May 5, 1988

Docket No.: STN 50-528

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Mr. E. E. Van Brunt, Jr.
Executive Vice President
Arizona Nuclear Power Project
Post Office Box 52034
Phoenix, Arizona 85072-2034

Dear Mr. Van Brunt:

SUBJECT: ISSUANCE OF AMENDMENT NO. 31 TO FACILITY OPERATING LICENSE NO. NPF-41 FOR THE PALO VERDE NUCLEAR GENERATING STATION, UNIT NO. 1 (TAC NOS. 67104 AND 67146)

The Commission has issued the subject Amendment, which is enclosed, to the Facility Operating License for Palo Verde Nuclear Generating Station, Unit 1. The Amendment consists of changes to the Technical Specifications in response to your applications dated January 21 and February 2, 1988.

The Amendment revises Specification 3/4.4.8.3 to add a footnote to the applicability of the Specification for RCS temperatures between 255°F and 295°F. The amendment also clarifies the surveillance requirements and bases section for Specification 4.11.2.5 dealing with the monitoring of hydrogen and oxygen gases in the waste gas holdup system and with the automatic control features of the system. These changes make those Specifications consistent with the current Specifications for Palo Verde, Units 2 and 3.

A copy of the related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's next regular bi-weekly Federal Register notice.

Enclosure 3 provides four errata technical specification pages (3/4 4-20, 3/4 6-38, B 3/4 1-1 and B 3/4 8-2) and their overleaf pages to Amendment No. 27 to NPF-41, issued on March 2, 1988. It also corrects that page 3/4 8-48 has been deleted instead of 3/4 8-38.

Sincerely,

/S/

E. A. Licitra, Senior Project Manager
Project Directorate V
Division of Reactor Projects - III, IV, V and Special Projects

8805230261 880505
PDR ADOCK 05000528
P PDR

Enclosures:
1. Amendment No. 31 to NPF-41
2. Safety Evaluation
3. Errata to Amendment No. 27 to NPF-41

cc: See next page
*See previous concurrence

*DRSP/PDV	*DRSP/PDV	*OGC-WF	DRSP/D:PDV
EALicitra:dr	JLee	JMoore	GWKnighton
04/07/88	04/12/88	04/15/88	05/3/88 5/3/88

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

May 5, 1988

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E A Licitra

E. A. Licitra, Senior Project Manager
Project Directorate V
Division of Reactor Projects - III,
IV, V and Special Projects

Enclosures:
1. Amendment No. 31 to NPF-41
2. Safety Evaluation
3. Errata to Amendment No. 27 to NPF-41

cc: See next page

Mr. E. E. Van Brunt, Jr.
Arizona Nuclear Power Project

Palo Verde

cc:
Arthur C. Gehr, Esq.
Snell & Wilmer
3100 Valley Center
Phoenix, Arizona 85073

Mr. James M. Flenner, Chief Counsel
Arizona Corporation Commission
1200 West Washington
Phoenix, Arizona 85007

Charles R. Kocher, Esq. Assistant
Council
James A. Boeletto, Esq.
Southern California Edison Company
P. O. Box 800
Rosemead, California 91770

Mr. Mark Ginsberg
Energy Director
Office of Economic Planning
and Development
1700 West Washington - 5th Floor
Phoenix, Arizona 85007

Mr. Wayne Shirley
Assistant Attorney General
Bataan Memorial Building
Santa Fe, New Mexico 87503

Mr. Tim Polich
U.S. Nuclear Regulatory Commission
P. O. Box 97
Tonopah, Arizona 85354-0097

Regional Administrator, Region V
U. S. Nuclear Regulatory Commission
1450 Maria Lane
Suite 210
Walnut Creek, California 94596

Ms. Lynee Bernabei
Government Accountability Project
of the Institute for Policy Studies
1901 Que Street, NW
Washington, DC 20009

Mr. Ron Rayner
P. O. Box 1509
Goodyear, AZ 85338

Mr. Charles B. Brinkman, Manager
Washington Nuclear Operations
Combustion Engineering, Inc.
7910 Woodmont Avenue Suite 1310
Bethesda, Maryland 20814

cc:

Chairman
Arizona Corporation Commission
Post Office Box 6019
Phoenix, Arizona 85003

Arizona Radiation Regulatory Agency
ATTN: Ms. Clara Palovic, Librarian
4814 South 40 Street
Phoenix, Arizona 85040

Mr. Charles Tedford, Director
Arizona Radiation Regulatory Agency
4814 South 40 Street
Phoenix, Arizona 85040

Chairman
Maricopa County Board of Supervisors
111 South Third Avenue
Phoenix, Arizona 85003

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

ARIZONA PUBLIC SERVICE COMPANY, ET AL.

