May 5, 2006

Mr. Richard M. Rosenblum
Senior Vice President and Chief Nuclear Officer
Southern California Edison Company
San Onofre Nuclear Generating Station
P.O. Box 128
San Clemente, CA 92674-0128

SUBJECT: SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 3 - ISSUANCE OF AMENDMENTS RE: REQUEST TO REPAIR SHUTDOWN COOLING LINE VENT LINE (TAC NOS. MD1446)

Dear Mr. Rosenblum:

The Commission has issued the enclosed Amendment No. 194 to Facility Operating License No. NPF-15 for San Onofre Nuclear Generating Station, Unit No. 3. The amendment is in response to your application dated May 4, 2006, to permit repair of vent line leakage 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 Final 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 vent line leakage 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,

**/RA/**

N. Kalyanam, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-362

Enclosures: 1. Amendment No. 194 to NPF-15
2. Safety Evaluation

cc w/encls: See next page

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SOUTHERN CALIFORNIA EDISON COMPANY

SAN DIEGO GAS AND ELECTRIC COMPANY

THE CITY OF RIVERSIDE, CALIFORNIA

THE CITY OF ANAHEIM, CALIFORNIA

DOCKET NO. 50-362

SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 194
License No. NPF-15

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 May 4, 2006, 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, by Amendment No. 194, License No. NPF-15 is amended to authorize 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 shutdown cooling line vent line repair is completed, whichever occurs first.

FOR THE NUCLEAR REGULATORY COMMISSION

**/RA/**

David Terao, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the UFSAR, Section 5.4.7.1.2

Date of Issuance: May 5, 2006

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 194

TO FACILITY OPERATING LICENSE NO. NPF-15

SOUTHERN CALIFORNIA EDISON COMPANY

SAN DIEGO GAS AND ELECTRIC COMPANY

THE CITY OF RIVERSIDE, CALIFORNIA

THE CITY OF ANAHEIM, CALIFORNIA

SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 3

DOCKET NO. 50-362

## 1.0 INTRODUCTION

By letter dated May 4, 2006, Southern California Edison Company (SCE or the licensee) requested an emergency amendment to the Updated Final Safety Analysis Report (UFSAR), Section 5.4.7.1.2, "Design Criteria," for the San Onofre Nuclear Generating Station (SONGS), Unit 3, relating to the Shutdown Cooling (SDC) system. The licensee also provided clarifications in a telephone call on May 5, 2006. The requested change would facilitate repair of a certain vent line in the SDC system and allow operation of Unit 3 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 repair on the vent line is completed. The licensee expects to complete the necessary repair by May 7, 2006.

## 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. 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.

During the current Unit 3 forced outage, evidence of leakage was discovered on a vent line off of the SDC discharge line to the Unit 3 RCS loop 2B. On May 3, 2006, this leak was determined to be due to a flaw in the SDC vent line. As a result, the SDC and LPSI systems have been determined to be inoperable. In order to repair this leak, a portion of the line must be drained. This will cause both trains of the SDC and LPSI systems to be unavailable for the duration of the repair. This is outside the licensing basis for SONGS, Unit 3.

The proposed activity is to restore the Injection Vent Valve S31204MR454 to a condition equivalent to the original design Unit 3 LPSI while in Mode 4. To perform the repairs, SCE will isolate the LPSI discharge header from the affected pipe, and will also isolate the downstream line using the LPSI Injection Isolation Valve 3HV9331.

## 3.0 TECHNICAL EVALUATION

### 3.1 Mitigation of Potential Events

The licensee has outlined the regular and backup equipment that will help mitigate any potential events. SONGS, Unit 3 is currently at the SDC system entry temperature in Mode 4 relying on two RCS loops for heat removal in accordance with Limiting Condition of Operation (LCO) 3.4.6. SONGS, Unit 3 has been shut down since March 30, 2006, for maintenance, and the decay heat has declined to below approximately 1.70E+07 British thermal units per hour (BTU/hr). The proposed activity temporarily removes SDC and LPSI from availability for service. LPSI is not required in Modes 4 or 5. With the SDC system unavailable in Mode 4, heat from the RCS is removed by the steam generators (SGs) and auxiliary feedwater (AFW) System and the Condensate Storage Tank (CST). A minimum of 504,000 gallons of water is maintained available in the CST as required by Technical Specification 3.7.6. The CST inventory is sufficient to remove decay heat for greater than 3 days with an operating reactor coolant pump (RCP) in each loop, and greater than 10 days with RCPs secured. The CST inventory is sufficient to provide continued cooling without additional makeup during any postulated emergency conditions with offsite power unavailable. Adequate on-site and off-site water makeup sources are available to replenish CST inventory during that timeframe.

