SOUTHERN CALIFORNIA
EDISON®
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Dwight E. Nunn
Vice President

September 15, 2003

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

Subject: **San Onofre Nuclear Generating Station Units 2 and 3**
Docket Nos. 50-361 and 50-362
Proposed Change Number (PCN) 546
Request to Revise Technical Specification 2.0 "Safety Limits (SLs)"

Gentlemen:

Pursuant to 10 CFR 50.90, Southern California Edison (SCE) hereby requests the following amendment: In Technical Specification (TS) 2.0, "Safety Limits (SLs)," Reactor Core SL 2.1.1.2, replace the peak linear heat rate SL with a peak fuel centerline temperature SL. This change is requested so SL 2.1.1.2 adequately conforms to 10 CFR 50.36(c)(1)(ii)(A), which requires that Limiting Safety System Settings prevent a Safety Limit from being exceeded. SCE has evaluated this request under the standards set forth in 10 CFR 50.92(c) and determined that a finding of "no significant hazards consideration" is justified.

SCE requests this amendment be issued effective as of the date of issuance, to be implemented within 60 days from the date of issuance.

SCE is making no formal commitments that would result from NRC approval of the proposed amendments.

If you have any questions or require additional information, please contact Mr. Jack Rainsberry at (949) 368-7420.

Sincerely,

Dwight E. Nunn

A001

P.O. Box 128
San Clemente, CA 92674-0128
949-368-1480
Fax 949-368-1490

Enclosures

1. Notarized Affidavits
2. Licensee's Evaluation of the Proposed Change
   Attachments:
   A. Existing Technical Specification page, Unit 2
   B. Existing Technical Specification page, Unit 3
   C. Markup of Technical Specification page, Unit 2
   D. Markup of Technical Specification page, Unit 3
   E. Retyped Technical Specification page, Unit 2
   F. Retyped Technical Specification page, Unit 3
3. Associated Bases Changes

cc: T. P. Gwynn, Acting Regional Administrator, NRC Region IV
B. M. Pham, NRC Project Manager, San Onofre Units 2, and 3
C. C. Osterholtz, NRC Senior Resident Inspector, San Onofre Units 2 and 3
S. Y. Hsu, Department of Health Services, Radiologic Health Branch

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

Application of SOUTHERN CALIFORNIA EDISON COMPANY, ET AL. for a Class 103 License to Acquire, Possess, and Use a Utilization Facility as Part of Unit No. 2 of the San Onofre Nuclear Generating Station	))))))	Docket No. 50-361 Amendment Application No. 223

SOUTHERN CALIFORNIA EDISON COMPANY, et al., pursuant to 10CFR50.90, hereby submit Amendment Application No. 223. This amendment application consists of Proposed Change Number (PCN) 546 to Facility Operating License NPF-10. PCN-546 is a request to replace Technical Specification "Peak Linear Heat Rate" Safety Limit 2.1.1.2, with a "Peak Fuel Centerline Temperature" Safety Limit for San Onofre Nuclear Generating Station Unit 2.

State of California
County of San Diego

Subscribed and sworn to (or affirmed) before me this 15th day of September, 2003.

By: [signature]
Dwight E. Nunn
Vice President

[signature]
Notary Public

MARIANE SANCHEZ
Commission # 1375470
Notary Public - California
San Diego County
My Comm. Expires Oct 14, 2006

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

Application of SOUTHERN CALIFORNIA EDISON COMPANY, ET AL. for a Class 103 License to Acquire, Possess, and Use a Utilization Facility as Part of Unit No. 3 of the San Onofre Nuclear Generating Station	))))))	Docket No. 50-362 Amendment Application No. 207

SOUTHERN CALIFORNIA EDISON COMPANY, et al., pursuant to 10CFR50.90, hereby submit Amendment Application No. 207. This amendment application consists of Proposed Change Number (PCN) 546 to Facility Operating License NPF-15. PCN-546 is a request to replace Technical Specification "Peak Linear Heat Rate" Safety Limit 2.1.1.2, with a "Peak Fuel Centerline Temperature" Safety Limit for San Onofre Nuclear Generating Station Unit 3.

State of California
County of San Diego

Subscribed and sworn to (or affirmed) before me this 15th day of September, 2003.

By: Dwight E. Nunn
Dwight E. Nunn
Vice President

Mariane Sanchez
Notary Public

MARIANE SANCHEZ
Commission # 1375470
Notary Public - California
San Diego County
My Comm. Expires Oct 14, 2006

ENCLOSURE 2

LICENSEE'S EVALUATION
Proposed Change Number 546

SUBJECT: Technical Specification 2.0, "Safety Limits (SLs)," Safety Limit 2.1.1.2

1.0 INTRODUCTION

This letter is a request to amend Operating Licenses NPF-10 and NPF-15 for San Onofre Nuclear Generating Station (SONGS) Units 2 and 3, respectively.

