April 30, 199

Mr. William L. Stewart
Executive Vice President, Nuclear
Arizona Public Service Company
Post Office Box 53999
(Station No. 7601)
Phoenix, Arizona 85072-3999

SUBJECT: ISSUANCE OF AMENDMENTS FOR THE PALO VERDE NUCLEAR GENERATING STATION UNIT NO. 1 (TAC NO. M94630), UNIT NO. 2 (TAC NO. M94631), AND UNIT NO. 3 (TAC NO. M94632)

Dear Mr. Stewart:

The Commission has issued the enclosed Amendment No. 106 to Facility Operating License No. NPF-41, Amendment No. 98 to Facility Operating License No. NPF-51, and Amendment No. 78 to Facility Operating License No. NPF-74 for the Palo Verde Nuclear Generating Station, Unit Nos. 1, 2, and 3, respectively. The amendments consist of changes to the Technical Specifications (TS) in response to your application dated February 1, 1996.

These amendments revise (1) TS Sections 3/4.1.1.1, 6.9.1.9, and 6.9.1.10 to relocate the shutdown margin (reactor trip breakers open) to the Core Operating Limits Report (COLR); (2) TS 3/4.3.2 (Tables 3.3-3 and 3.3-4) to specify an additional restriction for the allowed low-pressurizer-pressure trip setpoint when reducing reactor coolant system (RCS) pressure in Mode 3; (3) TS Section 2.2.1 (Table 2.2-1) to make it consistent with the footnote in TS Tables 3.3-3 and 3.3-4; and (4) revise TS Sections 3/4.5.2 and 3/4.5.3 to require two emergency core cooling system (ECCS) subsystems to be operable in Mode 3 whenever the RCS cold-leg temperature is equal to or above 485°F. The Table of Contents and the Bases have also been revised to reflect these changes.

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

Sincerely,

Original Signed By
Charles R. Thomas, Project Manager
Project Directorate IV-2
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket Nos. STN 50-528, STN 50-529
and STN 50-530

Enclosures:
1. Amendment No. 106 to NPF-41
2. Amendment No. 98 to NPF-51
3. Amendment No. 78 to NPF-74
4. Safety Evaluation

cc w/encls: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

April 30, 1996

Mr. William L. Stewart
Executive Vice President, Nuclear
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Post Office Box 53999
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A copy of the related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,

Charles R. Thomas, Project Manager
Project Directorate IV-2
Division of Reactor Projects III/IV
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Docket Nos. STN 50-528, STN 50-529
and STN 50-530

Enclosures:
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4. Safety Evaluation

cc w/encls: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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. 106
License No. NPF-41

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

   A. The application for amendment by the Arizona Public Service Company (APS or the licensee) 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 dated February 1, 1996, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, 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. NPF-41 is hereby amended to read as follows:

(2) <u>Technical Specifications and Environmental Protection Plan</u>

The Technical Specifications contained in Appendix A, as revised through Amendment No. 106, 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, except where otherwise stated in specific license conditions.

3. This license amendment is effective as of the date of issuance to be implemented within 45 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Charles R. Thomas

Charles R. Thomas, Project Manager
Project Directorate IV-2
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: April 30, 1996

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 106 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 marginal lines indicating the areas of change. The corresponding overleaf pages are also provided to maintain document completeness.

REMOVE	INSERT
V*	V
VI	VI
2-5	2-5
B 2-4	B 2-4
3/4 1-1	3/4 1-1
3/4 3-23	3/4 3-23
3/4 3-27	3/4 3-27
3/4 5-3	3/4 5-3
3/4 5-4*	3/4 5-4
3/4 5-7	3/4 5-7
B 3/4 3-3	B 3/4 3-3
B 3/4 3-4	B 3/4 3-4
B 3/4 5-2	B 3/4 5-2
6-20a	6-20a

*No changes were made to these pages, reissued to become overleaf pages.

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

TABLE NOTATIONS

(1) Trip may be manually bypassed above 10^{-4}% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is less than or equal to 10^{-4}% of RATED THERMAL POWER.

(2) In MODES 3-4, the value may be decreased manually, to a minimum of 100 psia, as pressurizer pressure is reduced, provided:

(a) the margin between the pressurizer pressure and this value is maintained at less than or equal to 400 psi; and

(b) when the RCS cold leg temperature is greater than or equal to 485 degrees F, this value is maintained at least 140 psi greater than the saturation pressure corresponding to the RCS cold leg temperature.

The setpoint shall be increased automatically as pressurizer pressure is increased until the trip setpoint is reached. Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is greater than or equal to 500 psia.

(3) In MODES 3-4, value may be decreased manually as steam generator pressure is reduced, provided the margin between the steam generator pressure and this value is maintained at less than or equal to 200 psi; the setpoint shall be increased automatically as steam generator pressure is increased until the trip setpoint is reached.

(4) % of the distance between steam generator upper and lower level wide range instrument nozzles.

(5) As stored within the Core Protection Calculator (CPC). Calculation of the trip setpoint includes measurement, calculational and processor uncertainties. Trip may be manually bypassed below 10^{-4}% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is greater than or equal to 10^{-4}% of RATED THERMAL POWER.

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

TABLE NOTATIONS (Continued)

(6) RATE is the maximum rate of decrease of the trip setpoint. There are no restrictions on the rate at which the setpoint can increase.
FLOOR is the minimum value of the trip setpoint.
BAND is the amount by which the trip setpoint is below the input signal unless limited by Rate or Floor.
Setpoints are based on steam generator differential pressure.

(7) The setpoint may be altered to disable trip function during testing pursuant to Specification 3.10.3.

(8) RATE is the maximum rate of increase of the trip setpoint. (The rate at which the setpoint can decrease is no slower than five percent per second.)
CEILING is the maximum value of the trip setpoint.
BAND is the amount by which the trip setpoint is above the steady state input signal unless limited by the rate or the ceiling.

(9) % of the distance between steam generator upper and lower level narrow range instrument nozzles.

REACTOR TRIP SETPOINTS (Continued)

The methodology for the calculation of the PVNGS trip setpoint values, plant protection system, is discussed in the CE Document No. CEN-286(V), Rev. 2, dated August 29, 1986.

Manual Reactor Trip

The Manual reactor trip is a redundant channel to the automatic protective instrumentation channels and provides manual reactor trip capability.

Variable Overpower Trip

A reactor trip on Variable Overpower is provided to protect the reactor core during rapid positive reactivity addition excursions. This trip function will trip the reactor when the indicated neutron flux power exceeds either a rate limited setpoint at a great enough rate or reaches a preset ceiling. The flux signal used is the average of three linear subchannel flux signals originating in each nuclear instrument safety channel. These trip setpoints are provided in Table 2.2-1.

Logarithmic Power Level - High

The Logarithmic Power Level - High trip is provided to protect the integrity of fuel cladding and the Reactor Coolant System pressure boundary in the event of an unplanned criticality from a shutdown condition. A reactor trip is initiated by the Logarithmic Power Level - High trip unless this trip is manually bypassed by the operator. The operator may manually bypass this trip when the THERMAL POWER level is above 10-4% of RATED THERMAL POWER; this bypass is automatically removed when the THERMAL POWER level decreases to 10-4% of RATED THERMAL POWER.

Pressurizer Pressure - High

The Pressurizer Pressure - High trip, in conjunction with the pressurizer safety valves and main steam safety valves, provides Reactor Coolant System protection against overpressurization in the event of loss of load without reactor trip. This trip's setpoint is below the nominal lift setting of the pressurizer safety valves and its operation minimizes the undesirable operation of the pressurizer safety valves.

Pressurizer Pressure - Low

The Pressurizer Pressure - Low trip is provided to trip the reactor and to assist the Engineered Safety Features System in the event of a decrease in Reactor Coolant System inventory and in the event of an increase in heat

Pressurizer Pressure - Low (Continued)

removal by the secondary system. During normal operation, this trip's setpoint may be manually decreased, to a minimum value of 100 psia, as pressurizer pressure is reduced during plant shutdowns, provided the margin between the pressurizer pressure and this trip's setpoint is maintained at less than or equal to 400 psi; this setpoint increases automatically as pressurizer pressure increases until the trip setpoint is reached. The setpoint must also be maintained at least 140 psi greater than the saturation pressure corresponding to the RCS cold leg temperature whenever the RCS cold leg temperature is equal to or greater than 485 degrees F. This will ensure safety injection actuation prior to reactor vessel upper head void formation in event of RCS depressurization caused by a steam line break. These are indicated values that include allowances for uncertainty. The operator may manually bypass this trip when pressurizer pressure is below 400 psia. This bypass is automatically removed when the pressurizer pressure increases to 500 psia.

Containment Pressure - High

The Containment Pressure - High trip provides assurance that a reactor trip is initiated in the event of containment building pressurization due to a pipe break inside the containment building. The setpoint for this trip is identical to the safety injection setpoint.

Steam Generator Pressure - Low

The Steam Generator Pressure - Low trip provides protection in the event of an increase in heat removal by the secondary system and subsequent cooldown of the reactor coolant. The setpoint is sufficiently below the full load operating point so as not to interfere with normal operation, but still high enough to provide the required protection in the event of excessively high steam flow. This trip's setpoint may be manually decreased as steam generator pressure is reduced during plant shutdowns, provided the margin between the steam generator pressure and this trip's setpoint is maintained at less than or equal to 200 psi; this setpoint increases automatically as steam generator pressure increases until the normal pressure trip setpoint is reached.

Steam Generator Level - Low

The Steam Generator Level - Low trip provides protection against a loss of feedwater flow incident and assures that the design pressure of the Reactor Coolant System will not be exceeded due to a decrease in heat removal by the secondary system. This specified setpoint provides allowance that there will be sufficient water inventory in the steam generator at the time of the trip to provide a margin of at least 10 minutes before auxiliary feedwater is required to prevent degraded core cooling.

Local Power Density - High

The Local Power Density - High trip is provided to prevent the linear heat rate (kW/ft) in the limiting fuel rod in the core from exceeding the fuel design limit in the event of any design bases anticipated operational occurrence. The local power density is calculated in the reactor protective system utilizing the following information:

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - REACTOR TRIP BREAKERS OPEN**

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to that specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: MODES 3, 4*, and 5* with the reactor trip breaker open**.

ACTION:

With the SHUTDOWN MARGIN less than that specified in the CORE OPERATING LIMITS REPORT, immediately initiate and continue boration at greater than or equal to 26 gpm to reactor coolant system of a solution containing greater than or equal to 4000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to that specified in the CORE OPERATING LIMITS REPORT at least once per 24 hours by consideration of at least the following factors:

1. Reactor Coolant System boron concentration,
2. CEA position,
3. Reactor Coolant System average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 1.0% delta k/k at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1, above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 EFPD after each fuel loading.

4.1.1.1.3 With the reactor trip breakers open** and any CEA(s) fully or partially withdrawn, the SHUTDOWN MARGIN shall be verified within one hour after detection of the withdrawn CEA(s) and at least once per 12 hours thereafter while the CEA(s) are withdrawn.

* See Special Test Exception 3.10.9.
**The CEA drive system not capable of CEA withdrawal.

SHUTDOWN MARGIN - REACTOR TRIP BREAKERS CLOSED**

LIMITING CONDITION FOR OPERATION

3.1.1.2

a. The SHUTDOWN MARGIN shall be greater than or equal to that specified in the CORE OPERATING LIMITS REPORT, and

b. For T_{cold} less than or equal to 500°F, K_{N-1} shall be less than 0.99.

c. Reactor criticality shall not be achieved with shutdown group CEA movement.

APPLICABILITY: MODES 1, 2*, 3*, 4*, and 5* with the reactor trip breakers closed.**

ACTION:

a. With the SHUTDOWN MARGIN less than that specified in the CORE OPERATING LIMITS REPORT, immediately initiate and continue boration at greater than or equal to 26 gpm to the reactor coolant system of a solution containing greater than or equal to 4000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored, and

b. With T_{cold} less than or equal to 500°F and K_{N-1} greater than or equal to 0.99, immediately vary CEA positions and/or initiate and continue boration at greater than or equal to 26 gpm to the reactor coolant system of a solution containing greater than or equal to 4000 ppm boron or equivalent until the required K_{N-1} is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2.1 With the reactor trip breakers closed**, the SHUTDOWN MARGIN shall be determined to be greater than or equal to that specified in the CORE OPERATING LIMITS REPORT:

a. Within 1 hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable.

* See Special Test Exceptions 3.10.1 and 3.10.9
**The CEA drive system capable of CEA withdrawal.

TABLE NOTATIONS

(a) In MODES 3-4, the value may be decreased manually, to a minimum of 100 psia, as pressurizer pressure is reduced, provided:

(i) the margin between the pressurizer pressure and this value is maintained at less than or equal to 400 psi; and

(ii) when the RCS cold leg temperature is greater than or equal to 485 degress F, this value is maintained at least 140 psi greater than the saturation pressure corresponding to the RCS cold leg temperature.

The setpoint shall be increased automatically as pressurizer pressure is increased until the trip setpoint is reached. Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is greater than or equal to 500 psia.

(b) In MODES 3-4, the value may be decreased manually as steam generator pressure is reduced, provided the margin between the steam generator pressure and this value is maintained at less than or equal to 200 psi; the setpoint shall be increased automatically as steam generator pressure is increased until the trip setpoint is reached.

(c) Four channels provided, arranged in a selective two-out-of-four configuration (i.e., one-out-of-two taken twice).

(d) The proper two-out-of-four combination.

(e) Input to channels.

* The provisions of Specification 3.0.4 are not applicable.