DOCKET NO. STN 50-528

PALO VERDE NUCLEAR GENERATING STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 31
License No. NPF-41

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

   A. The applications for amendment, dated January 21 and February 2, 1988, by the Arizona Public Service Company (APS) on behalf of itself and the Salt River Project Agricultural Improvement and Power District, El Paso Electric Company, Southern California Edison Company, Public Service Company of New Mexico, Los Angeles Department of Water and Power, and Southern California Public Power Authority (licensees), comply 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 applications, the provisions of 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;

   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 enclosure to this license amendment, and paragraph 2.C(2) of Facility Operating License No. NPF-41 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 31, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into this license. APS shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of issuance. The changes in the Technical Specifications are to become effective within 30 days of issuance of the amendment. In the period between issuance of the amendment and the effective date of the new Technical Specifications, the licensees shall adhere to the Technical Specifications existing at the time. The period of time during changeover shall be minimized.

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

Enclosure:
Changes to the Technical
Specifications

Date of Issuance: May 5, 1988

May 5, 1988

ENCLOSURE TO LICENSE AMENDMENT

AMENDMENT NO. 31 TO FACILITY OPERATING LICENSE NO. NPF-41

DOCKET NO. STN 50-528

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the areas of change. Also to be replaced are the following overleaf pages to the amended pages.

Amendment Pages	Overleaf Pages
3/4 4-32	3/4 4-31
3/4 11-14	3/4 11-13
B 3/4 11-5	B 3/4 11-6

PRESSURIZER HEATUP/COOLDOWN LIMITS

LIMITING CONDITION FOR OPERATION

3.4.8.2 The pressurizer temperature shall be limited to:

a. A maximum heatup rate of 200°F per hour, and

b. A maximum cooldown rate of 200°F per hour.

APPLICABILITY: At all times.

ACTION:

With the pressurizer temperature limits in excess of any of the above limits, restore the temperature to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the pressurizer; determine that the pressurizer remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the pressurizer pressure to less than 500 psig within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.8.2.1 The pressurizer temperatures shall be determined to be within the limits at least once per 30 minutes during system heatup or cooldown.

4.4.8.2.2 The spray water temperature differential shall be determined for use in Table 5.7-2 for each cycle of main spray with less than four reactor coolant pumps operating and for each cycle of auxiliary spray operation.

LIMITING CONDITION FOR OPERATION

3.4.8.3 Both shutdown cooling system (SCS) suction line relief valves with lift settings of less than or equal to 467 psig shall be OPERABLE and aligned to provide overpressure protection for the Reactor Coolant System.

APPLICABILITY: When the reactor vessel head is installed and the temperature of one or more of the RCS cold legs is less than or equal to:

a. 255°F during cooldown

b. 295°F* during heatup

ACTION:

a. With one SCS relief valve inoperable, restore the inoperable valve to OPERABLE status within seven days or reduce T_{cold} to less than 200°F and, depressurize and vent the RCS through a greater than or equal to 16 square inch vent(s) within the next eight hours. Do not start a reactor coolant pump if the steam generator secondary water temperature is greater than 100°F above any RCS cold leg temperature.

b. With both SCS relief valves inoperable, reduce T_{cold} to less than 200°F and, depressurize and vent the RCS through a greater than or equal to 16 square inch vent(s) within eight hours. Do not start a reactor coolant pump if the steam generator secondary water temperature is greater than 100°F above any RCS cold leg temperature.

c. In the event either the SCS suction line relief valves or an RCS vent(s) are used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the SCS suction line relief valves or RCS vent(s) on the transient and any corrective action necessary to prevent recurrence.

d. The provisions of Specification 3.0.4 are not applicable.

*255°F during heatup provided the heatup rate is limited to 10°F/hr or less for RCS temperature greater than 255°F and less than or equal to 295°F.

LIMITING CONDITION FOR OPERATION

3.11.2.4 The GASEOUS RADWASTE SYSTEM and the VENTILATION EXHAUST TREATMENT SYSTEM shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the projected gaseous effluent air doses due to gaseous effluent releases, from each reactor unit, from the site (see Figures 5.1-1 and 5.1-3), when averaged over 31 days, would exceed 0.2 mrad for gamma radiation and 0.4 mrad for beta radiation. The VENTILATION EXHAUST TREATMENT SYSTEM shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the projected doses due to gaseous effluent releases, from each reactor unit, to areas at and beyond the SITE BOUNDARY (see Figures 5.1-1 and 5.1-3) when averaged over 31 days would exceed 0.3 mrem to any organ of a MEMBER OF THE PUBLIC.

APPLICABILITY: At all times.

ACTION:

a. With radioactive gaseous waste being discharged without treatment and in excess of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which includes the following information:

   1. Identification of the inoperable equipment or subsystems and the reason for inoperability,

   2. Action(s) taken to restore the inoperable equipment to OPERABLE status, and

   3. Summary description of action(s) taken to prevent a recurrence.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.4 Doses due to gaseous releases from the site shall be projected at least once per 31 days, in accordance with the methodology and parameters in the ODCM.