The proposed activity does not adversely impact any other system required or credited to prevent or mitigate accidents or transients that may be postulated to occur during Mode 4 operation.

### 3.2 Compensatory Measures

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. The Nuclear Regulatory Commission (NRC) staff finds the compensatory measures reasonable and adequate.

1. The AFW system consists of two 100-percent-capacity, motor-driven AFW pumps and one 100-percent-capacity, steam-driven AFW pump. The two motor-driven pumps are fully OPERABLE. No maintenance activities will be performed on the motor-driven AFW system. Access to the AFW pump room will be restricted. (There is not currently enough steam to drive the steam-driven AFW pump).

2. No maintenance activities will be performed on the condensate pumps. The available volume of the condensate storage tanks will be maximized.

3. Alternate RCS injection capability will be maintained through the high-pressure safety injection system, which is fully OPERABLE.

4. No maintenance will be performed on secondary plant equipment relied upon for the condenser vacuum, including the condensate system which is an alternate source of makeup to the SGs.

5. No discretionary maintenance will be allowed on the on-site electrical distribution system or the Emergency Diesel Generators (EDGs) for the duration of the repair activities. The on-site electrical distribution system is currently OPERABLE, including the two EDGs on both Units 2 and 3.

6. Control Element Assemblies will be maintained fully inserted.

7. RCS Boron Concentation will be maintained at current value, which is approximately 1600 parts per million.

8. Switchyard access will be restricted.

## 3.3 Probabilistic Evaluation

### 3.3.1 Risk Assessment Evaluation

In evaluating the risk information submitted by the licensee, the NRC staff followed the three-tiered approach documented in Regulatory Guide (RG) 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications." Although RG 1.177 was developed for evaluating proposed changes to the Technical Specifications, it provides guidance on determining the acceptability of permitting equipment to remain out of service longer than the current licensing basis would permit.

Under the first tier, the staff determines if the proposed change is consistent with the NRC's Safety Goal Policy Statement, as documented in RG 1.174 for adequacy of plant protection from potential risk. Specifically, the first tier objective is to ensure that the plant risk does not increase unacceptably during the period the equipment is taken out of service.

The second tier addresses the need to preclude potentially high-risk plant configurations that could result if additional equipment, not associated with the proposed change, is taken out of service during the proposed 7 day amendment request.

The third tier addresses the establishment of a configuration risk management program for identifying risk-significant configurations resulting from maintenance or other operational activities, and taking appropriate compensatory measures to avoid such configurations.

### 3.3.2 Basis and Quality of Risk Assessment

The licensee used its probabilistic risk assessment (PRA) method and appropriate conservative assumptions to assess the risk increase associated with operation at Mode 4 for a period of 7 days without operable trains of shutdown cooling (SDC) and low pressure safety injection (LPSI) systems.

The licensee employed a plant-specific assessment of the proposed configuration during the 7 day amendment request. The corrective maintenance repair work would be performed outside of the containment and the Mode 4 configuration would be very similar to the at-power plant configuration. Consequently, the licensee risk insights in the submittal are based on the Mode 1 plant configuration model. The NRC's Standardized Plant Risk Analysis (SPAR) model at-power was employed, with full credit to the Auxiliary Feedwater System (AFW) to assess the licensee evaluation. The licensee PRA quality and assessment methodology was considered acceptable and meets current industry standards with continuous reviews and upgrading of the PRA model and the configuration risk management tool (i.e., Safety Monitor). Although the submittal did not use the PRA software explicitly nor address sensitivity/uncertainty analyses, the NRC staff concluded that the quality of the risk assessment is consistent with the licensee PRA quantification tools.