ACTION STATEMENTS

ACTION 12 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 13 - With the number of channels OPERABLE one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may continue provided the inoperable channel is placed in the bypassed or tripped condition within 1 hour. If the inoperable channel is bypassed, the desirability of maintaining this channel in the bypassed condition shall be reviewed in accordance with Specification 6.5.1.6.g. The channel shall be returned to OPERABLE status no later than during the next COLD SHUTDOWN.

With a channel process measurement circuit that affects multiple functional units inoperable or in test, bypass or trip all associated functional units as listed below.

Process Measurement Circuit	
1. Steam Generator Pressure - Low	Steam Generator Pressure - Low Steam Generator Level 1-Low (ESF) Steam Generator Level 2-Low (ESF)
2. Steam Generator Level (Wide Range)	Steam Generator Level - Low (RPS) Steam Generator Level 1-Low (ESF) Steam Generator Level 2-Low (ESF)

TABLE 3.3-3 (Continued)

ACTION STATEMENTS (Continued)

ACTION 14 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE, STARTUP and/or POWER OPERATION may continue provided the following conditions are satisfied:

a. Verify that one of the inoperable channels has been bypassed and place the other inoperable channel in the tripped condition within 1 hour.

b. All functional units affected by the bypassed/tripped channel shall also be placed in the bypassed/tripped condition as listed below:

Process Measurement Circuit	Functional Unit Bypassed/Tripped
1. Steam Generator Pressure - Low	Steam Generator Pressure - Low Steam Generator Level 1 - Low (ESF) Steam Generator Level 2 - Low (ESF)
2. Steam Generator Level - Low (Wide Range)	Steam Generator Level - Low (RPS) Steam Generator Level 1 - Low (ESF) Steam Generator Level 2 - Low (ESF)

STARTUP and/or POWER OPERATION may continue until the performance of the next required CHANNEL FUNCTIONAL TEST. Subsequent STARTUP and/or POWER OPERATION may continue if one channel is restored to OPERABLE status and the provisions of ACTION 13 are satisfied.

ACTION 15 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

ACTION 16 - With the number of OPERABLE channels one less than the Total Number of Channels, be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours; however, one channel may be bypassed for up to 1 hour for surveillance testing provided the other channel is OPERABLE.

ACTION 17 - With the number of OPERABLE channels one less than the Minimum Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 18 - With the number of OPERABLE channels one less than the Minimum Number of Channels, operation may continue for up to 6 hours. After 6 hours operation may continue provided at least 1 train of essential filtration is in operation, otherwise, be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

TABLE 3.3-4 (Continued)

TABLE NOTATIONS

(1) In MODES 3-4, the value may be decreased manually, to a minimum of 100 psia, as pressurizer pressure is reduced, provided:

(a) the margin between the pressurizer presure and this value is maintained at less than or equal to 400 psi; and

(b) when the RCS cold leg temperature is greater than or equal to 485 degrees F, this value is maintained at least 140 psi greater than the saturation pressure corresponding to the RCS cold leg temperature.

The setpoint shall be increased automatically as pressurizer pressure is increased until the trip setpoint is reached. Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is greater than or equal to 500 psia.

(2) % of the distance between steam generator upper and lower level narrow range instrument nozzles.

(3) In MODES 3-4, value may be decreased manually as steam generator pressure is reduced, provided the margin between the steam generator pressure and this value is maintained at less than or equal to 200 psi; the setpoint shall be increased automatically as steam generator pressure is increased until the trip setpoint is reached.

(4) % of the distance between steam generator upper and lower level wide range instrument nozzles.

3/4.5.2 ECCS SUBSYSTEMS - OPERATING

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent Emergency Core Cooling System (ECCS) subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE high-pressure safety injection pump,

b. One OPERABLE low-pressure safety injection pump, and

c. An independent OPERABLE flow path capable of taking suction from the refueling water tank on a safety injection actuation signal and automatically transferring suction to the containment sump on a recirculation actuation signal.

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

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*With pressurizer pressure greater than or equal to 1837 psia, or RCS cold leg temperature greater than or equal to 485 degrees F.

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the following valves are in the indicated positions with the valves key-locked shut:

Valve Number	Valve Function	Valve Position
1. SIA HV-604	1. HOT LEG INJECTION	1. SHUT
2. SIC HV-321	2. HOT LEG INJECTION	2. SHUT
3. SIB HV-609	3. HOT LEG INJECTION	3. SHUT
4. SID HV-331	4. HOT LEG INJECTION	4. SHUT

b. At least once per 31 days by:

1. Verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position, and

2. Verifying that the ECCS piping is full of water by venting the accessible discharge piping high points.

c. By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment sump and cause restriction of the pump suctions during LOCA conditions. This visual inspection shall be performed:

1. For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY, and

2. At least once daily of the affected areas within containment by containment entry and during the final entry when CONTAINMENT INTEGRITY is established.

d. At least once per 18 months by:

3/4.5.3 ECCS SUBSYSTEMS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.5.3 As a minimum, one ECCS subsystem comprised of the following shall be OPERABLE:

a. An OPERABLE high pressure safety injection pump, and

b. An OPERABLE flow path capable of taking suction from the refueling water tank on a safety injection actuation signal and automatically transferring suction to the containment sump on a recirculation actuation signal.

APPLICABILITY: MODES 3* AND 4.

ACTION:

a. With no ECCS subsystem OPERABLE, restore at least one ECCS subsystem to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 20 hours.

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected safety injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

SURVEILLANCE REQUIREMENTS

4.5.3 The ECCS subsystem shall be demonstrated OPERABLE per the applicable surveillance requirements of Specification 4.5.2.

*With pressurizer pressure less than 1837 psia and RCS cold leg temperature less than 485 degrees F.

Response time may be demonstrated by any series of sequential, overlapping, or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either (1) in place, onsite, or offsite test measurements or (2) utilizing replacement sensors with certified response times.

During normal operation, the low pressurizer pressure trip setpoint may be manually decreased, to a minimum value of 100 psia, as pressurizer pressure is reduced during plant shutdowns, provided the margin between the pressurizer pressure and this trip's setpoint is maintained at less than or equal to 400 psi; this setpoint increases automatically as pressurizer pressure increases until the trip setpoint is reached. This setpoint must also be maintained at least 140 psi greater than the saturation pressure corresponding to the RCS cold leg temperature whenever the RCS cold leg temperature is equal to or greater than 485 degrees F. This will ensure safety injection actuation prior to reactor vessel upper head void formation in event of RCS depressurization caused by a steam line break. These are indicated values that include allowances for uncertainty. The operator may manually bypass the low pressurizer pressure trip when pressurizer pressure is below 400 psia. This bypass is automatically removed when the pressurizer pressure increases to 500 psia.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring channels ensures that: (1) the radiation levels are continually measured in the areas served by the individual channels and (2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded.

3/4.3.3.2 INCORE DETECTORS

The OPERABILITY of the incore detectors with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the reactor core.

3/4.3.3.3 SEISMIC INSTRUMENTATION

The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the facility to determine if plant shutdown is required pursuant to Appendix A of 10 CFR Part 100. The instrumentation is consistent with the recommendations of Regulatory Guide 1.12, "Instrumentation for Earthquakes," April 1974 as identified in the PVNGS FSAR. The seismic instrumentation for the site is located in Table 3.3-7.

3/4.3.3.4 METEOROLOGICAL INSTRUMENTATION

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data are available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public and is consistent with the recommendations of Regulatory Guide 1.23 "Onsite Meteorological Programs," February 1972. Wind speeds less than 0.6 MPH cannot be measured by the meteorological instrumentation.

3/4.3.3.5 REMOTE SHUTDOWN SYSTEM

The OPERABILITY of the remote shutdown system ensures that sufficient capability is available to permit safe shutdown and maintenance of HOT STANDBY of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criterion 19 of 10 CFR Part 50.

The parameters selected to be monitored ensure that (1) the condition of the reactor is known, (2) conditions in the RCS are known, (3) the steam generators are available for residual heat removal, (4) a source of water is available for makeup to the RCS, and (5) the charging system is available to makeup water to the RCS.

The OPERABILITY of the remote shutdown system insures that a fire will not preclude achieving safe shutdown. The remote shutdown system instrumentation, control and power circuits and disconnect switches necessary to eliminate effects of the fire and allow operation of instrumentation, control and power circuits required to achieve and maintain a safe shutdown condition are independent of areas where a fire could damage systems normally used to shutdown the reactor. This capability is consistent with General Design Criterion 3 and Appendix R to 10 CFR 50.

The alternate disconnect methods or power or control circuits ensure that sufficient capability is available to permit shutdown and maintenance of cold shutdown of the facility by relying on additional operator actions at local control stations rather than at the RSP.

3/4.3.3.6 POST-ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the post-accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Plants to Assess Plant Conditions During and Following an Accident," December 1975 and NUREG 0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations."

The containment high range area monitors (RU-148 & RU-149) and the main steamline radiation monitors (RU-139 A&B and RU-140 A&B) are in Table 3.3-6. The high range effluent monitors and samplers (RU-144 and RU-146) are in the ODCM. The containment hydrogen monitors are in Specification 3/4.6.5.1. The Post Accident Sampling System (RCS coolant) is in Table 3.3-6.

The Subcooled Margin Monitor (SMM), the Heat Junction Thermocouple (HJTC), and the Core Exit Thermocouples (CET) comprise the Inadequate Core Cooling (ICC) instrumentation required by Item II.F.2 NUREG-0737, the Post TMI-2 Action Plan. The function of the ICC instrumentation is to enhance the ability of the plant operator to diagnose the approach to existance of, and recovery from ICC. Additionally, they aid in tracking reactor coolant inventory. These instruments are included in the Technical Specifications at the request of NRC Generic Letter 83-37. These are not required by the accident analysis, nor to bring the plant to Cold Shutdown.

3/4.5.1 SAFETY INJECTION TANKS

The OPERABILITY of each of the Safety Injection System (SIS) safety injection tanks ensures that a sufficient volume of borated water will be immediately forced into the reactor core through each of the cold legs in the event the RCS pressure falls below the pressure of the safety injection tanks. This initial surge of water into the RCS provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on safety injection tank volume, boron concentration, and pressure ensure that the safety injection tanks will adequately perform their function in the event of a LOCA in MODE 1, 2, 3, or 4.

A minimum of 25% narrow range corresponding to 1790 cubic feet and a maximum of 75% narrow range corresponding to 1927 cubic feet of borated water are used in the safety analysis as the volume in the SITs. To allow for instrument accuracy, 28% narrow range corresponding to 1802 cubic feet and 72% narrow range corresponding to 1914 cubic feet, are specified in the Technical Specification.

A minimum of 593 psig and a maximum pressure of 632 psig are used in the safety analysis. To allow for instrument accuracy 600 psig minimum and 625 psig maximum are specified in the Technical Specification.

A boron concentration of 2000 ppm minimum and 4400 ppm maximum are used in the safety analysis. The Technical Specification lower limit of 2300 ppm in the SIT assures that the backleakage from RCS will not dilute the SITs below the 2000 ppm limit assumed in the safety analysis prior to the time when draining of the SIT is necessary.

The SIT isolation valves are not single failure proof; therefore, whenever the valves are open power shall be removed from these valves and the switch keylocked open. These precautions ensure that the SITs are available during a Limiting Fault.

The SIT nitrogen vent valves are not single failure proof against depressurizing the SITs by spurious opening. Therefore, power to the valves is removed while they are closed to ensure the safety analysis assumption of four pressurized SITs.

All of the SIT nitrogen vent valves are required to be operable so that, given a single failure, all four SITs may still be vented during post-LOCA long-term cooling. Venting the SITs provides for SIT depressurization capability which ensures the timely establishment of shutdown cooling entry conditions as assumed by the safety analysis for small break LOCAs.

The limits for operation with a safety injection tank inoperable for any reason except an isolation valve closed minimizes the time exposure of the plant to a LOCA event occurring concurrent with failure of an additional safety injection tank which may result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be immediately opened, the full capability of one safety injection tank is not available and prompt action is required to place the reactor in a MODE where this capability is not required.

For MODES 3 and 4 operation with pressurizer pressure less than 1837 psia the Technical Specifications require a minimum of 57% wide range corresponding

SAFETY INJECTION TANKS (Continued)

to 1361 cubic feet and a maximum of 75% narrow range corresponding to 1927 cubic feet of borated water per tank, when three safety injection tanks are operable and a minimum of 36% wide range corresponding to 908 cubic feet and a maximum of 75% narrow range corresponding to 1927 cubic feet per tank, when four safety injection tanks are operable at a minimum pressure of 235 psig and a maximum pressure of 625 psig. To allow for instrument inaccuracy, 60% wide range instrument corresponding to 1415 cubic feet, and 72% narrow range instrument corresponding to 1914 cubic feet, when three safety injection tanks are operable, and 39% wide range instrument corresponding to 962 cubic feet, and 72% narrow range instrument corresponding to 1914 cubic feet, when four SITs are operable, are specified in the Technical Specifications. To allow for instrument inaccuracy 254 psig is specified in the Technical Specifications.

The instrumentation vs. volume correlation for the SITs is as follows:

Volume	Narrow Range	Wide Range
962 ft^3	<0%	39%
1415 ft^3	<0%	60%
1802 ft^3	28%	78%
1914 ft^3	72%	83%

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

The OPERABILITY of two separate and independent ECCS subsystems with the indicated RCS pressure greater than or equal to 1837 psia, or with the indicated RCS cold leg temperature greater than or equal to 485 degrees F ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. These indicated values include allowances for uncertainties. Either subsystem operating in conjunction with the safety injection tanks is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double-ended break of the largest RCS cold leg pipe downward. In addition, each ECCS subsystem provides long-term core cooling capability in the recirculation mode during the accident recovery period.

