10 CFR 50.90

APS A subsidiary of Pinnacle West Capital Corporation

Palo Verde Nuclear Generating Station

Dwight C. Mims
Vice President
Regulatory Affairs and Plant Improvement

Tel. 623-393-5403
Fax 623-393-6077

Mail Station 7605
P. O. Box 52034
Phoenix, Arizona 85072-2034

102-06088-DCM/RJR
October 30, 2009

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

Dear Sirs:

Subject: Palo Verde Nuclear Generating Station (PVNGS)
Units 1, 2, and 3
Docket Nos. STN 50-528, 50-529, and 50-530
Application for Amendment to License Condition C.(1) Maximum Power Level (Units 1 and 3 only), and Various Technical Specifications to Implement Administrative Changes

As permitted by 10 CFR 50.90, Arizona Public Service Company (APS) hereby applies for an amendment to Facility Operating Licenses NPF-41, NPF-51, and NPF-74 to remove requirements no longer applicable due to the completion of power uprate, replacement of steam generators, removal of part-length control rod assemblies, and completion of a core protection calculator upgrade. A minor administrative change to the nomenclature of the containment sump trash racks and screens is also proposed. The proposed changes are detailed in the enclosure to this letter.

Approval of the proposed amendment is requested within one year from the date of this letter. Once approved, the amendment shall be implemented within 90 days.

In accordance with the PVNGS Quality Assurance Program, the Plant Review Board and the Offsite Safety Review Committee have reviewed and concurred with this proposed amendment. By copy of this letter, this submittal is being forwarded to the Arizona Radiation Regulatory Agency (ARRA) pursuant to 10 CFR 50.91(b)(1).

No commitments are being made to the NRC by this letter. Should you need further information regarding this amendment request, please contact Russell A. Stroud, Licensing Section Leader, at (623) 393-5111.

A001
NRR

I declare under penalty of perjury that the foregoing is true and correct.

Executed on 10/30/09.

Sincerely,

R. C. Mims

DCM/RAS/RJR/gat

Enclosure: Arizona Public Service Company's Evaluation of the Proposed Change

cc:	E. E. Collins, Jr.	NRC Region IV Regional Administrator
	J. R. Hall	NRC NRR Project Manager
	R. I. Treadway	NRC Senior Resident Inspector for PVNGS
	A. V. Godwin	Arizona Radiation Regulatory Agency
	T. Morales	Arizona Radiation Regulatory Agency

ENCLOSURE

ARIZONA PUBLIC SERVICE COMPANY'S EVALUATION OF THE PROPOSED CHANGE

ATTACHMENTS:

1. Proposed Operating License and Technical Specification Changes (mark-up)
2. Proposed Technical Specification Changes (retyped)

1.0 SUMMARY DESCRIPTION

This evaluation supports the application to amend Operating Licenses NPF-41, NPF-51, and NPF-74 for Palo Verde Nuclear Generating Station (PVNGS) Units 1, 2, and 3, respectively. The proposed changes are administrative and would not result in any change to operating requirements. The proposed administrative changes remove requirements no longer applicable due to the completion of power uprate, replacement of steam generators, removal of part-length control rod assemblies, and completion of a core protection calculator upgrade. A minor administrative change to the nomenclature of the containment sump trash racks and screens is also proposed. These minor administrative changes are proposed for the following License Conditions and Technical Specifications (TS):

- Units 1 and 3 Facility Operating Licenses, License Condition C.(1) "Maximum Power Level"
- Units 1, 2 and 3 Technical Specifications:
  - Table of Contents,
  - 1.1 "Definitions,"
  - 3.1.5 "Control Element Assembly (CEA) Alignment,"
  - 3.1.8 "Part Length or Part Strength Control Element Assembly (CEA) Insertion Limits,"
  - 3.1.10 "Special Test Exceptions (STE) - MODES 1 and 2,"
  - 3.1.11 "Special Test Exceptions (STE) - Reactivity Coefficient Testing,"
  - 3.2.4 "Departure from Nucleate Boiling Ratio (DNBR),"
  - 3.3.1 "Reactor Protective System (RPS) Instrumentation - Operating,"
  - 3.3.2 "Reactor Protective System (RPS) Instrumentation - Shutdown,"
  - 3.3.3 "Control Element Assembly Calculators (CEACs),"
  - 3.3.5 "Engineered Safety Features Actuation System (ESFAS) Instrumentation,"
  - 3.4.1 "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits,"
  - 3.5.3 "ECCS - Operating,"
  - 3.7.1 "Main Steam Safety Valves (MSSVs),"
  - 5.4 "Procedures,"
  - 5.5.9 "Steam Generator (SG) Program," and
  - 5.6.5 "Core Operating Limits Report (COLR)."

2.0 DETAILED DESCRIPTION

The following changes are proposed:

A. Units 1 and 3 Operating Licenses, License Condition C. (1) "Maximum Power Level," and Units 1, 2, and 3, Technical Specifications 1.1 "Definitions," 3.3.1 "Reactor Protective System (RPS) Instrumentation - Operating," 3.3.2 "Reactor Protective System (RPS) Instrumentation - Shutdown," 3.3.5 "Engineered Safety Features Actuation System (ESFAS) Instrumentation," 3.4.1 "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," and 3.7.1 "Main Steam

Safety Valves (MSSVs)." Remove superseded references to 3876 megawatts thermal (MWt) and information related to this value.

This change is administrative and is necessary to complete the Power Uprate and Steam Generator Replacement Projects. Removal of the references to 3876 MWt and information related to this value corrects the Operating License and the affected TS sections.

B. Technical Specification "Table of Contents," 1.1 "Definitions," 3.1.5 "Control Element Assembly (CEA) Alignment," 3.1.8 "Part Length or Part Strength Control Element Assembly (CEA) Insertion Limits," 3.1.10 "Special Test Exceptions (STE) - MODES 1 and 2," 3.1.11 "Special Test Exceptions (STE) - Reactivity Coefficient Testing," 3.3.3 "Control Element Assembly Calculators (CEACs)," and 5.6.5 "Core Operating Limits Report (COLR)." Remove references to Part Length Control Element Assemblies (CEA).

   This change is administrative. APS has completed removal of all part length CEAs at Palo Verde Units 1, 2, and 3 and this change removes the respective references to part length CEAs from the affected TS sections.

C. Technical Specification 3.2.4 "Departure from Nucleate Boiling Ratio (DNBR)," 3.3.1 "Reactor Protective System (RPS) Instrumentation - Operating," 3.3.3 "Control Element Assembly Calculators (CEACs)," and 5.4 "Procedures." Remove pages from TS identified as "(Before CPC Upgrade)" or "(Before CPCS Upgrade)," and delete the designator "(After CPC Upgrade)" or "(After CPCS Upgrade)," from the affected TS pages. In addition, the indentation of the logical connectors AND and OR in TS 3.2.4, between Required Actions B.1, B.2.1, and B.2.2 is being adjusted to be consistent with the action numbers and with TS 1.2.

   These changes are administrative. APS has completed a core protection calculator (CPC) upgrade in all Palo Verde units. This change removes the redundant TS pages identified as "(Before CPC Upgrade) or (Before CPCS Upgrade)" and removes the reference to "(After CPC Upgrade) or (After CPCS Upgrade)" from various TS pages that will be renumbered and remain in place. The adjustment of the indentation of the logical connectors AND and OR in TS 3.2.4 is consistent with the action numbers and with TS 1.2.

D. Technical Specification 3.5.3 "ECCS - Operating," Surveillance Requirement (SR) 3.5.3.8. Change the nomenclature from "trash racks and screens" to "strainers."

   This change is administrative. The change from "trash racks and screens" to "strainers" does not change the intent of the Surveillance Requirement 3.5.3.8 to verify, by visual inspection, that each ECCS train containment sump suction inlet is not restricted by debris and the suction inlet strainers show no evidence of structural distress or abnormal corrosion.

E. Technical Specification 5.5.9 "Steam Generator (SG) Program." Delete SG Program Item 5.5.9d.2a and delete the statement "Replacement SGs with Alloy 690 TT tubes:" from Item 5.5.9d.2d and renumber item 5.5.9d.2b to 5.5.9d.2.

This change is administrative. The steam generators with Alloy 600 MA tubes have been replaced. Therefore the requirement to inspect SGs with Alloy 600 MA tubes is no longer applicable.

3.0 TECHNICAL EVALUATION

The proposed changes to the Operating License Condition and Technical Specifications are administrative and/or editorial and do not affect how plant equipment is operated or maintained. No changes to the physical plant or analytical methods are described and there are no impacts on the PVNGS Updated Final Safety Analysis Report (UFSAR) accident analysis.

List of Proposed Administrative Changes

Proposed Change	Justification	Technical Specification Sections
Remove superseded reference to 3876 megawatts thermal (MWt) and information related to this value.	Completed 3990 MWt power uprate.	1.1, 3.3.1, 3.3.2, 3.3.5, 3.4.1, and 3.7.1
Delete superseded part-length (PL) control element assembly (CEA) limits, specifications, and references.	Completed replacement of PL CEAs with part strength (PS) CEAs in all three PVNGS units.	1.1, 3.1.5, 3.1.8, 3.1.10, 3.1.11, 3.3.3, 5.6.5
Delete superseded core protection calculator (CPC) limits, specifications, and references.	Completed core protection calculator (CPC) replacement modification in all three PVNGS units.	3.2.4, 3.3.1, 3.3.3 and 5.4
Adjust the indentation of the logical connectors AND and OR in TS 3.2.4, between Required Actions B.1, B.2.1, and B.2.2.	The adjustment of the indentation of the logical connectors AND and OR in TS 3.2.4 is consistent with the Action numbers B.1, B.2.1, and B.2.2 and with TS 1.2	3.2.4

Proposed Change	Justification	Technical Specification Sections
Replace containment sump "trash racks and screens" with "strainers."	The new PVNGS containment sump screens/strainers function as trash racks and screens; however, this administrative change would clarify the PVNGS nomenclature to be consistent with the industry nomenclature. (STP ML080360209 03/25/08, DC Cook ML072780605 10/18/07).	TS SR 3.5.3.8
Delete Alloy-600 steam generator limits, specifications, and references.	Completed replacement of Alloy-600 steam generators in all three units.	5.5.9

4.0 REGULATORY EVLAUATION

4.1 Applicable Regulatory Requirements/Criteria

The proposed changes to the Operating License Condition and Technical Specifications are administrative and/or editorial and do not affect any regulatory requirements or criteria. These changes do not affect how plant equipment is operated or maintained and there are no changes to the physical plant or analytical methods. As a result, there are no impacts on the PVNGS UFSAR accident analysis.

4.2 No Significant Hazards Consideration Determination

Arizona Public Service (APS) has evaluated whether or not a significant hazards consideration is involved with the proposed amendments by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

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

   Response: No.

   The proposed amendment includes the following changes that are considered to be administrative and/or editorial changes:

   A. Remove superseded references to 3876 megawatts thermal (MWt) and related information to this value from Unit 1 and Unit 3 Operating Licenses and Unit 1, 2, and 3 Technical Specifications.

This change is administrative. The change only removes the references to 3876 MWt and related information to this value and leaves the references to 3990 MWt.

B. Remove references to Part Length Control Element Assemblies.

This change is administrative because it only removes references to part length CEAs which have been replaced by part strength CEAs.

C. Remove outdated pages and other references as a result of the CPC upgrade, and adjust the indentation of the logical connectors AND and OR in TS 3.2.4, between Required Actions B.1, B.2.1, and B.2.2.

This change is administrative because it removes the redundant TS pages identified as "(Before CPC Upgrade) or (Before CPCS Upgrade)" and removes the reference to "(After CPC Upgrade) or (After CPCS Upgrade)" from various TS pages that will be renumbered and remain in place. The CPC upgrade has been completed. The adjustment of the indentation of the logical connectors AND and OR in TS 3.2.4 is consistent with the Action numbers and with TS 1.2.

D. Change "trash racks and screens" to "strainers."

This change is administrative. The change from "trash racks and screens" to "strainers" does not change the intent of the Surveillance Requirement 3.5.3.8 to verify, by visual inspection, that each ECCS train containment sump suction inlet is not restricted by debris and the suction inlet strainers show no evidence of structural distress or abnormal corrosion.

E. Delete inspection requirements for Steam Generators (SG) with Alloy 600 MA tubes.

This change is administrative because APS has completed the SG replacement project which removed all SGs containing Alloy 600 MA tubes.

As discussed above, the proposed amendment involves administrative and/or editorial changes only. The proposed amendment does not impact any accident initiators, analyzed events, or assumed mitigation of accident or transient events. The proposed changes do not involve the addition or removal of any equipment or any design changes to the facility. The proposed changes do not affect any plant operations, design function, or analysis that verifies the capability of structures, systems, and components (SSCs) to perform a design function. The proposed changes do not change any of the accidents previously evaluated in the UFSAR. The proposed changes do not affect SSCs, operating procedures, and administrative controls that have the function of preventing or mitigating any of these accidents.