The proposed change will replace the Peak Linear Heat Rate (PLHR) Safety Limit (SL) with a Peak Fuel Centerline Temperature (PFCT) SL. This change is being undertaken so the SL more clearly conforms with 10 CFR 50.36(c)(1)(ii)(A), which requires that Limiting Safety System Settings prevent a Safety Limit from being exceeded. This change is consistent with the NRC Safety Evaluation transmitted from the NRC to the Technical Specification Task Force on December 23, 2002.

2.0 PROPOSED CHANGE

The proposed change replaces Technical Specification (TS) Safety Limit 2.1.1.2, "Peak Linear Heat Rate" with a "Peak Fuel Centerline Temperature" Safety Limit. This change is necessary to adequately address Anticipated Operational Occurrences (AOOs).

Attachments C and D contain the marked-up TS pages reflecting the proposed changes for SONGS Unit 2 and Unit 3, respectively. The Bases changes for Unit 2 associated with this TS change are also provided (Enclosure 3) for information only to reflect the new PFCT SL and provide a reference to the approved Topical Reports for determining the PFCT SL.

3.0 BACKGROUND

During review of the Waterford Steam Electric Station, Unit 3, 10 CFR 50, Appendix K Margin Recovery Power Uprate request (Reference 7.1), the NRC staff recognized that the PLHR SL of 21 KW/ft would be exceeded for an Anticipated Operational Occurrence (AOO). In accordance with 10 CFR 50.36(c)(1)(ii)(A), Limiting Safety System Settings must be chosen such that automatic action will prevent a SL from being exceeded. This assessment is applicable during steady state operations and AOOs. Therefore, conformance with 10 CFR 50.36 was not being clearly demonstrated. A similar condition exists for SONGS Units 2 and 3.

The current steady state limit of 21 KW/ft is exceeded during two AOOs at SONGS 2/3. However, the corresponding PFCT does not exceed the melting point during these events. The affected AOOs are the Control Element Assembly Withdrawal events from both Subcritical and at Low Power Startup conditions. The analysis for these events results in the 21 KW/ft limit being exceeded, although this had been previously reviewed and found to be acceptable by the NRC staff (Reference 7.2) for at least two other plants. The review and acceptance by the NRC staff for SONGS 2 and 3 is documented in SONGS Units 2 and 3, Cycle 3, NRC Safety Evaluation Reports (Reference 7.3).

By letter dated December 23, 2002 the NRC issued its Safety Evaluation (Reference 7.4) approving Nuclear Energy Institute Technical Specification Task Force Change Traveler (TSTF) 445, Revision 0, "Revision to Peak Linear Heat Rate Safety Limit," for plant-specific licensee amendment requests and for incorporation into NUREG-1432, Rev. 2, "Standard Technical Specifications Combustion Engineering Plants" (Reference 7.5). TSTF-445, Revision 1, dated February 3, 2003 (Reference 7.6) provided a minor editorial change to the wording of the Safety Limit. By letter dated March 18, 2003 (Reference 7.7), the NRC approved TSTF-445, Revision 1 to permit replacing the TS 2.1.1.2 PLHR SL with the PFCT SL. This proposed change is based on TSTF-445, Revision 1.

4.0 TECHNICAL ANALYSIS

The intent of the PLHR SL is to prevent the Fuel Centerline Temperature (FCT) from exceeding the melting point, which conservatively assures there will be no breach in cladding integrity. The current 21 KW/ft limit was historically chosen as a conservative limit at which the fuel can operate without causing the FCT to exceed the melting point and is a parameter that can be monitored directly by the operators in the Control Room.

For the two AOOs identified in Section 3.0 above, calculations have shown that fuel centerline temperature remains below the melt temperature at linear heat rates of 21 KW/ft. While the AOO analyses show that the peak linear heat rate may exceed 21 KW/ft, the fuel centerline temperature does not exceed the melt temperature, thereby fully satisfying the intent of the Safety Limit.