The Mode 3 safety analysis credits one HPSI pump to provide negative reactivity insertion to protect the core and RCS following a steam line break when RCS cold leg temperature is 485 degrees F or greater. Requiring two operable ECCS subsystems in the situation will ensure one HPSI pump is available assuming single failure of the other HPSI pump.

With the RCS cold leg temperature below 485 degrees F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

The trisodium phosphate dodecahydrate (TSP) stored in dissolving baskets located in the containment basement is provided to minimize the possibility of corrosion cracking of certain metal components during operation of the ECCS following a LOCA. The TSP provided this protection by dissolving in the sump water and causing its final pH to be raised to greater than or equal to 7.0.

The surveillance requirements provided to ensure OPERABILITY of each component ensure that at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. Surveillance requirements for throttle valve position stops and flow balance testing provide

CORE OPERATING LIMITS REPORT

6.9.1.9 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle for the following:

a. Shutdown Margin - Reactor Trip Breakers Open for Specification 3.1.1.1
b. Shutdown Margin - Reactor Trip Breakers Closed for Specification 3.1.1.2
c. Moderator Temperature Coefficient BOL and EOL limits for Specification 3.1.1.3
d. Boron Dilution Alarms for Specification 3.1.2.7
e. Movable Control Assemblies - CEA Position for Specification 3.1.3.1
f. Regulating CEA Insertion Limits for Specification 3.1.3.6
g. Part Length CEA Insertion Limits for Specification 3.1.3.7
h. Linear Heat Rate for Specification 3.2.1
i. Azimuthal Power Tilt - T_q for Specification 3.2.3
j. DNBR Margin for Specification 3.2.4
k. Axial Shape Index for Specification 3.2.7
l. Boron Concentration (Mode 6) for Specification 3.9.1

6.9.1.10 The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC in:

a. "CE Method for Control Element Assembly Ejection Analysis, "CENPD-0190-A, January 1976 (Methodology for Specification 3.1.3.6, Regulating CEA Insertion Limits).

b. "The ROCS and DIT Computer Codes for Nuclear Design," CENPD-266-P-A, April 1983 [Methodology for Specifications 3.1.1.1, Shutdown Margin - Reactor Trip Breakers Open; 3.1.1.2, Shutdown Margin - Reactor Trip Breakers Closed; 3.1.1.3, Moderator Temperature Coefficient BOL and EOL limits; 3.1.3.6, Regulating CEA Insertion Limits and 3.9.1, Boron Concentration (Mode 6)].

c. "Safety Evaluation Report related to the Final Design of the Standard Nuclear Steam Supply Reference Systems CESSAR System 80, Docket No. STN 50-470, "NUREG-0852 (November 1981), Supplements No. 1 (March 1983), No. 2 (September 1983), No. 3 (December 1987) (Methodology for Specifications 3.1.1.2, Reactor Trip Breakers Closed; 3.1.1.3, Moderator Temperature Coefficient BOL and EOL limits; 3.1.2.7, Boron Dilution Alarms; 3.1.3.1, Movable Control Assemblies - CEA Position; 3.1.3.6, Regulating CEA Insertion Limits; 3.1.3.7, Part Length CEA Insertion Limits and 3.2.3 Azimuthal Power Tilt - T_q).

d. "Modified Statistical Combination of Uncertainties," CEN-356(V)-P-A Revision 01-P-A, May 1988 and "System 80™ Inlet Flow Distribution," Supplement 1-P to Enclosure 1-P to LD-82-054, February 1993 (Methodology for Specification 3.2.4, DNBR Margin and 3.2.7 Axial Shape Index).

CORE OPERATING LIMITS REPORT (Continued)

e. "Calculative Methods for the CE Large Break LOCA Evaluation Model for the Analysis of CE and W Designed NSSS," CENPD-132, Supplement 3-P-A, June 1985 (Methodology for Specification 3.2.1, Linear Heat Rate).

f. "Calculative Methods for the CE Small Break LOCA Evaluation Model," CENPD-137-P, August 1974 (Methodology for Specification 3.2.1, Linear Heat Rate).

g. "Calculative Methods for the CE Small Break LOCA Evaluation Model," CENPD-137-P, Supplement 1P, January 1977 (Methodology for Specification 3.2.1, Linear Heat Rate).

h. Letter: O. D. Parr (NRC) to F. M. Stern (CE), dated June 13, 1975 (NRC Staff Review of the Combustion Engineering ECCS Evaluation Model). NRC approval for: 6.9.1.10f.

i. Letter: K. Kniel (NRC) to A. E. Scherer (CE), dated September 27, 1977 (Evaluation of Topical Reports CENPD-133, Supplement 3-P and CENPD-137, Supplement 1-P). NRC approval for 6.9.1.10.g.

j. "Fuel Rod Maximum Allowable Pressure," CEN-372-P-A, May 1990 (Methodology for Specification 3.2.1, Linear Heat Rate).

k. Letter: A. C. Thadani (NRC) to A. E. Scherer (CE), dated April 10, 1990, ("Acceptance for Reference CE Topical Report CEN-372-P"). NRC approval for 6.9.1.10.j.

The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and analysis limits) of the safety analysis are met.

The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

ARIZONA PUBLIC SERVICE COMPANY, ET AL.

DOCKET NO. STN 50-529

PALO VERDE NUCLEAR GENERATING STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 98
License No. NPF-51

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

   A. The application for amendment by the Arizona Public Service Company (APS or the licensee) 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 dated February 1, 1996, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, 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. NPF-51 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. 98 , 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, except where otherwise stated in specific license conditions.

3. This license amendment is effective as of the date of issuance to be implemented within 45 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Charles R. Thomas, Project Manager
Project Directorate IV-2
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: April 30, 1996

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 98 TO FACILITY OPERATING LICENSE NO. NPF-51

DOCKET NO. STN 50-529

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change. The corresponding overleaf pages are also provided to maintain document completeness.

REMOVE	INSERT
V*	V
VI	VI
2-5	2-5
B 2-4	B 2-4
3/4 1-1	3/4 1-1
3/4 3-23	3/4 3-23
3/4 3-27	3/4 3-27
3/4 5-3	3/4 5-3
3/4 5-4	3/4 5-4
3/4 5-7	3/4 5-7
B 3/4 3-2	B 3/4 3-2
B 3/4 5-2	B 3/4 5-2
B 3/4 5-3	B 3/4 5-3
6-20a	6-20a

*No changes were made to these pages, reissued to become overleaf pages.

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

TABLE NOTATIONS

(1) Trip may be manually bypassed above 10^{-4}% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is less than or equal to 10^{-4}% of RATED THERMAL POWER.

(2) In MODES 3-4, the value may be decreased manually, to a minimum of 100 psia, as pressurizer pressure is reduced, provided:

(a) the margin between the pressurizer pressure and this value is maintained at less than or equal to 400 psi; and

(b) when the RCS cold leg temperature is greater than or equal to 485 degrees F, this value is maintained at least 140 psi greater than the saturation pressure corresponding to the RCS cold leg temperature.

The setpoint shall be increased automatically as pressurizer pressure is increased until the trip setpoint is reached. Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is greater than or equal to 500 psia.

(3) In MODES 3-4, value may be decreased manually as steam generator pressure is reduced, provided the margin between the steam generator pressure and this value is maintained at less than or equal to 200 psi; the setpoint shall be increased automatically as steam generator pressure is increased until the trip setpoint is reached.

(4) % of the distance between steam generator upper and lower level wide range instrument nozzles.

(5) As stored within the Core Protection Calculator (CPC). Calculation of the trip setpoint includes measurement, calculational and processor uncertainties. Trip may be manually bypassed below 10^{-4}% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is greater than or equal to 10^{-4}% of RATED THERMAL POWER.

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

TABLE NOTATIONS (Continued)

(6) RATE is the maximum rate of decrease of the trip setpoint. There are no restrictions on the rate at which the setpoint can increase.
FLOOR is the minimum value of the trip setpoint.
BAND is the amount by which the trip setpoint is below the input signal unless limited by Rate or Floor.
Setpoints are based on steam generator differential pressure.

(7) The setpoint may be altered to disable trip function during testing pursuant to Specification 3.10.3.

(8) RATE is the maximum rate of increase of the trip setpoint. (The rate at which the setpoint can decrease is no slower than five percent per second.)
CEILING is the maximum value of the trip setpoint.
BAND is the amount by which the trip setpoint is above the steady state input signal unless limited by the rate or the ceiling.

(9) % of the distance between steam generator upper and lower level narrow range instrument nozzles.

REACTOR TRIP SETPOINTS (Continued)

The methodology for the calculation of the PVNGS trip setpoint values, plant protection system, is discussed in the CE Document No. CEN-286(V), Rev. 2, dated August 29, 1986.

Manual Reactor Trip

The Manual reactor trip is a redundant channel to the automatic protective instrumentation channels and provides manual reactor trip capability.

Variable Overpower Trip

A reactor trip on Variable Overpower is provided to protect the reactor core during rapid positive reactivity addition excursions. This trip function will trip the reactor when the indicated neutron flux power exceeds either a rate limited setpoint at a great enough rate or reaches a preset ceiling. The flux signal used is the average of three linear subchannel flux signals originating in each nuclear instrument safety channel. These trip setpoints are provided in Table 2.2-1.

Logarithmic Power Level - High

The Logarithmic Power Level - High trip is provided to protect the integrity of fuel cladding and the Reactor Coolant System pressure boundary in the event of an unplanned criticality from a shutdown condition. A reactor trip is initiated by the Logarithmic Power Level - High trip unless this trip is manually bypassed by the operator. The operator may manually bypass this trip when the THERMAL POWER level is above 10^{-4}% of RATED THERMAL POWER; this bypass is automatically removed when the THERMAL POWER level decreases to 10^{-4}% of RATED THERMAL POWER.

Pressurizer Pressure - High

The Pressurizer Pressure - High trip, in conjunction with the pressurizer safety valves and main steam safety valves, provides Reactor Coolant System protection against overpressurization in the event of loss of load without reactor trip. This trip's setpoint is below the nominal lift setting of the pressurizer safety valves and its operation minimizes the undesirable operation of the pressurizer safety valves.

Pressurizer Pressure - Low

The Pressurizer Pressure - Low trip is provided to trip the reactor and to assist the Engineered Safety Features System in the event of a decrease in Reactor Coolant System inventory and in the event of an increase in heat

Pressurizer Pressure - Low (Continued)

removal by the secondary system. During normal operation, this trip's setpoint may be manually decreased, to a minimum value of 100 psia, as pressurizer pressure is reduced during plant shutdowns, provided the margin between the pressurizer pressure and this trip's setpoint is maintained at less than or equal to 400 psi; this setpoint increases automatically as pressurizer pressure increases until the trip setpoint is reached. The setpoint must also be maintained at least 140 psi greater than the saturation pressure corresponding to the RCS cold leg temperature whenever the RCS cold leg temperature is equal to or greater than 485 degrees F. This will ensure safety injection actuation prior to reactor vessel upper head void formation in event of RCS depressurization caused by a steam line break. These are indicated values that include allowances for uncertainty. The operator may manually bypass this trip when pressurizer pressure is below 400 psia. This bypass is automatically removed when the pressurizer pressure increases to 500 psia.

Containment Pressure - High

The Containment Pressure - High trip provides assurance that a reactor trip is initiated in the event of containment building pressurization due to a pipe break inside the containment building. The setpoint for this trip is identical to the safety injection setpoint.

Steam Generator Pressure - Low

The Steam Generator Pressure - Low trip provides protection in the event of an increase in heat removal by the secondary system and subsequent cooldown of the reactor coolant. The setpoint is sufficiently below the full load operating point so as not to interfere with normal operation, but still high enough to provide the required protection in the event of excessively high steam flow. This trip's setpoint may be manually decreased as steam generator pressure is reduced during plant shutdowns, provided the margin between the steam generator pressure and this trip's setpoint is maintained at less than or equal to 200 psi; this setpoint increases automatically as steam generator pressure increases until the normal pressure trip setpoint is reached.

Steam Generator Level - Low

The Steam Generator Level - Low trip provides protection against a loss of feedwater flow incident and assures that the design pressure of the Reactor Coolant System will not be exceeded due to a decrease in heat removal by the secondary system. This specified setpoint provides allowance that there will be sufficient water inventory in the steam generator at the time of the trip to provide a margin of at least 10 minutes before auxiliary feedwater is required to prevent degraded core cooling.

Local Power Density - High

The Local Power Density - High trip is provided to prevent the linear heat rate (kW/ft) in the limiting fuel rod in the core from exceeding the fuel design limit in the event of any design bases anticipated operational occurrence. The local power density is calculated in the reactor protective system utilizing the following information:

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - REACTOR TRIP BREAKERS OPEN**

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to that specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: MODES 3, 4* and 5* with the reactor trip breakers open.**

ACTION:

With the SHUTDOWN MARGIN less than that specified in the CORE OPERATING LIMITS REPORT, immediately initiate and continue boration at greater than or equal to 26 gpm to reactor coolant system of a solution containing greater than or equal to 4000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to that specified in the CORE OPERATING LIMITS REPORT at least once per 24 hours by consideration of at least the following factors:

1. Reactor Coolant System boron concentration,
2. CEA position,
3. Reactor Coolant System average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 1.0% delta k/k at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1, above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 EFPD after each fuel loading.

4.1.1.1.3 With the reactor trip breakers open** and any CEA(s) fully or partially withdrawn, the SHUTDOWN MARGIN shall be verified within one hour after detection of the withdrawn CEA(s) and at least once per 12 hours thereafter while the CEA(s) are withdrawn.