LIMITING CONDITION FOR OPERATION

3.11.2.5 The concentration of oxygen in the waste gas holdup system shall be limited to less than or equal to 2% by volume whenever the hydrogen concentration exceeds 4% by volume.

APPLICABILITY: At all times.

ACTION:

a. With the concentration of oxygen in the waste gas holdup system greater than 2% by volume but less than or equal to 4% by volume, reduce the oxygen concentration to the above limit within 48 hours.

b. With the concentration of oxygen in the waste gas holdup system greater than 4% by volume, immediately suspend all additions of waste gases to the system and reduce the concentration of oxygen to less than 4% by volume within 6 hours.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.5 The concentration of hydrogen or oxygen in the waste gas holdup system shall be determined to be within the above limits by monitoring the waste gases in the waste gas holdup system in accordance with Specification 3.3.3.8.

GASEOUS RADWASTE TREATMENT (Continued)

This specification applies to the release of radioactive materials in gaseous effluents from each reactor unit at the site.

The minimum analysis frequency of 4/M (i.e. at least 4 times per month at intervals no greater than 9 days and a minimum of 48 times a year) is used for certain radioactive gaseous waste sampling in Table 4.11-2. This will eliminate taking double samples when quarterly and weekly samples are required at the same time.

3/4.11.2.5 EXPLOSIVE GAS MIXTURE

This specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the waste gas holdup system is maintained below the flammability limits of hydrogen and oxygen. (Automatic control features are included in the system to prevent the hydrogen and oxygen concentrations from reaching these flammability limits. These automatic control features include isolation of the source of hydrogen and/or oxygen, or injection of dilutants to reduce the concentration below the flammability limits.) Maintaining the concentration of hydrogen and oxygen below their flammability limits provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50.

3/4.11.2.6 GAS STORAGE TANKS

This specification considers postulated radioactive releases due to a waste gas system leak or failure, and limits the quantity of radioactivity contained in each pressurized gas storage tank in the GASEOUS RADWASTE SYSTEM to assure that a release would be substantially below the guidelines of 10 CFR Part 100 for a postulated event.

Restricting the quantity of radioactivity contained in each gas storage tank provides assurance that in the event of an uncontrolled release of the tank's contents, the resulting total body exposure to a MEMBER OF THE PUBLIC at the nearest exclusion area boundary will not exceed 0.5 rem. This is consistent with Standard Review Plan 11.3, Branch Technical Position ETSB 11-5, "Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure," in NUREG-0800, July 1981.

3/4.11.3 SOLID RADIOACTIVE WASTE

This specification addresses the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50. The process parameters included in establishing the PROCESS CONTROL PROGRAM may include, but are not limited to waste type, waste pH, waste/liquid/solidification agent/catalyst ratios, waste oil content, waste principal chemical constituents, and mixing and curing times.

3/4.11.4 TOTAL DOSE

This specification is provided to meet the dose limitations of 40 CFR Part 190 that have been incorporated into 10 CFR Part 20 by 46 FR 18525. The specification requires the preparation and submittal of a Special Report whenever the calculated doses from plant generated radioactive effluents and

TOTAL DOSE (Continued)

direct radiation exceed 25 mrems to the total body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrems. For sites containing up to four reactors, it is highly unlikely that the resultant dose to a MEMBER OF THE PUBLIC will exceed the dose limits of 40 CFR Part 190 if the individual reactors remain within twice the dose design objectives of Appendix I, and if direct radiation doses from the reactor units and outside storage tanks are kept small. The Special Report will describe a course of action that should result in the limitation of the annual dose to a MEMBER OF THE PUBLIC to within the 40 CFR Part 190 limits. For the purposes of the Special Report, it may be assumed that the dose commitment to the MEMBER OF THE PUBLIC from other uranium fuel cycle sources is negligible, with the exception that dose contributions from other nuclear fuel cycle facilities at the same site or within a radius of 8 km must be considered. If the dose to any MEMBER OF THE PUBLIC is estimated to exceed the requirements of 40 CFR Part 190, the Special Report with a request for a variance (provided the release conditions resulting in violation of 40 CFR Part 190 have not already been corrected), in accordance with the provisions of 40 CFR Part 190.11 and 10 CFR Part 20.405c, is considered to be a timely request and fulfills the requirements of 40 CFR Part 190 until NRC staff action is completed. The variance only relates to the limits of 40 CFR Part 190, and does not apply in any way to the other requirements for dose limitation of 10 CFR Part 20, as addressed in Specifications 3.11.1.1 and 3.11.2.1. An individual is not considered a MEMBER OF THE PUBLIC during any period in which he/she is engaged in carrying out any operation that is part of the nuclear fuel cycle.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 31 TO FACILITY OPERATING LICENSE NO. NPF-41

ARIZONA PUBLIC SERVICE COMPANY, ET AL.