The risk consideration included maintaining defense-in-depth and quantifying risk to determine the change in core damage frequency (CDF) and large early release frequency (LERF) as a result of the proposed 7 day amendment request. The NRC staff evaluated the quality of the PRA models, limited to the systems related to the proposed change, major assumptions, and data used in the risk assessment, and found it acceptable for this application. This evaluation compared the applicable findings from the NRC staff's review of the licensee's PRA with the NRC's SPAR model, Version 3.2, employing NRC PRA quantification tool, SAPHIRE Version 7, and NRC Manual Chapter 0609, Appendix H for LERF, as well as findings from similar evaluations of similar plants. The configuration risk model for the SPAR calculation is based on the configuration risk management model (zero maintenance model).

3.3.3 Risk Impact of the Proposed Change (Tier 1)

An acceptable approach to risk-informed decisionmaking is to show that the proposed change to the design basis meets several key principles. One of these principles is to show that the proposed change results in a small, but acceptable, increase in risk in terms of CDF and LERF, and is consistent with the NRC's Safety Goal Policy Statement. Acceptance guidelines for meeting this principle are presented in RG 1.174. The licensee evaluated the risk based on the realistic plant configuration similar to Mode 4 using the at-power configuration with full credit to AFW. Furthermore, both the incremental conditional core damage probability (ICCDP) and the incremental conditional large early release probability (ICLERP) were assessed, since the containment will be intact during the corrective maintenance. These quantities are a measure of the increase in probability of core damage and large early release, respectively, during a single outage that would last for the entire duration allowed by the proposed change. The acceptance guideline used for the license amendment is provided in RG 1.177 as 5.0E-7 and 5.0E-8 for ICCDP and ICLERP, respectively. However, the RG 1.177 guideline is for permanent technical specification changes and the reviewer has considered additional credits for the proposed one-time amendment request within the bound of adequate protection under the guideline in RG 1.174. Based on the one-time amendment request of 7 days, the incremental changes are summarized in the following table:

| | Baseline CDF | Incremental Change in CCDP | Baseline LERF | Incremental Change in ICLERP |
|---|---|---|---|---|
| Prior to Extension | 1.204E-05/yr | | 1.204E-06/yr | |

| | | | | | |
|---|---|---|---|---|---|
| Increase because of 7 day amendment request (Licensee Results) | | | 3.10E-07 | | 3.10E-08 |
| New Baseline CDF | | 1.23E-05/yr | | 1.23E-06/yr | |
| Increase because of 7 day amendment request | A. using NRC SPAR 3.2 Model | | 1.10E-06 | | 1.10E-7 |
| | B. Compensatory Measures* | | >1.0E-6 ( AFW) | | >1.0E-7 |
| Acceptance Guidelines** | | | 5E-7 | | 5E-8 |

* Quantifiable compensatory measures provided by the licensee
** Criteria for permanent change, flexibility considered for one-time changes.

Based on the staff's analysis using the SPAR model, the configuration risk increase with both shutdown LP trains (fail-to-start and fail-to-run) is 1.10E-6 in ICCDP, about 2 (two) times larger than the threshold value of 5.0E-07, the acceptance guideline in RG 1.177 for permanent Technical Specification changes. The LERF is calculated employing NRC Inspection Manual Chapter 0609, SDP Appendix H with the CDF-LERF conversion factor of 0.1. This conversion multiplier is a ratio of LERF-to-CDF to evaluate the LERF value conservatively for those plants without available Level 2 and 3 PRA models. Therefore, the ICLERP number also exceeds the RG 1.177 guideline.

During the amendment request period, the total CDF and LERF have been increased due to the incremental changes in ICCDP and ICLERP, respectively, beyond the RG 1.177 acceptance guidelines, resulting from the one-time 7 day amendment request. The SONGS Unit 3 containment is a large dry design and the design features (containment failure mechanisms) do not typically contribute to large early release (LERF). Similar to other large dry containment buildings, the major contributors to LERF are from containment bypass sequences, which include interfacing systems loss of coolant accidents and steam generator tube rupture events. However, without any bias toward containment bypass sequences, the LERF multiplier (based on CDF) is typically less than 0.1, and this is consistent with the licensee evaluation of the ICLERP.

**The impact of external events due to seismic, flood and high winds are evaluated qualitatively,** and the staff concluded that the incremental risk associated with these external events is negligible for the duration of the requested amendment. Concerning fire related issues, no new fire scenarios are introduced by the LPI unavailability. To increase sensitivity to fire prevention in these areas, the licensee will provide hourly fire watches during the proposed amendment request period.

