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

September 10, 2010

Mr. Randall K. Edington
Executive Vice President Nuclear/
Chief Nuclear Officer
Mail Station 7602
Arizona Public Service Company
P.O. Box 52034
Phoenix, AZ 85072-2034

SUBJECT: PALO VERDE NUCLEAR GENERATING STATION, UNITS 1, 2, AND 3 - ISSUANCE OF AMENDMENTS RE: ADMINISTRATIVE CHANGES TO FACILITY OPERATING LICENSES AND TECHNICAL SPECIFICATIONS (TAC NOS. ME2587, ME2588, AND ME2589)

Dear Mr. Edington:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 179 to Facility Operating License No. NPF-41, Amendment No. 179 to Facility Operating License No. NPF-51, and Amendment No. 179 to Facility Operating License No. NPF-74 for the Palo Verde Nuclear Generating Station, Units 1, 2, and 3, respectively. The amendments consist of changes to License Condition 2.C(1), "Maximum Power Level" (for Units 1 and 3 only), and Technical Specifications in response to your application dated October 30, 2009, as supplemented by letters dated April 29 and August 24, 2010.

The amendments remove requirements that are no longer applicable due to the completion of power uprates, the replacement of steam generators, the removal of part-length control rod element assemblies, and the completion of the core protection calculator upgrade. The amendment would also make a minor administrative change to the nomenclature of the containment sump trash racks and screens.

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

Sincerely,

James R. Hall

James R. Hall, Senior Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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

Enclosures:
1. Amendment No. 179 to NPF-41
2. Amendment No. 179 to NPF-51
3. Amendment No. 179 to NPF-74
4. Safety Evaluation

cc w/encls: Distribution via Listserv

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

ARIZONA PUBLIC SERVICE COMPANY, ET AL.

DOCKET NO. STN 50-528

PALO VERDE NUCLEAR GENERATING STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 179
License No. NPF-41

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

   A. The application for amendment by the Arizona Public Service Company (APS or the licensee) on behalf of itself and the Salt River Project Agricultural Improvement and Power District, El Paso Electric Company, Southern California Edison Company, Public Service Company of New Mexico, Los Angeles Department of Water and Power, and Southern California Public Power Authority dated October 30, 2009, as supplemented by letters dated April 29 and August 24, 2010, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraphs 2.C(1) and 2.C(2) of Facility Operating License No. NPF-41 are hereby amended to read as follows:

   (1) Maximum Power Level

   Arizona Public Service Company (APS) is authorized to operate the facility at reactor core power levels not in excess of 3990 megawatts thermal (100% power) 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. 179, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into this license. APS shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan, except where otherwise stated in specific license conditions.

3. This license amendment is effective as of the date of issuance and shall be implemented within 90 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T. Markley, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Facility Operating
License No. NPF-41 and
Technical Specifications

Date of Issuance: September 10, 2010

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

ARIZONA PUBLIC SERVICE COMPANY, ET AL.

DOCKET NO. STN 50-529

PALO VERDE NUCLEAR GENERATING STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 179
License No. NPF-51

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

   A. The application for amendment by the Arizona Public Service Company (APS or the licensee) on behalf of itself and the Salt River Project Agricultural Improvement and Power District, El Paso Electric Company, Southern California Edison Company, Public Service Company of New Mexico, Los Angeles Department of Water and Power, and Southern California Public Power Authority dated October 30, 2009, as supplemented by letters dated April 29 and August 24, 2010, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations set forth in 10 CFR Chapter I;

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

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

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

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

Enclosure 2

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C(2) of Facility Operating License No. NPF-51 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 179, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into this license. APS shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan, except where otherwise stated in specific license conditions.

3. This license amendment is effective as of the date of issuance and shall be implemented within 90 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Mohan Thadani for

Michael T. Markley, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Facility Operating License No. NPF-51 and Technical Specifications

Date of Issuance: September 10, 2010

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

ARIZONA PUBLIC SERVICE COMPANY, ET AL.

DOCKET NO. STN 50-530

PALO VERDE NUCLEAR GENERATING STATION, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 179
License No. NPF-74

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

   A. The application for amendment by the Arizona Public Service Company (APS or the licensee) on behalf of itself and the Salt River Project Agricultural Improvement and Power District, El Paso Electric Company, Southern California Edison Company, Public Service Company of New Mexico, Los Angeles Department of Water and Power, and Southern California Public Power Authority dated October 30, 2009, as supplemented by letters dated April 29 and August 24, 2010, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations set forth in 10 CFR Chapter I;

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

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

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

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

Enclosure 3

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraphs 2.C(1) and 2.C(2) of Facility Operating License No. NPF-74 are hereby amended to read as follows:

   (1) Maximum Power Level

   Arizona Public Service Company (APS) is authorized to operate the facility at reactor core power levels not in excess of 3990 megawatts thermal (100% power) 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. 179, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into this license. APS shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan, except where otherwise stated in specific license conditions.

