4, and not repairing the valves until the next scheduled refueling. The licensee determined that repairing the valves while in power operation, Mode 1, was the most prudent course of action. The licensee recognized that this would put the unit outside of its licensing basis. The staff finds the licensee conclusions reasonable. The repair activities have no effect on the emergency core cooling system injection and recirculation functions. Additionally, the licensee stated that for events that require transition into SDC (loss of coolant accidents (LOCAs) smaller than 0.01 ft^2) the areas required to restore shutdown cooling will remain habitable.

The licensee has outlined the regular and backup equipment that will help mitigate any potential events. These focus on abundant water supplies of condensate to keep the plant in hot shutdown while the valve repairs can be completed. There is 24 hours (over 500,000 gallons) of normal condensate storage when one of the compensatory measures discussed below is credited. There are an additional 535,000 gallons of non-seismic demineralized water available. A cross tie from the Unit 3 condensate is also available to Unit 2 (Unit 3 is shutdown and the condensate is not needed), and fire water is available if necessary. When restoring the valves to service, the valve repairs are only needed to restore pressure boundary integrity if problems are experienced during the repairs. The SDC function can be reestablished once the pressure boundary is restored.

The licensee has put the following compensatory measures in place to both reduce the likelihood of needing SDC and increase the time before SDC is needed;

1) The repair plan allows "backing out" of the repairs and restoring SDC within approximately 24 hours if determined necessary by plant operators or management. The contingency plan includes provisions for restoring the SDC path with MU200 and MU202 inoperable by restoring the valves' pressure boundary integrity.

2) A temporary instruction associated with Operating Instruction SO23-3-2.7.2, "Safety Injection Removal/Return To Service," will be in place to provide guidance to operators to perform the required actions to restore SDC if required.

3) Work activities will be controlled to minimize high risk activities during the repair period.

4) The available volume in condensate storage tank (CST) T-120 will be increased by isolating (or staging an operator to isolate) non-seismically qualified connections to the when the valve repair is initiated. This will preclude the loss of about 80,000 gallons of water assumed to occur following postulated seismic events. This volume of water is sufficient to steam at SDC entry conditions for about 8 hours.

The regulatory dose requirements continue to be met. The staff reviewed the licensee's evaluation of the increase in radiological consequences of a design basis accident occurring during the period that SDC would be unavailable. The licensee has estimated that the repairs could be completed in 30 to 40 hours, and that SDC function could be restored within 24 hours should it become necessary. The staff agrees with the licensee's evaluation that there would be no increase in the previously postulated doses for the design basis accident (DBA) LOCA since SDC is not required for long-term cooling in that event. The staff also agrees with the licensee's evaluation that there would be no increase in the previously postulated exclusion area boundary (EAB) doses since the SDC function would not be required prior to the end of the specified 2-hour exposure period for the EAB. For events requiring long-term cooling, e.g., steam generator tube rupture, there could be an increase in postulated doses for the low population zone (LPZ) and for the control room. The licensee's evaluation indicates that the increased doses would continue to meet radiological criteria in 10 CFR 100.11 and GDC-19, Appendix A, 10 CFR Part 50. The staff finds the radiological consequences of the licensee's proposal to be acceptable given (1) the licensee's evaluation of the postulated increase in consequences, (2) the temporary exigent condition. The staff's acceptance is limited to this temporary condition.

The licensee has proposed to repair the SDC valves while continuing to operate the plant. The licensee has concluded that this is the most prudent course of action. Because the licensee has demonstrated that normal and alternate equipment are available, including alternate sources of water, to mitigate any events, put compensatory measures in place, including contingency plans, and the licensee determined the dose consequences are acceptable. The staff finds the proposed one-time evolution acceptable.

4.0 EMERGENCY CIRCUMSTANCES

In its April 24, 1999 letter, the licensee requested that this amendment be treated as an emergency amendment. In accordance with 10 CFR 50.91(a)(5), the licensee provided information regarding why this emergency situation occurred and how it could not be avoided.