PALO VERDE NUCLEAR GENERATING STATION, UNIT NO. 1

DOCKET NO. STN 50-528

1.0 INTRODUCTION

By letters dated January 21 and February 2, 1988, the Arizona Public Service Company (APS) on behalf of itself, the Salt River Project Agricultural Improvement and Power District, Southern California Edison Company, El Paso Electric Company, Public Service Company of New Mexico, Los Angeles Department of Water and Power, and Southern California Public Power Authority (licensees), requested changes to the Technical Specifications for the Palo Verde Nuclear Generating Station, Unit 1 (Appendix A to Facility Operating License No. NPF-41). The proposed changes would revise Specification 3/4.4.8.3 with regard to the applicability of the Specification for reactor coolant system temperatures between 255°F and 295°F, Specification 4.11.2.5 with regard to the surveillance requirements for hydrogen and oxygen monitoring of the waste gas holdup system, and Bases Section 3/4.11.2.5 with regard to the automatic control features of the waste gas holdup system.

2.0 DISCUSSION

The proposed changes to the Technical Specifications, as requested in the January 21 and February 2, 1988 submittals, are discussed below.

A. Technical Specification 3/4.4.8.3

This specification provides the limiting conditions for operation, surveillance requirements and action statements for the low temperature overpressure protection (LTOP) of the reactor coolant system (RCS) during system heatup below 295°F and system cooldown below 255°F. In the January 21, 1988 request, the licensees propose to add a footnote to allow a change in the upper limit temperature (i.e., 295°F) applicability in the specification during heatup to read as follows:

> "*255°F during heatup provided the heatup rate is limited to 10°F/hr or less for RCS temperatures greater than 255°F and less than or equal to 295°F."

The purpose of the proposed change is to eliminate confusion as to whether LTOP is required to be in service for the situation where the plant is cooled down below 295°F but not below 255°F (while cooling down in this range, LTOP operability is not required) and then reheated prior to reaching 255°F.

B. Technical Specification 4.11.2.5 and Bases Section 3/4.11.2.5

This specification provides the surveillance requirements for monitoring the concentration of hydrogen and oxygen in the waste gas holdup system to be within the limits of Specification 3.11.2.5. Specification 4.11.2.5 currently states that the concentrations shall be determined by:

> "...continuously monitoring the waste gases in the waste gas holdup system with the hydrogen and oxygen monitors required OPERABLE by Table 3.3-12 of Specification 3.3.3.8."

In the February 2, 1988 request, the licensees propose to revise the statement within the above quotes to read as follows:

> "...monitoring the waste gases in the waste gas holdup system in accordance with Specification 3.3.3.8."

The purpose of the proposed change is to account for the situation when the gas monitoring equipment is inoperable since Specification 3.11.2.5 does not provide any specified action when the monitoring equipment is not operable. Specification 3.3.3.8 does provide operability requirements for the gas monitoring equipment when the gaseous radwaste system is in operation and also provides for grab sample monitoring when the monitoring equipment is inoperable.

The other proposed change is to delete from the bases section of Specification 3/4.11.2.5 the reference to automatic diversion to recombiners in the description of the automatic control features of the waste gas holdup system since the system design was never intended to provide automatic diversion to the recombiners.

3.0 EVALUATION

The evaluation of the proposed changes to the Technical Specifications is presented below.

A. Technical Specification 3/4.4.8.3

Figure 3/4 3.4-2 in Specification 3/4.4.8 provides acceptable pressure/temperature limits and heatup and cooldown rates for the RCS to assure compliance with the requirements of Appendix G to

10 CFR Part 50 for overpressure protection. These limits are not based on the availability of LTOP. As shown on Figure 3/4 3.4-2, a heatup rate of 10°F/hr is permissible for all RCS temperatures above 255°F.

The licensees' proposed footnote to Specification 3/4.4.8.3, to allow an RCS heatup rate of 10°F/hr between 255°F and 295°F, with or without LTOP in service, is consistent with Figure 3/4 3.4-2 and in compliance with Appendix G to 10 CFR Part 50. Also, the proposed change does not affect any accident previously evaluated nor create a different kind of accident since it does not affect any of the assumptions used in the safety analyses.

On the basis of the above evaluation, the staff finds the proposed change to Specification 3/4.4.8.3 to be acceptable.

B. Technical Specification 4.11.2.5 and Bases Section 3/4.11.2.5

The operability requirements for the hydrogen and oxygen monitors for the waste gas holdup system are stated in Table 3.3-12 of Specification 3.3.3.8. Specifically Table 3.3-12 requires two channels for each monitor to be operable during gaseous radwaste system operation. When one or more channels become inoperable, operation of the gaseous radwaste system may continue provided grab samples are taken in accordance with the frequency specified in Action Statement 39.

Specification 4.11.2.5, which describes the surveillance requirements for hydrogen and oxygen monitoring of the waste gas holdup system, refers to Specification 3.3.3.8 to identify the monitors which are required to be operable for performing the surveillances. Since Specification 3.3.3.8 does not require the hydrogen and oxygen monitors to be operable when the gaseous waste system is not in operation, the licensees' proposed change to Specification 4.11.2.5 would clarify the requirement for monitoring continuously to when the gaseous waste system is in operation. In addition, the proposed change would provide the appropriate Action Statement in Specification 3.3.3.8 for when the monitoring channels are inoperable.

