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

SACRAMENTO MUNICIPAL UTILITY DISTRICT

DOCKET NO. 50-312

RANCHO SECO NUCLEAR GENERATING STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 113
License No. DPR-54

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

   A. The application for amendment by Sacramento Municipal Utility District (the licensee) dated June 21, 1988, as supplemented February 28, 1989, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-54 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 113, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment shall become effective within 30 days of the issuance date. The implementation delay is provided to allow time for modification of affected procedures and promulgation of the changes to personnel.

FOR THE NUCLEAR REGULATORY COMMISSION

George W. Knighton

George W. Knighton, Director
Project Directorate V
Division of Reactor Projects III,
IV, V and Special Projects
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 29, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 113

FACILITY OPERATING LICENSE NO. DPR-54

DOCKET NO. 50-312

Replace the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

Remove	Insert
3-30a	3-30a
3-30b	3-30b
3-40a	3-40a
3-45	3-45
3-46	3-46
4-18a	4-18a
4-75	4-75

INSTRUMENTS OPERATING CONDITIONS

Functional Unit	(A) Total Number of Channels	(B) Minimum Channels Operable	(C) Operator Action if Conditions of Columns A and B Cannot be Met
9. Reactor Building Purge Isolation and Reactor Building Equalization Isolation on High Radiation	2	2	With the number of operable channels less than the Minimum Channels Operable, reactor power operation may continue provided the purge valves are closed in accordance with Specification 3.6.7, the equalization valves are closed, and the ACTION stated in Table 3.5.5-1 for Accident Monitoring Instrumentation Operability Requirements, item 1 is taken.
10. Borated Water Storage Tank Level	2	1	See Section 3.5.1.2.
<u>Emergency Feedwater Initiation and Control (EFIC) System</u>			
1. AFW Initiation			
a. Manual	2 (Note 1)	2 (Note 1)	See Actions 3 and 4.
b. Low Level, SGA or B (Note 2)	4/SG (Note 1)	3/SG	See Actions 1, 2 and 3. May be bypassed below 750 psig OTSG pressure.
c. Low Pressure, SGA or B	4/SG (Note 1)	3/SG	See Actions 1, 2 and 3. May be bypassed below 750 psig OTSG pressure.
d. Loss of MFW Anticipatory Reactor Trip	4 (Note 1)	3	See Actions 1, 2 and 3. Loss of MFW Anticipatory Reactor Trip is effectively bypassed in RPS below 20 percent power.
e. Loss of 4 RC Pumps	4 (Note 1)	3	See Actions 1, 2 and 3. May be bypassed below 750 psig OTSG pressure.
f. Automatic Trip Logic	2 (Note 1)	2 (Note 1)	See Actions 3 and 4.

Note 1 For channel testing, calibration, or maintenance the Total Number of Channels and/or the Minimum Channels Operable may be reduced by one for a maximum of 6 hours providing the remaining channels are OPERABLE.
Note 2 Low level AFW Initiation has a maximum of a 10.0 second delay.

Table 3.5.1-1 (Continued)

INSTRUMENTS OPERATING CONDITIONS

Functional Unit	(A) Total Number of Channels	(B) Minimum Channels Operable	(C) Operator Action if Conditions of Columns A and B Cannot be Met
2. SG-A Main Feedwater Isolation			
a. Manual	2 (Note 1)	2 (Note 1)	See Actions 3 and 4.
b. Low SGA Pressure (Note 3)	4 (Note 1)	3	See Actions 1, 2 and 3. May be bypassed below 750 psig OTSG pressure.
c. Automatic Trip Logic	2 (Note 1)	2 (Note 1)	See Actions 3 and 4.
3. SG-B Main Feedwater Isolation			
a. Manual	2 (Note 1)	2 (Note 1)	See Actions 3 and 4.
b. Low SGB Pressure (Note 3)	4 (Note 1)	3	See Actions 1, 2 and 3. May be bypassed below 750 psig OTSG pressure.
c. Automatic Trip Logic	2 (Note 1)	2 (Note 1)	See Actions 3 and 4.
4. AFW Valve Commands (Vector)			
a. Vector Enable	2 (Note 1)	2 (Note 1)	See Actions 3 and 4.
b. Vector Module (Note 4)	4 (Note 1)	3	See Actions 1 and 5.
c. Control Enable	2 (Note 1)	2 (Note 1)	See Actions 1 and 3.
d. Control Module	2 (Note 1)	2 (Note 1)	See Actions 1 and 3.