In accordance with 10 CFR 50, Appendix A, "General Design Criteria" (GDC) 10, "Reactor Design," and GDC 20, "Protection Systems Functions," the acceptance criteria for normal operation and AOOs is that the Specified Acceptable Fuel Design Limits (SAFDLs) not be exceeded. The SAFDL of interest, in this case, is the PFCT limit. This SAFDL is discussed in detail in Standard Review Plan (SRP) Section 4.2 (Reference 7.8), which states:

> *(II)(A)(2)(e) "Overheating of Fuel Pellets: It has also been traditional practice to assume that failure will occur if centerline melting takes place. ... For normal operation and anticipated operational occurrences, centerline melting is not permitted. ... The centerline melting criterion was established to assure that axial or radial relocation of molten fuel would neither allow molten fuel to come into contact with the cladding nor produce local hot spots. The assumption that centerline melting results in fuel failure is conservative."*

SONGS Units 2 and 3 comply with GDCs 10 and 20 as discussed in Updated Final Safety Analysis Report (UFSAR) Sections 3.1.2.1 and 3.1.3.1. Additionally, UFSAR Section 4.4.1 lists the SAFDLs utilized for the design of the SONGS Units 2 and 3 reactors. UFSAR Section 4.4.1.3, states:

> *"The peak temperature of the fuel shall be less than the melting point ... during steady state operation and anticipated operational occurrences."*

Therefore, a more representative SL would be one that is based upon the Peak Fuel Centerline Temperature. A PFCT SL would address both normal operation and AOOs. A PFCT SL would also be consistent with 10 CFR 50 Appendix A, the SRP (Reference 7.9), 10 CFR 50.36, and the SONGS Units 2 and 3 licensing basis.

The melting point of the fuel is dependent on fuel burnup and the amount and type of burnable poison used in the fuel. The design melting point of unirradiated fuel containing no burnable poison is 5080°F. The melting point is adjusted downward from this temperature depending on the amount of burnup and amount and type of burnable poison in the fuel. The adjustment for burnup of 58°F per 10,000 MWD/MTU is consistent with the Combustion Engineering (CE) standard TSs. The 58°F per 10,000 MWD/MTU was accepted by the NRC staff in Topical Report CEN-386-P-A (Reference 7.10). The burnable poison adjustments are determined in accordance with CENPD-382-P-A (Reference 7.11) for fuels containing erbium absorbers. The specific formula for adjustment for the erbium burnable poison is considered to be proprietary information and therefore is not included in this application. The mode of applicability and actions required if the limit is exceeded would be the same as they are for the current PLHR SL. Reference to CENPD-382-P-A (Reference 7.11) is included in the associated TS 2.1.1 Bases changes (Enclosure 3).

Therefore, a PFCT SL of less than 5080°F decreasing by 58°F per 10,0000 MWD/MTU for burnup and adjusting for burnable poisons per CENPD-382-P-A is more appropriate, from a verbatim compliance perspective, than the current PLHR SL. The PFCT SL will:

- address both normal operations and AOOs,
- be consistent with 10 CFR 50 Appendix A criteria,
- be consistent with SAFDLs,
- be consistent with SRP acceptance criteria,
- be consistent with the current licensing basis for SONGS Units 2 and 3,
- be determined using NRC approved methodologies, and
- clearly conform to 10 CFR 50.36(c)(1)(ii)(A).

5.0 REGULATORY ANALYSIS

5.1 Applicable Regulatory Requirements/Criteria

The proposed changes have been evaluated to determine whether applicable regulations and requirements continue to be met.

The proposed changes do not require any exemptions or relief from regulatory requirements, other than the Technical Specifications, and do not affect conformance with any General Design Criteria. The approval of this change will clearly establish conformance with 10 CFR 50.36.

5.2 No Significant Hazards Consideration

The proposed change will revise the operating licenses NPF-10 and NPF-15 for San Onofre Nuclear Generating Station (SONGS), Units 2 and 3, respectively, to replace the Peak Linear Heat Rate (PLHR) Safety Limit (SL), Technical Specification (TS) 2.1.1.2, with a Peak Fuel Centerline Temperature (PFCT) SL of 5080°F or less decreasing by 58°F per 10,000 MWD/MTU for burnup and adjusting for burnable poisons per CENPD-382-P-A. This change is necessary to more clearly conform with 10 CFR 50.36(c)(1)(ii)(A), which requires that Limiting Safety System Settings prevent a SL from being exceeded.

The proposed change has been evaluated as to whether or not a significant hazards consideration is involved with the proposed amendments by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

Response: No.

The proposed change does not require any physical change to any plant systems, structures, or components nor does it require any change in systems or plant operations. The proposed change does not require any change in safety analysis methods or results. The change to establish the PFCT as the SL is consistent with the Standard Review Plan (SRP) and the SONGS Units 2 and 3 licensing basis for ensuring that the fuel design limits are met. Operations and analysis will continue to be in compliance with NRC regulations.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?**

Response: No.

The SONGS Units 2 and 3 Updated Final Safety Analysis Report (UFSAR) Chapter 15 accident analysis for Anticipated Operational Occurrences (AOOs) where the peak linear heat rate may exceed the existing Safety Limit of 21 KW/ft is the Control Element Assembly (CEA) Withdrawal at subcritical and low power startup conditions.