Therefore, the proposed changes do not represent a significant increase in the probability or consequences of an accident previously evaluated.

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

   Response: No

   As stated in response to standard 1, the proposed amendment only involves administrative and/or editorial changes. No actual plant equipment or accident analyses will be affected by the proposed changes. The proposed changes will not change the design function or operation of any SSCs. The proposed changes will not result in any new failure mechanisms, malfunctions, or accident initiators not considered in the design and licensing bases. The proposed amendment does not impact any accident initiators, analyzed events, or assumed mitigation of accident or transient events.

   Therefore, this proposed change does not create the possibility of an accident of a new or different kind than previously evaluated.

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

   Response: No

   As stated in response to standard 1, the proposed amendment only involves administrative and/or editorial changes. The proposed change does not involve any physical changes to the plant or alter the manner in which plant systems are operated, maintained, modified, tested, or inspected. The proposed change does not alter the manner in which safety limits, limiting safety system settings or limiting conditions for operation are determined. The safety analysis acceptance criteria are not affected by this change. The proposed change will not result in plant operation in a configuration outside the design basis. The proposed change does not adversely affect systems that respond to safely shutdown the plant and to maintain the plant in a safe shutdown condition.

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

4.3 Conclusion

Based on the considerations discussed above:

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. Issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Therefore, APS concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c) and, accordingly, a finding of "no significant hazards consideration" is justified.

5.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would not change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or change an inspection or surveillance requirement. The proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or a significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

Proposed Operating License and Technical Specification Changes (mark-up)

U1 OL

(6)(a) Pursuant to an Order of the Nuclear Regulatory Commission dated December 12, 1985, the Public Service Company of New Mexico (PNM) was authorized to transfer a portion of its ownership share in Palo Verde, Unit 1 to certain institutional investors on December 31, 1985, and at the same time has leased back from such purchasers the same interest in the Palo Verde, Unit 1 facility. The term of the lease is to January 15, 2015, subject to a right of renewal. Additional sale and leaseback transactions (for a term expiring on January 15, 2015) of all or a portion of PNM's remaining ownership share in Palo Verde Unit 1 are hereby authorized until June 30, 1987. Any such sale and leaseback transaction is subject to the representations and conditions set forth in the aforementioned applications of October 19, 1985, February 5, 1986, October 16, 1986 and November 26, 1986, and the Commission's Order of December 12, 1985, consenting to such transactions. Specifically, the lessor and anyone else who may acquire an interest under this transaction are prohibited from exercising directly or indirectly any control over the licensees of the Palo Verde Nuclear Generating Station, Unit 1. For purposes of this condition, the limitations in 10 CFR 50.81, "Creditor Regulations," as now in effect and as they may be subsequently amended, are fully applicable to the lessor and any successor in interest to that lessor as long as the license for Palo Verde, Unit 1 remains in effect; this financial transaction shall have no effect on the license for the Palo Verde nuclear facility throughout the term of the license.

(b) Further, the licensees are also required to notify the NRC in writing prior to any change in: (i) the terms or conditions of any lease agreements executed as part of this transaction; (ii) the ANPP Participation Agreement, (iii) the existing property insurance coverage for the Palo Verde nuclear facility, Unit 1 as specified in license counsel's letter of November 26, 1985, and (iv) any action by the lessor or others that may have an adverse effect on the safe operation of the facility.

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

(1) Maximum Power Level

Arizona Public Service Company (APS) is authorized to operate the facility at reactor core power levels not in excess of ~~3876 megawatts thermal (100% power) through operating cycle 12, and~~ 3990 megawatts thermal (100% power) ~~after operating cycle 12~~, in accordance with the conditions specified herein.

U3 OL

(1) Maximum Power Level

Arizona Public Service Company (APS) is authorized to operate the facility at reactor core power levels not in excess of ~~3876 megawatts thermal (100% power) through operating cycle 13, and~~ 3990 megawatts thermal (100% power) ~~after operating cycle 13,~~ in accordance with the conditions specified herein.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 177, 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) Antitrust Conditions

This license is subject to the antitrust conditions delineated in Appendix C to this license.

(4) Initial Test Program (Section 14, SER and SSER 2)

Deleted

(5) Additional Conditions

The Additional Conditions contained in Appendix D, as revised through Amendment No. 171, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Additional Conditions.

(6) Mitigation Strategy License Condition

APS shall develop and maintain strategies for addressing large fires and explosions and that include the following key areas:

(a) Fire fighting response strategy with the following elements:

1. Pre-defined coordinated fire response strategy and guidance.
2. Assessment of mutual aid fire fighting assets.
3. Designated staging areas for equipment and materials.
4. Command and control.
5. Training of response personnel.

PALO VERDE NUCLEAR GENERATING STATION
IMPROVED TECHNICAL SPECIFICATIONS
TABLE OF CONTENTS

1.1 Definitions (continued)

RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of ~~3876 MWt for Unit 1 through operating cycle 12 and Unit 3 through operating cycle 13, and 3990 MWt for Unit 1 after operating cycle 12, Unit 2, and Unit 3 after operating cycle 13~~ 3990mwt.
REACTOR PROTECTIVE SYSTEM (RPS) RESPONSE TIME	The RPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until electrical power to the CEAs drive mechanism is interrupted. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.
SHUTDOWN MARGIN (SDM)	SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming: a. All full strength CEAs (shutdown and regulating) are fully inserted except for the single CEA of highest reactivity worth, which is assumed to be fully withdrawn. With any full strength CEAs not capable of being fully inserted, the withdrawn reactivity worth of these CEAs must be accounted for in the determination of SDM and b. There is no change in ~~part length or~~ part strength CEA position.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.5 Control Element Assembly (CEA) Alignment

LCO 3.1.5 All full strength CEAs shall be OPERABLE, and all full strength and ~~part length or~~ part strength CEAs shall be aligned to within 6.6 inches (indicated position) of all other CEAs in their respective groups.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more CEAs trippable and misaligned from its group by > 6.6 inches and ≤ 9.9 inches. OR One CEA trippable and misaligned from its group by > 9.9 inches.	A.1 Reduce THERMAL POWER in accordance with the limits in the COLR. AND A.2 Restore CEA alignment.	1 hour 2 hours

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.5.1	Verify the indicated position of each full strength and ~~part length or~~ part strength CEA is within 6.6 inches of all other CEAs in its group.	12 hours
SR 3.1.5.2	Verify that, for each CEA, its OPERABLE CEA position indicator channels indicate within 5.2 inches of each other.	12 hours
SR 3.1.5.3	Verify full strength CEA freedom of movement (trippability) by moving each individual full strength CEA that is not fully inserted in the core at least 5 inches.	92 days
SR 3.1.5.4	Perform a CHANNEL FUNCTIONAL TEST of each reed switch position transmitter channel.	18 months
SR 3.1.5.5	Verify each full strength CEA drop time ≤ 4.0 seconds.	Prior to reactor criticality, after each removal of the reactor head

3.1 REACTIVITY CONTROL SYSTEMS

3.1.8 ~~Part Length or~~ Part Strength Control Element Assembly (CEA) Insertion Limits

LCO 3.1.8 The ~~part length or~~ part strength CEA groups shall be limited to the insertion limits specified in the COLR.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ~~Part length or~~ part strength CEA groups inserted beyond the transient insertion limit.	A.1 Restore ~~part length or~~ part strength CEA groups to within the limit.	2 hours
	OR	
	A.2 Reduce THERMAL POWER to less than or equal to that fraction of RTP specified in the COLR.	2 hours
B. ~~Part length or~~ part strength CEA groups inserted between the long term steady state insertion limit and the transient insertion limit for intervals ≥ 7 effective full power days (EFPD) per 30 EFPD or ≥ 14 EFPD per 365 EFPD interval.	B.1 Restore ~~part length or~~ part strength CEA groups to within the long term steady state insertion limit.	2 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.8.1	Verify ~~part length or~~ part strength CEA group position.	12 hours

3.1 REACTIVITY CONTROL SYSTEMS

3.1.10 Special Test Exceptions (STE) – MODES 1 and 2

LCO 3.1.10 During performance of PHYSICS TESTS, the requirements of:

LCO 3.1.4, "Moderator Temperature Coefficient (MTC)";
LCO 3.1.5, "Control Element Assembly (CEA) Alignment";
LCO 3.1.6, "Shutdown Control Element Assembly (CEA) Insertion Limits";
LCO 3.1.7, "Regulating Control Element Assembly (CEA) Insertion Limits";
LCO 3.1.8, "~~Part Length or~~ Part Strength CEA Insertion Limits";
LCO 3.2.2, "Planar Radial Peaking Factors (Fxy)";
LCO 3.2.3, "AZIMUTHAL POWER TILT (Tq)";
LCO 3.2.5, "AXIAL SHAPE INDEX (ASI)"; and
LCO 3.3.3, "Control Element Assembly Calculators (CEACs)"

may be suspended, provided:

a. THERMAL POWER is restricted to the test power plateau, which shall not exceed 85% RTP; and

b. Shutdown reactivity equivalent to at least the highest estimated CEA worth (of those CEAs actually withdrawn) is available for trip insertion.

APPLICABILITY: MODES 1 and 2 during PHYSICS TESTS.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Test power plateau exceeded.	A.1 Reduce THERMAL POWER to less than or equal to the test power plateau.	15 minutes
B. Required Action and associated Completion Time not met.	B.1 Suspend PHYSICS TESTS.	1 hour

3.1 REACTIVITY CONTROL SYSTEMS

3.1.11 Special Test Exceptions (STE) – Reactivity Coefficient Testing

LCO 3.1.11 During performance of PHYSICS TESTS, the requirements of:

LCO 3.1.7, "Regulating Control Element Assembly (CEA) Insertion Limits";
LCO 3.1.8, "~~Part Length or~~ Part Strength Control Element Assembly (CEA) Insertion Limits;" and
LCO 3.4.1, "RCS Pressure, Temperature and Flow limits" (LCO 3.4.1.b, RCS Cold Leg Temperature only)

may be suspended, provided LHR and DNBR do not exceed the limits in the COLR.

APPLICABILITY: MODE 1 with Thermal Power > 20% RTP during PHYSICS TESTS.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LHR or DNBR outside the limits specified in the COLR.	A.1 Reduce THERMAL POWER to restore LHR and DNBR to within limits.	15 minutes
B. Required Action and associated Completion Time not met.	B.1 Suspend PHYSICS TESTS.	1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.11.1	Verify LHR and DNBR do not exceed limits by performing SR 3.2.1.1 and SR 3.2.4.1.	Continuously

3.2 POWER DISTRIBUTION LIMITS

3.2.4 Departure From Nucleate Boiling Ratio (DNBR)

Delete Page

LCO 3.2.4 The DNBR shall be maintained by one of the following methods:

a. Maintaining Core Operating Limit Supervisory System (COLSS) calculated core power less than or equal to COLSS calculated core power operating limit based on DNBR (when COLSS is in service, and either one or both Control Element Assembly Calculators (CEACs) are OPERABLE);

b. Maintaining COLSS calculated core power less than or equal to COLSS calculated core power operating limit based on DNBR decreased by the allowance specified in the COLR (when COLSS is in service and neither CEAC is OPERABLE);

c. Operating within the region of acceptable operation specified in the COLR using any OPERABLE Core Protection Calculator (CPC) channel (when COLSS is out of service and either one or both CEACs are OPERABLE); or

d. Operating within the region of acceptable operation specified in the COLR using any OPERABLE CPC channel (when COLSS is out of service and neither CEAC is OPERABLE).