During the proposed amendment request period, the total CDF and LERF have been increased due to the incremental changes in ICCDP and ICLERP, respectively, resulting from the one-time 7 day amendment request. However, the licensee employed several conservative assumptions with separate compensatory measures during the maintenance activities to reduce the plant risk. The specifics of risk quantification (qualitative and quantitative) of the proposed compensatory measures are documented in the proposed request letter and supplemented in the RAI response letter. Not all of the proposed compensatory measures are credited in the incremental risk figures, and the risk increases under the proposed amendment request are well within the acceptable guidelines for this one-time request.

In conclusion, a one-time 7 day amendment request at Mode 4 to perform appropriate maintenance work would be acceptable.

3.3.4 Avoidance of High Risk Plant Configurations (Tier 2)

The licensee's PRA will identify and estimate major risk contributors of plant configurations, contributing event sequences, and associated cutsets. Potential major risk contributors include plant equipment failures, human errors and common cause failures. Insights from the risk assessment will be used in identifying and monitoring the plant configurations or conditions that may lead to significant risk increases during the proposed amendment request. The NRC staff finds that the proposed precautions, as well as the proposed compensatory measures identified in the licensee's submittal and the clarifications provided in a conference call of May 5, 2..6, are adequate for preventing plant configurations or conditions that may increase risk significantly. In conclusion, there is reasonable assurance that high risk plant configurations will not occur during the proposed 7 day amendment request.

3.3.5 Risk-Informed Configuration Risk Management (Tier 3)

The intent of risk-informed configuration risk management is to ensure that plant safety is maintained and monitored. A formal commitment to maintain a configuration risk management program is necessary on the part of a utility, and is available employing the Safety Monitor. This program can support the licensee's decision-making regarding the appropriate actions to control risk whenever a risk-informed Technical Specification LCO is entered. The staff finds that the licensee has an adequate configuration management program.

3.3.6 Summary

The NRC staff has developed risk insights, associated with conducting the repair to the shutdown cooling system during Mode 4, and qualitatively compared the risk with the total risk of performing the maintenance activities. The staff concludes that the shutdown risk under the proposed configuration is no greater than at-power risk and, thus, the proposed one-time 7 day amendment request with an inoperable LPI is acceptable.

3.4 Radiological Assessment

For those loss-of-coolant accidents (LOCAs) that could fail fuel, the radiological consequences of the design-basis LOCA as evaluated in the UFSAR remain bounding, given the reduced Mode 4 source term and the absence of any new activity release path. All other non-LOCA events would merely release primary or secondary side activity.

The ability to establish SDC following a steam generator tube rupture (SGTR) is the same as that described for the smaller Small Break LOCAs (SBLOCAs) (UFSAR Section 15.6.3.2).

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 alternating current (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. With SDC unavailable, it will be necessary for the Operators to continue to use the ADVs to maintain RCS temperature stable until SDC is restored.

The Exclusion Area Boundary (EAB) doses are evaluated for only the first 2 hours of a transient, and, therefore, the EAB doses are not affected by initiating SDC later in the events.

When explicitly evaluated, the Low Population (LPZ) and Control Room doses are evaluated for the event duration. These doses are not expected to increase as a consequence of the increased event duration for the following reasons:

a) The core source term has significantly decreased due to Unit 3 being shutdown for over 35 days. The I-131 inventory is 4.5 percent of the analysis value; the remainder of the iodine isotopes have decayed to insignificant levels. The Xe-131m inventory is 27 percent of the analysis value, Xe-133 is 1 percent, and Kr-85 is essentially unchanged (due to its long half life). All other noble gases have decayed to insignificant levels.

b) In the absence of fuel failure, the non-LOCA UFSAR radiological dose consequences are based on the Technical Specification coolant activity limits. The actual primary and secondary activities at this time are less than the Technical Specification limit, providing substantial margin to the calculated doses.

c) Pre-accident iodine spiking is not currently present, and accident initiated iodine spiking would be less severe than assumed in a design-basis analysis given the present initial low primary coolant and core iodine activity profiles.

d) 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, any additional activity releases due to a delay in SDC initiation will still yield dose consequences that are significantly less than EAB dose consequences.

e) A relatively large portion of the limiting control room thyroid dose is attributed to iodine 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, and the current core iodine activity profile available for release to the environment is significantly reduced because Unit 3 has been shut down for at least 35 days.