3. This license amendment is effective as of the date of issuance and shall be implemented within 90 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T. Markley, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Facility Operating License No. NPF-74 and Technical Specifications

Date of Issuance: September 10, 2010

ATTACHMENT TO LICENSE AMENDMENT NOS. 179, 179, AND 179

FACILITY OPERATING LICENSE NOS. NPF-41, NPF-51, AND NPF-74

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

Replace the following pages of the Facility Operating Licenses Nos. NPF-41, NPF-51, and NPF-74, and Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Facility Operating License No. NPF-41

REMOVE	INSERT
-4-	-4-
-5-	-5-

Facility Operating License No. NPF-51

REMOVE	INSERT
-6-	-6-

Facility Operating License No. NPF-74

REMOVE	INSERT
-4-	-4-

Technical Specifications

REMOVE	INSERT	REMOVE	INSERT
TS TOC page i	TS TOC page i	3.3.3-5 to 3.3.3-8	3.3.3-1 to 3.3.3-4
1.1-6	1.1-6	3.3.5-4	3.3.5-4
3.1.5-1	3.1.5-1	3.4.1-3	--
3.1.5-3	3.1.5-3	3.4.1-4	3.4.1-3
3.1.8-1	3.1.8-1	3.5.3-3	3.5.3-3
3.1.8-2	3.1.8-2	3.7.1-4	3.7.1-4
3.1.10-1	3.1.10-1	5.4-1	--
3.1.11-1	3.1.11-1	5.4-2	5.4-1
3.2.4-1 to 3.2.4-2	--	5.5-7	5.5-7
3.2.4-3 to 3.2.4-4	3.2.4-1 to 3.2.4-2	5.5-8	5.5-8
3.3.1-1 to 3.3.1-10	--	5.6-3	5.6-3
3.3.1-11 to 3.3.1-19	3.3.1-1 to 3.3.1-9	5.6-4	5.6-4
3.3.2-5	3.3.2-5	5.6-5	5.6-5
3.3.3-1 to 3.3.3-4	--		

(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 3990 megawatts thermal (100% power), 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. 179, 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) Operating Staff Experience Requirements

Deleted

(5) Post-Fuel-Loading Initial Test Program (Section 14, SER and SSER 2)*

Deleted

(6) Environmental Qualification

Deleted

(7) Fire Protection Program

APS shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report for the facility, as supplemented and amended, and as approved in the SER through Supplement 11, subject to the following provision:

APS may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

(8) Emergency Preparedness

Deleted

*The parenthetical notation following the title of many license conditions denotes the section of the Safety Evaluation Report and/or its supplements wherein the license condition is discussed.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 179, 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) Operating Staff Experience Requirements (Section 13.1.2, SSER 9)*

Deleted

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

Deleted

(6) Fire Protection Program

APS shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report for the facility, as supplemented and amended, and as approved in the SER through Supplement 11, subject to the following provision:

APS may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

(7) Inservice Inspection Program (Sections 5.2.4 and 6.6, SER and SSER 9)

Deleted

(8) Supplement No. 1 to NUREG-0737 Requirements

Deleted

*The parenthetical notation following the title of many license conditions denotes the section of the Safety Evaluation Report and/or its supplements wherein the license condition is discussed.

(1) Maximum Power Level

Arizona Public Service Company (APS) is authorized to operate the facility at reactor core power levels not in excess of 3990 megawatts thermal (100% power), 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. 179, 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)

PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	The PTLR is the site specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.9.
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
	<u>AND</u>		
	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
	<u>OR</u>		
	C.2.1	Place affected automatic trip channel in bypass or trip.	1 hour
	<u>AND</u>		
	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
	<u>OR</u>		
	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 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	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. OR Both CEACs inoperable in one or more CPC channels.	B.1 Declare the affected CPC channel(s) inoperable. OR	Immediately (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.	4 hours
	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.	4 hours
	AND	
		(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
	<u>AND</u>		
	B.2.5	Perform SR 3.1.5.1.	Once per 4 hours
	<u>AND</u>		
	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.