The accident analyses indicate that the peak linear heat rate may exceed the Limiting Safety System Setpoint of 21 KW/ft during Control Element Assembly Withdrawal Events at Subcritical and Hot Zero Power conditions. The analyses for these AOOs indicate that the PFCT is not approached or exceeded. The existing analyses remain unchanged and do not affect any accident initiators that would create a new accident.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. **Does the proposed change involve a significant reduction in a margin of safety?**

Response: No.

The proposed change does not require any change in accident analysis methods or results. Therefore, by changing the SL from PLHR to Peak Fuel Centerline Temperature, the margin as established in the current license basis remains unchanged.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, the proposed amendments present no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

5.3 Environmental Considerations

The proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 PRECEDENCE

The proposed "PFCT SL" is consistent with the "Peak Fuel Centerline Temperature" Safety Limit in the Standard Technical Specifications (STS) for CE plants (Reference 7.5) as approved by the NRC (Reference 7.7). In addition, the NRC has also approved a similar change for Waterford (Reference 7.12) and Palo Verde (Reference 7.13).

7.0 REFERENCES

7.1 Entergy letter dated September 21, 2001, Technical Specification Change Request, NPF-38-238, "Appendix K Margin Recovery - Power Uprate Request"

7.2 Issuance of Amendment No. 138 to Facility Operating License No. NPF-6 – Arkansas Nuclear One, Unit No. 2 (TAC No. M84098) dated July 22, 1992 and Waterford Steam Electric Station, Unit 3, Cycle 2 Safety Evaluation Report, Section 5.4, dated January 16, 1987

7.3 NRC Safety Evaluation Supporting Amendment No. 47 to Facility Operating License NPF-10 and Amendment No. 36 to Facility Operating License NPF-15, San Onofre Nuclear Generating Station, Units 2 and 3, May 16, 1986

7.4 NRC Safety Evaluation approving changes to NUREG-1432, Standard Technical Specifications, Combustion Engineering Plants, dated December 23, 2002
7.3

7.5 NUREG-1432, "Standard Technical Specifications, Combustion Engineering Plants," Revision 2

7.6 Industry/TSTF Standard Technical Specification Change Traveler, TSTF-445, Rev. 1, "Revision to Peak Linear Heat Rate Safety Limit," February 3, 2003

7.7 NRC letter to Mr. Anthony Pietrangelo dated March 18, 2003 regarding "the Nuclear Energy Institute Technical Specification Change Traveler, TSTF-445, Rev. 1, "Revision to Peak Linear Heat Rate Safety Limit," dated February 3, 2003"

7.8 NUREG-0800, Standard Review Plan, Section 4.2, "Fuel System Design," Rev. 2, July 1981

7.9 NUREG-0800, "Standard Review Plan," Section 15.4.1, "Uncontrolled Control Rod Assembly Withdrawal From A Subcritical or Low Power Startup Condition," Rev. 2, July 1981

7.10 CEN-386-P-A, "Verification of the Acceptability of a 1-Pin Burnup Limit of 60 MWD/kgU for Combustion Engineering 16x16 PWR Fuel," August 1992

7.11 Topical Report, CENPD-382-P-A, "Methodology for Core Designs Containing Erbium Burnable Absorbers," Revision 0, August 1993

7.12 March 5, 2002 letter from N. Kalyanam (NRC) to Joseph E. Venable (Entergy), Subject: Waterford Steam Electric Station, Unit 3 – Issuance of Amendment Re: Revision to Peak Linear Heat Rate Safety Limit (TAC No. MB3926)

7.13 December 2, 2002 letter from Jack Donohew (NRC) to Gregg R. Overbeck (APS), Subject: Palo Verde Nuclear Generating Station, Units 1, 2, and 3 – Issuance of Amendments on Peak Fuel Centerline Temperature Safety Limit (TAX Nos. MB6328, MB6329, MB6330)

Attachment A

Existing Technical Specification Page

SONGS Unit 2

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 In MODES 1 and 2, departure from nucleate boiling ratio (DNBR) shall be maintained at $\geq$ 1.31.

2.1.1.2 In MODES 1 and 2, the peak linear heat rate (LHR) (adjusted for fuel rod dynamics) shall be maintained at $\leq$ 21.0 kW/ft.

2.1.2 Reactor Coolant System (RCS) Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained at $\leq$ 2750 psia.

2.2 SL Violations

2.2.1 If SL 2.1.1.1 or SL 2.1.1.2 is violated, restore compliance and be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.2.2.2 In MODE 3, 4, or 5, restore compliance within 5 minutes.

2.2.3 Within 1 hour, notify the NRC Operations Center, in accordance with 10 CFR 50.72.

2.2.4 Within 24 hours, notify the Vice President - Nuclear Generation and the Nuclear Safety Group (NSG) Supervisor.

2.2.5 Within 30 days of the violation, a Licensee Event Report (LER) shall be prepared pursuant to 10 CFR 50.73. The LER shall be submitted to the NRC, the NSG Supervisor, and the Vice President - Nuclear Generation.

