SONGS UNIT 3 SINGLE-SIDED

Operating License & Technical Specification/Licensee Controlled Specification

AMD 206/198 & AMD 207/199 B06-07

Revise Unit 3 Operating License by removing pages and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and/or dates and contain marginal/lines indicating the areas of change. **Please direct any questions regarding filing of information to: (949) 368-6557 or email CDM Controlled Manuals Desk.**

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Filing Instructions Reviewed & Approved:

[signature] 1/2/07

Signature/NRA Date

LaDonna Miller 1/2/07

Signature/CDM Processor Date

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Rec'd w/out letter

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B 3.4-83	3	116		B 3.5-32	3	116	
B 3.4-84	3	116		B 3.5-33	3	116	
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B 3.4-93	3	196		B 3.6-8	3	116	
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B 3.6-46	3	116		B 3.7-40	3	116	
B 3.6-47	3	116		B 3.7-41	3	116	04/03/06
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B 3.7-77	3	116		B 3.8-42	3	116	03/25/02
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(3) SCE, pursuant to the Act and 10 CFR Part 70, to receive, possess, and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;

(4) SCE, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use at any time any byproduct, source and special nuclear materials as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(5) SCE, pursuant to the Act and 10 CFR Parts 30, 40, and 70 to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(6) SCE, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of San Onofre Nuclear Generating Station, Units 1 and 3 and by the decommissioning of San Onofre Nuclear Generating Station Unit 1.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

Southern California Edison Company (SCE) is authorized to operate the facility at reactor core power levels not in excess of full power (3438 megawatts thermal).

(2) Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 199, are hereby incorporated in the license. Southern California Edison Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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.

(continued)

2.0 SLs

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5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

5.1.1 The corporate officer with direct responsibility for the plant shall be responsible for overall unit operation and maintenance of Units 2 and 3 at San Onofre Nuclear Generating Station, and all site support functions. He shall delegate in writing the succession to this responsibility during his absence.

5.1.2 The Shift Manager shall be responsible for the ultimate command decision authority for all unit activities and operations which affect the safety of the plant, site personnel, and/or the general public. A management directive to this effect, signed by the corporate officer with direct responsibility for the plant shall be reissued to all site/station personnel on an annual basis.

5.1.3 The Control Room Supervisor (CRS) shall be responsible for the Control Room command function. A management directive to this effect, signed by the corporate officer with direct responsibility for the plant, shall be issued annually to all site/station personnel. The confines of the Control Room Area shall be defined as depicted in the Licensee Controlled Specification (LCS). During any absence of the CRS from the Control Room Area while the Unit is in MODE 1, 2, 3, or 4, an individual with an active Senior Reactor Operator's (SRO) license shall be designated to assume the Control Room command function. During any absence of the CRS from the Control Room Area while the Unit is in MODE 5 or 6, an individual with an active SRO license or Reactor Operator's license shall be designated to assume the Control Room command function.

5.0 ADMINISTRATIVE CONTROLS

5.2 Organization

5.2.1 Onsite and Offsite Organizations

Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting the safety of the nuclear power plant.

a. Lines of authority, responsibility, and communication shall be established and defined for the highest management levels through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These relationships, including the plant-specific titles of those personnel fulfilling the responsibilities for the positions delineated in these Technical Specifications, are documented in the UFSAR.

b. The corporate officer with direct responsibility for the plant shall be responsible for overall unit safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

c. A specified corporate officer (or officers) shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.

d. The individuals who train the operating staff and those who carry out health physics and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

(continued)

5.2 Organization (continued)

5.2.2 UNIT STAFF

The unit staff organization shall include the following:

a. A non-Licensed Operator shall be assigned to each reactor containing fuel and an additional non-Licensed Operator shall be assigned for each unit when a reactor is operating in MODES 1, 2, 3, or 4.