Note 1 For channel testing, calibration, or maintenance the Total Number of Channels and/or the Minimum Channels Operable may be reduced by one for a maximum of 6 hours providing the remaining channels are OPERABLE.
Note 3 Low pressure AFW Initiation has a maximum of a 3.0 second delay.
Note 4 SG Pressure Difference AFW Valve Command (Vector) has a maximum of a 10.0 second delay.

3.6.7 The Reactor Building purge valves, SFV 53503, SFV 53504, SFV 53604, and SFV 53605, shall be closed with their respective breakers de-energized, except during cold shutdown or refueling. Valves SFV 53503 and SFV 53604 shall be verified to be in the above condition at least monthly. The breakers/disconnects on valves SFV 53504 and SFV 53605 shall be verified to be de-energized at least monthly.

3.6.8 The Reactor Building purge valves and Reactor Building pressure equalization valves shall isolate on high containment radiation level. See Table 3.5.1-1 for operability requirements.

Bases

The reactor coolant system conditions of cold shutdown assure that no steam will be formed and hence no pressure buildup in the containment if the reactor coolant system ruptures.

The selected shutdown conditions are based on the type of activities that are being carried out and will preclude criticality in any occurrence.

The Reactor Building is designed for an internal pressure of 59 psig and an external pressure 2.0 psi greater than the internal pressure. The design external pressure corresponds to the differential pressure that could be developed if the building is sealed with an internal temperature of 120°F with a barometric pressure of 29.0 inches of Hg and the building is subsequently cooled to an internal temperature of 80°F with a concurrent rise in barometric pressure to 31.0 inches of Hg.

When containment integrity is established, the limits of 10 CFR 100 will not be exceeded should the maximum hypothetical accident occur.

The OPERABILITY of the containment isolation ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere by pressurization of the containment. Containment isolation within the time limits specified ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for LOCA.

Specifications 3.6.7 and 3.6.8 are in response to NUREG 0737, item II.E.4.2.

REFERENCES

(1) USAR, section 5

3.8.8 When two irradiated fuel assemblies are being handled simultaneously within the fuel transfer canal, a minimum of 10 feet separation shall be maintained between the assemblies at all times. Irradiated fuel assemblies may be handled with the auxiliary bridge crane provided no other irradiated fuel assembly is being handled in the fuel transfer canal.

3.8.9 If any of the above specified limiting conditions for fuel loading and refueling are not met, movement of fuel into the reactor core shall cease; action shall be initiated to correct the conditions so that the specified limits are met, and no operations which may increase the reactivity of the core shall be made.

3.8.10 The Reactor Building purge system, including the Reactor Building Stack radiation monitor, shall be tested and verified to be operable within 100 hours prior to REFUELING OPERATIONS and once per 7 days during REFUELING OPERATIONS.

3.8.11 With the Reactor Building purge system or the Reactor Building Stack radiation monitor inoperable, close each of the Reactor Building purge system penetrations which provide direct access from the Reactor Building atmosphere to the outside atmosphere.

3.8.12 No loads will be handled over irradiated fuel stored in the spent fuel pool, except the fuel assemblies themselves. A dead weight load test at the rated load will be performed on the Fuel Storage Building handling bridge prior to each refueling.