The proposed change to the bases section of Specification 3/4.11.2.5 is purely administrative to correct an inconsistency regarding the design capability of the waste gas holdup system.

On the basis of the above evaluation, the staff finds the proposed changes to Specification 4.11.2.5 and its bases section to be acceptable.

In summary, the staff has completed its review of the licensees' proposed Technical Specification changes submitted by letters dated

January 21 and February 2, 1988 and finds the changes to be acceptable. These changes also make these specifications consistent with the specifications for Palo Verde Units 2 and 3 which were previously reviewed and accepted by the staff prior to licensing of Units 2 and 3.

4.0 CONTACT WITH STATE OFFICIAL

The Arizona Radiation Regulatory Agency was advised of the proposed determination of no significant hazards consideration with regard to these changes. No comments were received.

5.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the installation or use of facility components located within the restricted area as defined in 10 CFR 20 and clarifies surveillance requirements and limiting conditions for operations. The staff has determined that this amendment involve no significant increase in the amount, and no significant change in the type, of any effluent that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued proposed findings that the amendment involves no significant hazard consideration, and there has been no public comment on such finding. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance of this amendment.

6.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public. We, therefore, conclude that the proposed changes are acceptable.

Principal contributor: E. A. Licitra

Dated: May 5, 1988

ELECTRIC HYDROGEN RECOMBINERS

LIMITING CONDITION FOR OPERATION

3.6.4.2 Two portable independent containment hydrogen recombiner systems shared among the three units shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

With one hydrogen recombiner system inoperable, restore the inoperable system to OPERABLE status within 30 days or meet the requirements of Specification 3.6.4.3, or be in at least HOT STANDBY within the next 6 hours.*

SURVEILLANCE REQUIREMENTS

4.6.4.2 Each hydrogen recombiner system shall be demonstrated OPERABLE:

a. At least once per 6 months by:

1. Verifying through a visual examination that there is no evidence of abnormal conditions within the recombiner enclosure and control console.

2. Operating the recombiner to include the air blast heat exchanger fan motor and enclosed blower motor continuously for at least 30 minutes at a temperature of approximately 800°F reaction chamber temperature.

b. At least once per year by performing a CHANNEL CALIBRATION of recombiner instrumentation to include a functional test of the recombiner at 1200°F (± 50°F) for at least four hours.

*Prior to March 30, 1986 or until the completion of the environmental qualification modifications to the hydrogen recombiner system, whichever occurs first, the provisions of Specification 3.0.4 are not applicable during implementation of the environmental qualification modifications to the hydrogen recombiner system when the containment hydrogen purge cleanup system described in Specification 3.6.4.3 is OPERABLE.

CONTAINMENT SYSTEMS

HYDROGEN PURGE CLEANUP SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.4.3 A containment hydrogen purge cleanup system, shared among the three units, shall be OPERABLE and capable of being powered from a minimum of one OPERABLE emergency bus.

APPLICABILITY: MODES 1* and 2*.

ACTION:

With the containment hydrogen purge cleanup system inoperable and one hydrogen recombiner OPERABLE as determined by Specification 4.6.4.2, restore the hydrogen purge cleanup system to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.4.3 The hydrogen purge cleanup system shall be demonstrated OPERABLE:

a. At least once per 31 days by initiating flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 15 minutes.

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:

   1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 50 scfm ± 10%.

   2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978,** meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.**

*With less than two hydrogen recombiners OPERABLE.
**ANSI N509-1980 is applicable for this specification.

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.5.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 gpm UNIDENTIFIED LEAKAGE,

c. 1 gpm total primary-to-secondary leakage through all steam generators, and 720 gallons per day through any one steam generator,

d. 10 gpm IDENTIFIED LEAKAGE from the Reactor Coolant System, and

e. 1 gpm leakage at a Reactor Coolant System pressure of 2250 ± 20 psia from any Reactor Coolant System pressure isolation valve specified in Table 3.4-1.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b.* With any Reactor Coolant System leakage greater than any one of the limits, excluding PRESSURE BOUNDARY LEAKAGE and leakage from Reactor Coolant System pressure isolation valves, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any Reactor Coolant System pressure isolation valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least one closed manual or deactivated automatic valve, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. With RCS leakage alarmed and confirmed in a flow path with no flow rate indicators, commence an RCS water inventory balance within 1 hour to determine the leak rate.

SURVEILLANCE REQUIREMENTS

4.4.5.2.1 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by:

a. Monitoring the containment atmosphere gaseous and particulate radioactivity monitor at least once per 12 hours.

*As a one time only extension during the power ascension program, an additional 72 hours is granted to cold shutdown. During this 72 hours if the unidentified leakage exceeds 2.0 gpm, an immediate cooldown will be initiated. The RCS leakage (Surveillance Requirement 4.4.5.2.1.c) will be calculated at least once per eight hours during this 72-hour extension.