The staff concludes that an emergency condition exists in that failure to act in a timely way would result in a shutdown of SONGS Unit 2. In addition, the staff has assessed the licensee's reasons for failing to file an application sufficiently in advance to preclude an emergency, and concludes that the licensee promptly performed the inspection and identified the deficiency, promptly notified the staff of the deficiency, and promptly proposed this amendment to remedy the situation. Thus, the staff concludes that the licensee has not abused the emergency provisions by failing to make timely application for the amendment. Thus, the conditions needed to satisfy 10 CFR 50.91(a)(5) exist, and the amendment is being processed on an emergency basis.

5.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The Commission's regulations in 10 CFR 50.92(c) state that the Commission may make a final determination that a license amendment involves no significant hazards consideration if operation of the facility in accordance with the amendment would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or,

(2) Create the possibility of a new or different kind of accident from any previously evaluated; or,

(3) Involve a significant reduction in a margin of safety.

The following analysis was provided by the licensee in its April 24, 1999 letter.

(1) Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated.

No.

Initiating events for accidents and transients evaluated in the Updated Final Safety Analysis Report (UFSAR) are listed in Chapter 15, Table 15.0-2, Initiating Events. Except for a Shutdown Cooling (SDC) line break in Mode 4, both SDC and Low Pressure Safety Injection (LPSI) systems are accident mitigators and not accident initiators. The proposed activity will not change the probability of occurrence of any of the listed initiating events. The SDC piping involved in the proposed activity is isolated from the piping associated with initiating events. The proposed activity will preclude a SDC line break because SDC will not be initiated with MU015 and MU018 closed.

Therefore, this amendment request does not significantly increase the probability of an accident previously evaluated.

Evaluations of accidents are described in UFSAR Chapter 15, Accident Analysis. LPSI and SDC are used to mitigate the consequences of accidents and transients evaluated in the UFSAR. The proposed activity does not impact the operability of LPSI for safety injection. Restoration of SDC system operability prior to needing SDC for Reactor Coolant System (RCS)/decay heat removal assures that this activity will not adversely affect SDC's ability to provide long term core cooling.

For accident evaluations considering inoperable SDC, the most limiting accidents are Loss of Coolant Accidents (LOCAs). UFSAR Figure 6.3-24 shows the spectrum of LOCAs evaluated in the UFSAR. For certain size LOCAs (breaks larger than 0.01 ft^2), SDC is not required for long term cooling and accident mitigation and thus, this repair does not affect does consequences. Long term cooling is provided by simultaneous hot leg/cold leg High Pressure Safety Injection (HPSI).

For very small break LOCAs (0.01 ft^2 or smaller), SDC is required for long term cooling. The major assumptions used in performing the long term cooling analysis are listed in UFSAR Section 6.3.3.4.2. The proposed activity does not change any of those assumptions. However, the analysis credited in the UFSAR only takes credit for the volume in T-121 and does not take credit for T-120 inventory. The proposed activity does take credit for T-120's water inventory (including compensatory actions to increase its useful volume above the Technical Specification minimum limit) to extend the water inventory available to reach SDC entry conditions and to maintain that condition prior to SDC initiation. The additional time provides reasonable assurance that SDC can be returned to operable prior to the time it is required for accident mitigation.

The plant can be maintained on auxiliary feedwater using T-120 and T-121 until SDC has been returned to service. Reactor coolant inventory can be maintained using HPSI, either from the Refueling Water Storage Tank (RWST) or recirculation.

As shown in UFSAR Figure 15.6-126, for LOCAs 0.01 ft^2 or smaller, core uncovery (and fuel damage) is not postulated. Therefore, the areas required to restore SDC operability, and locally operate the Atmospheric Dump Valves (ADVs) (required to control the steam generators on auxiliary feedwater should offsite power and normal plant support systems be unavailable) will remain habitable.

The ability to establish SDC following a Steam Generator Tube Rupture (SGTR) is the same as that described for the 0.01 ft^2 or smaller LOCAs.

Various UFSAR Chapter 15 non-LOCA transients, including seismic events, are evaluated for the assumed scenario of either a loss of condenser vacuum or a loss of normal AC power, either of which requires use of one or both Atmospheric Dump Valves (ADVs) to effect plant cooldown prior to placing SDC into service. As long as the ADV from the affected steam generator is open, secondary side steaming provides an activity release path to the environment. In accordance with the UFSAR, these non-LOCA transient event scenarios terminate several hours into the event with the initiation of SDC and the coincident Operator closure of the ADVs to isolate the activity release path.