3.8.13 Irradiated fuel shall not be removed from the reactor until the unit has been subcritical for at least 72 hours.

Bases

Detailed written procedures will be available for use by refueling personnel. These procedures, the above specifications, and the design of the fuel handling equipment, as described in Section 9.8 of the USAR, incorporating built-in interlocks and safety features, provide assurance that no incident could occur during the refueling operations that would result in a hazard to public health and safety. If no change is being made in core geometry, one flux monitor is sufficient. This permits maintenance on the instrumentation. Continuous monitoring of radiation levels and neutron flux provides immediate indication of an unsafe condition. The decay heat removal pump is used to maintain a uniform boron concentration.[1] The refueling boron concentration indicated in Specification 3.8.4 will be maintained to ensure that the more restrictive of the following reactivity conditions is met:

1. Either a k_{eff} of 0.95 or less with all control rods removed from the core.

2. A boron concentration of $\geq$1800 ppm.

The actual calculated boron concentration for item (1) above is 1974 ppm boron. Specification 3.8.5 allows the control room operator to inform the Reactor Building personnel of any impending unsafe condition detected from the main control board indicators during fuel movement.

The specification requiring testing Reactor Building purge termination is to verify that these components will function as required should a fuel handling accident occur that results in the release of significant fission products.

Specification 3.8.13 is required because the safety analysis for the fuel handling accident was based on the assumption that the reactor had been shut down for 72 hours and all 208 fuel pins in the hottest fuel assembly fail, releasing all gap activity.[2]

The requirement that at least one DHR loop be in operation ensures that (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 140°F as required during REFUELING OPERATION, and (2) sufficient coolant circulation is maintained through the reactor core to minimize the effect of a boron dilution incident and prevent boron stratification.

The requirement to have two DHR loops OPERABLE when there is less than 37 feet of water above the core ensures that a single failure of the operating DHR loop will not result in a complete loss of decay heat removal capability. With the reactor vessel head removed and 37 feet of water above the core, a large heat sink is available for core cooling. Thus, in the event of a failure of the operating DHR loop, adequate time is provided to initiate emergency procedures to cool the core.

REFERENCES

(1) USAR, Section 9.5

(2) USAR, paragraph 14.2.2.3.2

4.4.1.2.3 (Continued)

(d) The Containment purge valves shall be tested at least once every 6 months. The Containment equalization valves shall be tested at least once every 3 months.

(e) The Containment purge valves shall be tested prior to the initial purge on each cold shutdown and prior to reaching hot shutdown during heatup for a return to operation. A test conducted for this section may be applied to satisfy the requirement for a 6-month test of section (d) above if it is conducted within that interval. If the equalization valves are not tested with the purge valves under this section, their 3-month test requirement must still be met.

(f) At least once per 31 days by verifying that all penetrations not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions.

Exceptions to this are those valves listed in Table 3.6-1, and any other valves, blind flanges, and deactivated automatic valves which are located inside the containment and are locked, sealed or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except that such verification need not be performed more often than once per 92 days.

Table 4.22-1

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

Gaseous Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (a) (uCi/ml)
A. Waste Gas Storage Tank	P Each Tank Grab Sample	P Each Tank	Principal Gamma Emitters (f)	1×10^{-4}
B. Reactor Building Purge and Equalization Vent	P Each Purge and Equalization Vent Grab Sample(b,e,i)	P Each Purge and Equalization Vent (b,e,i)	Principal Gamma Emitters (f)	1×10^{-4}
			H-3	1×10^{-6}
C. Auxiliary Building Stack	M(b,c,e) Grab Sample	M(b)	Principal Gamma Emitters (f)	1×10^{-4}
			H-3	1×10^{-6}
D. Auxiliary Building Grade Level Vent	M(b) Grab Sample	M(b)	Principal Gamma Emitters (f)	1×10^{-4}
			H-3	1×10^{-6}
E. All Release Types as listed in A,B,C,D above	Continuous	W(d) Charcoal Sample	I-131	1×10^{-12}
	Continuous	W(d) Particulate Sample	Principal Gamma Emitters (f) (I-131, Others)	1×10^{-11}
	Continuous	M Composite Particulate Sample	Gross Alpha(h)	1×10^{-11}
			Sr-89, Sr-90(g)	1×10^{-11}
	Continuous	Noble Gas Monitor	Noble Gases Gross Beta and Gamma	1×10^{-4} as Xe-133