With both units shutdown or defueled, a total of three non-Licensed operators are required for the two units.

b. At least one licensed Reactor Operator (RO) shall be in the Control Room when fuel is in the reactor. In addition, while the unit is in MODE 1, 2, 3 or 4, at least one licensed Senior Reactor Operator (SRO) shall be in the Control Room Area.

c. Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i) and 5.2.2.a for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements.

d. A radiation protection technician shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

e. Administrative controls shall be developed and implemented to limit the working hours of personnel who perform safety-related functions (e.g., senior reactor operators, reactor operators, auxiliary operators, health physicists, and key maintenance personnel). The controls shall include guidelines on working hours that ensure that adequate shift coverage is maintained without routine heavy use of overtime for individuals.

Any deviation from the working hour guidelines shall be authorized in advance by the cognizant corporate officer, or designees, in accordance with approved administrative procedures, or by higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation.

(continued)

5.2 Organization (continued)

5.2.2 UNIT STAFF (continued)

Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the cognizant corporate officer, or designees, to ensure that excessive hours have not been assigned. Routine deviation from the above guidelines shall not be authorized.

f. The Manager, Plant Operations (at time of appointment), Shift Managers, and Control Room Supervisors shall hold a Senior Reactor Operator's license.

g. The Shift Technical Advisor (STA) shall provide advisory technical support to the Shift Manager in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. The STA shall have a Bachelor's Degree or equivalent in a scientific or engineering discipline with specific training in plant design and in the response and analysis of the plant for transients and accidents.

(continued)

5.0 ADMINISTRATIVE CONTROLS

5.3 Unit Staff Qualifications

5.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except a) the radiation protection manager who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975, and b) multi-discipline supervisors who shall meet or exceed the qualifications listed below.

In addition, the Shift Technical Advisor shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift.

Multi-discipline supervisors shall meet or exceed the following requirements:

a. Education: Minimum of a high school diploma or equivalent.

b. Experience: Minimum of four years of related technical experience which shall include three years power plant experience of which one year is at a nuclear plant.

c. Training: Complete the multi-discipline supervisor training program.

5.5 Procedures, Programs, and Manuals (continued)

5.5.2 Programs and Manuals

The following programs and manuals shall be established, implemented, and maintained.

5.5.2.1 Offsite Dose Calculation Manual (ODCM)

a. The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the Radiological Environmental Monitoring Program;

b. The ODCM shall also contain the Radioactive Effluent Controls required by Specification 5.5.2.3 and the Radiological Environmental Monitoring programs required by the LCS, and descriptions of the information that should be included in the Annual Radiological Environmental Operating Report and the Radioactive Effluent Release Report required by Specification 5.7.1.2 and Specification 5.7.1.3.

5.5.2.1.1 Licensee-initiated changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

   1. Sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s);

   2. A determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.106, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

   3. Documentation of the fact that the change has been reviewed and found acceptable.

b. Shall become effective upon review and approval by the corporate officer with direct responsibility for the plant or designee.

(continued)

5.5 Procedures, Programs, and Manuals (continued)

5.5.2.11 Steam Generator (SG) Program (continued)

c. Provisions for SG tube repair criteria.

1. Tubes shall be plugged or repaired if the non-sleeved region of a tube is found by inservice inspection to contain flaws with a depth equal to or exceeding 44% of the nominal tube wall thickness, at a location that is not addressed in Technical Specification 5.5.2.11.c.2.

2. Tubes shall be plugged or repaired if the non-sleeved region of a tube is found by inservice inspection to contain flaws at either of the following locations:

a) below the bottom of the hot leg expansion transition or hot leg top of the tubesheet, whichever is higher, or

b) below the bottom of the cold leg expansion transition or cold leg top of the tubesheet, whichever is higher.

3. Tubes shall be plugged if the sleeved region of a tube is found to contain flaws in the:

a) sleeve, or

b) sleeve or original tube wall at a sleeve-to-tube joint.

4. The following C* methodology may be applied in a portion of the expanded tube in the tubesheet region, as an alternative to the repair criteria of Technical Specification 5.5.2.11.c.2. Flaws, in the locations described below, may remain in service regardless of size.

a) For tubes that have not been repaired in the hot leg tubesheet region: Greater than 10.6 inches below the bottom of the hot leg expansion transition or top of the hot leg tubesheet, whichever is lower.

b) For tubes that have not been repaired in the cold leg tubesheet region: Greater than 11.0 inches below the bottom of the cold leg expansion transition or top of the cold leg tubesheet, whichever is lower.