SURVEILLANCE REQUIREMENTS (Continued)

b. Monitoring the containment sump inventory and discharge at least once per 12 hours.

c. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours**.

d. Monitoring the reactor head flange leakoff system at least once per 24 hours.

4.4.5.2.2 Each Reactor Coolant System pressure isolation valve specified in Table 3.4-1 shall be demonstrated OPERABLE by verifying leakage to be within its limit**:

a. At least once per 18 months,

b.* Prior to entering MODE 2 whenever the plant has been in COLD SHUTDOWN for 72 hours or more and if leakage testing has not been performed in the previous 9 months,

c. Prior to returning the valve to service following maintenance, repair or replacement work on the valve,

d.* Within 24 hours following valve actuation due to automatic or manual action or flow through the valve,

e.* Within 72 hours following a system response to an Engineered Safety Feature actuation signal.

*The provisions of Specifications 4.4.5.2.2.b, 4.4.5.2.2.d, and 4.4.5.2.2.e are not applicable for valves UV 651, UV 652, UV 653 and UV 654 due to position indication of valves in the control room.

**The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN AND K_{N-1}

The function of SHUTDOWN MARGIN is to ensure that the reactor remains subcritical following a design basis accident or anticipated operational occurrence. The function of K_{N-1} is to maintain sufficient subcriticality to preclude inadvertent criticality following ejection of a single control element assembly (CEA). During operation in MODES 1 and 2, with k_{eff} greater than or equal to 1.0, the transient insertion limits of Specification 3.1.3.6 ensure that sufficient SHUTDOWN MARGIN is available.

SHUTDOWN MARGIN is the amount by which the core is subcritical, or would be subcritical immediately following a reactor trip, considering a single malfunction resulting in the highest worth CEA failing to insert. K_{N-1} is a measure of the core's reactivity, considering a single malfunction resulting in the highest worth inserted CEA being ejected.

SHUTDOWN MARGIN requirements vary throughout the core life as a function of fuel depletion and reactor coolant system (RCS) cold leg temperature (T_{cold}). The most restrictive condition occurs at EOL, with T_{cold} at no-load operating temperature, and is associated with a postulated steam line break accident and the resulting uncontrolled RCS cooldown. In the analysis of this accident, the specified SHUTDOWN MARGIN is required to control the reactivity transient and ensure that the fuel performance and offsite dose criteria are satisfied. As (initial) T_{cold} decreases, the potential RCS cooldown and the resulting reactivity transient are less severe and, therefore, the required SHUTDOWN MARGIN also decreases. Below T_{cold} of about 210°F, the inadvertent deboration event becomes limiting with respect to the SHUTDOWN MARGIN requirements. Below 210°F, the specified SHUTDOWN MARGIN ensures that sufficient time for operator actions exists between the initial indication of the deboration and the total loss of shutdown margin. Accordingly, with at least one CEA partially or fully withdrawn, the SHUTDOWN MARGIN requirements are based upon these limiting conditions.

Additional events considered in establishing requirements on SHUTDOWN MARGIN that are not limiting with respect to the Specification limits are single CEA withdrawal and startup of an inactive reactor coolant pump.

K_{N-1} requirements vary with the amount of positive reactivity that would be introduced assuming the CEA with the highest inserted worth ejects from the core. In the analysis of the CEA ejection event, the K_{N-1} requirement ensures that the radially averaged enthalpy acceptance criterion is satisfied, considering power redistribution effects. Above T_{cold} of 500°F, Doppler reactivity feedback is sufficient to preclude the need for a specific K_{N-1} requirement. With all CEAs fully inserted, K_{N-1} and SHUTDOWN MARGIN requirements are equivalent in terms of minimum acceptable core boron concentration.

3/4.1.2 BORATION SYSTEMS

The boron injection system ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include (1) borated water sources, (2) charging pumps, (3) separate flow paths, (4) an emergency power supply from OPERABLE diesel generators, and (5) the volume control tank (VCT) outlet valve CH-UV-501, capable of isolating the VCT from the charging pump suction line. The nominal capacity of each charging pump is 44 gpm at its discharge. Up to 16 gpm of this may be diverted to the volume control tank via the RCP control bleedoff. Instrument inaccuracies and pump performance uncertainties are limited to 2 gpm yielding the 26 gpm value.

With the RCS temperature above 210°F, a minimum of two separate and redundant boron injection systems are provided to ensure single functional capability in the event an assumed failure renders one of the systems inoperable. Allowable out-of-service periods ensure that minor component repair or corrective action may be completed without undue risk to overall facility safety from injection system failures during the repair period.

Each system is capable of providing boration equivalent to a SHUTDOWN MARGIN of 4% delta k/k after xenon decay and cooldown to 210°F. Therefore, the boration capacity of either system is more than sufficient to satisfy the SHUTDOWN MARGIN and/or K_{N-1} requirements of the specifications. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires 23,800 gallons of 4000 ppm borated water from either the refueling water tank or the spent fuel pool.