APPLICABILITY: MODE 1 with THERMAL POWER > 20% RTP. **(Before CPC Upgrade)**

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. COLSS calculated core power not within limit.	A.1 Restore the DNBR to within limit.	1 hour

(continued)

Delete page

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. DNBR outside the region of acceptable operation when COLSS is out of service.	B.1 Determine trend in DNBR.	Once per 15 minutes
	AND	
	B.2.1 With an adverse trend, restore DNBR to within limit.	1 hour
	OR	
	B.2.2 With no adverse trend, restore DNBR to within limit.	4 hours
C. Required Action and associated Completion Time not met.	C.1 Reduce THERMAL POWER to ≤ 20% RTP.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.4.1	----------NOTE---------- 1. Only applicable when COLSS is out of service. With COLSS in service, this parameter is continuously monitored. 2. Not required to be performed until 2 hours after MODE 1 with THERMAL POWER > 20% RTP. ---------- Verify DNBR, as indicated on any OPERABLE DNBR channels, is within the limit of the COLR, as applicable.	2 hours
SR 3.2.4.2	Verify COLSS margin alarm actuates at a THERMAL POWER level equal to or less than the core power operating limit based on DNBR.	31 days

3.2 POWER DISTRIBUTION LIMITS

3.2.4 Departure From Nucleate Boiling Ratio (DNBR)

LCO 3.2.4 The DNBR shall be maintained by one of the following methods:

a. Core Operating Limit Supervisory System (COLSS) In Service:

1. Maintaining COLSS calculated core power less than or equal to COLSS calculated core power operating limit based on DNBR when at least one Control Element Assembly Calculator (CEAC) is OPERABLE in each OPERABLE Core Protection Calculator (CPC) channel; or

2. Maintaining COLSS calculated core power less than or equal to COLSS calculated core power operating limit based on DNBR decreased by the allowance specified in the COLR when the CEAC requirements of LCO 3.2.4.a.1 are not met.

b. COLSS Out of Service:

1. Operating within the region of acceptable operation specified in the COLR using any OPERABLE Core Protection Calculator (CPC) channel when at least one Control Element Assembly Calculator (CEAC) is OPERABLE in each OPERABLE CPC channel; or

2. Operating within the region of acceptable operation specified in the COLR using any OPERABLE CPC channel (with both CEACs inoperable) when the CEAC requirements of LCO 3.2.4.b.1 are not met.

APPLICABILITY: MODE 1 with THERMAL POWER > 20% RTP. ~~(After CPC Upgrade)~~

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. COLSS calculated core power not within limit.	A.1 Restore the DNBR to within limit.	1 hour

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. DNBR outside the region of acceptable operation when COLSS is out of service.	B.1 Determine trend in DNBR.	Once per 15 minutes
	AND	
	B.2.1 With an adverse trend, restore DNBR to within limit.	1 hour
	OR	
	B.2.2 With no adverse trend, restore DNBR to within limit.	4 hours
C. Required Action and associated Completion Time not met.	C.1 Reduce THERMAL POWER to ≤ 20% RTP.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.4.1	----------NOTE---------- 1. Only applicable when COLSS is out of service. With COLSS in service, this parameter is continuously monitored. 2. Not required to be performed until 2 hours after MODE 1 with THERMAL POWER > 20% RTP. ---------- Verify DNBR, as indicated on any OPERABLE DNBR channels, is within the limit of the COLR, as applicable.	2 hours
SR 3.2.4.2	Verify COLSS margin alarm actuates at a THERMAL POWER level equal to or less than the core power operating limit based on DNBR.	31 days

3.3 INSTRUMENTATION

3.3.1 Reactor Protective System (RPS) Instrumentation – Operating

LCO 3.3.1 Four RPS trip and bypass removal channels for each Function in Table 3.3.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1-1. **(Before CPC Upgrade)**

ACTIONS

----------NOTE----------
Separate Condition entry is allowed for each RPS Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one automatic RPS trip channel inoperable.	A.1 Place channel in bypass or trip. AND A.2 Restore channel to OPERABLE status.	1 hour Prior to entering MODE 2 following next MODE 5 entry
B. One or more Functions with two automatic RPS trip channels inoperable.	B.1 Place one channel in bypass and the other in trip.	1 hour

(continued)

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One or more Functions with one automatic bypass removal channel inoperable.	C.1 Disable bypass channel.	1 hour
	OR	
	C.2.1 Place affected automatic trip channel in bypass or trip.	1 hour
	AND	
	C.2.2 Restore bypass removal channel and associated automatic trip channel to OPERABLE status.	Prior to entering MODE 2 following next MODE 5 entry
D. One or more Functions with two automatic bypass removal channels inoperable.	D.1 Disable bypass channels.	1 hour
	OR	
	D.2 Place one affected automatic trip channel in bypass and place the other in trip.	1 hour
E. One or more core protection calculator (CPC) channels with a cabinet high temperature alarm.	E.1 Perform CHANNEL FUNCTIONAL TEST on affected CPC.	12 hours

(continued)

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. One or more CPC channels with three or more autorestarts during a 12 hour period.	F.1 Perform CHANNEL FUNCTIONAL TEST on affected CPC.	24 hours
G. Required Action and associated Completion Time not met.	G.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

----------NOTE----------
Refer to Table 3.3.1-1 to determine which SR shall be performed for each RPS Function.

SURVEILLANCE		FREQUENCY
SR 3.3.1.1	Perform a CHANNEL CHECK of each RPS instrument channel.	12 hours

(continued)

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SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.2	----------NOTE---------- Not required to be performed until 12 hours after THERMAL POWER ≥ 70% RTP. ---------- Verify total Reactor Coolant System (RCS) flow rate as indicated by each CPC is less than or equal to the RCS total flow rate. If necessary, adjust the CPC addressable constant flow coefficients such that each CPC indicated flow is less than or equal to the RCS flow rate.	12 hours
SR 3.3.1.3	Check the CPC autorestart count.	12 hours

(continued)

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SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.4	----------NOTES---------- 1. Not required to be performed until 12 hours after THERMAL POWER ≥ 20% RTP. 2. The daily calibration may be suspended during PHYSICS TESTS, provided the calibration is performed upon reaching each major test power plateau and prior to proceeding to the next major test power plateau. ---------- Perform calibration (heat balance only) and adjust the linear power level signals and the CPC addressable constant multipliers to make the CPC ΔT power and CPC nuclear power calculations agree with the calorimetric, if the absolute difference is ≥ 2% when THERMAL POWER is ≥ 80% RTP. Between 20% and 80% RTP the maximum difference is -0.5% to 10%.	24 hours
SR 3.3.1.5	----------NOTE---------- Not required to be performed until 12 hours after THERMAL POWER ≥ 70% RTP. ---------- Verify total RCS flow rate indicated by each CPC is less than or equal to the RCS flow determined either using the reactor coolant pump differential pressure instrumentation and the ultrasonic flow meter adjusted pump curves or by calorimetric calculations.	31 days

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(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.6	----------NOTE---------- Not required to be performed until 12 hours after THERMAL POWER ≥ 15% RTP. ---------- Verify linear power subchannel gains of the excore detectors are consistent with the values used to establish the shape annealing matrix elements in the CPCs.	31 days
SR 3.3.1.7	----------NOTES---------- 1. The CPC CHANNEL FUNCTIONAL TEST shall include verification that the correct values of addressable constants are installed in each OPERABLE CPC. 2. Not required to be performed for logarithmic power level channels until 2 hours after reducing logarithmic power below 1E-4% NRTP. ---------- Perform CHANNEL FUNCTIONAL TEST on each channel.	92 days
SR 3.3.1.8	----------NOTE---------- Neutron detectors are excluded from the CHANNEL CALIBRATION. ---------- Perform CHANNEL CALIBRATION of the power range neutron flux channels.	92 days

(continued)

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SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.9	----------NOTE---------- Neutron detectors are excluded from CHANNEL CALIBRATION. ---------- Perform CHANNEL CALIBRATION on each channel, including bypass removal functions.	18 months
SR 3.3.1.10	Perform a CHANNEL FUNCTIONAL TEST on each CPC channel.	18 months
SR 3.3.1.11	Using the incore detectors, verify the shape annealing matrix elements to be used by the CPCs.	Once after each refueling prior to exceeding 70% RTP
SR 3.3.1.12	Perform a CHANNEL FUNCTIONAL TEST on each automatic bypass removal function.	Once within 92 days prior to each reactor startup
SR 3.3.1.13	----------NOTE---------- Neutron detectors are excluded. ---------- Verify RPS RESPONSE TIME is within limits.	18 months on a STAGGERED TEST BASIS

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Table 3.3.1-1 (page 1 of 3)
Reactor Protective System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Variable Over Power	1,2	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.6 SR 3.3.1.7 SR 3.3.1.8 SR 3.3.1.9 SR 3.3.1.13	Ceiling ≤ 111.0% RTP Band ≤ 9.9% RTP Incr. Rate ≤ 11.0%/min RTP Decr. Rate > 5%/sec RTP
2. Logarithmic Power Level – High[a]	2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.12 SR 3.3.1.13	≤ 0.011% NRTP
3. Pressurizer Pressure – High	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	≤ 2388 psia
4. Pressurizer Pressure – Low	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.12 SR 3.3.1.13	≥ 1821 psia
5. Containment Pressure – High	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	≤ 3.2 psig
6. Steam Generator #1 Pressure – Low	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	3876 MWt RTP: ≥ 890 psia 3990 MWt RTP: ≥ 955 psia[aa]
7. Steam Generator #2 Pressure – Low	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	3876 MWt RTP: ≥ 890 psia 3990 MWt RTP: ≥ 955 psia[aa]

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(continued)

(a) Trip may be bypassed when logarithmic power is > 1E-4% NRTP. Bypass shall be automatically removed when logarithmic power is ≤ 1E-4% NRTP.

(aa) 1. If the as-found channel setpoint is conservative with respect to the Allowable Value but outside its predetermined as-found acceptance criteria band, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service. If the as-found instrument channel setpoint is not conservative with respect to the Allowable Value, the channel shall be declared inoperable.

2. The instrument channel setpoint shall be reset to a value that is within the as-left tolerance of the UFSAR Trip Setpoint, or within the as left tolerance of a setpoint that is more conservative than the UFSAR Trip Set Point; otherwise the channel shall be declared inoperable. The UFSAR Trip Setpoint and the methodology used to determine 1) the UFSAR Trip Setpoint, 2) the predetermined as found acceptance criteria band, and 3) the as-left setpoint tolerance band are specified in the UFSAR.

Table 3.3.1-1 (page 2 of 3)
Reactor Protective System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
8. Steam Generator #1 Level – Low	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	≥ 43.7%
9. Steam Generator #2 Level – Low	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	≥ 43.7%
10. Steam Generator #1 Level – High	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	≤ 91.5%
11. Steam Generator #2 Level – High	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	≤ 91.5%
12. Reactor Coolant Flow, Steam Generator #1-Low	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	Ramp: ≤ 0.115 psid/sec. Floor: ≥ 12.49 psid Step: ≤ 17.2 psid
13. Reactor Coolant Flow, Steam Generator #2-Low	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	Ramp: ≤ 0.115 psid/sec. Floor: ≥ 12.49 psid Step: ≤ 17.2 psid

(continued)

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Table 3.3.1-1 (page 3 of 3)
Reactor Protective System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
14. Local Power Density – High[b]	1,2	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.3 SR 3.3.1.4 SR 3.3.1.5 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.10 SR 3.3.1.11 SR 3.3.1.12 SR 3.3.1.13	≤ 21.0 kW/ft
15. Departure From Nucleate Boiling Ratio (DNBR) – Low[b]	1,2	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.3 SR 3.3.1.4 SR 3.3.1.5 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.10 SR 3.3.1.11 SR 3.3.1.12 SR 3.3.1.13	≥ 1.34

(b) Trip may be bypassed when logarithmic power is < 1E-4% NRTP. Bypass shall be automatically removed when logarithmic power is ≥ 1E-4% NRTP.

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3.3 INSTRUMENTATION

3.3.1 Reactor Protective System (RPS) Instrumentation – Operating

LCO 3.3.1 Four RPS trip and bypass removal channels for each Function in Table 3.3.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1-1. ~~(After CPC Upgrade)~~

ACTIONS

----------NOTE----------
Separate Condition entry is allowed for each RPS Function.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more Functions with one automatic RPS trip channel inoperable.	A.1	Place channel in bypass or trip.	1 hour
	AND		
	A.2	Restore channel to OPERABLE status.	Prior to entering MODE 2 following next MODE 5 entry
B. One or more Functions with two automatic RPS trip channels inoperable.	B.1	Place one channel in bypass and the other in trip.	1 hour

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One or more Functions with one automatic bypass removal channel inoperable.	C.1 Disable bypass channel. OR C.2.1 Place affected automatic trip channel in bypass or trip. AND C.2.2 Restore bypass removal channel and associated automatic trip channel to OPERABLE status.	1 hour 1 hour Prior to entering MODE 2 following next MODE 5 entry
D. One or more Functions with two automatic bypass removal channels inoperable.	D.1 Disable bypass channels. OR D.2 Place one affected automatic trip channel in bypass and place the other in trip.	1 hour 1 hour
E. Required Action and associated Completion Time not met.	E.1 Be in MODE 3	6 hours

(continued)

SURVEILLANCE REQUIREMENTS

----------NOTE----------
Refer to Table 3.3.1-1 to determine which SR shall be performed for each RPS Function.