(continued)

5.5 Procedures, Programs, and Manuals (continued)

5.5.2.11 Steam Generator (SG) Program (continued)

c) For tubes that have been repaired in the hot leg tubesheet region: Below the bottom of the lower sleeve-to-tube joint or greater than 10.6 inches below the bottom of the hot leg expansion transition or greater than 10.6 inches below the top of the hot leg tubesheet, whichever of these three is lowest.

d) For tubes that have been repaired in the cold leg tubesheet region: Below the bottom of the lower sleeve-to-tube joint or greater than 11.0 inches below the bottom of the cold leg expansion transition or greater than 11.0 inches below the top of the cold leg tubesheet, whichever of these three is lowest.

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. The tube-to-tubesheet weld is not part of the tube. In tubes repaired by sleeving, the portion of the original tube wall between the sleeve's joints is not an area requiring re-inspection. In addition to meeting the requirements of d.1, d.2, d.3, and d.4 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

1. Inspect 100% of the tubes in each SG during the first refueling outage following SG replacement.

2. Inspect 100% of the tubes at sequential periods of 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. No SG shall operate for more than 24 effective full power months or one refueling outage (whichever is less) without being inspected.

(continued)

5.8. High Radiation Area (continued)

5.8.2 In addition, areas that are accessible to personnel and that have radiation levels greater than 1.0 rem (but less than 500 rads at 1 meter) in 1 hour at 30 cm from the radiation source, or from any surface penetrated by the radiation, shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the shift manager on duty or radiation protection supervisor. Doors shall remain locked except during periods of access by personnel under an approved REP that specifies the dose rates in the immediate work areas and the maximum allowable stay time for individuals in that area. In lieu of a stay time specification on the REP, direct or remote continuous surveillance (such as closed circuit TV cameras) may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area.

5.8.3 Individual high radiation areas that are accessible to personnel, that could result in radiation doses greater than 1.0 rem in 1 hour, and that are within large areas, where no enclosure exists to enable locking and where no enclosure can be reasonably constructed around the individual area shall be barricaded and conspicuously posted. A flashing light shall be activated as a warning device whenever the dose rate in such an area exceeds or is expected to exceed 1.0 rem in 1 hour at 30 cm from the radiation source or from any surface penetrated by the radiation.

BASES (continued)

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.

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 3. Adjustments for burnable poisons are established based on NRC approved Topical Report CENPD-382-P-A, Reference 4.

(continued)

BASES (continued)

SAFETY LIMIT VIOLATIONS

The following violation responses are applicable to the reactor core SLs.

2.2.1

If SL 2.1.1.1 or SL 2.1.1.2 is violated, the requirement to go to MODE 3 places the unit in a MODE in which this SL is not applicable.

The allowed Completion Time of 1 hour recognizes the importance of bringing the unit to a MODE where this SL is not applicable and reduces the probability of fuel damage.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.
2. UFSAR, Section 15.0.3.2, "Initial Conditions."
3. 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.
4. CENPD-382-P-A, "Methodology for Core Designs Containing Erbium Burnable Absorbers," August 1993.

(continued)

BASES (continued)

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SAFETY LIMIT VIOLATIONS

The following SL violation responses are applicable to the RCS pressure SLs.

2.2.2.1

If the RCS pressure SL is violated when the reactor is in MODE 1 or 2, the requirement is to restore compliance and be in MODE 3 within 1 hour.

With RCS pressure greater than the value specified in SL 2.1.2 in MODE 1 or 2, the pressure must be reduced to below this value. A pressure greater than the value specified in SL 2.1.2 exceeds 110% of the RCS design pressure and may challenge system integrity.

The allowed Completion Time of 1 hour provides the operator time to complete the necessary actions to reduce RCS pressure by terminating the cause of the pressure increase, removing mass or energy from the RCS, or a combination of these actions, and to establish MODE 3 conditions.

2.2.2.2

If the RCS pressure SL is exceeded in MODE 3, 4, or 5, RCS pressure must be restored to within the SL value within 5 minutes.