Should SDC be unavailable, it will be necessary for the Operators to continue use of the ADVs to effect plant cooldown. Consequently additional radioactivity may be released to the environment thereby increasing offsite and control room operator event duration dose exposures. However, the Exclusion Area Boundary (EAB) doses which are evaluated for only the first two hours of a transient are not affected by initiating SDC later in the events.

When explicitly evaluated, the Low Population [Zone] (LPZ) and Control Room doses are evaluated for the event duration. These doses will increase as a consequence of the increased event duration. However, the increased doses will be acceptable (i.e., below Standard Review Plan, 10CFR100.11 and General Design Criterion [GDC] 19 dose acceptance criteria) for the following reasons:

1) LPZ doses are typically one or more orders of magnitude less than EAB doses due to the additional atmospheric dispersion between the activity release points and the dose receptor. Consequently, increased activity releases due to a delay in SDC initiation will still yield dose consequences that are significantly less than EAB dose consequences. For example, in the event of a SGTR with a pre-existing iodine spike, the LPZ thyroid dose assuming SDC is placed into service at 3.12 hours into the event is 0.081 rem, while the 2-hour EAB dose is 2.8 rem. Even with a hypothetical factor of ten increase (> 31.2 hours prior to SDC), the LPZ dose will still be less than that evaluated at the EAB.

2) A relatively large portion of the control room thyroid dose is attributed to activity entering the control room prior to the initiation of the Control Room Emergency Air Cleanup System (CREACUS). The proposed activity does not affect initiating CREACUS.

   As such, increased activity releases due to delays in SDC initiation will not yield dose consequences that are significantly greater than currently calculated on a per hour basis. For example, for SGTR with a pre-existing iodine spike, the Control Room thyroid dose assuming SDC is placed into service at 3.12 hours into the event is 0.67 rem while the 3-minute control room dose is 0.11 rem. Even with a hypothetical factor of ten increase (> 31.2 hours prior to (SDC) of the additional 0.56 rem dose occurring after 3 minutes, the control room dose would increase to 5.7 rem, which is significantly less than the 30 rem GDC 19 dose criterion.

3) In the case of SGTR concurrent with the primary side temperature decreasing below 350° F, the primary to secondary side pressure gradient forcing additional radioactivity across the Technical Specification leaking steam generator tubes and into the secondary side will be reduced. As such the rate of radioactivity release to the environment is greatly reduced.

   The above doses are based on design RCS activities. The Technical Specification coolant activity limits are lower, and would result in lower doses. The actual RCS activity at this time is less than the Technical Specification limit, providing substantial margin to the calculated doses.

   In the case of a seismic event, SDC will be able to perform its decay heat removal function discussed in UFSAR section 5.4.7.1.2, based on the Technical Specification volume in T-120 and T-121 is sufficient to allow steaming for 24 hours, compensatory measures which will increase the available CST volume to allow an additional 8 hours of steaming, and the repair plan includes provisions to back out of the repair and restore SDC within approximately 24 hours.

Therefore, this amendment request does not significantly increase the probability or consequences of any accident previously evaluated.

2) Does the amendment request create the possibility of a new or different kind of accident from any accident previously evaluated?

No.

UFSAR Section 15.0.1, Identification of Causes and Frequency Classification, describes how incidents are considered in the UFSAR. The initiating events are each placed in one of the categories of process variable perturbations listed in Table 15.0-1. The initiating events for which analyses are presented are listed in Table 15.0-2 along with their respective section designations. Certain initiating events which are suggested for consideration are not explicitly analyzed. These initiating events, along with the reasons for omission of their analyses, are provided in the appropriate paragraphs in Chapter 15.

The components involved in the proposed activity are passive in nature, and do not interact with other Systems, Structures or Components (SSC) in such a way as to cause any of the initiating event categories listed in Table 15.0-1.

With isolation valves MU015 and MU018 open, the possible events are bounded by existing analyses. With the isolation valves closed, the SDC system becomes inoperable, but this does not create the possibility of a new or difference kind of accident.

Therefore, this amendment request does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3) Does this amendment request involve a significant reduction in a margin of safety?

No.

SCE completed a probabilistic risk assessment (PRA) of the proposed repair plan. The assessment included all events requiring the shutdown cooling function to mitigate core damage and large early release: small break LOCAs, SGTRs, and seismic events. The increase in core damage and large early release risk are estimated to be 7.1E-6 and 1.7E-7, respectively.