SURVEILLANCE		FREQUENCY
SR 3.3.1.1	Perform a CHANNEL CHECK of each RPS instrument channel.	12 hours
SR 3.3.1.2	----------NOTE---------- Not required to be performed until 12 hours after THERMAL POWER ≥ 70% RTP. --- Verify total Reactor Coolant System (RCS) flow rate as indicated by each CPC is less than or equal to the RCS total flow rate. If necessary, adjust the CPC addressable constant flow coefficients such that each CPC indicated flow is less than or equal to the RCS flow rate.	12 hours
SR 3.3.1.3	Check the CPC System Event Log.	12 hours

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.4	----------NOTES---------- 1. Not required to be performed until 12 hours after THERMAL POWER ≥ 20% RTP. 2. The daily calibration may be suspended during PHYSICS TESTS, provided the calibration is performed upon reaching each major test power plateau and prior to proceeding to the next major test power plateau. ---------- Perform calibration (heat balance only) and adjust the linear power level signals and the CPC addressable constant multipliers to make the CPC ΔT power and CPC nuclear power calculations agree with the calorimetric, if the absolute difference is ≥ 2% when THERMAL POWER is ≥ 80% RTP. Between 20% and 80% RTP the maximum difference is -0.5% to 10%.	24 hours
SR 3.3.1.5	----------NOTE---------- Not required to be performed until 12 hours after THERMAL POWER ≥ 70% RTP. ---------- Verify total RCS flow rate indicated by each CPC is less than or equal to the RCS flow determined either using the reactor coolant pump differential pressure instrumentation and the ultrasonic flow meter adjusted pump curves or by calorimetric calculations.	31 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.6	----------NOTE---------- Not required to be performed until 12 hours after THERMAL POWER ≥ 15% RTP. ---------- Verify linear power subchannel gains of the excore detectors are consistent with the values used to establish the shape annealing matrix elements in the CPCs.	31 days
SR 3.3.1.7	----------NOTES---------- 1. The CPC CHANNEL FUNCTIONAL TEST shall include verification that the correct values of addressable constants are installed in each OPERABLE CPC. 2. Not required to be performed for logarithmic power level channels until 2 hours after reducing logarithmic power below 1E-4% NRTP. ---------- Perform CHANNEL FUNCTIONAL TEST on each channel.	92 days
SR 3.3.1.8	----------NOTE---------- Neutron detectors are excluded from the CHANNEL CALIBRATION. ---------- Perform CHANNEL CALIBRATION of the power range neutron flux channels.	92 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.9	----------NOTE---------- Neutron detectors are excluded from CHANNEL CALIBRATION. ---------- Perform CHANNEL CALIBRATION on each channel, including bypass removal functions.	18 months
SR 3.3.1.10	Perform a CHANNEL FUNCTIONAL TEST on each CPC channel.	18 months
SR 3.3.1.11	Using the incore detectors, verify the shape annealing matrix elements to be used by the CPCs.	Once after each refueling prior to exceeding 70% RTP
SR 3.3.1.12	Perform a CHANNEL FUNCTIONAL TEST on each automatic bypass removal function.	Once within 92 days prior to each reactor startup
SR 3.3.1.13	----------NOTE---------- Neutron detectors are excluded. ---------- Verify RPS RESPONSE TIME is within limits.	18 months on a STAGGERED TEST BASIS

Table 3.3.1-1 (page 1 of 3)
Reactor Protective System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Variable Over Power	1,2	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.6 SR 3.3.1.7 SR 3.3.1.8 SR 3.3.1.9 SR 3.3.1.13	Ceiling ≤ 111.0% RTP Band ≤ 9.9% RTP Incr. Rate ≤ 11.0%/min RTP Decr. Rate > 5%/sec RTP
2. Logarithmic Power Level – High[a]	2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.12 SR 3.3.1.13	≤ 0.011% NRTP
3. Pressurizer Pressure – High	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	≤ 2388 psia
4. Pressurizer Pressure – Low	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.12 SR 3.3.1.13	≥ 1821 psia
5. Containment Pressure – High	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	≤ 3.2 psig
6. Steam Generator #1 Pressure – Low	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	~~3876 MWt RTP: ≥ 890 psia~~ 3990 MWt RTP: ≥ 955 psia[aa]
7. Steam Generator #2 Pressure – Low	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	~~3876 MWt RTP: ≥ 890 psia~~ 3990 MWt RTP: ≥ 955 psia[aa]

(continued)

(a) Trip may be bypassed when logarithmic power is > 1E-4% NRTP. Bypass shall be automatically removed when logarithmic power is ≤ 1E-4% NRTP.

(aa) 1. If the as-found channel setpoint is conservative with respect to the Allowable Value but outside its predetermined as-found acceptance criteria band, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service. If the as-found instrument channel setpoint is not conservative with respect to the Allowable Value, the channel shall be declared inoperable.

2. The instrument channel setpoint shall be reset to a value that is within the as-left tolerance of the UFSAR Trip Setpoint, or within the as left tolerance of a setpoint that is more conservative than the UFSAR Trip Set Point; otherwise the channel shall be declared inoperable. The UFSAR Trip Setpoint and the methodology used to determine 1) the UFSAR Trip Setpoint, 2) the predetermined as found acceptance criteria band, and 3) the as-left setpoint tolerance band are specified in the UFSAR.

Table 3.3.1-1 (page 2 of 3)
Reactor Protective System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
8. Steam Generator #1 Level – Low	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	≥ 43.7%
9. Steam Generator #2 Level – Low	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	≥ 43.7%
10. Steam Generator #1 Level – High	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	≤ 91.5%
11. Steam Generator #2 Level – High	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	≤ 91.5%
12. Reactor Coolant Flow, Steam Generator #1-Low	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	Ramp: ≤ 0.115 psid/sec. Floor: ≥ 12.49 psid Step: ≤ 17.2 psid
13. Reactor Coolant Flow, Steam Generator #2-Low	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	Ramp: ≤ 0.115 psid/sec. Floor: ≥ 12.49 psid Step: ≤ 17.2 psid

(continued)

Table 3.3.1-1 (page 3 of 3)
Reactor Protective System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
14. Local Power Density – High[b]	1,2	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.3 SR 3.3.1.4 SR 3.3.1.5 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.10 SR 3.3.1.11 SR 3.3.1.12 SR 3.3.1.13	≤ 21.0 kW/ft
15. Departure From Nucleate Boiling Ratio (DNBR) – Low[b]	1,2	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.3 SR 3.3.1.4 SR 3.3.1.5 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.10 SR 3.3.1.11 SR 3.3.1.12 SR 3.3.1.13	≥ 1.34

(b) Trip may be bypassed when logarithmic power is < 1E-4% NRTP. Bypass shall be automatically removed when logarithmic power is ≥ 1E-4% NRTP.

Table 3.3.2-1
Reactor Protective System Instrumentation - Shutdown

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALVE
1. Logarithmic Power Level-High[d]	3[a], 4[a], 5[a]	SR 3.3.2.1 SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.4 SR 3.3.2.5	≤ 0.011% NRTP[c]
2. Steam Generator #1 Pressure-Low[b]	3[a]	SR 3.3.2.1 SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.5	~~3876 MWt RTP: ≥ 890 psia~~ 3990 MWt RTP: ≥ 955 psia[e]
3. Steam Generator #2 Pressure-Low[b]	3[a]	SR 3.3.2.1 SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.5	~~3876 MWt RTP: ≥ 890 psia~~ 3990 MWt RTP: ≥ 955 psia[e]

(a) With any Reactor Trip Circuit Breakers (RTCBs) closed and any control element assembly capable of being withdrawn.

(b) The setpoint may be decreased as steam pressure is reduced, provided the margin between steam pressure and the setpoint is maintained ≤ 200 psig. The setpoint shall be automatically increased to the normal setpoint as steam pressure is increased.

(c) The setpoint must be reduced to ≤ 1E-4% NRTP when less than 4 RCPs are running.

(d) Trip may be bypassed when logarithmic power is > 1E-4% NRTP. Bypass shall be automatically removed when logarithmic power is ≤ 1E-4% NRTP.

(e) 1. If the as-found channel setpoint is conservative with respect to the Allowable Value but outside its predetermined as-found acceptance criteria band, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service. If the as-found instrument channel setpoint is not conservative with respect to the Allowable Value, the channel shall be declared inoperable.

2. The instrument channel setpoint shall be reset to a value that is within the as-left tolerance of the UFSAR Trip Setpoint, or within the as left tolerance of a setpoint that is more conservative than the UFSAR Trip Set Point; otherwise the channel shall be declared inoperable. The UFSAR Trip Setpoint and the methodology used to determine 1) the UFSAR Trip Setpoint, 2) the predetermined as found acceptance criteria band, and 3) the as-left setpoint tolerance band are specified in the UFSAR.

3.3 INSTRUMENTATION

3.3.3 Control Element Assembly Calculators (CEACs)

LCO 3.3.3 Two CEACs shall be OPERABLE.

APPLICABILITY: MODES 1 and 2. **(Before CPC Upgrade)**

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CEAC inoperable.	A.1 Perform SR 3.1.5.1.	Once per 4 hours
	AND	
	A.2 Restore CEAC to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met. OR Both CEACs inoperable.	B.1 Verify the departure from nucleate boiling ratio requirement of LCO 3.2.4, "Departure from Nucleate Boiling Ratio (DNBR)," is met.	4 hours
	AND	(continued)

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ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. (continued)	B.2	Verify all full strength and part length or part strength control element assembly (CEA) groups are fully withdrawn and maintained fully withdrawn, except during Surveillance testing pursuant to SR 3.1.5.3 or for control, when CEA group #5 may be inserted to a maximum of 127.5 inches withdrawn.	4 hours
	AND		
	B.3	Verify the "RSPT/CEAC Inoperable" addressable constant in each core protection calculator (CPC) is set to indicate that both CEACs are inoperable.	4 hours
	AND		
	~~B.4~~	Verify the Control Element Drive Mechanism Control System is placed in "STANDBY MODE" and maintained in "STANDBY MODE," except during CEA motion permitted by Required Action B.2.	4 hours
	AND		
	B.5	Perform SR 3.1.5.1.	Once per 4 hours
	AND		(continued)

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.6 Disable the Reactor Power Cutback System (RPCS)	4 hours
C. Receipt of a CPC channel B or C cabinet high temperature alarm.	C.1 Perform CHANNEL FUNCTIONAL TEST on affected CEAC(s).	12 hours
D. One or two CEACs with three or more auto restarts during a 12 hour period.	D.1 Perform CHANNEL FUNCTIONAL TEST on affected CEAC.	24 hours
E. Required Action and associated Completion Time of Condition B, C, or D not met.	E.1 Be in MODE 3.	6 hours

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.3.1	Perform a CHANNEL CHECK.	12 hours
SR 3.3.3.2	Check the CEAC auto restart count.	12 hours
SR 3.3.3.3	Perform a CHANNEL FUNCTIONAL TEST.	92 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.3.4	Perform a CHANNEL CALIBRATION.	18 months
SR 3.3.3.5	Perform a CHANNEL FUNCTIONAL TEST.	18 months
SR 3.3.3.6	Verify the isolation characteristics of each CEAC isolation amplifier.	18 months

Remove Recap

3.3 INSTRUMENTATION

3.3.3 Control Element Assembly Calculators (CEACs)

LCO 3.3.3 Two CEACs shall be OPERABLE in each CPC channel

APPLICABILITY: MODES 1 and 2. ~~(After CPC Upgrade)~~

ACTIONS

----------NOTE----------
Separate condition entry is allowed for each CPC channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CEAC inoperable in one or more CPC channels.	A.1 Declare the affected CPC channel(s) inoperable.	Immediately
	OR	
	A.2.1 Perform SR 3.1.5.1	Once per 4 hours
	AND	
	A.2.2 Restore CEAC to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met.	B.1 Declare the affected CPC channel(s) inoperable.	Immediately
OR	OR	
Both CEACs inoperable in one or more CPC channels.		(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.2.1 Verify the departure from nucleate boiling ratio requirement of LCO 3.2.4, "Departure from Nucleate Boiling Ratio (DNBR)," is met. AND	4 hours
	B.2.2 Verify all full strength and ~~part length or~~ part strength control element assembly (CEA) groups are fully withdrawn and maintained fully withdrawn, except during Surveillance testing pursuant to SR 3.1.5.3 or for control, when CEA group #5 may be inserted to a maximum of 127.5 inches withdrawn. AND	
	B.2.3 Verify the "RSPT/CEAC Inoperable" addressable constant in each affected core protection calculator (CPC) is set to indicate that both CEACs are inoperable. AND	4 hours
		(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	B.2.4 Verify the Control Element Drive Mechanism Control System is placed in "STANDBY MODE" and maintained in "STANDBY MODE," except during CEA motion permitted by Required Action B.2.2.	4 hours
	AND	
	B.2.5 Perform SR 3.1.5.1.	Once per 4 hours
	AND	
	B.2.6 Disable the Reactor Power Cutback System (RPCS)	4 hours
C. Required Action and associated Completion Time of Condition B not met.	C.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.3.1	Perform a CHANNEL CHECK.	12 hours
SR 3.3.3.2	Deleted	
SR 3.3.3.3	Perform a CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.3.4	Perform a CHANNEL CALIBRATION.	18 months
SR 3.3.3.5	Perform a CHANNEL FUNCTIONAL TEST.	18 months