Exceeding the RCS pressure SL in MODE 3, 4, or 5 is potentially more severe than exceeding this SL in MODE 1 or 2, since the reactor vessel temperature may be lower and the vessel material, consequently, less ductile. As such, pressure must be reduced to less than the SL within 5 minutes. This action does not require reducing MODES, since this would require reducing temperature, which would compound the problem by adding thermal gradient stresses to the existing pressure stress.

(continued)

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"

BASES (continued)

LCO 3.0.3 (continued)

Voluntary entry into LCO 3.0.3 is permissible but requires prior approval (approval may be verbal) from either the Operations Manager, Station Manager or corporate officer with direct responsibility for the plant. The approval must subsequently be documented in written retrievable manner. Inadvertent entry still allows for the one hour preparation period before Actions to change MODES must begin.

A unit shutdown required in accordance with LCO 3.0.3 may be terminated and LCO 3.0.3 exited if any of the following occurs:

a. The LCO is now met.

b. A Condition exists for which the Required Actions have now been performed.

c. ACTIONS exist that do not have expired Completion Times. These Completion Times are applicable from the point in time that the Condition is initially entered and not from the time LCO 3.0.3 is exited.

The time limits of Specification 3.0.3 allow 37 hours for the unit to be in MODE 5 when a shutdown is required during MODE 1 operation. If the unit is in a lower MODE of operation when a shutdown is required, the time limit for reaching the next lower MODE applies. If a lower MODE is reached in less time than allowed, however, the total allowable time to reach MODE 5, or other applicable MODE, is not reduced. For example, if MODE 3 is reached in 2 hours, then the time allowed for reaching MODE 4 is the next 11 hours, because the total time for reaching MODE 4 is not reduced from the allowable limit of 13 hours. Therefore, if remedial measures are completed that would permit a return to MODE 1, a penalty is not incurred by having to reach a lower MODE of operation in less than the total time allowed.

In MODES 1, 2, 3, and 4, LCO 3.0.3 provides actions for Conditions not covered in other Specifications. The requirements of LCO 3.0.3 do not apply in MODES 5 and 6 because the unit is already in the most restrictive Condition required by LCO 3.0.3.

The requirements of LCO 3.0.3 do not apply in other specified conditions of the Applicability (unless in MODE 1, 2, 3, or 4) because the ACTIONS of individual Specifications

(continued)

SONGS UNIT 2 SINGLE-SIDED
Operating License & Technical Specification/Licensee Controlled Specification
AMD 206/198 & AMD 207/199 B06-07

Revise Unit 2 Operating License by removing pages and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and/or dates and contain marginal/lines indicating the areas of change. **Please direct any questions regarding filing of information to: (949) 368-6557 or email CDM Controlled Manuals Desk.**

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(3) SCE, pursuant to the Act and 10 CFR Part 70, to receive, possess, and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;

(4) SCE, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(5) SCE, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(6) SCE, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of San Onofre Nuclear Generating Station, Units 1 and 2 and by the decommissioning of San Onofre Nuclear Generating Station Unit 1.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) <u>Maximum Power Level</u>

Southern California Edison Company (SCE) is authorized to operate the facility at reactor core power levels not in excess of full power (3438 megawatts thermal).

(2) <u>Technical Specifications</u>

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 207, are hereby incorporated in the license. Southern California Edison Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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.

(continued)

2.0 SLs

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5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

5.1.1 The corporate officer with direct responsibility for the plant shall be responsible for overall unit operation and maintenance of Units 2 and 3 at San Onofre Nuclear Generating Station, and all site support functions. He shall delegate in writing the succession to this responsibility during his absence.

5.1.2 The Shift Manager shall be responsible for the ultimate command decision authority for all unit activities and operations which affect the safety of the plant, site personnel, and/or the general public. A management directive to this effect, signed by the corporate officer with direct responsibility for the plant shall be reissued to all site/station personnel on an annual basis.