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* See Special Test Exception 3.10.9.
**The CEA drive system not capable of CEA withdrawal.

SHUTDOWN MARGIN - REACTOR TRIP BREAKERS CLOSED**

LIMITING CONDITION FOR OPERATION

3.1.1.2

a. The SHUTDOWN MARGIN shall be greater than or equal to that specified in the CORE OPERATING LIMITS REPORT, and

b. For T_{cold} less than or equal to 500°F, K_{N-1} shall be less than 0.99.

c. Reactor criticality shall not be achieved with shutdown group CEA movement.

APPLICABILITY: MODES 1, 2*, 3*, 4*, and 5* with the reactor trip breakers closed**.

ACTION:

a. With the SHUTDOWN MARGIN less than that specified in the CORE OPERATING LIMITS REPORT, immediately initiate and continue boration at greater than or equal to 26 gpm to the reactor coolant system of a solution containing greater than or equal to 4000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored, and

b. With T_{cold} less than or equal to 500°F and K_{N-1} greater than or equal to 0.99, immediately vary CEA positions and/or initiate and continue boration at greater than or equal to 26 gpm to the reactor coolant system of a solution containing greater than or equal to 4000 ppm boron or equivalent until the required K_{N-1} is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2.1 With the reactor trip breakers closed**, the SHUTDOWN MARGIN shall be determined to be greater than or equal to that specified in the CORE OPERATING LIMITS REPORT:

a. Within 1 hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable.

* See Special Test Exceptions 3.10.1 and 3.10.9.
**The CEA drive system capable of CEA withdrawal.

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

TABLE NOTATIONS

(a) In MODES 3-4, the value may be decreased manually, to a minimum of 100 psia, as pressurizer pressure is reduced, provided:

(i) the margin between the pressurizer pressure and this value is maintained at less than or equal to 400 psi; and

(ii) when the RCS cold leg temperature is greater than or equal to 485 degrees F, this value is maintained at least 140 psi greater than the saturation pressure corresponding to the RCS cold leg temperature.

The setpoint shall be increased automatically as pressurizer pressure is increased until the trip setpoint is reached. Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is greater than or equal to 500 psia.

(b) In MODES 3-4, the value may be decreased manually as steam generator pressure is reduced, provided the margin between the steam generator pressure and this value is maintained at less than or equal to 200 psi; the setpoint shall be increased automatically as steam generator pressure is increased until the trip setpoint is reached.

(c) Four channels provided, arranged in a selective two-out-of-four configuration (i.e., one-out-of-two taken twice).

(d) The proper two-out-of-four combination.

(e) Input to channels.

* The provisions of Specification 3.0.4 are not applicable.

ACTION STATEMENTS

ACTION 12 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 13 - With the number of channels OPERABLE one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may continue provided the inoperable channel is placed in the bypassed or tripped condition within 1 hour. If the inoperable channel is bypassed, the desirability of maintaining this channel in the bypassed condition shall be reviewed in accordance with Specification 6.5.1.6.g. The channel shall be returned to OPERABLE status no later than during the next COLD SHUTDOWN.

With a channel process measurement circuit that affects multiple functional units inoperable or in test, bypass or trip all associated functional units as listed below.

Process Measurement Circuit	
1. Steam Generator Pressure - Low	Steam Generator Pressure - Low Steam Generator Level 1-Low (ESF) Steam Generator Level 2-Low (ESF)
2. Steam Generator Level (Wide Range)	Steam Generator Level - Low (RPS) Steam Generator Level 1-Low (ESF) Steam Generator Level 2-Low (ESF)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

ACTION STATEMENTS

ACTION 14 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE, STARTUP and/or POWER OPERATION may continue provided the following conditions are satisfied:

a. Verify that one of the inoperable channels has been bypassed and place the other inoperable channel in the tripped condition within 1 hour.

b. All functional units affected by the bypassed/tripped channel shall also be placed in the bypassed/tripped condition as listed below:

Process Measurement Circuit	Functional Unit Bypassed/Tripped
1. Steam Generator Pressure - Low	Steam Generator Pressure - Low Steam Generator Level 1 - Low (ESF) Steam Generator Level 2 - Low (ESF)
2. Steam Generator Level - Low (Wide Range)	Steam Generator Level - Low (RPS) Steam Generator Level 1 - Low (ESF) Steam Generator Level 2 - Low (ESF)

STARTUP and/or POWER OPERATION may continue until the performance of the next required CHANNEL FUNCTIONAL TEST. Subsequent STARTUP and/or POWER OPERATION may continue if one channel is restored to OPERABLE status and the provisions of ACTION 13 are satisfied.

ACTION 15 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

ACTION 16 - With the number of OPERABLE channels one less than the Total Number of Channels, be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours; however, one channel may be bypassed for up to 1 hour for surveillance testing provided the other channel is OPERABLE.

ACTION 17 - With the number of OPERABLE channels one less than the Minimum Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 18 - With the number of OPERABLE channels one less than the Minimum Number of Channels, operation may continue for up to 6 hours. After 6 hours operation may continue provided at least 1 train of essential filtration is in operation, otherwise, be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

TABLE NOTATIONS

(1) In MODES 3-4, the value may be decreased manually, to a minimum of 100 psia, as pressurizer pressure is reduced, provided:

(a) the margin between the pressurizer pressure and this value is maintained at less than or equal to 400 psi; and

(b) when the RCS cold leg temperature is greater than or equal to 485 degrees F, this value is maintained at least 140 psi greater than the saturation pressure corresponding to the RCS cold leg temperature.

The setpoint shall be increased automatically as pressurizer pressure is increased until the trip setpoint is reached. Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is greater than or equal to 500 psia.

(2) % of the distance between steam generator upper and lower level narrow range instrument nozzles.

(3) In MODES 3-4, value may be decreased manually as steam generator pressure is reduced, provided the margin between the steam generator pressure and this value is maintained at less than or equal to 200 psi; the setpoint shall be increased automatically as steam generator pressure is increased until the trip setpoint is reached.

(4) % of the distance between steam generator upper and lower level wide range instrument nozzles.

3/4.5.2 ECCS SUBSYSTEMS - OPERATING

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent Emergency Core Cooling System (ECCS) subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE high-pressure safety injection pump,

b. One OPERABLE low-pressure safety injection pump, and

c. An independent OPERABLE flow path capable of taking suction from the refueling water tank on a safety injection actuation signal and automatically transferring suction to the containment sump on a recirculation actuation signal.

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

*With pressurizer pressure greater than or equal to 1837 psia, or RCS cold leg temperature greater than or equal to 485 degrees F.

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the following valves are in the indicated positions with the valves key-locked shut:

Valve Number	Valve Function	Valve Position
1. SIA HV-604	1. HOT LEG INJECTION	1. SHUT
2. SIC HV-321	2. HOT LEG INJECTION	2. SHUT
3. SIB HV-609	3. HOT LEG INJECTION	3. SHUT
4. SID HV-331	4. HOT LEG INJECTION	4. SHUT

b. At least once per 31 days by:

1. Verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position, and

2. Verifying that the ECCS piping is full of water by venting the accessible discharge piping high points.

c. By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment sump and cause restriction of the pump suctions during LOCA conditions. This visual inspection shall be performed:

1. For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY, and

2. At least once daily of the affected areas within containment by containment entry and during the final entry when CONTAINMENT INTEGRITY is established.

d. At least once per 18 months by:

3/4.5.3 ECCS SUBSYSTEMS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.5.3 As a minimum, one ECCS subsystem comprised of the following shall be OPERABLE:

a. An OPERABLE high pressure safety injection pump, and

b. An OPERABLE flow path capable of taking suction from the refueling water tank on a safety injection actuation signal and automatically transferring suction to the containment sump on a recirculation actuation signal.

APPLICABILITY: MODES 3* AND 4.

ACTION:

a. With no ECCS subsystem OPERABLE, restore at least one ECCS subsystem to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 20 hours.

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected safety injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

SURVEILLANCE REQUIREMENTS

4.5.3 The ECCS subsystem shall be demonstrated OPERABLE per the applicable surveillance requirements of Specification 4.5.2.

*With pressurizer pressure less than 1837 psia and RCS cold leg temperature less than 485 degrees F.

3/4.3.1 and 3/4.3.2 REACTOR PROTECTIVE AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

The OPERABILITY of the reactor protective and Engineered Safety Features Actuation Systems instrumentation and bypasses ensures that (1) the associated Engineered Safety Features Actuation action and/or reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its setpoint, (2) the specified coincidence logic is maintained, (3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and (4) sufficient system functional capability is available from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy, and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the safety analyses.

The quarterly frequency for the channel functional tests for these systems is based on the analyses presented in the NRC approved topical report CEN-327-A, "RPS/ESFAS Extended Test Interval Evaluation," and CEN-327-A, Supplement 1, and calculation 13-JC-SB-200-Rev. 01.

Response time testing of resistance temperature devices, which are a part of the reactor protective system, shall be performed by using in-situ loop current test techniques or another NRC approved method.

The Core Protection Calculator (CPC) addressable constants are provided to allow calibration of the CPC system to more accurate indications of power level, RCS flow rate, axial flux shape, radial peaking factors and CEA deviation penalties. Administrative controls on changes and periodic checking of addressable constant values (see also Technical Specifications 3.3.1 and 6.8.1) ensure that inadvertent misloading of addressable constants into the CPCs is unlikely.

The design of the Control Element Assembly Calculators (CEAC) provides reactor protection in the event one or both CEACs become inoperable. If one CEAC is in test or inoperable, verification of CEA position is performed at least every 4 hours. If the second CEAC fails, the CPCs in conjunction with plant Technical Specifications will use DNBR and LPD penalty factors and increased DNBR and LPD margin to restrict reactor operation to a power level that will ensure safe operation of the plant. If the margins are not maintained, a reactor trip will occur.

The value of the DNBR in Specification 2.1 is conservatively compensated for measurement uncertainties. Therefore, the actual RCS total flow rate determined by the reactor coolant pump differential pressure instrumentation or by calorimetric calculations does not have to be conservatively compensated for measurement uncertainties.

REACTOR PROTECTIVE AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

The measurement of response time at the specified frequencies provides assurance that the protective and ESF action function associated with each channel is completed within the time limit assumed in the safety analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable. The instrumentation response times are made up of the time to generate the trip signal at the detector (sensor response time) and the time for the signal to interrupt power to the CEA drive mechanism (signal or trip delay time).

Response time may be demonstrated by any series of sequential, overlapping, or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either (1) in place, onsite, or offsite test measurements or (2) utilizing replacement sensors with certified response times.

During normal operation, the low pressurizer pressure trip setpoint may be manually decreased, to a minimum value of 100 psia, as pressurizer pressure is reduced during plant shutdowns, provided the margin between the pressurizer pressure and this trip's setpoint is maintained at less than or equal to 400 psi; this setpoint increases automatically as pressurizer pressure increases until the trip setpoint is reached. This setpoint must also be maintained at least 140 psi greater than the saturation pressure corresponding to the RCS cold leg temperature whenever the RCS cold leg tempeature is equal to or greater than 485 degrees F. This will ensure safety injection actuation prior to reactor vessel upper head void formation in event of RCS depressurization caused by a steam line break. These are indicated values that include allowances for uncertainty. The operator may manually bypass the low pressurizer pressure trip when pressurizer pressure is below 400 psia. This bypass is automatically removed when the pressurizer pressure increases to 500 psia.

3/4.5.1 SAFETY INJECTION TANKS

The OPERABILITY of each of the Safety Injection System (SIS) safety injection tanks ensures that a sufficient volume of borated water will be immediately forced into the reactor core through each of the cold legs in the event the RCS pressure falls below the pressure of the safety injection tanks. This initial surge of water into the RCS provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on safety injection tank volume, boron concentration, and pressure ensure that the safety injection tanks will adequately perform their function in the event of a LOCA in MODE 1, 2, 3, or 4.

A minimum of 25% narrow range corresponding to 1790 cubic feet and a maximum of 75% narrow range corresponding to 1927 cubic feet of borated water are used in the safety analysis as the volume in the SITs. To allow for instrument accuracy, 28% narrow range corresponding to 1802 cubic feet and 72% narrow range corresponding to 1914 cubic feet, are specified in the Technical Specification.

A minimum of 593 psig and a maximum pressure of 632 psig are used in the safety analysis. To allow for instrument accuracy 600 psig minimum and 625 psig maximum are specified in the Technical Specification.

A boron concentration of 2000 ppm minimum and 4400 ppm maximum are used in the safety analysis. The Technical Specification lower limit of 2300 ppm in the SIT assures that the backleakage from RCS will not dilute the SITs below the 2000 ppm limit assumed in the safety analysis prior to the time when draining of the SIT is necessary

The SIT isolation valves are not single failure proof; therefore, whenever the valves are open power shall be removed from these valves and the switch keylocked open. These precautions ensure that the SITs are available during a Limiting Fault.

The SIT nitrogen vent valves are not single failure proof against depressurizing the SITs by spurious opening. Therefore, power to the valves is removed while they are closed to ensure the safety analysis assumption of four pressurized SITs.

All of the SIT nitrogen vent valves are required to be operable so that, given a single failure, all four SITs may still be vented during post-LOCA long-term cooling. Venting the SITs provides for SIT depressurization capability which ensures the timely establishment of shutdown cooling entry conditions as assumed by the safety analysis for small break LOCAs.

The limits for operation with a safety injection tank inoperable for any reason except an isolation valve closed minimizes the time exposure of the plant to a LOCA event occurring concurrent with failure of an additional safety injection tank which may result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be immediately opened, the full capability of one safety injection tank is not available and prompt action is required to place the reactor in a MODE where this capability is not required.