Table 3.3.5-1 (page 1 of 1)
Engineered Safety Features Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	ALLOWABLE VALUE
1. Safety Injection Actuation Signal		
a. Containment Pressure - High	1,2,3	≤ 3.2 psig
b. Pressurizer Pressure - Low[a]		≥ 1821 psia
2. Containment Spray Actuation Signal		
a. Containment Pressure - High High	1,2,3	≤ 8.9 psig
3. Containment Isolation Actuation Signal		
a. Containment Pressure - High	1,2,3	≤ 3.2 psig
b. Pressurizer Pressure - Low[a]		≥ 1821 psia
4. Main Steam Isolation Signal[c]		
a. Steam Generator #1 Pressure-Low[b]	1,2,3	~~2876 MWt RTP: ≥ 800 psia~~ 3990 MWt RTP: ≥ 955 psia[d]
b. Steam Generator #2 Pressure-Low[b]		~~2876 MWt RTP: ≥ 800 psia~~ 3990 MWt RTP: ≥ 955 psia[d]
c. Steam Generator #1 Level-High		≤ 91.5%
d. Steam Generator #2 Level-High		≤ 91.5%
e. Containment Pressure-High		≤ 3.2 psig
5. Recirculation Actuation Signal		
a. Refueling Water Storage Tank Level-Low	1,2,3	≥ 6.9 and ≤ 7.9%
6. Auxiliary Feedwater Actuation Signal SG #1 (AFAS-1)		
a. Steam Generator #1 Level-Low	1,2,3	≥ 25.3%
b. SG Pressure Difference-High		≤ 192 psid
7. Auxiliary Feedwater Actuation Signal SG #2 (AFAS-2)		
a. Steam Generator #2 Level-Low	1,2,3	≥ 25.3%
b. SG Pressure Difference-High		≤ 192 psid

(a) The setpoint may be decreased to a minimum value of 100 psia, as pressurizer pressure is reduced, provided the margin between pressurizer pressure and the setpoint is maintained ≤ 400 psia or ≥ 140 psia greater than the saturation pressure of the RCS cold leg when the RCS cold leg temperature is ≥ 485°F. Trips may be bypassed when pressurizer pressure is < 400 psia. Bypass shall be automatically removed when pressurizer pressure is ≥ 500 psia. The setpoint shall be automatically increased to the normal setpoint as pressurizer pressure is increased.

(b) The setpoint may be decreased as steam pressure is reduced, provided the margin between steam pressure and the setpoint is maintained ≤ 200 psig. The setpoint shall be automatically increased to the normal setpoint as steam pressure is increased.

(c) The Main Steam Isolation Signal (MSIS) Function (Steam Generator Pressure - Low, Steam Generator Level-High and Containment Pressure - High signals) is not required to be OPERABLE when all associated valves isolated by the MSIS Function are closed.

(d) 1. If the as-found channel setpoint is conservative with respect to the Allowable Value but outside its predetermined as-found acceptance criteria band, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service. If the as-found instrument channel setpoint is not conservative with respect to the Allowable Value, the channel shall be declared inoperable.

2. The instrument channel setpoint shall be reset to a value that is within the as-left tolerance of the UFSAR Trip Setpoint, or within the as left tolerance of a setpoint that is more conservative than the UFSAR Trip Set Point; otherwise the channel shall be declared inoperable. The UFSAR Trip Setpoint and the methodology used to determine 1) the UFSAR Trip Setpoint, 2) the predetermined as found acceptance criteria band, and 3) the as-left setpoint tolerance band are specified in the UFSAR.

Figure 3.4.1-1, (Page 1 of 2)
Reactor Coolant Cold Leg Temperature vs. Core Power Level

3876 MWt RTP

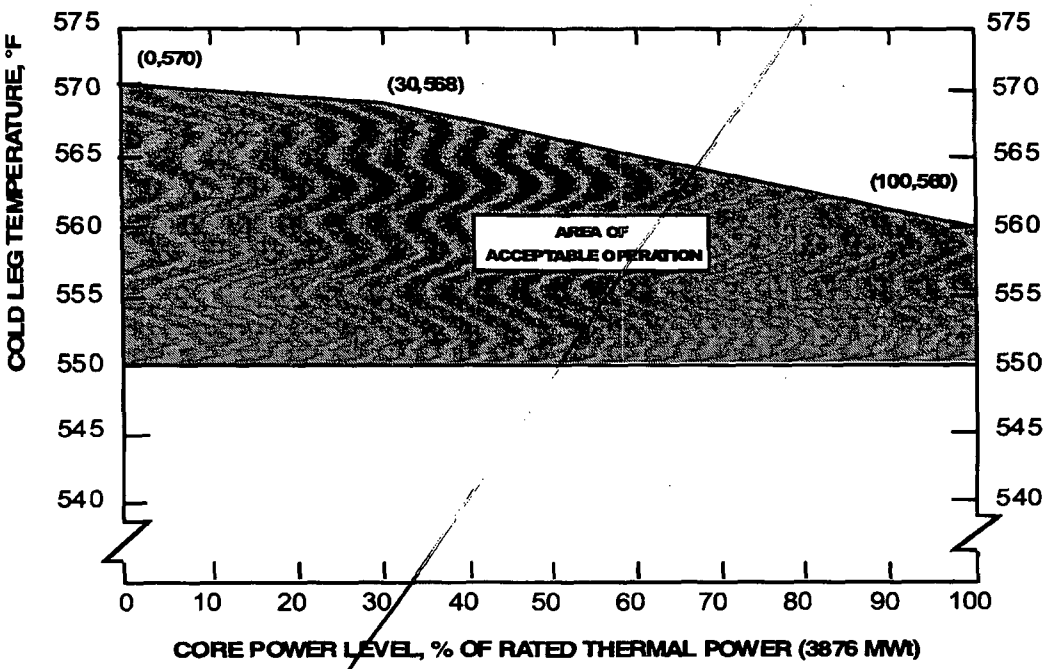

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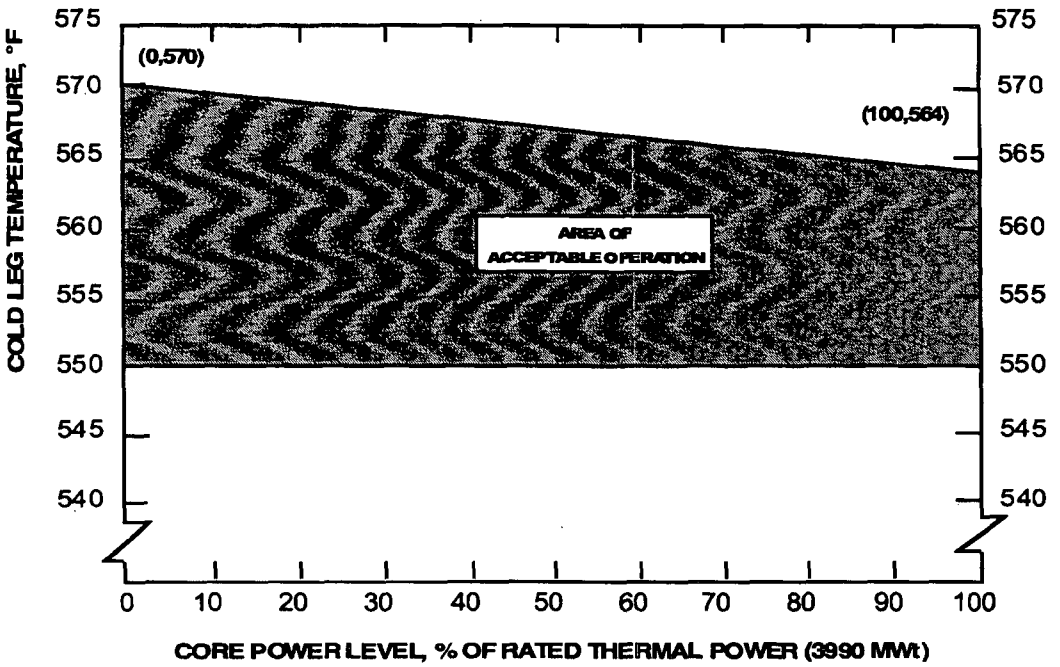
Figure 3.4.1-1, ~~(Page 2 of 2)~~
Reactor Coolant Cold Leg Temperature vs. Core Power Level

~~3990 MWt RTP~~



SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.3.7	Verify, for each ECCS throttle valve listed below, each position stop is in the correct position. LPSI System Valve Number: SIB-UV 615, SIB-UV 625, SIA-UV 635, SIA-UV 645, SIA-HV 306, SIB-HV 307 Hot Leg Injection Valve Numbers: SIC-HV 321, SID-HV 331	18 months
SR 3.5.3.8	Verify, by visual inspection, each ECCS train containment sump suction inlet is not restricted by debris and the suction inlet ~~trash racks and screens~~ Strainers show no evidence of structural distress or abnormal corrosion.	18 months

Table 3.7.1-1 (page 1 of 1)
Variable Overpower Trip Setpoint versus
OPERABLE Main Steam Safety Valves

MINIMUM NUMBER OF MSSVs PER STEAM GENERATOR REQUIRED OPERABLE	NUMBER OF INOPERABLE MSSVs PER STEAM GENERATOR	MAXIMUM POWER (% RTP) or HIGHEST MODE		MAXIMUM ALLOWABLE VARIABLE OVERPOWER TRIP SETPOINT[a] (% RTP)	
		~~3876 MWt RTP~~	~~3990 MWt RTP~~	~~3876 MWt RTP~~	~~3990 MWt RTP~~
10	0	100.0	100.0	111.0	111.0
9	1	98.2	90.0	108.0	99.7
8	2	87.3	80.0	97.1	89.7
7	3	76.4	68.0	86.2	77.7
6	4	65.5	56.0	75.3	65.7
5	5	MODE 3	MODE 3	NA	NA
4	6	MODE 3	MODE 3	NA	NA
3	7	MODE 3	MODE 3	NA	NA
2	8	MODE 3	MODE 3	NA	NA

(a) The VOPT setpoint is not required to be reset in MODE 3.

5.0 ADMINISTRATIVE CONTROLS

5.4 Procedures

5.4.1 Written procedures shall be established, implemented, and maintained covering the following activities:

a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978;

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and to NUREG-0737, Supplement 1, as stated in Generic Letter 82-33;

c. Quality assurance for effluent and environmental monitoring;

d. Fire Protection Program implementation; and

e. All programs specified in Specification 5.5.

f. Modification of core protection calculator (CPC) addressable constants. These procedures shall include provisions to ensure that sufficient margin is maintained in CPC type I addressable constants to avoid excessive operator interaction with CPCs during reactor operation.

Modifications to the CPC software (including changes of algorithms and fuel cycle specific data) shall be performed in accordance with the most recent version of "CPC Protection Algorithm Software Change Procedure," CEN-39(A)-P, which has been determined to be applicable to the facility. Additions or deletions to CPC addressable constants or changes to addressable constant software limit values shall not be implemented without prior NRC approval.

Revised Pag

5.0 ADMINISTRATIVE CONTROLS

5.4 Procedures

5.4.1 Written procedures shall be established, implemented, and maintained covering the following activities:

a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978;

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and to NUREG-0737, Supplement 1, as stated in Generic Letter 82-33;

c. Quality assurance for effluent and environmental monitoring;

d. Fire Protection Program implementation; and

e. All programs specified in Specification 5.5.

f. Modification of core protection calculator (CPC) addressable constants. These procedures shall include provisions to ensure that sufficient margin is maintained in CPC type I addressable constants to avoid excessive operator interaction with CPCs during reactor operation.

Modifications to the CPC software (including changes of algorithms and fuel cycle specific data) shall be performed in accordance with the most recent version of the Software Program Manual for Common Q Systems", CE-CES-195, which has been determined to be applicable to the facility. Additions or deletions to CPC addressable constants or changes to addressable constant software limit values shall not be implemented without prior NRC approval.

5.5 Programs and Manuals (continued)

5.5.9 Steam Generator (SG) Program (continued)

shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 0.5 gpm per SG and 1 gpm through both SGs.

3. The operational LEAKAGE performance criterion is specified in LCO 3.4.14, "RCS Operational LEAKAGE."

c. Provisions for SG tube repair criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.

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

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

2a. ~~Original SGs with Alloy 600 MA tubes: Inspect 100%~~ of the tu~~bes at sequential periods of 60 effecti~~ve full power months. The first sequential period shall be

(continued)

5.5 Programs and Manuals (continued)

5.5.9 Steam Generator (SG) Program (continued)

~~considered to begin after the first inservice inspection of the SGs. No SG shall operate for more than 24 effective full power months or one refueling outage (whichever is less) without being inspected.~~

2~~b~~. ~~(Replacement SGs with Alloy 690 TT tubes:~~ Inspect 100% of the tubes at sequential periods of 144, 108, 72, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected.