(continued)

Attachment B

Existing Technical Specification Page

SONGS Unit 3

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 In MODES 1 and 2, departure from nucleate boiling ratio (DNBR) shall be maintained at ≥ 1.31.

2.1.1.2 In MODES 1 and 2, the peak linear heat rate (LHR) (adjusted for fuel rod dynamics) shall be maintained at ≤ 21.0 kW/ft.

2.1.2 Reactor Coolant System (RCS) Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained at ≤ 2750 psia.

2.2 SL Violations

2.2.1 If SL 2.1.1.1 or SL 2.1.1.2 is violated, restore compliance and be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.2.2.2 In MODE 3, 4, or 5, restore compliance within 5 minutes.

2.2.3 Within 1 hour, notify the NRC Operations Center, in accordance with 10 CFR 50.72.

2.2.4 Within 24 hours, notify the Vice President - Nuclear Generation and the Nuclear Safety Group (NSG) Supervisor.

2.2.5 Within 30 days of the violation, a Licensee Event Report (LER) shall be prepared pursuant to 10 CFR 50.73. The LER shall be submitted to the NRC, the NSG Supervisor, and the Vice President - Nuclear Generation.

(continued)

Attachment C

Proposed Technical Specification Page

(Redline and Strikeout)

SONGS Unit 2

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 In MODES 1 and 2, departure from nucleate boiling ratio (DNBR) shall be maintained at ≥ 1.31.

2.1.1.2 In MODES 1 and 2, ~~the peak linear heat rate (LHR) (adjusted for fuel rod dynamics) shall be maintained at ≤ 21.0 kW/ft.~~ peak fuel centerline temperature shall be maintained at < 5080°F, decreasing by 58°F per 10,000 MWD/MTU and adjusted for burnable poison per CENPD-382-P-A.

2.1.2 Reactor Coolant System (RCS) Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained at ≤ 2750 psia.

2.2 SL Violations

2.2.1 If SL 2.1.1.1 or SL 2.1.1.2 is violated, restore compliance and be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.2.2.2 In MODE 3, 4, or 5, restore compliance within 5 minutes.

2.2.3 Within 1 hour, notify the NRC Operations Center, in accordance with 10 CFR 50.72.

2.2.4 Within 24 hours, notify the Vice President - Nuclear Generation and the Nuclear Safety Group (NSG) Supervisor.

2.2.5 Within 30 days of the violation, a Licensee Event Report (LER) shall be prepared pursuant to 10 CFR 50.73. The LER shall be submitted to the NRC, the NSG Supervisor, and the Vice President - Nuclear Generation.

(continued)

Attachment D

Proposed Technical Specification Page

(Redline and Strikeout)

SONGS Unit 3

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 In MODES 1 and 2, departure from nucleate boiling ratio (DNBR) shall be maintained at ≥ 1.31.

2.1.1.2 In MODES 1 and 2, ~~the peak linear heat rate (LHR) (adjusted for fuel rod dynamics) shall be maintained at ≤ 21.0 kW/ft.~~ peak fuel centerline temperature shall be maintained at < 5080°F, decreasing by 58°F per 10,000 MWD/MTU and adjusted for burnable poison per CENPD-382-P-A.

2.1.2 Reactor Coolant System (RCS) Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained at ≤ 2750 psia.

2.2 SL Violations

2.2.1 If SL 2.1.1.1 or SL 2.1.1.2 is violated, restore compliance and be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.2.2.2 In MODE 3, 4, or 5, restore compliance within 5 minutes.

2.2.3 Within 1 hour, notify the NRC Operations Center, in accordance with 10 CFR 50.72.

2.2.4 Within 24 hours, notify the Vice President - Nuclear Generation and the Nuclear Safety Group (NSG) Supervisor.

2.2.5 Within 30 days of the violation, a Licensee Event Report (LER) shall be prepared pursuant to 10 CFR 50.73. The LER shall be submitted to the NRC, the NSG Supervisor, and the Vice President - Nuclear Generation.

(continued)

Attachment E

Proposed Technical Specification Page

SONGS Unit 2

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 In MODES 1 and 2, departure from nucleate boiling ratio (DNBR) shall be maintained at ≥ 1.31.

2.1.1.2 In MODES 1 and 2, peak fuel centerline temperature shall be maintained at < 5080°F, decreasing by 58°F per 10,000 MWD/MTU and adjusted for burnable poison per CENPD-382-P-A.