For MODES 3 and 4 operation with pressurizer pressure less than 1837 psia

SAFETY INJECTION TANKS (Continued)

the Technical Specifications require a minimum of 57% wide range corresponding to 1361 cubic feet and a maximum of 75% narrow range corresponding to 1927 cubic feet of borated water per tank, when three safety injection tanks are operable and a minimum of 36% wide range corresponding to 908 cubic feet and a maximum of 75% narrow range corresponding to 1927 cubic feet per tank, when four safety injection tanks are operable at a minimum pressure of 235 psig and a maximum pressure of 625 psig. To allow for instrument inaccuracy, 60% wide range instrument corresponding to 1415 cubic feet, and 72% narrow range instrument corresponding to 1914 cubic feet, when three safety injection tanks are operable, and 39% wide range instrument corresponding to 962 cubic feet, and 72% narrow range instrument corresponding to 1914 cubic feet, when four SITs are operable, are specified in the Technical Specifications. To allow for instrument inaccuracy 254 psig is specified in the Technical Specifications.

The instrumentation vs. volume correlation for the SITs is as follows:

Volume	Narrow Range	Wide Range
962 ft^3	<0%	39%
1415 ft^3	<0%	60%
1802 ft^3	28%	78%
1914 ft^3	72%	83%

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

The OPERABILITY of two separate and independent ECCS subsystems with the indicated RCS pressure greater than or equal to 1837 psia, or with the indicated RCS cold leg temperature greater than or equal to 485 degrees F ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. These indicated values include allowances for uncertainties. Either subsystem operating in conjunction with the safety injection tanks is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double-ended break of the largest RCS cold leg pipe downward. In addition, each ECCS subsystem provides long-term core cooling capability in the recirculation mode during the accident recovery period.

The Mode 3 safety analysis credits one HPSI pump to provide negative reactivity insertion to protect the core and RCS following a steam line break when RCS cold leg temperature is 485 degrees F or greater. Requiring two operable ECCS subsystems in the situation will ensure one HPSI pump is available assuming single failure of the other HPSI pump.

With the RCS cold leg temperature below 485 degrees F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

The trisodium phosphate dodecahydrate (TSP) stored in dissolving baskets located in the containment basement is provided to minimize the possibility of corrosion cracking of certain metal components during operation of the ECCS following a LOCA. The TSP provided this protection by dissolving in the sump water and causing its final pH to be raised to greater than or equal to 7.0.

ECCS SUBSYSTEMS (Continued)

The surveillance requirements provided to ensure OPERABILITY of each component ensure that at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. Surveillance requirements for throttle valve position stops and flow balance testing provide assurance that proper ECCS flows will be maintained in the event of a LOCA.* Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses. In specification 4.5.2.h, the specified flows include instrumentation uncertainties. The requirement to dissolve a representative sample of TSP in a sample of RWT water provides assurance that the stored TSP will dissolve in borated water at the postulated post-LOCA temperatures.

The term "minimum bypass recirculation flow," as used in Specification 4.5.2e.3. and 4.5.2f., refers to that flow directed back to the RWT from the ECCS pumps for pump protection. Testing of the ECCS pumps under the condition of minimum bypass recirculation flow in Specification 4.5.2f. verifies that the performance of the ECCS pumps supports the safety analysis minimum RCS pressure assumption at zero delivery to the RCS.

3/4.5.4 REFUELING WATER TANK

The OPERABILITY of the refueling water tank (RWT) as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on RWT minimum volume and boron concentration ensure that (1) sufficient water plus 10% margin is available to permit 20 minutes of engineered safety features pump operation, and (2) the reactor will remain subcritical in the cold condition following mixing of the RWT and the RCS water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses.

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*The following test conditions, which apply during flow balance tests, ensure that the ECCS subsystems are adequately tested.

1. The pressurizer pressure is at atmospheric pressure.
2. The miniflow bypass recirculation lines are aligned for injection.
3. For LPSI system, (add/subtract) 6.4 gpm (to/from) the 4800 gpm requirement for every foot by which the difference of RWT water level above the RWT RAS setpoint level (exceeds/is less than) the difference of RCS water level above the cold leg centerline.

REFUELING WATER TANK (Continued)

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

The limits on contained water volume and boron concentration of the RWT also ensure a pH value of between 7.0 and 8.5 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

The limit on the RWT solution temperature ensures that the assumptions used in the LOCA analyses remain valid.

CORE OPERATING LIMITS REPORT

6.9.1.9 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle for the following:

a. Shutdown Margin - Reactor Trip Breakers Open for Specification 3.1.1.1
b. Shutdown Margin - Reactor Trip Breakers Closed for Specification 3.1.1.2
c. Moderator Temperature Coefficient BOL and EOL limits for Specification 3.1.1.3
d. Boron Dilution Alarms for Specification 3.1.2.7
e. Movable Control Assemblies - CEA Position for Specification 3.1.3.1
f. Regulating CEA Insertion Limits for Specification 3.1.3.6
g. Part Length CEA Insertion Limits for Specification 3.1.3.7
h. Linear Heat Rate for Specification 3.2.1
i. Azimuthal Power Tilt - T_q for Specification 3.2.3
j. DNBR Margin for Specification 3.2.4
k. Axial Shape Index for Specification 3.2.7
l. Boron Concentration (Mode 6) for Specification 3.9.1

6.9.1.10 The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC in:

a. "CE Method for Control Element Assembly Ejection Analysis," CENPD-0190-A, January 1976 (Methodology for Specification 3.1.3.6, Regulating CEA Insertion Limits).

b. "The ROCS and DIT Computer Codes for Nuclear Design," CENPD-266-P-A, April 1983 [Methodology for Specifications 3.1.1.1, Shutdown Margin - Reactor Trip Breakers Open; 3.1.1.2, Shutdown Margin - Reactor Trip Breakers Closed; 3.1.1.3, Moderator Temperature Coefficient BOL and EOL limits; 3.1.3.6, Regulating CEA Insertion Limits and 3.9.1, Boron Concentration (Mode 6)].

c. "Safety Evaluation Report related to the Final Design of the Standard Nuclear Steam Supply Reference Systems CESSAR System 80, Docket No. STN 50-470, "NUREG-0852 (November 1981), Supplements No. 1 (March 1983), No. 2 (September 1983), No. 3 (December 1987) (Methodology for Specifications 3.1.1.2, Shutdown Margin - Reactor Trip Breakers Closed; 3.1.1.3, Moderator Temperature Coefficient BOL and EOL limits; 3.1.2.7, Boron Dilution Alarms; 3.1.3.1, Movable Control Assemblies - CEA Position; 3.1.3.6, Regulating CEA Insertion Limits; 3.1.3.7, Part Length CEA Insertion Limits and 3.2.3 Azimuthal Power Tilt - T_q).

d. "Modified Statistical Combination of Uncertainties," CEN-356(V)-P-A Revision 01-P-A, May 1988 and "System 80™ Inlet Flow Distribution," Supplement 1-P to Enclosure 1-P to LD-82-054, February 1993 (Methodology for Specification 3.2.4, DNBR Margin and 3.2.7 Axial Shape Index).

CORE OPERATING LIMITS REPORT (Continued)

e. "Calculative Methods for the CE Large Break LOCA Evaluation Model for the Analysis of CE and W Designed NSSS," CENPD-132, Supplement 3-P-A, June 1985 (Methodology for Specification 3.2.1, Linear Heat Rate).

f. "Calculative Methods for the CE Small Break LOCA Evaluation Model," CENPD-137-P, August 1974 (Methodology for Specification 3.2.1, Linear Heat Rate).

g. "Calculative Methods for the CE Small Break LOCA Evaluation Model," CENPD-137-P, Supplement 1P, January 1977 (Methodology for Specification 3.2.1, Linear Heat Rate).

h. Letter: O. D. Parr (NRC) to F. M. Stern (CE), dated June 13, 1975 (NRC Staff Review of the Combustion Engineering ECCS Evaluation Model). NRC approval for: 6.9.1.10f.

i. Letter: K. Kniel (NRC) to A. E. Scherer (CE), dated September 27, 1977 (Evaluation of Topical Reports CENPD-133, Supplement 3-P and CENPD-137, Supplement 1-P). NRC approval for 6.9.1.10.g.

j. "Fuel Rod Maximum Allowable Pressure," CEN-372-P-A, May 1990 (Methodology for Specification 3.2.1, Linear Heat Rate).

k. Letter: A. C. Thadani (NRC) to A. E. Scherer (CE), dated April 10, 1990, ("Acceptance for Reference CE Topical Report CEN-372-P"). NRC approval for 6.9.1.10.j.

The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and analysis limits) of the safety analysis are met.

The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

ARIZONA PUBLIC SERVICE COMPANY, ET AL.

DOCKET NO. STN 50-530

PALO VERDE NUCLEAR GENERATING STATION, UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 78
License No. NPF-74

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

   A. The application for amendment by the Arizona Public Service Company (APS or the licensee) 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 dated February 1, 1996, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, 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. NPF-74 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. 78 , 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, except where otherwise stated in specific license conditions.

3. This license amendment is effective as of its date of issuance to be implemented within 45 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Charles R. Thomas

Charles R. Thomas, Project Manager
Project Directorate IV-2
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: April 30, 1996

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 78 TO FACILITY OPERATING LICENSE NO. NPF-74

DOCKET NO. STN 50-530

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change. The corresponding overleaf pages are also provided to maintain document completeness.

REMOVE	INSERT
V*	V
VI	VI
2-5	2-5
B 2-4	B 2-4
3/4 1-1	3/4 1-1
3/4 3-23	3/4 3-23
3/4 3-27	3/4 3-27
3/4 5-3	3/4 5-3
3/4 5-4*	3/4 5-4
3/4 5-7	3/4 5-7
B 3/4 3-2	B 3/4 3-2
B 3/4 3-3	B 3/4 3-3
B 3/4 3-4	B 3/4 3-4
---	B 3/4 3-5
B 3/4 5-2	B 3/4 5-2
6-20a	6-20a

*No changes were made to these pages, reissued to become overleaf pages.

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

TABLE NOTATIONS

(1) Trip may be manually bypassed above 10^{-4}% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is less than or equal to 10^{-4}% of RATED THERMAL POWER.

(2) In MODES 3-4, the value may be decreased manually, to a minimum of 100 psia, as pressurizer pressure is reduced, provided:

(a) the margin between the pressurizer pressure and this value is maintained at less than or equal to 400 psi; and

(b) when the RCS cold leg temperature is greater than or equal to 485 degrees F, this value is maintained at least 140 psi greater than the saturation pressure corresponding to the RCS cold leg temperature.

The setpoint shall be increased automatically as pressurizer pressure is increased until the trip setpoint is reached. Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is greater than or equal to 500 psia.

(3) In MODES 3-4, value may be decreased manually as steam generator pressure is reduced, provided the margin between the steam generator pressure and this value is maintained at less than or equal to 200 psi; the setpoint shall be increased automatically as steam generator pressure is increased until the trip setpoint is reached.

(4) % of the distance between steam generator upper and lower level wide range instrument nozzles.

(5) As stored within the Core Protection Calculator (CPC). Calculation of the trip setpoint includes measurement, calculational and processor uncertainties. Trip may be manually bypassed below 10^{-4}% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is greater than or equal to 10^{-4}% of RATED THERMAL POWER.

(6) <u>RATE</u> is the maximum rate of decrease of the trip setpoint. There are no restrictions on the rate at which the setpoint can increase.
<u>FLOOR</u> is the minimum value of the trip setpoint.
<u>BAND</u> is the amount by which the trip setpoint is below the input signal unless limited by Rate or Floor.
Setpoints are based on steam generator differential pressure.

(7) The setpoint may be altered to disable trip function during testing pursuant to Specification 3.10.3.

(8) <u>RATE</u> is the maximum rate of increase of the trip setpoint. (The rate at which the setpoint can decrease is no slower than five percent per second.)
<u>CEILING</u> is the maximum value of the trip setpoint.
<u>BAND</u> is the amount by which the trip setpoint is above the steady state input signal unless limited by the rate or the ceiling.

(9) % of the distance between steam generator upper and lower level narrow range instrument nozzles.

REACTOR TRIP SETPOINTS (Continued)

The methodology for the calculation of the PVNGS trip setpoint values, plant protection system, is discussed in the CE Document No. CEN-286(V), Rev. 2, dated August 29, 1986.

Manual Reactor Trip

The Manual reactor trip is a redundant channel to the automatic protective instrumentation channels and provides manual reactor trip capability.

Variable Overpower Trip

A reactor trip on Variable Overpower is provided to protect the reactor core during rapid positive reactivity addition excursions. This trip function will trip the reactor when the indicated neutron flux power exceeds either a rate limited setpoint at a great enough rate or reaches a preset ceiling. The flux signal used is the average of three linear subchannel flux signals originating in each nuclear instrument safety channel. These trip setpoints are provided in Table 2.2-1.

Logarithmic Power Level - High

The Logarithmic Power Level - High trip is provided to protect the integrity of fuel cladding and the Reactor Coolant System pressure boundary in the event of an unplanned criticality from a shutdown condition. A reactor trip is initiated by the Logarithmic Power Level - High trip unless this trip is manually bypassed by the operator. The operator may manually bypass this trip when the THERMAL POWER level is above 10^{-4}% of RATED THERMAL POWER; this bypass is automatically removed when the THERMAL POWER level decreases to 10^{-4}% of RATED THERMAL POWER.