5.1.3 The Control Room Supervisor (CRS) shall be responsible for the Control Room command function. A management directive to this effect, signed by the corporate officer with direct responsibility for the plant, shall be issued annually to all site/station personnel. The confines of the Control Room Area shall be defined as depicted in the Licensee Controlled Specification (LCS). During any absence of the CRS from the Control Room Area while the Unit is in MODE 1, 2, 3, or 4, an individual with an active Senior Reactor Operator's (SRO) license shall be designated to assume the Control Room command function. During any absence of the CRS from the Control Room Area while the Unit is in MODE 5 or 6, an individual with an active SRO license or Reactor Operator's license shall be designated to assume the Control Room command function.

5.0 ADMINISTRATIVE CONTROLS

5.2 Organization

5.2.1 Onsite and Offsite Organizations

Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting the safety of the nuclear power plant.

a. Lines of authority, responsibility, and communication shall be established and defined for the highest management levels through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These relationships, including the plant-specific titles of those personnel fulfilling the responsibilities for the positions delineated in these Technical Specifications, are documented in the UFSAR.

b. The corporate officer with direct responsibility for the plant shall be responsible for overall unit safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

c. A specified corporate officer (or officers) shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.

d. The individuals who train the operating staff and those who carry out health physics and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

(continued)

5.2 Organization (continued)

5.2.2 UNIT STAFF

The unit staff organization shall include the following:

a. A non-Licensed Operator shall be assigned to each reactor containing fuel and an additional non-Licensed Operator shall be assigned for each unit when a reactor is operating in MODES 1, 2, 3, or 4.

With both units shutdown or defueled, a total of three non-Licensed operators are required for the two units.

b. At least one licensed Reactor Operator (RO) shall be in the Control Room when fuel is in the reactor. In addition, while the unit is in MODE 1, 2, 3 or 4, at least one licensed Senior Reactor Operator (SRO) shall be in the Control Room Area.

c. Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i) and 5.2.2.a for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements.

d. A radiation protection technician shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

e. Administrative controls shall be developed and implemented to limit the working hours of personnel who perform safety-related functions (e.g., senior reactor operators, reactor operators, auxiliary operators, health physicists, and key maintenance personnel). The controls shall include guidelines on working hours that ensure that adequate shift coverage is maintained without routine heavy use of overtime for individuals.

Any deviation from the working hour guidelines shall be authorized in advance by the cognizant corporate officer, or designees, in accordance with approved administrative procedures, or by higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation.

(continued)

5.2 Organization (continued)

5.2.2 UNIT STAFF (continued)

Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the cognizant corporate officer, or designees, to ensure that excessive hours have not been assigned. Routine deviation from the above guidelines shall not be authorized.

f. The Manager, Plant Operations (at time of appointment), Shift Managers, and Control Room Supervisors shall hold a Senior Reactor Operator's license.

g. The Shift Technical Advisor (STA) shall provide advisory technical support to the Shift Manager in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. The STA shall have a Bachelor's Degree or equivalent in a scientific or engineering discipline with specific training in plant design and in the response and analysis of the plant for transients and accidents.

(continued)

5.0 ADMINISTRATIVE CONTROLS

5.3 Unit Staff Qualifications

5.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except a) the radiation protection manager who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975, and b) multi-discipline supervisors who shall meet or exceed the qualifications listed below.

In addition, the Shift Technical Advisor shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift.

Multi-discipline supervisors shall meet or exceed the following qualifications:

a. Education: Minimum of a high school diploma or equivalent.

b. Experience: Minimum of four years of related technical experience which shall include three years power plant experience of which one year is at a nuclear plant.

c. Training: Complete the multi-discipline supervisor training program.

5.5 Procedures, Programs, and Manuals (continued)

5.5.2 Programs and Manuals

The following programs and manuals shall be established, implemented, and maintained.

5.5.2.1 Offsite Dose Calculation Manual (ODCM)

a. The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the Radiological Environmental Monitoring Program;

b. The ODCM shall also contain the Radioactive Effluent Controls required by Specification 5.5.2.3 and the Radiological Environmental Monitoring programs required by the LCS, and descriptions of the information that should be included in the Annual Radiological Environmental Operating Report and the Radioactive Effluent Release Report required by Specification 5.7.1.2 and Specification 5.7.1.3.