2.1.2 Reactor Coolant System (RCS) Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained at ≤ 2750 psia.

2.2 SL Violations

2.2.1 If SL 2.1.1.1 or SL 2.1.1.2 is violated, restore compliance and be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.2.2.2 In MODE 3, 4, or 5, restore compliance within 5 minutes.

2.2.3 Within 1 hour, notify the NRC Operations Center, in accordance with 10 CFR 50.72.

2.2.4 Within 24 hours, notify the Vice President - Nuclear Generation and the Nuclear Safety Group (NSG) Supervisor.

2.2.5 Within 30 days of the violation, a Licensee Event Report (LER) shall be prepared pursuant to 10 CFR 50.73. The LER shall be submitted to the NRC, the NSG Supervisor, and the Vice President - Nuclear Generation.

(continued)

Attachment F

Proposed Technical Specification Page

SONGS Unit 3

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 In MODES 1 and 2, departure from nucleate boiling ratio (DNBR) shall be maintained at ≥ 1.31.

2.1.1.2 In MODES 1 and 2, peak fuel centerline temperature shall be maintained at < 5080°F, decreasing by 58°F per 10,000 MWD/MTU and adjusted for burnable poison per CENPD-382-P-A.

2.1.2 Reactor Coolant System (RCS) Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained at ≤ 2750 psia.

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2.2.1 If SL 2.1.1.1 or SL 2.1.1.2 is violated, restore compliance and be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.2.2.2 In MODE 3, 4, or 5, restore compliance within 5 minutes.

2.2.3 Within 1 hour, notify the NRC Operations Center, in accordance with 10 CFR 50.72.

2.2.4 Within 24 hours, notify the Vice President - Nuclear Generation and the Nuclear Safety Group (NSG) Supervisor.

2.2.5 Within 30 days of the violation, a Licensee Event Report (LER) shall be prepared pursuant to 10 CFR 50.73. The LER shall be submitted to the NRC, the NSG Supervisor, and the Vice President - Nuclear Generation.

(continued)

ENCLOSURE 3

ASSOCIATED BASES CHANGES
Proposed Change Number 546

San Onofre Unit 2

B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

BASES

BACKGROUND

GDC 10 (Ref. 1) requires and SLs ensure that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). This is accomplished by having a departure from nucleate boiling (DNB) design basis, which corresponds to a 95% probability at a 95% confidence level (95/95 DNB criterion) that DNB will not occur and by requiring that fuel centerline temperature stays below the melting temperature.

The restrictions of this SL prevent overheating of the fuel and cladding and possible cladding perforation that would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state, ~~peak linear heat rate (LHR) below the level at which fuel centerline melting occurs~~peak Centerline Temperature below the melting point. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime, where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Fuel centerline melting occurs when the local peak linear heat rate (LHR), or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant.

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of DNB and the resultant sharp reduction in the heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding water (zirconium water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

h. Local Power Density - High trip;

i. DNBR - Low trip;

j. Reactor Coolant Flow - Low trip; and

k. Steam Generator Safety Valves.

The SL represents a design requirement for establishing the protection system trip setpoint allowable values identified previously. LCO 3.2.1, "Linear Heat Rate (LHR)," and LCO 3.2.4, "Departure From Nucleate Boiling Ratio (DNBR)," or the assumed initial conditions of the safety analyses (as indicated in the UFSAR, Ref. 2) provide more restrictive limits to ensure that the SLs are not exceeded.

SAFETY LIMITS

SL 2.1.1.1 and SL 2.1.1.2 ensure that the minimum DNBR is not less than the safety analyses limit and that fuel centerline temperature remains below melting.

The minimum value of the DNBR during normal operation and design basis AOOs is limited to 1.31, based on a statistical combination of CE-1 CHF correlation and engineering factor uncertainties, and is established as an SL. Additional factors such as rod bow and spacer grid size and placement will determine the limiting safety system settings required to ensure that the SL is maintained. ~~Maintaining the dynamically adjusted peak LHR to ≤ 21 kW/ft ensures that fuel centerline melt will not occur during normal operating conditions or design AOOs.~~

A steady state peak linear heat rate of 21 KW/ft has been established as the Limiting Safety System Setting to prevent fuel centerline melting during normal steady state operation. Following design basis anticipated operational occurrences, the transient linear heat rate may exceed 21 KW/ft provided the fuel centerline melt temperature is not exceeded.

The design melting point of new fuel with no burnable poison is 5080°F. The melting point is adjusted downward from this temperature depending on the amount of burnup and amount and type of burnable poison in the fuel. The 58°F per 10,000 MWD/MTU adjustment for burnup was accepted by the NRC in Topical Report CEN-386-P-A, Reference 8. Adjustments for burnable poisons are established based on NRC approved Topical Report CENPD-382-P-A, Reference 9.