3. If crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary to secondary LEAKAGE.

(continued)

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1. Shutdown Margin - Reactor Trip Breakers Open for Specification 3.1.1.

2. Shutdown Margin - Reactor Trip Breakers Closed for Specification 3.1.2.

3. Moderator Temperature Coefficient BOL and EOL limits for Specification 3.1.4.

4. Boron Dilution Alarm System for Specification 3.3.12.

5. CEA Alignment for Specification 3.1.5.

6. Regulating CEA Insertion Limits for Specification 3.1.7.

7. ~~Part Length or~~ Part Strength CEA Insertion Limits for ~~Specification~~ 3.1.8.

8. Linear Heat Rate for Specification 3.2.1.

9. Azimuthal Power Tilt - T_q for Specification 3.2.3.

10. DNBR for Specification 3.2.4.

11. Axial Shape Index for Specification 3.2.5.

12. Boron Concentration (Mode 6) for Specification 3.9.1.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

----------NOTE----------

The COLR will contain the complete identification for each of the Technical Specification referenced topical reports used to prepare the COLR (i.e., report number, title, revision, date, and any supplements).

(continued)

5.6 Reporting Requirements (continued)

5.6.5 Core Operating Limits Report (COLR) (continued)

1. "CE Method for Control Element Assembly Ejection Analysis," CENPD-0190-A, (Methodology for Specification 3.1.7, Regulating CEA Insertion Limits).

2. "The ROCS and DIT Computer Codes for Nuclear Design," CENPD-266-P-A, [Methodology for Specifications 3.1.1, Shutdown Margin - Reactor Trip Breakers Open; 3.1.2, Shutdown Margin - Reactor Trip Breakers Closed; 3.1.4, Moderator Temperature Coefficient BOL and EOL limits; 3.1.7, Regulating CEA Insertion Limits and 3.9.1, Boron Concentration (Mode 6)].

3. "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 3.1.2, Shutdown Margin - Reactor Trip Breakers Closed; 3.1.4, Moderator Temperature Coefficient BOL and EOL limits; 3.3.12, Boron Dilution Alarm System; 3.1.5, CEA Alignment; 3.1.7, Regulating CEA Insertion Limits; 3.1.8, ~~Part Length or~~ Part Strength CEA Insertion Limits and 3.2.3, Azimuthal Power Tilt - T_q].

4. "Modified Statistical Combination of Uncertainties," CEN-356(V)-P-A and "System 80™ Inlet Flow Distribution," Supplement 1-P to Enclosure 1-P to LD-82-054, (Methodology for Specification 3.2.4, DNBR and 3.2.5 Axial Shape Index).

5. "Calculative Methods for the CE Large Break LOCA Evaluation Model," CENPD-132, (Methodology for Specification 3.2.1, Linear Heat Rate).

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

(continued)

5.6 Reporting Requirements

5.6.5 Core Operating Limits Report (COLR) (continued)

7. 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: 5.6.5.b.6.

8. 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 5.6.5.b.6.

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

10. 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 5.6.5.b.9.

11. "Arizona Public Service Company PWR Reactor Physics Methodology Using CASMO-4/SIMULATE-3," [Methodology for Specifications 3.1.1, Shutdown Margin - Reactor Trip Breakers Open; 3.1.2, Shutdown Margin - Reactor Trip Breakers Closed; 3.1.4, Moderator Temperature Coefficient; 3.1.7, Regulating CEA Insertion Limits and 3.9.1, Boron Concentration (Mode 6)].

12. "Technical Manual for the CENTS Code," CE-NPD 282-P-A, Volumes 1-3, [Methodology for Specifications 3.1.2, Shutdown Margin-Reactor Trip Breakers Closed; 3.1.4, Moderator Temperature Coefficient; 3.1.5, CEA Alignment; 3.1.7, Regulating CEA Insertion Limits; 3.1.8, ~~Part Length or~~ Part Strength CEA Insertion Limits and 3.2.3, Azimuthal Power Tilt- T_q].

13. CENPD-404-P-A, "Implementation of ZIRLO™ Cladding Material in CE Nuclear Power Fuel Assembly Designs.

14. CENPD-188-A, "HERMITE, A Multi-Dimensional Space-Time Kinetics Code for PWR Transients." [Methodology for Specifications 3.1.2, Shutdown Margin-Reactor Trip Breakers Closed; 3.1.4, Moderator Temperature Coefficient; 3.2.1, Linear Heat Rate; 3.2.3, Azimuthal Power Tilt; 3.2.4, DNBR; and 3.2.5, Axial Shape Index.]

15. CENPD-161-P-A, "TORC Code, A Computer Code for Determining the Thermal Margin of a Reactor Core." [Methodology for Specifications 3.1.1, Shutdown Margin-Reactor Trip Breakers Open; 3.1.2, Shutdown Margin-Reactor Trip Breakers Closed; 3.1.4, Moderator Temperature Coefficient; 3.2.1, Linear Heat

(continued)

Proposed Technical Specification Changes (retyped)

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PALO VERDE NUCLEAR GENERATING STATION
IMPROVED TECHNICAL SPECIFICATIONS
TABLE OF CONTENTS

1.1 Definitions (continued)

RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3990 MWt.
REACTOR PROTECTIVE SYSTEM (RPS) RESPONSE TIME	The RPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until electrical power to the CEAs drive mechanism is interrupted. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.
SHUTDOWN MARGIN (SDM)	SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming: a. All full strength CEAs (shutdown and regulating) are fully inserted except for the single CEA of highest reactivity worth, which is assumed to be fully withdrawn. With any full strength CEAs not capable of being fully inserted, the withdrawn reactivity worth of these CEAs must be accounted for in the determination of SDM and b. There is no change in part strength CEA position.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.5 Control Element Assembly (CEA) Alignment

LCO 3.1.5 All full strength CEAs shall be OPERABLE, and all full strength and part strength CEAs shall be aligned to within 6.6 inches (indicated position) of all other CEAs in their respective groups.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more CEAs trippable and misaligned from its group by > 6.6 inches and ≤ 9.9 inches. OR One CEA trippable and misaligned from its group by > 9.9 inches.	A.1 Reduce THERMAL POWER in accordance with the limits in the COLR. AND A.2 Restore CEA alignment.	1 hour 2 hours

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.5.1	Verify the indicated position of each full strength and part strength CEA is within 6.6 inches of all other CEAs in its group.	12 hours
SR 3.1.5.2	Verify that, for each CEA, its OPERABLE CEA position indicator channels indicate within 5.2 inches of each other.	12 hours
SR 3.1.5.3	Verify full strength CEA freedom of movement (trippability) by moving each individual full strength CEA that is not fully inserted in the core at least 5 inches.	92 days
SR 3.1.5.4	Perform a CHANNEL FUNCTIONAL TEST of each reed switch position transmitter channel.	18 months
SR 3.1.5.5	Verify each full strength CEA drop time ≤ 4.0 seconds.	Prior to reactor criticality, after each removal of the reactor head

3.1 REACTIVITY CONTROL SYSTEMS

3.1.8 Part Strength Control Element Assembly (CEA) Insertion Limits

LCO 3.1.8 The part strength CEA groups shall be limited to the insertion limits specified in the COLR.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Part strength CEA groups inserted beyond the transient insertion limit.	A.1 Restore part strength CEA groups to within the limit.	2 hours
	OR	
	A.2 Reduce THERMAL POWER to less than or equal to that fraction of RTP specified in the COLR.	2 hours
B. Part strength CEA groups inserted between the long term steady state insertion limit and the transient insertion limit for intervals ≥ 7 effective full power days (EFPD) per 30 EFPD or ≥ 14 EFPD per 365 EFPD interval.	B.1 Restore part strength CEA groups to within the long term steady state insertion limit.	2 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.8.1 Verify part strength CEA group position.	12 hours

3.1 REACTIVITY CONTROL SYSTEMS

3.1.10 Special Test Exceptions (STE) – MODES 1 and 2

LCO 3.1.10 During performance of PHYSICS TESTS, the requirements of:

LCO 3.1.4, "Moderator Temperature Coefficient (MTC)";
LCO 3.1.5, "Control Element Assembly (CEA) Alignment";
LCO 3.1.6, "Shutdown Control Element Assembly (CEA) Insertion Limits";
LCO 3.1.7, "Regulating Control Element Assembly (CEA) Insertion Limits";
LCO 3.1.8, "Part Strength CEA Insertion Limits";
LCO 3.2.2, "Planar Radial Peaking Factors (Fxy)";
LCO 3.2.3, "AZIMUTHAL POWER TILT (Tq)";
LCO 3.2.5, "AXIAL SHAPE INDEX (ASI)"; and
LCO 3.3.3, "Control Element Assembly Calculators (CEACs)"

may be suspended, provided:

a. THERMAL POWER is restricted to the test power plateau, which shall not exceed 85% RTP: and

b. Shutdown reactivity equivalent to at least the highest estimated CEA worth (of those CEAs actually withdrawn) is available for trip insertion.

APPLICABILITY: MODES 1 and 2 during PHYSICS TESTS.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Test power plateau exceeded.	A.1 Reduce THERMAL POWER to less than or equal to the test power plateau.	15 minutes
B. Required Action and associated Completion Time not met.	B.1 Suspend PHYSICS TESTS.	1 hour

3.1 REACTIVITY CONTROL SYSTEMS

3.1.11 Special Test Exceptions (STE) – Reactivity Coefficient Testing

LCO 3.1.11 During performance of PHYSICS TESTS, the requirements of:

LCO 3.1.7, "Regulating Control Element Assembly (CEA) Insertion Limits";
LCO 3.1.8, "Part Strength Control Element Assembly (CEA) Insertion Limits;" and
LCO 3.4.1, "RCS Pressure, Temperature and Flow limits" (LCO 3.4.1.b, RCS Cold Leg Temperature only)

may be suspended, provided LHR and DNBR do not exceed the limits in the COLR.

APPLICABILITY: MODE 1 with Thermal Power > 20% RTP during PHYSICS TESTS.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LHR or DNBR outside the limits specified in the COLR.	A.1 Reduce THERMAL POWER to restore LHR and DNBR to within limits.	15 minutes
B. Required Action and associated Completion Time not met.	B.1 Suspend PHYSICS TESTS.	1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.11.1	Verify LHR and DNBR do not exceed limits by performing SR 3.2.1.1 and SR 3.2.4.1.	Continuously

3.2 POWER DISTRIBUTION LIMITS

3.2.4 Departure From Nucleate Boiling Ratio (DNBR)

LCO 3.2.4 The DNBR shall be maintained by one of the following methods:

a. Core Operating Limit Supervisory System (COLSS) In Service:

1. Maintaining COLSS calculated core power less than or equal to COLSS calculated core power operating limit based on DNBR when at least one Control Element Assembly Calculator (CEAC) is OPERABLE in each OPERABLE Core Protection Calculator (CPC) channel; or

2. Maintaining COLSS calculated core power less than or equal to COLSS calculated core power operating limit based on DNBR decreased by the allowance specified in the COLR when the CEAC requirements of LCO 3.2.4.a.1 are not met.

b. COLSS Out of Service:

1. Operating within the region of acceptable operation specified in the COLR using any OPERABLE Core Protection Calculator (CPC) channel when at least one Control Element Assembly Calculator (CEAC) is OPERABLE in each OPERABLE CPC channel; or

2. Operating within the region of acceptable operation specified in the COLR using any OPERABLE CPC channel (with both CEACs inoperable) when the CEAC requirements of LCO 3.2.4.b.1 are not met.

APPLICABILITY: MODE 1 with THERMAL POWER > 20% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. COLSS calculated core power not within limit.	A.1 Restore the DNBR to within limit.	1 hour

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. DNBR outside the region of acceptable operation when COLSS is out of service.	B.1 Determine trend in DNBR.	Once per 15 minutes
	AND	
	B.2.1 With an adverse trend, restore DNBR to within limit.	1 hour
	OR	
	B.2.2 With no adverse trend, restore DNBR to within limit.	4 hours
C. Required Action and associated Completion Time not met.	C.1 Reduce THERMAL POWER to ≤ 20% RTP.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.4.1	----------NOTE---------- 1. Only applicable when COLSS is out of service. With COLSS in service, this parameter is continuously monitored. 2. Not required to be performed until 2 hours after MODE 1 with THERMAL POWER > 20% RTP. ---------- Verify DNBR, as indicated on any OPERABLE DNBR channels, is within the limit of the COLR, as applicable.	2 hours
SR 3.2.4.2	Verify COLSS margin alarm actuates at a THERMAL POWER level equal to or less than the core power operating limit based on DNBR.	31 days

3.3 INSTRUMENTATION

3.3.1 Reactor Protective System (RPS) Instrumentation – Operating

LCO 3.3.1 Four RPS trip and bypass removal channels for each Function in Table 3.3.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1-1.