Pressurizer Pressure - High

The Pressurizer Pressure - High trip, in conjunction with the pressurizer safety valves and main steam safety valves, provides Reactor Coolant System protection against overpressurization in the event of loss of load without reactor trip. This trip's setpoint is below the nominal lift setting of the pressurizer safety valves and its operation minimizes the undesirable operation of the pressurizer safety valves.

Pressurizer Pressure - Low

The Pressurizer Pressure - Low trip is provided to trip the reactor and to assist the Engineered Safety Features System in the event of a decrease in Reactor Coolant System inventory and in the event of an increase in heat

Pressurizer Pressure - Low (Continued)

removal by the secondary system. During normal operation, this trip's setpoint may be manually decreased, to a minimum value of 100 psia, as pressurizer pressure is reduced during plant shutdowns, provided the margin between the pressurizer pressure and this trip's setpoint is maintained at less than or equal to 400 psi; this setpoint increases automatically as pressurizer pressure increases until the trip setpoint is reached. The setpoint must also be maintained at least 140 psi greater than the saturation pressure corresponding to the RCS cold leg temperature whenever the RCS cold leg temperature is equal to or greater than 485 degrees F. This will ensure safety injection actuation prior to reactor vessel upper head void formation in event of RCS depressurization caused by a steam line break. These are indicated values that include allowances for uncertainty. The operator may manually bypass this trip when pressurizer pressure is below 400 psia. This bypass is automatically removed when the pressurizer pressure increases to 500 psia.

Containment Pressure - High

The Containment Pressure - High trip provides assurance that a reactor trip is initiated in the event of containment building pressurization due to a pipe break inside the containment building. The setpoint for this trip is identical to the safety injection setpoint.

Steam Generator Pressure - Low

The Steam Generator Pressure - Low trip provides protection in the event of an increase in heat removal by the secondary system and subsequent cooldown of the reactor coolant. The setpoint is sufficiently below the full load operating point so as not to interfere with normal operation, but still high enough to provide the required protection in the event of excessively high steam flow. This trip's setpoint may be manually decreased as steam generator pressure is reduced during plant shutdowns, provided the margin between the steam generator pressure and this trip's setpoint is maintained at less than or equal to 200 psi; this setpoint increases automatically as steam generator pressure increases until the normal pressure trip setpoint is reached.

Steam Generator Level - Low

The Steam Generator Level - Low trip provides protection against a loss of feedwater flow incident and assures that the design pressure of the Reactor Coolant System will not be exceeded due to a decrease in heat removal by the secondary system. This specified setpoint provides allowance that there will be sufficient water inventory in the steam generator at the time of the trip to provide a margin of at least 10 minutes before auxiliary feedwater is required to prevent degraded core cooling.

Local Power Density - High

The Local Power Density - High trip is provided to prevent the linear heat rate (kW/ft) in the limiting fuel rod in the core from exceeding the fuel design limit in the event of any design bases anticipated operational occurrence. The local power density is calculated in the reactor protective system utilizing the following information:

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - REACTOR TRIP BREAKERS OPEN**

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to that specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: MODES 3, 4* and 5* with the reactor trip breakers open**.

ACTION:

With the SHUTDOWN MARGIN less than that specified in the CORE OPERATING LIMITS REPORT, immediately initiate and continue boration at greater than or equal to 26 gpm to reactor coolant system of a solution containing greater than or equal to 4000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to that specified in the CORE OPERATING LIMITS REPORT at least once per 24 hours by consideration of at least the following factors:

1. Reactor Coolant System boron concentration,
2. CEA position,
3. Reactor Coolant System average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 1.0% delta k/k at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1, above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 EFPD after each fuel loading.

4.1.1.1.3 With the reactor trip breakers open** and any CEA(s) fully or partially withdrawn, the SHUTDOWN MARGIN shall be verified within one hour after detection of the withdrawn CEA(s) and at least once per 12 hours thereafter while the CEA(s) are withdrawn.

* See Special Test Exception 3.10.9.
**The CEA drive system not capable of CEA withdrawal.

SHUTDOWN MARGIN - REACTOR TRIP BREAKERS CLOSED**

LIMITING CONDITION FOR OPERATION

3.1.1.2

a. The SHUTDOWN MARGIN shall be greater than or equal to that specified in the CORE OPERATING LIMITS REPORT, and

b. For T_{cold} less than or equal to 500°F, K_{N-1} shall be less than 0.99.

c. Reactor criticality shall not be achieved with shutdown group CEA movement.

APPLICABILITY: MODES 1, 2*, 3*, 4*, and 5* with the reactor trip breakers closed**.

ACTION:

a. With the SHUTDOWN MARGIN less than that specified in the CORE OPERATING LIMITS REPORT, immediately initiate and continue boration at greater than or equal to 26 gpm to the reactor coolant system of a solution containing greater than or equal to 4000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored, and

b. With T_{cold} less than or equal to 500°F and K_{N-1} greater than or equal to 0.99, immediately vary CEA positions and/or initiate and continue boration at greater than or equal to 26 gpm to the reactor coolant system of a solution containing greater than or equal to 4000 ppm boron or equivalent until the required K_{N-1} is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2.1 With the reactor trip breakers closed**, the SHUTDOWN MARGIN shall be determined to be greater than or equal to that specified in the CORE OPERATING LIMITS REPORT:

a. Within 1 hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable.

* See Special Test Exceptions 3.10.1 and 3.10.9.
**The CEA drive system capable of CEA withdrawal.

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

TABLE NOTATIONS

(a) In MODES 3-4, the value may be decreased manually, to a minimum of 100 psia, as pressurizer pressure is reduced, provided:

(i) the margin between the pressurizer pressure and this value is maintained at less than or equal to 400 psi; and

(ii) when the RCS cold leg temperature is greater than or equal to 485 degrees F, this value is maintained at least 140 psi greater than the saturation pressure corresponding to the RCS cold leg temperature.

The setpoint shall be increased automatically as pressurizer pressure is increased until the trip setpoint is reached. Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is greater than or equal to 500 psia.

(b) In MODES 3-4, the value may be decreased manually as steam generator pressure is reduced, provided the margin between the steam generator pressure and this value is maintained at less than or equal to 200 psi; the setpoint shall be increased automatically as steam generator pressure is increased until the trip setpoint is reached.

(c) Four channels provided, arranged in a selective two-out-of-four configuration (i.e., one-out-of-two taken twice).

(d) The proper two-out-of-four combination.

(e) Input to channels.

* The provisions of Specification 3.0.4 are not applicable.

ACTION STATEMENTS

ACTION 12 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 13 - With the number of channels OPERABLE one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may continue provided the inoperable channel is placed in the bypassed or tripped condition within 1 hour. If the inoperable channel is bypassed, the desirability of maintaining this channel in the bypassed condition shall be reviewed in accordance with Specification 6.5.1.6.g. The channel shall be returned to OPERABLE status no later than during the next COLD SHUTDOWN.

With a channel process measurement circuit that affects multiple functional units inoperable or in test, bypass or trip all associated functional units as listed below.

Process Measurement Circuit

1.	Steam Generator Pressure - Low	Steam Generator Pressure - Low Steam Generator Level 1-Low (ESF) Steam Generator Level 2-Low (ESF)
2.	Steam Generator Level (Wide Range)	Steam Generator Level - Low (RPS) Steam Generator Level 1-Low (ESF) Steam Generator Level 2-Low (ESF)

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

ACTION STATEMENTS

ACTION 14 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE, STARTUP and/or POWER OPERATION may continue provided the following conditions are satisfied:

a. Verify that one of the inoperable channels has been bypassed and place the other inoperable channel in the tripped condition within 1 hour.

b. All functional units affected by the bypassed/tripped channel shall also be placed in the bypassed/tripped condition as listed below:

Process Measurement Circuit	Functional Unit Bypassed/Tripped
1. Steam Generator Pressure - Low	Steam Generator Pressure - Low Steam Generator Level 1 - Low (ESF) Steam Generator Level 2 - Low (ESF)
2. Steam Generator Level - Low (Wide Range)	Steam Generator Level - Low (RPS) Steam Generator Level 1 - Low (ESF) Steam Generator Level 2 - Low (ESF)

STARTUP and/or POWER OPERATION may continue until the performance of the next required CHANNEL FUNCTIONAL TEST. Subsequent STARTUP and/or POWER OPERATION may continue if one channel is restored to OPERABLE status and the provisions of ACTION 13 are satisfied.

ACTION 15 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

ACTION 16 - With the number of OPERABLE channels one less than the Total Number of Channels, be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours; however, one channel may be bypassed for up to 1 hour for surveillance testing provided the other channel is OPERABLE.

ACTION 17 - With the number of OPERABLE channels one less than the Minimum Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 18 - With the number of OPERABLE channels one less than the Minimum Number of Channels, operation may continue for up to 6 hours. After 6 hours operation may continue provided at least 1 train of essential filtration is in operation, otherwise, be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

TABLE NOTATIONS

(1) In MODES 3-4, the value may be decreased manually, to a minimum of 100 psia, as pressurizer pressure is reduced, provided:

(a) the margin between the pressurizer pressure and this value is maintained at less than or equal to 400 psi; and

(b) when the RCS cold leg temperature is greater than or equal to 485 degrees F, this value is maintained at least 140 psi greater than the saturation pressure corresponding to the RCS cold leg temperature.

The setpoint shall be increased automatically as pressurizer pressure is increased until the trip setpoint is reached. Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is greater than or equal to 500 psia.

(2) % of the distance between steam generator upper and lower level narrow range instrument nozzles.

(3) In MODES 3-4, value may be decreased manually as steam generator pressure is reduced, provided the margin between the steam generator pressure and this value is maintained at less than or equal to 200 psi; the setpoint shall be increased automatically as steam generator pressure is increased until the trip setpoint is reached.

(4) % of the distance between steam generator upper and lower level wide range instrument nozzles.

EMERGENCY CORE COOLING SYSTEMS

3/4.5.2 ECCS SUBSYSTEMS - OPERATING

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent Emergency Core Cooling System (ECCS) subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE high-pressure safety injection pump,

b. One OPERABLE low-pressure safety injection pump, and

c. An independent OPERABLE flow path capable of taking suction from the refueling water tank on a safety injection actuation signal and automatically transferring suction to the containment sump on a recirculation actuation signal.

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

*With pressurizer pressure greater than or equal to 1837 psia, or RCS cold leg temperature greater than or equal or 485 degrees F.

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the following valves are in the indicated positions with the valves key-locked shut:

Valve Number	Valve Function	Valve Position
1. SIA HV-604	1. HOT LEG INJECTION	1. SHUT
2. SIC HV-321	2. HOT LEG INJECTION	2. SHUT
3. SIB HV-609	3. HOT LEG INJECTION	3. SHUT
4. SID HV-331	4. HOT LEG INJECTION	4. SHUT

b. At least once per 31 days by:

1. Verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position, and

2. Verifying that the ECCS piping is full of water by venting the accessible discharge piping high points.

c. By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment sump and cause restriction of the pump suctions during LOCA conditions. This visual inspection shall be performed:

1. For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY, and

2. At least once daily of the affected areas within containment by containment entry and during the final entry when CONTAINMENT INTEGRITY is established.

d. At least once per 18 months by:

EMERGENCY CORE COOLING SYSTEMS

3/4.5.3 ECCS SUBSYSTEMS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.5.3 As a minimum, one ECCS subsystem comprised of the following shall be OPERABLE:

a. An OPERABLE high pressure safety injection pump, and

b. An OPERABLE flow path capable of taking suction from the refueling water tank on a safety injection actuation signal and automatically transferring suction to the containment sump on a recirculation actuation signal.

APPLICABILITY: MODES 3* AND 4.

ACTION:

a. With no ECCS subsystem OPERABLE, restore at least one ECCS subsystem to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 20 hours.

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected safety injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

SURVEILLANCE REQUIREMENTS

4.5.3 The ECCS subsystem shall be demonstrated OPERABLE per the applicable surveillance requirements of Specification 4.5.2.

*With pressurizer pressure less than 1837 psia and RCS cold leg temperature less than 485 degrees F.

3/4.3.1 and 3/4.3.2 REACTOR PROTECTIVE AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

The OPERABILITY of the reactor protective and Engineered Safety Features Actuation Systems instrumentation and bypasses ensures that (1) the associated Engineered Safety Features Actuation action and/or reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its setpoint, (2) the specified coincidence logic is maintained, (3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and (4) sufficient system functional capability is available from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy, and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the safety analyses.

The quarterly frequency for the channel functional tests for these systems is based on the analyses presented in the NRC approved topical report CEN-327-A, "RPS/ESFAS Extended Test Interval Evaluation," and CEN-327-A, Supplement 1, and calculation 13-JC-SB-200-Rev. 01.

Response time testing of resistance temperature devices, which are a part of the reactor protective system, shall be performed by using in-situ loop current test techniques or another NRC approved method.

The Core Protection Calculator (CPC) addressable constants are provided to allow calibration of the CPC system to more accurate indications of power level, RCS flow rate, axial flux shape, radial peaking factors and CEA deviation penalties. Administrative controls on changes and periodic checking of addressable constant values (see also Technical Specifications 3.3.1 and 6.8.1) ensure that inadvertent misloading of addressable constants into the CPCs is unlikely.

The design of the Control Element Assembly Calculators (CEAC) provides reactor protection in the event one or both CEACs become inoperable. If one CEAC is in test or inoperable, verification of CEA position is performed at least every 4 hours. If the second CEAC fails, the CPCs in conjunction with plant Technical Specifications will use DNBR and LPD penalty factors and increased DNBR and LPD margin to restrict reactor operation to a power level that will ensure safe operation of the plant. If the margins are not maintained, a reactor trip will occur.