With the RCS temperature below 210°F one injection system is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity changes in the event the single injection system becomes inoperable. The restrictions of one and only one operable charging pump whenever reactor coolant level is below the bottom of the pressurizer is based on the assumptions used in the analysis of the boron dilution event.

Each system is capable of providing boration equivalent to a SHUTDOWN MARGIN of 4% delta k/k. Therefore, the boration capacity of the system required below 210°F is more than sufficient to satisfy the shutdown margin and/or K_{N-1} requirements of the specifications. This condition requires 9,700 gallons of 4000 ppm borated water from either the refueling water tank or the spent fuel pool.

3/4.8.1, 3/4.8.2 and 3/4.8.3 A.C. SOURCES, D.C SOURCES and ONSITE POWER DISTRIBUTION SYSTEMS

The OPERABILITY of the A.C. and D.C. power sources and associated distribution systems during operation ensures that sufficient power will be available to supply the safety-related equipment required for (1) the safe shutdown of the facility and (2) the mitigation and control of accident conditions within the facility. The minimum specified independent and redundant A.C. and D.C. power sources and distribution systems satisfy the requirements of General Design Criterion 17 of Appendix "A" to 10 CFR 50.

The ACTION requirements specified for the levels of degradation of the power sources provide restriction upon continued facility operation commensurate with the level of degradation. The OPERABILITY of the power sources are consistent with the initial condition assumptions of the safety analyses and are based upon maintaining at least one redundant set of onsite A.C. and D.C. power sources and associated distribution systems OPERABLE during accident conditions coincident with an assumed loss-of-offsite power and single failure of the other onsite A.C. source.

The required steady state frequency for the emergency diesels is 60 + 1.2/ -0.3 Hz to be consistent with the safety analysis to provide adequate safety injection flow.

The OPERABILITY of the minimum specified A.C. and D.C. power sources and associated distribution systems during shutdown and refueling ensures that (1) the facility can be maintained in the shutdown or refueling condition for extended time periods and (2) sufficient instrumentation and control capability is available for monitoring and maintaining the unit status.

The surveillance requirements for demonstrating the OPERABILITY of the diesel generators are in accordance with the recommendations of Regulatory Guides 1.9 "Selection of Diesel Generator Set Capacity for Standby Power Supplies," March 10, 1971, and 1.108 "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants," Revision 1, August 1977.

A.C. SOURCES, D.C. SOURCES AND ONSITE POWER DISTRIBUTION SYSTEMS (Continued)

The surveillance requirement for demonstrating the OPERABILITY of the Station batteries are based on the recommendations of Regulatory Guide 1.129, "Maintenance Testing and Replacement of Large Lead Storage Batteries for Nuclear Power Plants," February 1978, and IEEE Std 450-1980, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations."

Verifying average electrolyte temperature above the minimum for which the battery was sized, total battery terminal voltage on float charge, connection resistance values and the performance of battery service and discharge tests ensures the effectiveness of the charging system, the ability to handle high discharge rates and compares the battery capacity at that time with the rated capacity.

Table 4.8-2 specifies the normal limits for each designated pilot cell and each connected cell for electrolyte level, float voltage and specific gravity. The limits for the designated pilot cells float voltage and specific gravity, greater than 2.13 volts and 0.010 below the manufacturer's full charge specific gravity or a battery charger current that had stabilized at a low value, is characteristic of a charged cell with adequate capacity. The normal limits for each connected cell for float voltage and specific gravity, greater than 2.13 volts and not more than 0.020 below the manufacturer's full charge specific gravity with an average specific gravity of all the connected cells not more than 0.010 below the manufacturer's full charge specific gravity, ensures the OPERABILITY and capability of the battery.

Operation with a battery cell's parameter outside the normal limit but within the allowable value specified in Table 4.8-2 is permitted for up to 7 days. During this 7-day period: (1) the allowable values for electrolyte level ensures no physical damage to the plates with an adequate electron transfer capability; (2) the allowable value for the average specific gravity of all the cells, not more than 0.020 below the manufacturer's recommended full charge specific gravity, ensures that the decrease in rating will be less than the safety margin provided in sizing; (3) the allowable value for an individual cell's specific gravity, ensures that an individual cell's specific gravity will not be more than 0.040 below the manufacturer's full charge specific gravity and that the overall capability of the battery will be maintained within an acceptable limit; and (4) the allowable value for an individual cell's float voltage, greater than 2.07 volts, ensures the battery's capability to perform its design function.

If any other metallic structures (e.g., buildings, new or modified piping systems, conduit) are placed in the ground in the vicinity of the fuel oil storage system or if the original system is modified, the adequacy and frequency of inspections of the cathodic protection system shall be re-evaluated and adjusted in accordance with Regulatory Guide 1.137.