(continued)

BASES

APPLICABILITY

SL 2.1.1.1 and SL 2.1.1.2 only apply in MODES 1 and 2 because these are the only MODES in which the reactor is critical. Automatic protection functions are required to be OPERABLE during MODES 1 and 2 to ensure operation within the reactor core SLs. The steam generator safety valves or automatic protection actions serve to prevent RCS heatup to the reactor core SL conditions or to initiate a reactor trip function, which forces the unit into MODE 3.

In MODES 3, 4, 5, and 6, Applicability is not required, since the reactor is not generating significant fraction of rated thermal power (RTP).

(continued)

BASES

SAFETY LIMIT VIOLATIONS (continued)

2.2.4

If the RCS pressure SL is violated, the appropriate senior management of the nuclear plant and the utility shall be notified within 24 hours. This 24 hour period provides time for the plant operators and staff to take the appropriate immediate action and to assess the condition of the unit before reporting to the senior management.

2.2.5

If the RCS pressure SL is violated, a Licensee Event Report shall be prepared and submitted within 30 days to the NRC, Vice President - Nuclear Generation, and the NSG Supervisor. This requirement is in accordance with 10 CFR 50.73 (Ref. 7).

2.2.6

If the RCS pressure SL is violated, restart of the unit shall not commence until authorized by the NRC. This requirement ensures the NRC that all necessary reviews, analyses, and actions are completed before the unit begins its restart to normal operation.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 14, GDC 15, and GDC 28.
2. ASME, Boiler and Pressure Vessel Code, Section III, Article NB-7000.
3. ASME, Boiler and Pressure Vessel Code, Section XI, Article IWX-5000.
4. 10 CFR 100.
5. UFSAR, Section 7.2, "Reactor Protective Systems"
6. 10 CFR 50.72.
7. 10 CFR 50.73.
8. CEN-386-P-A, "Verification of the Acceptability of a 1-Pin Burnup Limit of 60 MWD/MTU for Combustion Engineering 16x16 PWR Fuel," August 1992.
9. CENPD-382-P-A, "Methodology for Core Designs Containing Erbium Burnable Absorbers," August 1993.

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

b. During a loss of flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition (Ref. 4);

c. During an ejected CEA accident, the fission energy input to the fuel must not exceed 280 cal/gm (Ref. 1); and

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (GDC 27, Ref. 4).

The power density at any point in the core must be limited to maintain the fuel design criteria (Refs. 4 and 5). This is accomplished by maintaining the power distribution and reactor coolant conditions so that the peak ~~LHR~~ Fuel Centerline Temperature and DNB parameters are within operating limits supported by the accident analyses (Ref. 1) with due regard for the correlations between measured quantities, the power distribution, and uncertainties in determining the power distribution.

Fuel cladding failure during a LOCA is limited by restricting the maximum linear heat generation rate so that the peak cladding temperature does not exceed 2200°F (Ref. 5). Peak cladding temperatures exceeding 2200°F cause severe cladding failure by oxidation due to a Zircaloy water reaction.

The LCOs governing the LHR, ASI, and RCS ensure that these criteria are met as long as the core is operated within the ASI and F_{xy} limits specified in the COLR, and within the T_q limits. The latter are process variables that characterize the three dimensional power distribution of the reactor core.

Operation within the limits for these variables ensures that their actual values are within the ranges used in the accident analyses.

Fuel cladding damage does not occur from conditions outside the limits of these LCOs during normal operation. However, fuel cladding damage could result if an accident occurs from initial conditions outside the limits of these LCOs. This

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

b. During a loss of flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition (Ref. 4);

c. During an ejected CEA accident, the fission energy input to the fuel must not exceed 280 cal/gm (Ref. 1); and

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (GDC 26, Ref. 4).

The power density at any point in the core must be limited to maintain the fuel design criteria (Refs. 4 and 5). This result is accomplished by maintaining the power distribution and reactor coolant conditions so that the peak ~~LHR~~ Fuel Centerline Temperature and DNB parameters are within operating limits supported by the accident analyses (Ref. 1) with due regard for the correlations between measured quantities, the power distribution, and the uncertainties in the determination of power distribution.

Fuel cladding failure during a LOCA is limited by restricting the maximum linear heat generation rate so that the peak cladding temperature does not exceed 2200°F (Ref. 5). Peak cladding temperatures exceeding 2200°F cause severe cladding failure by oxidation due to a Zircaloy water reaction.

The LCOs governing LHR, ASI, and RCS ensure that these criteria are met as long as the core is operated within the ASI and F_{xy} limits specified in the COLR, and within the T_q limits. The latter are process variables that characterize the three dimensional power distribution of the reactor core. Operation within the limits for these variables ensures that their actual values are within the ranges used in the accident analyses.