5.5.2.1.1 Licensee-initiated changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

   1. Sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s);

   2. A determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.106, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

   3. Documentation of the fact that the change has been reviewed and found acceptable.

b. Shall become effective upon review and approval by the corporate officer with direct responsibility for the plant or designee.

(continued)

5.5 Procedures, Programs, and Manuals (continued)

5.5.2.11 Steam Generator (SG) Program (continued)

c. Provisions for SG tube repair criteria.

1. Tubes shall be plugged or repaired if the non-sleeved region of a tube is found by inservice inspection to contain flaws with a depth equal to or exceeding 44% of the nominal tube wall thickness, at a location that is not addressed in Technical Specification 5.5.2.11.c.2.

2. Tubes shall be plugged or repaired if the non-sleeved region of a tube is found by inservice inspection to contain flaws at either of the following locations:

   a) below the bottom of the hot leg expansion transition or hot leg top of the tubesheet, whichever is higher, or

   b) below the bottom of the cold leg expansion transition or cold leg top of the tubesheet, whichever is higher.

3. Tubes shall be plugged if the sleeved region of a tube is found to contain flaws in the:

   a) sleeve, or

   b) sleeve or original tube wall at a sleeve-to-tube joint.

4. The following C* methodology may be applied in a portion of the expanded tube in the tubesheet region, as an alternative to the repair criteria of Technical Specification 5.5.2.11.c.2. Flaws, in the locations described below, may remain in service regardless of size.

   a) For tubes that have not been repaired in the hot leg tubesheet region: Greater than 10.6 inches below the bottom of the hot leg expansion transition or top of the hot leg tubesheet, whichever is lower.

   b) For tubes that have not been repaired in the cold leg tubesheet region: Greater than 11.0 inches below the bottom of the cold leg expansion transition or top of the cold leg tubesheet, whichever is lower.

(continued)

5.5 Procedures, Programs, and Manuals (continued)

5.5.2.11 Steam Generator (SG) Program (continued)

c) For tubes that have been repaired in the hot leg tubesheet region: Below the bottom of the lower sleeve-to-tube joint or greater than 10.6 inches below the bottom of the hot leg expansion transition or greater than 10.6 inches below the top of the hot leg tubesheet, whichever of these three is lowest.

d) For tubes that have been repaired in the cold leg tubesheet region: Below the bottom of the lower sleeve-to-tube joint or greater than 11.0 inches below the bottom of the cold leg expansion transition or greater than 11.0 inches below the top of the cold leg tubesheet, whichever of these three is lowest.

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. The tube-to-tubesheet weld is not part of the tube. In tubes repaired by sleeving, the portion of the original tube wall between the sleeve's joints is not an area requiring re-inspection. In addition to meeting the requirements of d.1, d.2, d.3, and d.4 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

1. Inspect 100% of the tubes in each SG during the first refueling outage following SG replacement.

2. Inspect 100% of the tubes at sequential periods of 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. No SG shall operate for more than 24 effective full power months or one refueling outage (whichever is less) without being inspected.

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5.8. High Radiation Area (continued)

5.8.2 In addition, areas that are accessible to personnel and that have radiation levels greater than 1.0 rem (but less than 500 rads at 1 meter) in 1 hour at 30 cm from the radiation source, or from any surface penetrated by the radiation, shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the shift manager on duty or radiation protection supervisor. Doors shall remain locked except during periods of access by personnel under an approved REP that specifies the dose rates in the immediate work areas and the maximum allowable stay time for individuals in that area. In lieu of a stay time specification on the REP, direct or remote continuous surveillance (such as closed circuit TV cameras) may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area.

5.8.3 Individual high radiation areas that are accessible to personnel, that could result in radiation doses greater than 1.0 rem in 1 hour, and that are within large areas where no enclosure exists to enable locking and where no enclosure can be reasonably constructed around the individual area shall be barricaded and conspicuously posted. A flashing light shall be activated as a warning device whenever the dose rate in such an area exceeds or is expected to exceed 1.0 rem in 1 hour at 30 cm from the radiation source or from any surface penetrated by the radiation.