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B 3/4 8-1	-
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B 3/4 9-1	B 3/4 9-2

*3/4 8-48 has been deleted.

TABLE 3.8-2 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	SERVICE DESCRIPTION
E-ZAA-C05 (FUSE)	E-PKA-D2114	STEAM GEN BLOWDOWN CTMT ISOLATION VALVE J-SGA-UV-500P
E-ZAA-C05 (FUSE)	E-PKA-D2114	BLOWDOWN SAMPLE CTMT ISOLATION VALVE J-SGA-UV-204
E-ZAA-C05 (FUSE)	E-PKA-D2114	BLOWDOWN SAMPLE CTMT ISOLATION VALVE J-SGA-UV-211
E-ZAA-C05 (FUSE)	E-PKA-D2114	BLOWDOWN SAMPLE CTMT ISOLATION VALVE J-SGA-UV-220
E-ZAA-C06 (FUSE)	E-PKA-D2121	SAFETY INJ TANK NITROGEN SUPPLY VALVE J-SIA-HV-619
E-ZAA-C06 (FUSE)	E-PKA-D2121	SAFETY INJ TANK NITROGEN SUPPLY VALVE J-SIA-HV-629
E-ZAA-C06 (FUSE)	E-PKA-D2121	SAFETY INJ TANK VENT VALVE J-SIA-HV-605
E-ZAA-C06 (FUSE)	E-PKA-D2121	SAFETY INJ TANK VENT VALVE J-SIA-HV-606
E-ZAA-C06 (FUSE)	E-PKA-D2121	SAFETY INJ TANK VENT VALVE J-SIA-HV-607
E-ZAA-C06 (FUSE)	E-PKA-D2121	SAFETY INJ TANK VENT VALVE J-SIA-HV-608
E-ZAA-C06 (FUSE)	E-PKA-D2121	RC SYSTEM VENT TO CTMT VALVE J-RCA-HV-106
E-ZAB-C03 (FUSE)	E-PKB-D2209	REGEN HEAT EXCH TO AUX SPRAY VALVE J-CHB-HV-203
E-ZAB-C03 (FUSE)	E-PKB-D2209	REACTOR COOLANT VENT VALVE J-RCB-HV-102
E-ZAB-C03 (FUSE)	E-PKB-D2209	SAFETY INJ TANK FILL AND DRAIN VALVE J-SIB-UV-611
E-ZAB-C03 (FUSE)	E-PKB-D2209	SI TANK CHECK VALVE LEAKAGE LINE ISO VALVE J-SIB-UV-618

TABLE 3.8-2 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	SERVICE DESCRIPTION
E-ZAB-C01 (FUSE)	E-PKB-D2210	CTMT ATM RADIATION MONITORING ISO VALVE J-HCB-UV-44
E-ZAB-C01 (FUSE)	E-PKB-D2210	CTMT ATM RADIATION MONITORING ISO VALVE J-HCB-UV-47
E-ZAB-C01 (FUSE)	E-PKB-D2210	CONTAINMENT POWER ACCESS PURGE MODE ISOLATION VALVE J-CPB-UV-5A
E-ZAB-C01 (FUSE)	E-PKB-D2210	CONTAINMENT POWER ACCESS PURGE MODE ISOLATION VALVE J-CPB-UV-5B
E-ZAB-C04 (FUSE)	E-PKB-D2202	REACTOR COOLANT VENT VALVE J-RCB-HV-108
E-ZAB-C04 (FUSE)	E-PKB-D2202	SAFETY INJ TANK FILL AND DRAIN VALVE J-SIB-UV-621
E-ZAB-C04 (FUSE)	E-PKB-D2202	SI TANK CHECK VALVE LEAKAGE LINE ISO VALVE J-SIB-UV-628
E-ZAB-C05 (FUSE)	E-PKB-D2214	REACTOR COOLANT VENT VALVE J-RCB-HV-109
E-ZAB-C05 (FUSE)	E-PKB-D2214	STEAM GEN BLOWDOWN CTMT ISOLATION VALVE J-SGB-UV-500R
E-ZAB-C05 (FUSE)	E-PKB-D2214	BLOWDOWN SAMPLE CTMT ISOLATION VALVE J-SGB-UV-222
E-ZAB-C05 (FUSE)	E-PKB-D2214	BLOWDOWN SAMPLE CTMT ISOLATION VALVE J-SGB-UV-224
E-ZAB-C05 (FUSE)	E-PKB-D2214	BLOWDOWN SAMPLE CTMT ISOLATION VALVE J-SGB-UV-226
E-ZAN-C01 (FUSE)	E-NKN-D4226	SEAL INJECT VALVES TO RCP J-CHE-FV-241
E-ZAN-C01 (FUSE)	E-NKN-D4224	SEAL INJECT VALVES TO RCP J-CHE-FV-242
E-ZAN-C01 (FUSE)	E-NKN-D4222	SEAL INJECT VALVES TO RCP J-CHE-FV-244