ACTIONS

----------NOTE----------
Separate Condition entry is allowed for each RPS Function.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more Functions with one automatic RPS trip channel inoperable.	A.1	Place channel in bypass or trip.	1 hour
	AND		
	A.2	Restore channel to OPERABLE status.	Prior to entering MODE 2 following next MODE 5 entry
B. One or more Functions with two automatic RPS trip channels inoperable.	B.1	Place one channel in bypass and the other in trip.	1 hour

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One or more Functions with one automatic bypass removal channel inoperable.	C.1 Disable bypass channel.	1 hour
	OR	
	C.2.1 Place affected automatic trip channel in bypass or trip.	1 hour
	AND	
	C.2.2 Restore bypass removal channel and associated automatic trip channel to OPERABLE status.	Prior to entering MODE 2 following next MODE 5 entry.
D. One or more Functions with two automatic bypass removal channels inoperable.	D.1 Disable bypass channels.	1 hour
	OR	
	D.2 Place one affected automatic trip channel in bypass and place the other in trip.	1 hour
E. Required Action and associated Completion Time not met.	E.1 Be in MODE 3	6 hours

(continued)

SURVEILLANCE REQUIREMENTS

----------NOTE----------
Refer to Table 3.3.1-1 to determine which SR shall be performed for each RPS Function.

SURVEILLANCE		FREQUENCY
SR 3.3.1.1	Perform a CHANNEL CHECK of each RPS instrument channel.	12 hours
SR 3.3.1.2	----------NOTE---------- Not required to be performed until 12 hours after THERMAL POWER ≥ 70% RTP. --- Verify total Reactor Coolant System (RCS) flow rate as indicated by each CPC is less than or equal to the RCS total flow rate. If necessary, adjust the CPC addressable constant flow coefficients such that each CPC indicated flow is less than or equal to the RCS flow rate.	12 hours
SR 3.3.1.3	Check the CPC System Event Log.	12 hours

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.4	----------NOTES---------- 1. Not required to be performed until 12 hours after THERMAL POWER ≥ 20% RTP. 2. The daily calibration may be suspended during PHYSICS TESTS, provided the calibration is performed upon reaching each major test power plateau and prior to proceeding to the next major test power plateau. ---------- Perform calibration (heat balance only) and adjust the linear power level signals and the CPC addressable constant multipliers to make the CPC ΔT power and CPC nuclear power calculations agree with the calorimetric, if the absolute difference is ≥ 2% when THERMAL POWER is ≥ 80% RTP. Between 20% and 80% RTP the maximum difference is -0.5% to 10%.	24 hours
SR 3.3.1.5	----------NOTE---------- Not required to be performed until 12 hours after THERMAL POWER ≥ 70% RTP. ---------- Verify total RCS flow rate indicated by each CPC is less than or equal to the RCS flow determined either using the reactor coolant pump differential pressure instrumentation and the ultrasonic flow meter adjusted pump curves or by calorimetric calculations.	31 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.6	----------NOTE---------- Not required to be performed until 12 hours after THERMAL POWER ≥ 15% RTP. ---------- Verify linear power subchannel gains of the excore detectors are consistent with the values used to establish the shape annealing matrix elements in the CPCs.	31 days
SR 3.3.1.7	----------NOTES---------- 1. The CPC CHANNEL FUNCTIONAL TEST shall include verification that the correct values of addressable constants are installed in each OPERABLE CPC. 2. Not required to be performed for logarithmic power level channels until 2 hours after reducing logarithmic power below 1E-4% NRTP. ---------- Perform CHANNEL FUNCTIONAL TEST on each channel.	92 days
SR 3.3.1.8	----------NOTE---------- Neutron detectors are excluded from the CHANNEL CALIBRATION. ---------- Perform CHANNEL CALIBRATION of the power range neutron flux channels.	92 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.9	----------NOTE---------- Neutron detectors are excluded from CHANNEL CALIBRATION. ---------- Perform CHANNEL CALIBRATION on each channel, including bypass removal functions.	18 months
SR 3.3.1.10	Perform a CHANNEL FUNCTIONAL TEST on each CPC channel.	18 months
SR 3.3.1.11	Using the incore detectors, verify the shape annealing matrix elements to be used by the CPCs.	Once after each refueling prior to exceeding 70% RTP
SR 3.3.1.12	Perform a CHANNEL FUNCTIONAL TEST on each automatic bypass removal function.	Once within 92 days prior to each reactor startup
SR 3.3.1.13	----------NOTE---------- Neutron detectors are excluded. ---------- Verify RPS RESPONSE TIME is within limits.	18 months on a STAGGERED TEST BASIS

Table 3.3.1-1 (page 1 of 3)
Reactor Protective System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Variable Over Power	1,2	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.6 SR 3.3.1.7 SR 3.3.1.8 SR 3.3.1.9 SR 3.3.1.13	Ceiling ≤ 111.0% RTP Band ≤ 9.9% RTP Incr. Rate ≤ 11.0%/min RTP Decr. Rate > 5%/sec RTP
2. Logarithmic Power Level – High[a]	2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.12 SR 3.3.1.13	≤ 0.011% NRTP
3. Pressurizer Pressure – High	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	≤ 2388 psia
4. Pressurizer Pressure – Low	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.12 SR 3.3.1.13	≥ 1821 psia
5. Containment Pressure – High	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	≤ 3.2 psig
6. Steam Generator #1 Pressure – Low	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	3990 MWt RTP: ≥ 955 psia[aa]
7. Steam Generator #2 Pressure – Low	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	3990 MWt RTP: ≥ 955 psia[aa]

(continued)

(a) Trip may be bypassed when logarithmic power is > 1E-4% NRTP. Bypass shall be automatically removed when logarithmic power is ≤ 1E-4% NRTP.

(aa) 1. If the as-found channel setpoint is conservative with respect to the Allowable Value but outside its predetermined as-found acceptance criteria band, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service. If the as-found instrument channel setpoint is not conservative with respect to the Allowable Value, the channel shall be declared inoperable.

2. The instrument channel setpoint shall be reset to a value that is within the as-left tolerance of the UFSAR Trip Setpoint, or within the as left tolerance of a setpoint that is more conservative than the UFSAR Trip Set Point; otherwise the channel shall be declared inoperable. The UFSAR Trip Setpoint and the methodology used to determine 1) the UFSAR Trip Setpoint, 2) the predetermined as found acceptance criteria band, and 3) the as-left setpoint tolerance band are specified in the UFSAR.

Table 3.3.1-1 (page 2 of 3)
Reactor Protective System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
8. Steam Generator #1 Level – Low	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	≥ 43.7%
9. Steam Generator #2 Level – Low	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	≥ 43.7%
10. Steam Generator #1 Level – High	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	≤ 91.5%
11. Steam Generator #2 Level – High	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	≤ 91.5%
12. Reactor Coolant Flow, Steam Generator #1-Low	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	Ramp: ≤ 0.115 psid/sec. Floor: ≥ 12.49 psid Step: ≤ 17.2 psid
13. Reactor Coolant Flow, Steam Generator #2-Low	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	Ramp: ≤ 0.115 psid/sec. Floor: ≥ 12.49 psid Step: ≤ 17.2 psid

(continued)

Table 3.3.1-1 (page 3 of 3)
Reactor Protective System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
14. Local Power Density – High[b]	1,2	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.3 SR 3.3.1.4 SR 3.3.1.5 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.10 SR 3.3.1.11 SR 3.3.1.12 SR 3.3.1.13	≤ 21.0 kW/ft
15. Departure From Nucleate Boiling Ratio (DNBR) – Low[b]	1,2	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.3 SR 3.3.1.4 SR 3.3.1.5 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.10 SR 3.3.1.11 SR 3.3.1.12 SR 3.3.1.13	≥ 1.34

(b) Trip may be bypassed when logarithmic power is < 1E-4% NRTP. Bypass shall be automatically removed when logarithmic power is ≥ 1E-4% NRTP.

Table 3.3.2-1
Reactor Protective System Instrumentation - Shutdown

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Logarithmic Power Level-High[d]	3[a], 4[a], 5[a]	SR 3.3.2.1 SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.4 SR 3.3.2.5	≤ 0.011% NRTP[c]
2. Steam Generator #1 Pressure-Low[b]	3[a]	SR 3.3.2.1 SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.5	3990 MWt RTP: ≥ 955 psia[e]
3. Steam Generator #2 Pressure-Low[b]	3[a]	SR 3.3.2.1 SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.5	3990 MWt RTP: ≥ 955 psia[e]

(a) With any Reactor Trip Circuit Breakers (RTCBs) closed and any control element assembly capable of being withdrawn.

(b) The setpoint may be decreased as steam pressure is reduced, provided the margin between steam pressure and the setpoint is maintained ≤ 200 psig. The setpoint shall be automatically increased to the normal setpoint as steam pressure is increased.

(c) The setpoint must be reduced to ≤ 1E-4% NRTP when less than 4 RCPs are running.

(d) Trip may be bypassed when logarithmic power is > 1E-4% NRTP. Bypass shall be automatically removed when logarithmic power is ≤ 1E-4% NRTP.

(e) 1. If the as-found channel setpoint is conservative with respect to the Allowable Value but outside its predetermined as-found acceptance criteria band, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service. If the as-found instrument channel setpoint is not conservative with respect to the Allowable Value, the channel shall be declared inoperable.

2. The instrument channel setpoint shall be reset to a value that is within the as-left tolerance of the UFSAR Trip Setpoint, or within the as left tolerance of a setpoint that is more conservative than the UFSAR Trip Set Point; otherwise the channel shall be declared inoperable. The UFSAR Trip Setpoint and the methodology used to determine 1) the UFSAR Trip Setpoint, 2) the predetermined as found acceptance criteria band, and 3) the as-left setpoint tolerance band are specified in the UFSAR.

3.3 INSTRUMENTATION

3.3.3 Control Element Assembly Calculators (CEACs)

LCO 3.3.3 Two CEACs shall be OPERABLE in each CPC channel

APPLICABILITY: MODES 1 and 2.

ACTIONS

----------NOTE----------
Separate condition entry is allowed for each CPC channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CEAC inoperable in one or more CPC channels.	A.1 Declare the affected CPC channel(s) inoperable.	Immediately
	OR	
	A.2.1 Perform SR 3.1.5.1	Once per 4 hours
	AND	
	A.2.2 Restore CEAC to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met.	B.1 Declare the affected CPC channel(s) inoperable.	Immediately
OR	OR	
Both CEACs inoperable in one or more CPC channels.		(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.2.1 Verify the departure from nucleate boiling ratio requirement of LCO 3.2.4, "Departure from Nucleate Boiling Ratio (DNBR)," is met. AND B.2.2 Verify all full strength and part strength control element assembly (CEA) groups are fully withdrawn and maintained fully withdrawn, except during Surveillance testing pursuant to SR 3.1.5.3 or for control, when CEA group #5 may be inserted to a maximum of 127.5 inches withdrawn. AND B.2.3 Verify the "RSPT/CEAC Inoperable" addressable constant in each affected core protection calculator (CPC) is set to indicate that both CEACs are inoperable. AND	4 hours 4 hours (continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	B.2.4 Verify the Control Element Drive Mechanism Control System is placed in "STANDBY MODE" and maintained in "STANDBY MODE," except during CEA motion permitted by Required Action B.2.2.	4 hours
	AND	
	B.2.5 Perform SR 3.1.5.1.	Once per 4 hours
	AND	
	B.2.6 Disable the Reactor Power Cutback System (RPCS)	4 hours
C. Required Action and associated Completion Time of Condition B not met.	C.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.3.1	Perform a CHANNEL CHECK.	12 hours
SR 3.3.3.2	Deleted	
SR 3.3.3.3	Perform a CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.3.4	Perform a CHANNEL CALIBRATION.	18 months
SR 3.3.3.5	Perform a CHANNEL FUNCTIONAL TEST.	18 months

Table 3.3.5-1 (page 1 of 1)
Engineered Safety Features Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	ALLOWABLE VALUE
1. Safety Injection Actuation Signal		
a. Containment Pressure – High	1,2,3	≤ 3.2 psig
b. Pressurizer Pressure – Low[a]		≥ 1821 psia
2. Containment Spray Actuation Signal		
a. Containment Pressure – High High	1,2,3	≤ 8.9 psig
3. Containment Isolation Actuation Signal		
a. Containment Pressure – High	1,2,3	≤ 3.2 psig
b. Pressurizer Pressure – Low[a]		≥ 1821 psia
4. Main Steam Isolation Signal[c]		
a. Steam Generator #1 Pressure–Low[b]	1,2,3	3990 MWt RTP: ≥ 955 psia[d]
b. Steam Generator #2 Pressure–Low[b]		3990 MWt RTP: ≥ 955 psia[d]
c. Steam Generator #1 Level-High		≤ 91.5%
d. Steam Generator #2 Level-High		≤ 91.5%
e. Containment Pressure-High		≤ 3.2 psig
5. Recirculation Actuation Signal		
a. Refueling Water Storage Tank Level–Low	1,2,3	≥ 6.9 and ≤ 7.9%
6. Auxiliary Feedwater Actuation Signal SG #1 (AFAS-1)		
a. Steam Generator #1 Level–Low	1,2,3	≥ 25.3%
b. SG Pressure Difference–High		≤ 192 psid
7. Auxiliary Feedwater Actuation Signal SG #2 (AFAS-2)		
a. Steam Generator #2 Level–Low	1,2,3	≥ 25.3%
b. SG Pressure Difference–High		≤ 192 psid

(a) The setpoint may be decreased to a minimum value of 100 psia, as pressurizer pressure is reduced, provided the margin between pressurizer pressure and the setpoint is maintained ≤ 400 psia or ≥ 140 psia greater than the saturation pressure of the RCS cold leg when the RCS cold leg temperature is ≥ 485°F. Trips may be bypassed when pressurizer pressure is < 400 psia. Bypass shall be automatically removed when pressurizer pressure is ≥ 500 psia. The setpoint shall be automatically increased to the normal setpoint as pressurizer pressure is increased.