The dominant contributor to core damage risk during the repair is from a seismic event of a magnitude greater than 0.3g pga. A seismic event of this magnitude or greater is assumed to fail the condensate makeup function to the condensate storage tanks. In this case, the condensate storage tank inventory limits the time available for restoring shutdown cooling to service. The compensatory measures such as use of firewater to replenish the condensate tanks are not credited in the risk assessment. The dominant contributor to the large early release risk during the repair is from a steam generator tube rupture event assuming unsuccessful depressurization of the reactor coolant system prior to refueling water storage tank inventory depletion.

Other repair options, such as performing the repair in Mode 4 (decay heat removal via steaming at reduced reactor coolant system pressure) and in a defueled condition during the next refueling outage, were considered. The risk of repairing the valves in Mode 4 is on the same order of magnitude as repair in Mode 1. Long term plant operation without repairing the valves until the next refueling outage was considered undesirable due to the degraded condition of the valves and the desire to do the repair in a planned and controlled manner, rather than attempt recovery actions in the unlikely event of an event requiring SDC.

Based upon the PRA results and planned contingency measures (not considered explicitly in the PRA), the overall risk of the repair plan is small. Based upon Regulatory Guide 1.174, these increases in risk are also characterized as "small."

For very small break LOCAs (0.01 ft^2 or smaller), SDC is required for long term cooling. The major assumptions used in performing the long term cooling plan analysis are listed in UFSAR Section 6.3.3.4.2. The proposed activity does not change any of those assumptions. However, the analysis credited in the UFSAR only takes credit for the volume in the condensate storage tank (CST) T-121 and does not take credit for T-120 inventory. The proposed activity does take credit for T-120's water inventory (including compensatory actions to increase its useful volume above the Technical Specification minimum limit) to extend the water inventory available to reach SDC entry conditions and to maintain that condition prior to SDC initiation. The additional time provides reasonable assurance that SDC can returned to operable prior to the time it is required for accident mitigation.

The ability to establish SDC following a Steam Generator Tube Rupture (SGTR) is the same as that described for the 0.01 ft^2 or smaller LOCAs.

Should SDC be unavailable, it will be necessary for the Operators to continue use of the ADVs to effect plant cooldown. Consequently additional radioactivity may be released to the environment thereby increasing offsite and control room operator event duration dose exposures. However, the Exclusion Area Boundary (EAB) doses which are evaluated for only the first two hours of a transient are not affected by initiating SDC later in the events.

When explicitly evaluated, the Low Population [Zone] (LPZ) and Control Room doses are evaluated for the event duration. These doses will increase as a consequence of the increased event duration. However, the increased doses will be acceptable (i.e., below SRP, 10CFR100.11 and General Design Criterion [GDC] 19 dose acceptance criteria).

The calculated doses are based on design RCS activities. The Technical Specification coolant activity limits are lower, and would result in lower doses. The actual RCS activity at this time is less than the Technical Specification limit, providing substantial margin to the calculated doses.

In the case of a seismic event, SDC will be able to perform its decay heat removal function discussed in UFSAR section 5.4.7.1.2, based on the Technical Specification volume in T-120 and T-121 is sufficient to allow steaming for 24 hours, compensatory measures which will increase the available CST volume to allow an additional 8 hours of steaming, and the repair plan includes provisions to back out of the repair and restore SDC within approximately 24 hours.

Therefore, this amendment request does not involve a significant reduction in a margin of safety.

Based on the negative responses to these three Commission criteria, SCE concludes that the proposed amendment involves no significant hazards consideration.

6.0 STATE CONSULTATION

In accordance with the Commission's regulations, the California State official was notified of the proposed issuance of the amendment. The State official had no comments.

7.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has made a final no significant hazards finding with respect to this amendment. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

8.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that (1) the amendment does not: (a) involve a significant increase in the probability or consequences of an accident previously evaluated; or, (b) create the possibility of a new or different kind of accident from any previously evaluated; or, (c) involve a significant reduction in a margin of safety and therefore, the amendment does not involve a significant hazards consideration; (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (3) such activities will be conducted in compliance with the Commission's regulations, and (4) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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Date: April 26, 1999