Other Design-Basis Accidents

In Mode 4, the only other UFSAR Chapter 15 accidents of potential concern are uncontrolled control element assembly (CEA) bank withdrawal, inadvertent boron dilution, and inadvertent opening of a pressurizer safety valve.

With all of the full-length CEAs inserted and the shutdown margin at the Technical Specification 3.1.1 limit of 5.15 percent Δk/k, the analysis of the uncontrolled CEA bank withdrawal shows that the core would not reach criticality. The actual shutdown margin of approximately 10 percent Δk/k is significantly higher than the technical specification requirement of 5.15 percent Δk/k. Therefore, this event would have no adverse consequences were it to occur.

The inadvertent boron dilution event is not impacted by the loss of LPSI or SDC, and is, thus, unchanged from the current analysis of record.

The inadvertent opening of a pressurizer safety valve is bounded by the small break LOCA (SBLOCA) analysis, and, thus, the previous SBLOCA discussion applies to this event.

The NRC 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, and (2) the temporary condition. The NRC staff's acceptance is limited to the temporary condition.

## 4.0 STATEMENT OF EMERGENCY CIRCUMSTANCES

In its May 4, 2006, letter, the licensee requested that this amendment be treated on an emergency basis. In accordance with paragraph 50.91(a)(5) of Title 10 of the *Code of Federal Regulations* (10 CFR), the licensee provided information regarding why this emergency situation occurred and how it could not be avoided.

The NRC staff concludes that an emergency condition exists in that failure to act in a timely way would prevent resumption of operation of SONGS, Unit 3. In addition, the NRC 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 during startup, promptly notified the NRC staff of the deficiency, and promptly proposed this amendment to remedy the situation. Thus, the NRC 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 accident previously evaluated; or

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

The following analysis was provided by the licensee in its May 4, 2006 letter.

1) Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated[?]

Response: 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[n] SDC line break because SDC will not be initiated during the proposed maintenance activity.

Southern California Edison (SCE) completed a Probabilistic Risk Assessment (PRA) of the plant configuration during the proposed activity for the requested 7[-]day period. The assessment included all events (internal and external) which could call upon the shutdown cooling safety function or LPSI function to avert core damage and/or large, early release. The increase in core damage and large early release risk are estimated to be 3.1E-7 and 3.1E-8, respectively.

The dominant contributor to core damage and large, early release risk during the repair is from a loss of offsite power (internal or externally induced, i.e., fire or seismic) since emergency condensate would not be available should Auxiliary Feedwater (AFW) fail. The next dominant contributor was the total loss of AFW initiating event. Planned compensatory measures to reduce the risk were credited in the risk assessment where applicable.

Based upon the PRA results and the planned compensatory measures, the overall risk over the requested 7-day period is “very small” as characterized in Regulatory Guide 1.174.

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

LPSI and SDC are used to mitigate the consequences of accidents and transients evaluated in UFSAR Chapter 15.

<u>Loss of Low Pressure Safety Injection Function</u>

With the plant in Mode 4, LPSI is not required to perform an accident mitigation Emergency Core Cooling System (ECCS) function. This is reflected in Technical Specification 3.5.3.

<u>Loss of Shutdown Cooling Function</u>

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 of 0.012 $ft^2$ and larger), SDC is not required for long[-]term cooling and accident mitigation. For breaks of 0.012 $ft^2$ and larger long[-]term cooling is provided by simultaneous hot leg/cold leg High Pressure Safety Injection (HPSI) injection.

For very small break LOCAs (smaller than 0.012 $ft^2$), 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. As shown in UFSAR Section 6.3.3.4.3, the analysis is based on an initial core power of 3458 MWt. As Unit 3 has currently been shutdown for over 35 days, reducing the actual core power to less than 0.2 percent. This lower power level significantly reduces the required makeup water flow rate from that assumed in the LOCA analysis of record. This means that the existing volume in T-121 will take much longer to be depleted. Additionally, 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 to allow the plant to be cooled down to cold shutdown. Reactor coolant inventory can be maintained using HPSI, either from the Refueling Water Storage Tank (RWST) or recirculation.

The LOCAs of the break sizes that require SDC operation do not result in core uncovery, and thus there is no potential for fuel damage. Therefore, the plant 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 for those LOCAs that required SDC operation.

For those LOCAs that could fail fuel, the radiological consequences of the design basis LOCA as evaluated in the UFSAR remain bounding, given the reduced Mode 4 source term and the absence of any new activity release path. All other non-LOCA events would merely release primary or secondary side activity.