(b) The setpoint may be decreased as steam pressure is reduced, provided the margin between steam pressure and the setpoint is maintained ≤ 200 psig. The setpoint shall be automatically increased to the normal setpoint as steam pressure is increased.

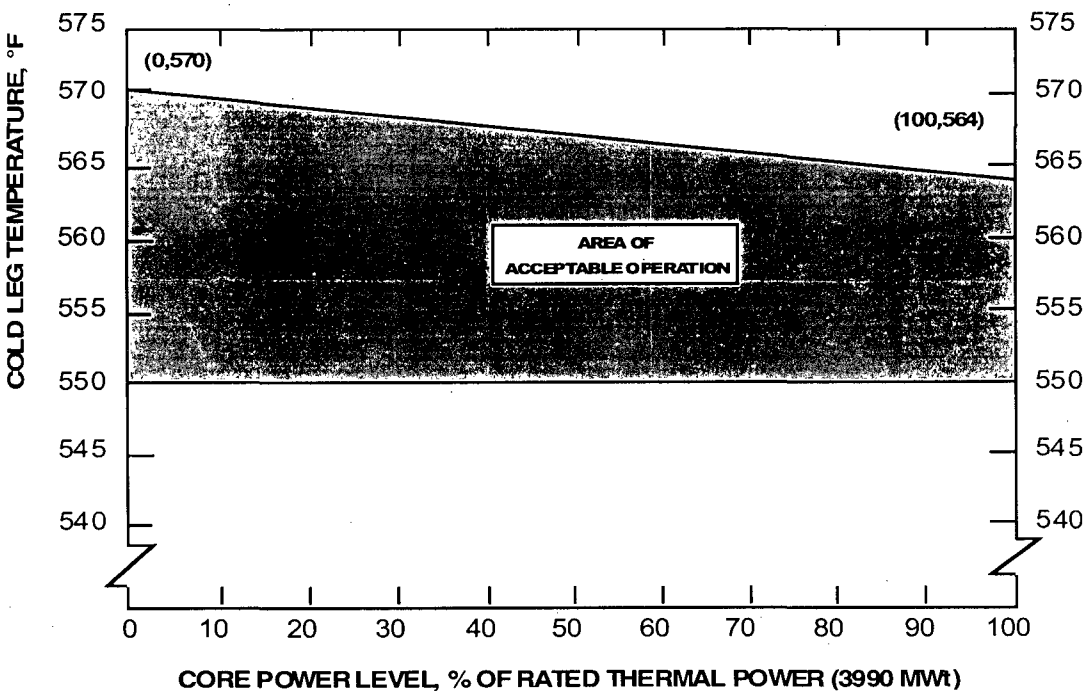
(c) The Main Steam Isolation Signal (MSIS) Function (Steam Generator Pressure – Low, Steam Generator Level-High and Containment Pressure – High signals) is not required to be OPERABLE when all associated valves isolated by the MSIS Function are closed.

(d) 1. If the as-found channel setpoint is conservative with respect to the Allowable Value but outside its predetermined as-found acceptance criteria band, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service. If the as-found instrument channel setpoint is not conservative with respect to the Allowable Value, the channel shall be declared inoperable.

2. The instrument channel setpoint shall be reset to a value that is within the as-left tolerance of the UFSAR Trip Setpoint, or within the as left tolerance of a setpoint that is more conservative than the UFSAR Trip Set Point; otherwise the channel shall be declared inoperable. The UFSAR Trip Setpoint and the methodology used to determine 1) the UFSAR Trip Setpoint, 2) the predetermined as found acceptance criteria band, and 3) the as-left setpoint tolerance band are specified in the UFSAR.

Figure 3.4.1-1

Reactor Coolant Cold Leg Temperature vs. Core Power Level



SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.3.7	Verify, for each ECCS throttle valve listed below, each position stop is in the correct position. LPSI System Valve Number: SIB-UV 615, SIB-UV 625, SIA-UV 635, SIA-UV 645, SIA-HV 306, SIB-HV 307 Hot Leg Injection Valve Numbers: SIC-HV 321, SID-HV 331	18 months
SR 3.5.3.8	Verify, by visual inspection, each ECCS train containment sump suction inlet is not restricted by debris and the suction inlet strainers show no evidence of structural distress or abnormal corrosion.	18 months

Table 3.7.1-1 (page 1 of 1)
Variable Overpower Trip Setpoint versus
OPERABLE Main Steam Safety Valves

MINIMUM NUMBER OF MSSVs PER STEAM GENERATOR REQUIRED OPERABLE	NUMBER OF INOPERABLE MSSVs PER STEAM GENERATOR	MAXIMUM POWER (% RTP) or HIGHEST MODE	MAXIMUM ALLOWABLE VARIABLE OVERPOWER TRIP SETPOINT[a] (% RTP)
10	0	100.0	111.0
9	1	90.0	99.7
8	2	80.0	89.7
7	3	68.0	77.7
6	4	56.0	65.7
5	5	MODE 3	NA
4	6	MODE 3	NA
3	7	MODE 3	NA
2	8	MODE 3	NA

(a) The VOPT setpoint is not required to be reset in MODE 3.

5.0 ADMINISTRATIVE CONTROLS

5.4 Procedures

5.4.1 Written procedures shall be established, implemented, and maintained covering the following activities:

a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978;

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and to NUREG-0737, Supplement 1, as stated in Generic Letter 82-33;

c. Quality assurance for effluent and environmental monitoring;

d. Fire Protection Program implementation; and

e. All programs specified in Specification 5.5.

f. Modification of core protection calculator (CPC) addressable constants. These procedures shall include provisions to ensure that sufficient margin is maintained in CPC type I addressable constants to avoid excessive operator interaction with CPCs during reactor operation.

Modifications to the CPC software (including changes of algorithms and fuel cycle specific data) shall be performed in accordance with the most recent version of the Software Program Manual for Common Q Systems", CE-CES-195, which has been determined to be applicable to the facility. Additions or deletions to CPC addressable constants or changes to addressable constant software limit values shall not be implemented without prior NRC approval.

5.5 Programs and Manuals (continued)

5.5.9 Steam Generator (SG) Program (continued)

shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 0.5 gpm per SG and 1 gpm through both SGs.

3. The operational LEAKAGE performance criterion is specified in LCO 3.4.14, "RCS Operational LEAKAGE."

c. Provisions for SG tube repair criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.

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

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

2. Inspect 100% of the tubes at sequential periods of 144, 108, 72, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected.

(continued)

5.5 Programs and Manuals (continued)

5.5.9 Steam Generator (SG) Program (continued)

3. If crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary to secondary LEAKAGE.

(continued)

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1. Shutdown Margin - Reactor Trip Breakers Open for Specification 3.1.1.

2. Shutdown Margin - Reactor Trip Breakers Closed for Specification 3.1.2.

3. Moderator Temperature Coefficient BOL and EOL limits for Specification 3.1.4.

4. Boron Dilution Alarm System for Specification 3.3.12.

5. CEA Alignment for Specification 3.1.5.

6. Regulating CEA Insertion Limits for Specification 3.1.7.

7. Part Strength CEA Insertion Limits for Specification 3.1.8.

8. Linear Heat Rate for Specification 3.2.1.

9. Azimuthal Power Tilt - T_q for Specification 3.2.3.

10. DNBR for Specification 3.2.4.

11. Axial Shape Index for Specification 3.2.5.

12. Boron Concentration (Mode 6) for Specification 3.9.1.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

----------NOTE----------
The COLR will contain the complete identification for each of the Technical Specification referenced topical reports used to prepare the COLR (i.e., report number, title, revision, date, and any supplements).

(continued)

5.6 Reporting Requirements (continued)

5.6.5 Core Operating Limits Report (COLR) (continued)

1. "CE Method for Control Element Assembly Ejection Analysis," CENPD-0190-A, (Methodology for Specification 3.1.7, Regulating CEA Insertion Limits).

2. "The ROCS and DIT Computer Codes for Nuclear Design," CENPD-266-P-A, [Methodology for Specifications 3.1.1, Shutdown Margin - Reactor Trip Breakers Open; 3.1.2, Shutdown Margin - Reactor Trip Breakers Closed; 3.1.4, Moderator Temperature Coefficient BOL and EOL limits; 3.1.7, Regulating CEA Insertion Limits and 3.9.1, Boron Concentration (Mode 6)].

3. "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 3.1.2, Shutdown Margin - Reactor Trip Breakers Closed; 3.1.4, Moderator Temperature Coefficient BOL and EOL limits; 3.3.12, Boron Dilution Alarm System; 3.1.5, CEA Alignment; 3.1.7, Regulating CEA Insertion Limits; 3.1.8, Part Strength CEA Insertion Limits and 3.2.3, Azimuthal Power Tilt - T_q].

4. "Modified Statistical Combination of Uncertainties," CEN-356(V)-P-A and "System 80™ Inlet Flow Distribution," Supplement 1-P to Enclosure 1-P to LD-82-054, (Methodology for Specification 3.2.4, DNBR and 3.2.5 Axial Shape Index).

5. "Calculative Methods for the CE Large Break LOCA Evaluation Model," CENPD-132, (Methodology for Specification 3.2.1, Linear Heat Rate).

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

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5.6 Reporting Requirements

5.6.5 Core Operating Limits Report (COLR) (continued)

7. 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: 5.6.5.b.6.

8. 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 5.6.5.b.6.

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

10. 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 5.6.5.b.9.

11. "Arizona Public Service Company PWR Reactor Physics Methodology Using CASMO-4/SIMULATE-3," [Methodology for Specifications 3.1.1, Shutdown Margin - Reactor Trip Breakers Open; 3.1.2, Shutdown Margin - Reactor Trip Breakers Closed; 3.1.4, Moderator Temperature Coefficient; 3.1.7, Regulating CEA Insertion Limits and 3.9.1, Boron Concentration (Mode 6)].

12. "Technical Manual for the CENTS Code," CE-NPD 282-P-A, Volumes 1-3, [Methodology for Specifications 3.1.2, Shutdown Margin-Reactor Trip Breakers Closed; 3.1.4, Moderator Temperature Coefficient; 3.1.5, CEA Alignment; 3.1.7, Regulating CEA Insertion Limits; 3.1.8, Part Strength CEA Insertion Limits and 3.2.3, Azimuthal Power Tilt- T_q].

13. CENPD-404-P-A, "Implementation of ZIRLO™ Cladding Material in CE Nuclear Power Fuel Assembly Designs.

14. CENPD-188-A, "HERMITE, A Multi-Dimensional Space-Time Kinetics Code for PWR Transients." [Methodology for Specifications 3.1.2, Shutdown Margin-Reactor Trip Breakers Closed; 3.1.4, Moderator Temperature Coefficient; 3.2.1, Linear Heat Rate; 3.2.3, Azimuthal Power Tilt; 3.2.4, DNBR; and 3.2.5, Axial Shape Index.]

15. CENPD-161-P-A, "TORC Code, A Computer Code for Determining the Thermal Margin of a Reactor Core." [Methodology for Specifications 3.1.1, Shutdown Margin-Reactor Trip Breakers Open; 3.1.2, Shutdown Margin-Reactor Trip Breakers Closed; 3.1.4, Moderator Temperature Coefficient; 3.2.1, Linear Heat

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