Fuel cladding damage does not occur because of conditions outside the limits of these LCOs for ASI, F_{xy}, and T_q during normal operation. However, fuel cladding damage may result if an accident occurs with initial conditions outside the limits of these LCOs. This potential for fuel cladding damage exists because changes in the power distribution can

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

b. During a loss of flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition (Ref. 4);

c. During a CEA ejection accident, the fission energy input to the fuel must not exceed 280 cal/gm (Ref. 5); and

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 6).

The power density at any point in the core must be limited to maintain the fuel design criteria (Ref. 1). This result is accomplished by maintaining the power distribution and reactor coolant conditions so that the peak ~~LHR~~ Fuel Centerline Temperature and DNB parameters are within operating limits supported by the accident analysis (Ref. 2) with due regard for the correlations between measured quantities, the power distribution, and uncertainties in the determination of power distribution.

Fuel cladding failure during a LOCA is limited by restricting the maximum linear heat generation rate (LHGR) so that the peak cladding temperature does not exceed 2200°F (Ref. 1). Peak cladding temperatures exceeding 2200°F cause severe cladding failure by oxidation due to a Zircaloy water reaction.

The LCOs governing LHR, ASI, and RCS ensure that these criteria are met as long as the core is operated within the ASI and F_{xy} limits specified in the COLR, and within the T_q limits. The latter are process variables that characterize the three dimensional power distribution of the reactor core. Operation within the limits of these variables ensures that their actual values are within the range used in the accident analyses.

Fuel cladding damage does not occur from conditions outside the limits of these LCOs during normal operation. However, fuel cladding damage could result if an accident occurs due to initial conditions outside the limits of these LCOs. The potential for fuel cladding damage exists because changes in the power distribution can cause increased power peaking and correspondingly increased local LHRs.

T_q satisfies Criterion 2 of the NRC Policy Statement.

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

a. During a LOCA, peak cladding temperature must not exceed 2200°F (Ref. 5);

b. During a loss of flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition (Ref. 4);

c. During an ejected CEA accident, the fission energy input to the fuel must not exceed 280 cal/gm (Ref. 1); and

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 6).

The power density at any point in the core must be limited to maintain the fuel design criteria (Ref. 4). This is accomplished by maintaining the power distribution and reactor coolant conditions so that the peak ~~LHR~~ Fuel Centerline Temperature and DNB parameters are within operating limits supported by the accident analyses (Ref. 1) with due regard for the correlations between measured quantities, the power distribution, and uncertainties in the determination of power distribution.

Fuel cladding failure during a LOCA is limited by restricting the maximum linear heat generation rate so that the peak cladding temperature does not exceed 2200°F (Ref. 4). Peak cladding temperatures exceeding 2200°F may cause severe cladding failure by oxidation due to a Zircaloy water reaction.

The LCOs governing LHR, ASI, and RCS ensure that these criteria are met as long as the core is operated within the ASI and F_{xy} limits specified in the COLR, and within the T_q limits. The latter are process variables that characterize the three dimensional power distribution of the reactor core. Operation within the limits for these variables ensures that their actual values are within the range used in the accident analyses (Ref. 1).

Fuel cladding damage does not occur from conditions outside the limits of these LCOs during normal operation. However, fuel cladding damage could result if an accident occurs from initial conditions outside the limits of these LCOs. This

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

b. During a loss of flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition (Ref. 4);

c. During an ejected CEA accident, the fission energy input to the fuel must not exceed 280 cal/gm (Ref. 6);

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 7).

The power density at any point in the core must be limited to maintain the fuel design criteria (Refs. 4 and 5). This is accomplished by maintaining the power distribution and reactor coolant conditions so that the peak ~~LHR~~ Fuel Centerline Temperature and DNB parameters are within operating limits supported by the accident analyses (Ref. 1) with due regard for the correlations among measured quantities, the power distribution, and uncertainties in the determination of power distribution.

Fuel cladding failure during a LOCA is limited by restricting the maximum so that the peak cladding temperature does not exceed 2200°F (Ref. 5). Peak cladding temperatures exceeding 2200°F may cause severe cladding failure by oxidation due to a Zircaloy water reaction.

The LCOs governing LHR, ASI, and RCS ensure that these criteria are met as long as the core is operated within the ASI and F_{xy} limits specified in the COLR, and within the T_q limits. The latter are process variables that characterize the three dimensional power distribution of the reactor core. Operation within the limits for these variables ensures that their actual values are within the range used in the accident analysis.

Fuel cladding damage does not occur from conditions outside these LCOs during normal operation. However, fuel cladding damage results when an accident occurs due to initial conditions outside the limits of these LCOs. This potential for fuel cladding damage exists because changes in the power distribution can cause increased power peaking and correspondingly increased local LHRs.

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