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.

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 3. Adjustments for burnable poisons are established based on NRC approved Topical Report CENPD-382-P-A, Reference 4.

(continued)

BASES (continued)

SAFETY LIMIT VIOLATIONS

The following violation responses are applicable to the reactor core SLs.

2.2.1

If SL 2.1.1.1 or SL 2.1.1.2 is violated, the requirement to go to MODE 3 places the unit in a MODE in which this SL is not applicable.

The allowed Completion Time of 1 hour recognizes the importance of bringing the unit to a MODE where this SL is not applicable and reduces the probability of fuel damage.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.
2. UFSAR, Section 15.0.3.2, "Initial Conditions."
3. 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.
4. CENPD-382-P-A, "Methodology for Core Designs Containing Erbium Burnable Absorbers," August 1993.

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BASES

SAFETY LIMIT VIOLATIONS

The following SL violation responses are applicable to the RCS pressure SLs.

2.2.2.1

If the RCS pressure SL is violated when the reactor is in MODE 1 or 2, the requirement is to restore compliance and be in MODE 3 within 1 hour.

With RCS pressure greater than the value specified in SL 2.1.2 in MODE 1 or 2, the pressure must be reduced to below this value. A pressure greater than the value specified in SL 2.1.2 exceeds 110% of the RCS design pressure and may challenge system integrity.

The allowed Completion Time of 1 hour provides the operator time to complete the necessary actions to reduce RCS pressure by terminating the cause of the pressure increase, removing mass or energy from the RCS, or a combination of these actions, and to establish MODE 3 conditions.

2.2.2.2

If the RCS pressure SL is exceeded in MODE 3, 4, or 5, RCS pressure must be restored to within the SL value within 5 minutes.

Exceeding the RCS pressure SL in MODE 3, 4, or 5 is potentially more severe than exceeding this SL in MODE 1 or 2, since the reactor vessel temperature may be lower and the vessel material, consequently, less ductile. As such, pressure must be reduced to less than the SL within 5 minutes. This action does not require reducing MODES, since this would require reducing temperature, which would compound the problem by adding thermal gradient stresses to the existing pressure stress.

(continued)

BASES

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"

BASES (continued)

LCO 3.0.3 (continued)

Voluntary entry into LCO 3.0.3 is permissible but requires prior approval (approval may be verbal) from either the Operations Manager, Station Manager or corporate officer with direct responsibility for the plant. The approval must subsequently be documented in written retrievable manner. Inadvertent entry still allows for the one hour preparation period before Actions to change MODES must begin.

A unit shutdown required in accordance with LCO 3.0.3 may be terminated and LCO 3.0.3 exited if any of the following occurs:

a. The LCO is now met.

b. A Condition exists for which the Required Actions have now been performed.

c. ACTIONS exist that do not have expired Completion Times. These Completion Times are applicable from the point in time that the Condition is initially entered and not from the time LCO 3.0.3 is exited.

The time limits of Specification 3.0.3 allow 37 hours for the unit to be in MODE 5 when a shutdown is required during MODE 1 operation. If the unit is in a lower MODE of operation when a shutdown is required, the time limit for reaching the next lower MODE applies. If a lower MODE is reached in less time than allowed, however, the total allowable time to reach MODE 5, or other applicable MODE, is not reduced. For example, if MODE 3 is reached in 2 hours, then the time allowed for reaching MODE 4 is the next 11 hours, because the total time for reaching MODE 4 is not reduced from the allowable limit of 13 hours. Therefore, if remedial measures are completed that would permit a return to MODE 1, a penalty is not incurred by having to reach a lower MODE of operation in less than the total time allowed.

In MODES 1, 2, 3, and 4, LCO 3.0.3 provides actions for Conditions not covered in other Specifications. The requirements of LCO 3.0.3 do not apply in MODES 5 and 6 because the unit is already in the most restrictive Condition required by LCO 3.0.3.

The requirements of LCO 3.0.3 do not apply in other specified conditions of the Applicability (unless in MODE 1, 2, 3, or 4) because the ACTIONS of individual Specifications

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