The value of the DNBR in Specification 2.1 is conservatively compensated for measurement uncertainties. Therefore, the actual RCS total flow rate determined by the reactor coolant pump differential pressure instrumentation or by calorimetric calculations does not have to be conservatively compensated for measurement uncertainties.

REACTOR PROTECTIVE AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

The measurement of response time at the specified frequencies provides assurance that the protective and ESF action function associated with each channel is completed within the time limit assumed in the safety analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable. The instrumentation response times are made up of the time to generate the trip signal at the detector (sensor response time) and the time for the signal to interrupt power to the CEA drive mechanism (signal or trip delay time).

Response time may be demonstrated by any series of sequential, overlapping, or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either (1) in place, onsite, or offsite test measurements or (2) utilizing replacement sensors with certified response times.

During normal operation, the low pressurizer pressure trip setpoint may be manually decreased, to a minimum value of 100 psia, as pressurizer pressure is reduced during plant shutdowns, provided the margin between the pressurizer pressure and this trip's setpoint is maintained at less than or equal to 400 psi; this setpoint increases automatically as pressurizer pressure increases until the trip setpoint is reached. This setpoint must also be maintained at least 140 psi greater than the saturation pressure corresponding to the RCS cold leg temperature whenever the RCS cold leg temperature is equal to or greater than 485 degrees F. This will ensure safety injection actuation prior to reactor vessel upper head void formation in event of RCS depressurization caused by a steam line break. These are indicated values that include allowances for uncertainty. The operator may manually bypass the low pressurizer pressure trip when pressurizer pressure is below 400 psia. This bypass is automatically removed when the pressurizer pressure increases to 500 psia.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring channels ensures that: (1) the radiation levels are continually measured in the areas served by the individual channels and (2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded.

3/4.3.3.2 INCORE DETECTORS

The OPERABILITY of the incore detectors with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the reactor core.

3/4.3.3.3 SEISMIC INSTRUMENTATION

The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the facility to determine if plant shutdown is required pursuant to Appendix A of 10 CFR Part 100. The instrumentation is consistent with the recommendations of Regulatory Guide 1.12, "Instrumentation for Earthquakes," April 1974 as identified in the PVNGS FSAR. The seismic instrumentation for the site is listed in Table 3.3-7.

3/4.3.3.4 METEOROLOGICAL INSTRUMENTATION

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data are available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public and is consistent with the recommendations of Regulatory Guide 1.23 "Onsite Meteorological Programs," February 1972. Wind speeds less than 0.6 MPH cannot be measured by the meteorological instrumentation.

3/4.3.3.5 REMOTE SHUTDOWN SYSTEM

The OPERABILITY of the remote shutdown system ensures that sufficient capability is available to permit safe shutdown and maintenance of HOT STANDBY of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criterion 19 of 10 CFR Part 50.

The parameters selected to be monitored ensure that (1) the condition of the reactor is known, (2) conditions in the RCS are known, (3) the steam generators are available for residual heat removal, (4) a source of water is available for makeup to the RCS, and (5) the charging system is available to makeup water to the RCS.

The OPERABILITY of the remote shutdown system insures that a fire will not preclude achieving safe shutdown. The remote shutdown system instrumentation, control and power circuits and disconnect switches necessary to eliminate effects of the fire and allow operation of instrumentation, control and power circuits required to achieve and maintain a safe shutdown condition are independent of areas where a fire could damage systems normally used to shutdown the reactor. This capability is consistent with General Design Criterion 3 and Appendix R to 10 CFR 50.

The alternate disconnect methods or power or control circuits ensure that sufficient capability is available to permit shutdown and maintenance of cold shutdown of the facility by relying on additional operator actions at local control stations rather than at the RSP.

3/4.3.3.6 POST-ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the post-accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Plants to Assess Plant Conditions During and Following an Accident," December 1975 and NUREG 0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations."

The containment high range area monitors (RU-148 & RU-149) and the main steamline radiation monitors (RU-139 A&B and RU-140 A&B) are in Table 3.3-6. The high range effluent monitors and samplers (RU-144 and RU-146) are in the ODCM. The containment hydrogen monitors are in Specification 3/4.6.4.1. The Post Accident Sampling System (RCS coolant) is in Table 3.3-6.

The Subcooled Margin Monitor (SMM), the Heat Junction Thermocouple (HJTC), and the Core Exit Thermocouples (CET) comprise the Inadequate Core Cooling (ICC) instrumentation required by Item II.F.2 NUREG-0737, the Post TMI-2 Action Plan. The function of the ICC instrumentation is to enhance the ability of the plant operator to diagnose the approach to existance of, and recovery from ICC. Additionally, they aid in tracking reactor coolant inventory. These instruments are included in the Technical Specifications at the request of NRC Generic Letter 83-37. These are not required by the accident analysis, nor to bring the plant to Cold Shutdown.

In the event more than four sensors in a Reactor Vessel Level channel are inoperable, repairs may only be possible during the next refueling outage. This is because the sensors are accessible only after the missile shield and reactor vessel head are removed. It is not feasible to repair a channel except during a refueling outage when the missile shield and reactor vessel head are removed to refuel the core. If both channels are inoperable, the channels shall be restored to OPERABLE status in the nearest refueling outage. If only one channel is inoperable, it is intended that this channel be restored to OPERABLE status in a refueling outage as soon as reasonably possible.

3/4.3.3.7 LOOSE-PART DETECTION INSTRUMENTATION

The OPERABILITY of the loose-part detection instrumentation ensures that sufficient capability is available to detect loose metallic parts in the primary system and avoid or mitigate damage to primary system components. The allowable out-of-service times and surveillance requirements are consistent with the recommendations of Regulatory Guide 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors," May 1981.

3/4.3.3.8 EXPLOSIVE GAS MONITORING INSTRUMENTATION

The explosive gas instrumentation is provided for monitoring (and controlling) the concentrations of potentially explosive gas mixtures in the GASEOUS RADWASTE SYSTEM. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

BASES

3/4.5.1 SAFETY INJECTION TANKS

The OPERABILITY of each of the Safety Injection System (SIS) safety injection tanks ensures that a sufficient volume of borated water will be immediately forced into the reactor core through each of the cold legs in the event the RCS pressure falls below the pressure of the safety injection tanks. This initial surge of water into the RCS provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on safety injection tank volume, boron concentration, and pressure ensure that the safety injection tanks will adequately perform their function in the event of a LOCA in MODE 1, 2, 3, or 4.

A minimum of 25% narrow range corresponding to 1790 cubic feet and a maximum of 75% narrow range corresponding to 1927 cubic feet of borated water are used in the safety analysis as the volume in the SITs. To allow for instrument accuracy, 28% narrow range corresponding to 1802 cubic feet and 72% narrow range corresponding to 1914 cubic feet, are specified in the Technical Specification.

A minimum of 593 psig and a maximum pressure of 632 psig are used in the safety analysis. To allow for instrument accuracy, 600 psig minimum and 625 psig maximum are specified in the Technical Specification.

A boron concentration of 2000 ppm minimum and 4400 ppm maximum are used in the safety analysis. The Technical Specification lower limit of 2300 ppm in the SIT assures that the backleakage from RCS will not dilute the SITs below the 2000 ppm limit assumed in the safety analysis prior to the time when draining of the SIT is necessary.

The SIT isolation valves are not single failure proof; therefore, whenever the valves are open power shall be removed from these valves and the switch keylocked open. These precautions ensure that the SITs are available during a Limiting Fault.

The SIT nitrogen vent valves are not single failure proof against depressurizing the SITs by spurious opening. Therefore, power to the valves is removed while they are closed to ensure the safety analysis assumption of four pressurized SITs.

All of the SIT nitrogen vent valves are required to be operable so that, given a single failure, all four SITs may still be vented during post-LOCA long-term cooling. Venting the SITs provides for SIT depressurization capability which ensures the timely establishment of shutdown cooling entry conditions as assumed by the safety analysis for small break LOCAs.

The limits for operation with a safety injection tank inoperable for any reason except an isolation valve closed minimizes the time exposure of the plant to a LOCA event occurring concurrent with failure of an additional safety injection tank which may result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be immediately opened, the full capability of one safety injection tank is not available and prompt action is required to place the reactor in a MODE where this capability is not required.

For MODES 3 and 4 operation with pressurizer pressure less than 1837 psia the Technical Specifications require a minimum of 57% wide range corresponding

SAFETY INJECTION TANKS (Continued)

to 1361 cubic feet and a maximum of 75% narrow range corresponding to 1927 cubic feet of borated water per tank, when three safety injection tanks are operable and a minimum of 36% wide range corresponding to 908 cubic feet and a maximum of 75% narrow range corresponding to 1927 cubic feet per tank, when four safety injection tanks are operable at a minimum pressure of 235 psig and a maximum pressure of 625 psig. To allow for instrument inaccuracy, 60% wide range instrument corresponding to 1415 cubic feet, and 72% narrow range instrument corresponding to 1914 cubic feet, when three safety injection tanks are operable, and 39% wide range instrument corresponding to 962 cubic feet, and 72% narrow range instrument corresponding to 1914 cubic feet, when four SITs are operable, are specified in the Technical Specifications. To allow for instrument inaccuracy 254 psig is specified in the Technical Specifications.

The instrumentation vs. volume correlation for the SITs is as follows:

Volume	Narrow Range	Wide Range
962 ft^3	<0%	39%
1415 ft^3	<0%	60%
1802 ft^3	28%	78%
1914 ft^3	72%	83%

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

The OPERABILITY of two separate and independent ECCS subsystems with the indicated RCS pressure greater than or equal to 1837 psia, or with the indicated RCS cold leg temperature greater than or equal to 485 degrees F ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. These indicated values include allowances for uncertainties. Either subsystem operating in conjunction with the safety injection tanks is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double-ended break of the largest RCS cold leg pipe downward. In addition, each ECCS subsystem provides long-term core cooling capability in the recirculation mode during the accident recovery period.

The Mode 3 safety analysis credits one HPSI pump to provide negative reactivity insertion to protect the core and RCS following a steam line break when RCS cold leg temperature is 485 degrees F or greater. Requiring two operable ECCS subsystems in the situation will ensure one HPSI pump is available assuming single failure of the other HPSI pump.

With the RCS cold leg temperature below 485 degrees F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

The trisodium phosphate dodecahydrate (TSP) stored in dissolving baskets located in the containment basement is provided to minimize the possibility of corrosion cracking of certain metal components during operation of the ECCS following a LOCA. The TSP provided this protection by dissolving in the sump water and causing its final pH to be raised to greater than or equal to 7.0.

The surveillance requirements provided to ensure OPERABILITY of each component ensure that at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. Surveillance requirements for throttle valve position stops and flow balance testing provide

CORE OPERATING LIMITS REPORT

6.9.1.9 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle for the following:

a. Shutdown Margin - Reactor Trip Breakers Open for Specification 3.1.1.1
b. Shutdown Margin - Reactor Trip Breakers Closed for Specification 3.1.1.2
c. Moderator Temperature Coefficient BOL and EOL limits for Specification 3.1.1.3
d. Boron Dilution Alarms for Specification 3.1.2.7
e. Movable Control Assemblies - CEA Position for Specification 3.1.3.1
f. Regulating CEA Insertion Limits for Specification 3.1.3.6
g. Part Length CEA Insertion Limits for Specification 3.1.3.7
h. Linear Heat Rate for Specification 3.2.1
i. Azimuthal Power Tilt - T_q for Specification 3.2.3
j. DNBR Margin for Specification 3.2.4
k. Axial Shape Index for Specification 3.2.7
l. Boron Concentration (Mode 6) for Specification 3.9.1

6.9.1.10 The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC in:

a. "CE Method for Control Element Assembly Ejection Analysis," CENPD-0190-A, January 1976 (Methodology for Specification 3.1.3.6, Regulating CEA Insertion Limits).

b. "The ROCS and DIT Computer Codes for Nuclear Design," CENPD-266-P-A, April 1983 [Methodology for Specifications 3.1.1.1, Shutdown Margin - Reactor Trip Breakers Open; 3.1.1.2, Shutdown Margin - Reactor Trip Breakers Closed; 3.1.1.3, Moderator Temperature Coefficient BOL and EOL limits; 3.1.3.6, Regulating CEA Insertion Limits and 3.9.1, Boron Concentration (Mode 6)].

c. "Safety Evaluation Report related to the Final Design of the Standard Nuclear Steam Supply Reference Systems CESSAR System 80, Docket No. STN 50-470, "NUREG-0852 (November 1981), Supplements No. 1 (March 1983), No. 2 (September 1983), No. 3 (December 1987) (Methodology for Specifications 3.1.1.2, Shutdown Margin - Reactor Trip Breakers Closed; 3.1.1.3, Moderator Temperature Coefficient BOL and EOL limits; 3.1.2.7, Boron Dilution Alarms; 3.1.3.1, Movable Control Assemblies - CEA Position; 3.1.3.6, Regulating CEA Insertion Limits; 3.1.3.7, Part Length CEA Insertion Limits and 3.2.3 Azimuthal Power Tilt - T_q).

d. "Modified Statistical Combination of Uncertainties," CEN-356(V)-P-A Revision 01-P-A, May 1988 and "System 80™ Inlet Flow Distribution," Supplement 1-P to Enclosure 1-P to LD-82-054, February 1993 (Methodology for Specification 3.2.4, DNBR Margin and 3.2.7 Axial Shape Index).