The ability to establish SDC following a Steam Generator Tube Rupture (SGTR) is the same as that described for the smaller Small Break LOCAs (SBLOCAs) (UFSAR Section 15.6.3.2).

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 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. With SDC unavailable, it will be necessary for the Operators to continue use of the ADVs to maintain a stable RCS temperature until SDC is restored.

The Exclusion Area Boundary (EAB) doses are evaluated for only the first two hours of a transient, and therefore the EAB doses are not affected by initiating SDC later in the events.

When explicitly evaluated, the Low Population (LPZ) and Control Room doses are evaluated for the event duration. These doses are not expected to increase as a consequence of the increased event duration for the following reasons:

a) The core source term has significantly decreased due to Unit 3 being shutdown for over 35 days. The I-131 inventory is 4.5 percent of the analysis value, the remainder of the iodine isotopes have decayed to insignificant levels. The Xe-131m inventory is 27 percent of the analysis value, Xe-133 is 1 percent, and Kr-85 is essentially unchanged (due to its long half life). All other noble gases have decayed to insignificant levels.

b) In the absence of fuel failure, the non-LOCA UFSAR radiological dose consequences are based on the Technical Specification coolant activity limits. The actual primary and secondary activities at this time are less than the Technical Specification limits, providing substantial margin to the calculated doses.

c) Pre-accident iodine spiking is not currently present, and accident initiated iodine spiking would be less severe than assumed in a design basis analysis given the present initial low primary coolant and core iodine activity profiles.

d) 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, any additional activity releases due to a delay in SDC initiation will still yield dose consequences that are significantly less than EAB dose consequences.

e) A relatively large portion of the limiting control room thyroid dose is attributed to iodine 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, and the current core iodine activity profile available for release to the environment is significantly reduced because Unit 3 has been shut down for at least 35 days.

<u>Other Design[-]Basis Accidents</u>

In Mode 4, the only other UFSAR Chapter 15 accidents of potential concern are Uncontrolled Control Element Assembly (CEA) Bank Withdrawal, Inadvertent Boron Dilution, and Inadvertent Opening of a Pressurizer Safety Valve.

With all of the full length CEAs inserted and Shutdown Margin at the Technical Specification 3.1.1 limit of 5.15 percent$\Delta$k/k, the analysis of the Uncontrolled CEA Bank Withdrawal shows that the core would not reach criticality. Note that the actual Shutdown Margin of approximately 10 percent$\Delta$k/k is significantly higher than the Technical Specification requirement of 5.15 percent$\Delta$k/k. Therefore, this event would have no adverse consequences were it to occur.
The Inadvertent Boron Dilution event is not impacted by the loss of LPSI or SDC, and is thus unchanged from the current analysis of record.

The Inadvertent Opening of a Pressurizer Safety Valve is bounded by the small break LOCA analysis, and thus the previous small break LOCA discussion applies to this event.

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?

Response: 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 do not interact with other Systems, Structures or Components (SSCs) in such a way as to cause any of the initiating event categories listed in Table 15.0-1.

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?

Response: No

For very small break LOCAs (0.012 $ft^2$ 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, if an accident were to occur, the initial power level and core activity inventory are significantly reduced due to Unit 3 being shutdown for over 35 days. The proposed activity takes credit for T-120's water inventory 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 ability to establish SDC following a Steam Generator Tube Rupture (SGTR) is the same as that described for the 0.012 $ft^2$ or smaller LOCAs.
With SDC unavailable, it will be necessary for the Operators to use the ADVs for a longer period of time to effect plant cooldown. While the release duration will increase from that in the current analysis of record, the activity release rate will be significantly less due to Unit 3 being shut down for over 35 days. Therefore, the control room (CR) and low population zone (LPZ) doses are not expected to increase beyond those in the current analysis of record. The longer ADV operation will not impact the Exclusion Area Boundary (EAB) doses, as they are only evaluated for the first two hours of an event.

The UFSAR radiological dose consequences are based on the Technical Specification coolant activity limits. The actual RCS activity at this time is less than the Technical Specification limit. The actual core inventory is significantly less than that used in the analyses due to Unit 3 being shutdown for over 35 days. Thus, there is substantial margin to the calculated doses.

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.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment requests involve 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 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) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

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