CORE OPERATING LIMITS REPORT (Continued)

e. "Calculative Methods for the CE Large Break LOCA Evaluation Model for the Analysis of CE and W Designed NSSS," CENPD-132, Supplement 3-P-A, June 1985 (Methodology for Specification 3.2.1, Linear Heat Rate).

f. "Calculative Methods for the CE Small Break LOCA Evaluation Model," CENPD-137-P, August 1974 (Methodology for Specification 3.2.1, Linear Heat Rate).

g. "Calculative Methods for the CE Small Break LOCA Evaluation Model," CENPD-137-P, Supplement 1P, January 1977 (Methodology for Specification 3.2.1, Linear Heat Rate).

h. Letter: O. D. Parr (NRC) to F. M. Stern (CE), dated June 13, 1975 (NRC Staff Review of the Combustion Engineering ECCS Evaluation Model). NRC approval for: 6.9.1.10f.

i. Letter: K. Kniel (NRC) to A. E. Scherer (CE), dated September 27, 1977 (Evaluation of Topical Reports CENPD-133, Supplement 3-P and CENPD-137, Supplement 1-P). NRC approval for 6.9.1.10.g.

The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and analysis limits) of the safety analysis are met.

The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 106 TO FACILITY OPERATING LICENSE NO. NPF-41,

AMENDMENT NO. 98 TO FACILITY OPERATING LICENSE NO. NPF-51,

AND AMENDMENT NO. 78 TO FACILITY OPERATING LICENSE NO. NPF-74

ARIZONA PUBLIC SERVICE COMPANY, ET AL.

PALO VERDE NUCLEAR GENERATING STATION, UNIT NOS. 1, 2, AND 3

DOCKET NOS. STN 50-528, STN 50-529, AND STN 50-530

1.0 INTRODUCTION

By application dated February 1, 1996, the Arizona Public Service Company (APS or the licensee) requested changes to the Technical Specifications (TS) (Appendix A to Facility Operating License Nos. NPF-41, NPF-51, and NPF-74, respectively) for the Palo Verde Nuclear Generating Station (PVNGS), Units 1, 2, and 3. APS submitted this request 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.

These amendments would (1) revise TS Sections 3/4.1.1.1, 6.9.1.9, and 6.9.1.10 to relocate the shutdown margin (reactor trip breakers open) to the Core Operating Limits Report (COLR); (2) revise TS 3/4.3.2 (Tables 3.3-3 and 3.3-4) to specify an additional restriction for the allowed low-pressurizer-pressure trip setpoint when reducing reactor coolant system (RCS) pressure in Mode 3; (3) revise TS Section 2.2.1 (Table 2.2-1) to make it consistent with the footnote in TS Tables 3.3-3 and 3.3-4; and (4) revise TS Sections 3/4.5.2 and 3/4.5.3 to require two emergency core cooling system (ECCS) subsystems to be operable in Mode 3 whenever the RCS cold-leg temperature is equal to or above 485°F. The Table of Contents and the Bases would also be revised to reflect these changes.

2.0 EVALUATION

The licensee's proposed changes would result in TS requirements that reflect the more conservative assumptions of the updated PVNGS Mode 3 safety analysis, as described in Licensee Event Report (LER) 95-002-01, which the licensee submitted to the U.S. Nuclear Regulatory Commission (NRC) on August 25, 1995. As described in the LER, the licensee has administrative controls in place to ensure plant operation within the safety analysis assumptions. The changes requested by the licensee are as follows.

Revise TS 3/4.1.1.1, TS 6.9.1.9, and TS 6.9.1.10 to relocate the "shutdown margin - reactor trip breakers open" to the COLR.

The revision of TS 3/4.1.1.1 relocates the "shutdown margin - reactor trip breakers open" to the COLR. This revision also ensures a minimum shutdown margin that is sufficient to avoid unacceptable accident consequences to the fuel or the reactor coolant system (RCS) as a result of a design-basis accident or an anticipated operational occurrence when the reactor trip breakers are open. TS 6.9.1.9 lists the core operating limits that must be established and documented in the COLR. TS 6.9.1.10 lists the analytical methods used to determine the core operating limits that are listed in TS 6.9.1.9. TS 6.9.1.9 and 6.9.1.10 would also be revised to add this shutdown margin limit to the list of limits required in the COLR and to specify the analytical method used to determine this limit.

Following relocation to the COLR, the "shutdown margin - reactor trip breakers open" would be updated to reflect the more restrictive limit identified in the updated Mode 3 steamline break analysis. This analysis was performed using the NRC-approved analytical method listed in TS 6.9.1.10.b, "The ROCS and DIT Computer Codes for Nuclear Design," CENPD-266-P-A, April 1983. This change would also allow the licensee to revise this shutdown margin to reflect future cycle-specific limits determined using NRC-approved methods without the need for a license amendment. Also, the shutdown margin requirement of TS 3.1.1.2 was relocated to the COLR in TS Amendments 69, 55, and 42 for PVNGS Units 1, 2, and 3, respectively, dated December 30, 1992.

Generic Letter (GL) 88-16 identified the shutdown margin as one of the cycle-specific parameters that would be appropriately relocated to the COLR. GL 88-16 stated that if these limits are developed using an NRC-approved methodology, it would ensure that the values of cycle-specific parameters are determined consistent with plant safety analyses and design basis and provide for safe operation. The staff has reviewed the licensee's proposed changes to TS Sections TS 3/4.1.1.1, TS 6.9.1.9, and TS 6.9.1.10, relocating the "shutdown margin - reactor trip breakers open" to the COLR, and finds the changes consistent with GL 88-16 and acceptable.

Revise the footnote in TS 3/4.3.2, Tables 3.3-3, and 3.3-4 to add a new requirement that the "pressurizer pressure - low trip setpoint" must be maintained higher than 140 psi above the saturation pressure corresponding to the RCS cold-leg temperature when the cold-leg temperature is equal to or above 485°F while pressurizer pressure is reduced in Modes 3 and 4.

TS 3/4.3.2, Table 3.3-3, specifies the engineered safety features actuation system (ESFAS) instrumentation required by TS 3.3.2 to protect against violating core design limits and RCS pressure boundary during anticipated operational occurrences and to limit consequences during accidents.

TS 3/4.3.2, Table 3.3-4, specifies the trip setpoint values for the ESFAS instrumentation listed in Table 3.3-3.

Footnote a of Table 3.3-3 and footnote 1 of Table 3.3-4 (the footnotes are identical) permit the "pressurizer pressure - low setpoint value" to be decreased manually as the RCS is depressurized in Modes 3 and 4 to allow for controlled depressurization of the RCS without causing an unnecessary reactor trip, a containment isolation actuation signal (CIAS), or a safety injection actuation signal. The footnotes also require the setpoint to be automatically increased when repressurizing to ensure protection is appropriately restored.

These identical footnotes apply to the low-pressurizer-pressure setpoint, which provides the trip signals for the reactor protective system, safety injection actuation, and containment isolation actuation. Specifically, the proposed changes would add to these footnotes a restriction that requires the low-pressurizer-pressure trip setpoint be at least 140 psi greater than the saturation pressure corresponding to the RCS cold-leg temperature, when the RCS cold-leg temperature is greater than or equal to 485°F. The Bases for these TS sections would also be revised to describe the basis for the new restriction. In Unit 1 TS, footnote 1 of Table 3.3-4 should be changed from "In Modes 3-6..." to "In Modes 3-4..." This change will correct a typographical error to make this footnote consistent with related footnotes in Unit 1 TS and the same footnote in the TS of the other units.

These changes impose additional, more restrictive requirements to ensure that safety injection will be actuated before the RCS pressure drops to the reactor vessel upper head saturation pressure during RCS depressurization following a steamline break, as credited in the Mode 3 steamline break safety analysis. This requirement is necessary to protect the core and the RCS by ensuring that negative reactivity is added to the core by safety injection to counter the positive reactivity being added as a result of the cooling of the RCS during a steamline break. A steamline break in Mode 3 when the RCS cold-leg temperature is less than 485°F would not require the insertion of negative reactivity from safety injection, since the positive reactivity added as a result of cooldown in this situation would not be enough to overcome the shutdown margin and would have unacceptable consequences.

The "pressurizer pressure - low trip signal" also initiates CIAS (the revised footnotes are referenced by the CIAS ESFAS instrumentation in TS Tables 3.3-3 and 3.3-4). The additional restriction to keep the trip setpoint higher than 140 psi above the saturation pressure corresponding to the cold-leg temperature will not prevent the initiation of CIAS when needed according to the safety analyses and will not cause an initiation of CIAS that would be contrary to the safety analyses.

The 485°F lower threshold for the proposed requirement is an indicated value that accounts for a 15°F instrumentation uncertainty. The 140-psi requirement is also an indicated value that accounts for uncertainties in both pressure and temperature instrumentation.

This change is required to reflect the Mode 3 safety analysis assumption that safety injection would be actuated at a pressure above the reactor vessel upper head saturation pressure during RCS depressurization following a steamline break. This actuation is necessary because if a steamline break occurred in Mode 3 and the RCS depressurized below the RCS cold-leg saturation temperature before initiation of safety injection, a steam void could form in the reactor vessel upper head region. This steam void could act to keep the RCS pressurized above the safety injection setpoint, even though the RCS coolant would continue to be cooled. The RCS cooling would add positive reactivity as a result of the negative moderator temperature coefficient. This situation could result in potentially unacceptable consequences if safety injection was not actuated to provide negative reactivity. The staff has reviewed the proposed changes to the footnote in TS 3/4.3.2, Tables 3.3-3 and 3.3-4, and on the basis of its evaluation finds the changes acceptable.

<u>Revise TS Table 2.2-1</u>, the "pressurizer pressure - low" footnote, to make it consistent with the revised footnotes in TS Tables 3.3-3 and 3.3-4.

TS Table 2.2-1 specifies the reactor protective instrumentation trip setpoint limits required by TS 2.2.1 to protect against violating core design limits and the RCS pressure boundary during anticipated operational occurrences and to assist the engineered safety features systems in mitigating accidents. Specifically, footnote 2 of this table permits the "pressurizer pressure - low setpoint value" to be decreased as the RCS is depressurized in Modes 3 and 4. This process allows for controlled depressurization of the RCS while still maintaining an active trip setpoint until the trip signal is no longer needed to protect the plant. The footnote also requires the setpoint to be automatically increased when repressurizing to ensure protection is appropriately restored. The "pressurizer pressure - low trip signal" also actuates safety injection and containment isolation.

The "pressurizer pressure - low trip signal" also initiates a reactor trip from the reactor protection system. The additional restriction to keep the trip setpoint higher than 140 psi above the saturation pressure corresponding to the cold-leg temperature will not prevent a reactor trip when needed according to the safety analyses and will not cause a reactor trip that would be contrary to the safety analyses.

This change is required to make this footnote consistent with the footnotes on the ESFAS Instrumentation Tables. The source of the "pressurizer pressure - low trip signal" for the reactor protective instrumentation is the same as that for the ESFAS instrumentation, and any operational requirements should be the same. The staff has reviewed this proposed change and finds it acceptable.

<u>Revise TS 3/4.5.2 and TS 3/4.5.3</u> footnotes for applicability and add the requirement that two ECCS subsystems must be operable in Mode 3 when the RCS cold-leg temperature is greater than or equal to 485°F.

These changes are required so that the TS titles more accurately reflect the TS applicability. TS 3/4.5.2 is applicable in Modes 1 and 2 at all times and in Mode 3 when the pressurizer pressure is greater than or equal to 1837 psia and when the RCS cold-leg temperature is greater than or equal to 485°F. Since the applicability of these two TS is split at 1837 psia in Mode 3 and 485°F, the 350°F statement in the TS titles is not appropriate and is inconsistent with its applications. The terms "operating" and "shutdown," as requested for the new titles, will increase clarity and are consistent with the titles specified for those sections in NUREG-1432, "Standard Technical Specifications—Combustion Engineering Plants."

These changes are more restrictive and will ensure that if one high-pressure safety injection (HPSI) pump should fail during a steamline break in Mode 3 when the RCS is less than 1837 psia and greater than 485°F, one HPSI pump would still be available for safety injection. The Mode 3 safety analysis credits one HPSI pump to provide negative reactivity insertion to protect the core and the RCS following a steamline break.

These changes will also reflect bounding analysis parameters used in the Mode 3 steamline break analysis. The Mode 3 safety analysis credits one HPSI pump to provide negative reactivity insertion to protect the core and the RCS following a steamline break when the RCS cold-leg temperature is 485°F or greater. Requiring two ECCS subsystems to be operable will ensure that one HPSI pump is available, assuming a single failure of the other HPSI pump. The staff has reviewed the proposed changes to TS 3/4.5.2 and TS 3/4.5.3 and, on the basis of its evaluation, finds them acceptable.

Administrative Changes for TS 3/4.5.2 and TS 3/4.5.3

Additionally, the title of TS 3/4.5.2 would be revised to "ECCS Subsystems - Operating," and the title of TS 3/4.5.3 would be revised to "ECCS Subsystems - Shutdown." These editorial changes clarify the titles of the TS and have no effect on the operation of the plant or on any plant structures, systems, or components. The licensee will revise the Bases for these TS sections to reflect the new restrictions. Also, footnote ** would be deleted from Section 3.5.2 of the Unit 2 TS. This footnote has expired. These changes are administrative in nature, thus not affecting the health and safety of the public. The staff has reviewed the proposed changes and finds them acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Arizona State official was notified of the proposed issuance of the amendments. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (61 FR 13522). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). The amendments also involve changes in recordkeeping, reporting or administrative procedures or requirements. Accordingly, with respect to these items, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(10). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

5.0 CONCLUSION

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

Principal Contributor: C. Thomas

Date: April 30, 1996