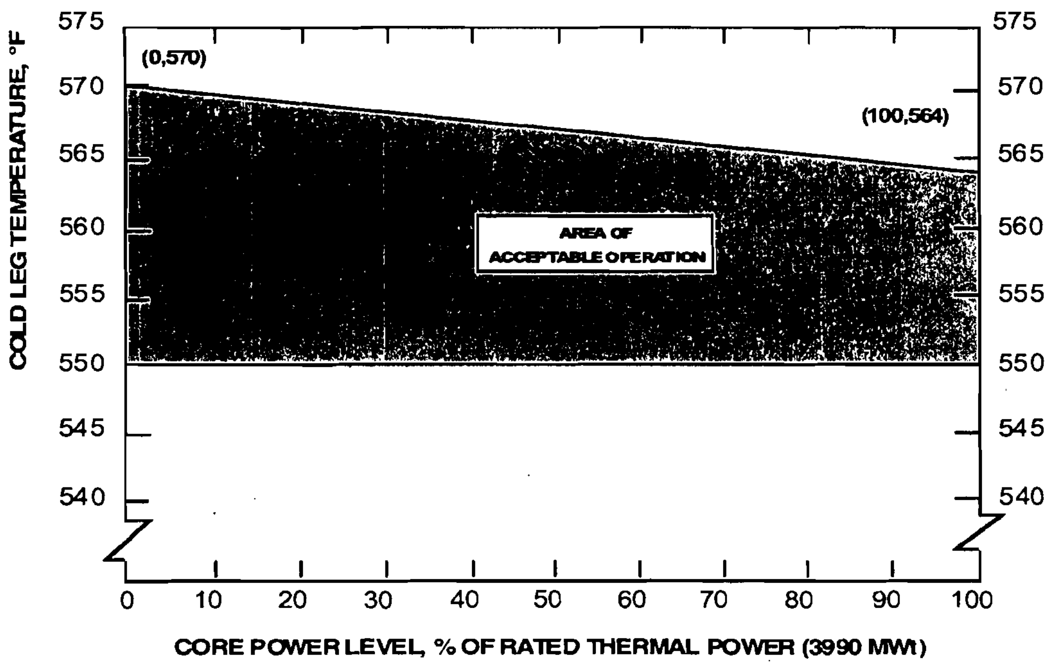
(c) The Main Steam Isolation Signal (MSIS) Function (Steam Generator Pressure - Low, Steam Generator Level-High and Containment Pressure - High signals) is not required to be OPERABLE when all associated valves isolated by the MSIS Function are closed.

(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).

(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 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)

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 179 TO FACILITY OPERATING LICENSE NO. NPF-41,

AMENDMENT NO. 179 TO FACILITY OPERATING LICENSE NO. NPF-51, AND

AMENDMENT NO. 179 TO FACILITY OPERATING LICENSE NO. NPF-74

ARIZONA PUBLIC SERVICE COMPANY, ET AL.

PALO VERDE NUCLEAR GENERATING STATION, UNITS 1, 2, AND 3

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

1.0 INTRODUCTION

By application dated October 30, 2009 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML093170318), as supplemented by letters dated April 29 and August 24, 2010 (ADAMS Accession Nos. ML101310225 and ML102440121, respectively), Arizona Public Service Company (APS, the licensee) requested changes to the Technical Specifications (TSs) for the Palo Verde Nuclear Generating Station (PVNGS), Units 1, 2, and 3, and changes to License Condition 2.C(1) in the Facility Operating Licenses for PVNGS, Units 1 and 3. The proposed amendments would remove requirements that are no longer applicable due to the completion of power uprates, the replacement of steam generators, the removal of part-length control element assemblies, and the completion of the core protection calculator upgrade. The amendment also would make a minor administrative change to the nomenclature of the containment sump trash racks and screens.

The supplemental letters dated April 29 and August 24, 2010, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on January 26, 2010 (75 FR 4113).

2.0 REGULATORY EVALUATION

Section 182.a. of the Atomic Energy Act of 1954, as amended (AEA) requires applicants for licenses to operate nuclear power plants to include technical specifications (TSs) as part of the license application. These TSs become part of any license issued and are derived from the plant safety analyses.

The regulations in Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.36, "Technical specifications," contain the requirements for the content of TSs. Pursuant to 10 CFR 50.36, TSs are required to include items in the following five specific categories related to station operation: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation (LCOs); (3) surveillance requirements (SRs); (4) design features; and (5) administrative controls.

The regulations in 10 CFR 50.36 do not specify each particular requirement to be included in a plant's TSs, nor does it specify the format of a plant's TSs. Rather, the NRC publishes generic guidance on TS format and content. The NRC published a set of Standard Technical Specifications (STS) in NUREG-1432, Revision 3, "Standard Technical Specifications, Combustion Engineering Plants," dated March 2004. The STS are a guide to what a plant's TS should contain with regard to format and content. The STS are not requirements in a regulatory sense, but licensees adopting portions of the improved STS to existing TSs should adopt all related requirements, as applicable, to achieve a high degree of standardization and consistency.

Licensees may propose revisions to the TSs to adopt improved STS format and content provided that the plant-specific review supports a determination of continued adequate safety because: (1) the change is editorial, administrative or provides clarification (i.e., no requirements are materially altered), (2) the change is more restrictive than the licensee's current requirement, or (3) the change is less restrictive than the licensee's current requirement, but nonetheless still affords adequate assurance of safety when judged against current regulatory standards. The NRC staff reviews the proposed revision and determines whether or not the proposed revision is acceptable. If the staff determines that the proposed revision is acceptable, the staff changes the licensee's TSs. The detailed application of this general framework, and additional specialized guidance, are discussed below in the context of specific proposed changes.

3.0 TECHNICAL EVALUATION

The licensee proposes TS revisions that are editorial, administrative, or provide clarification. In order for these changes to be acceptable, the staff must determine that the changes are compatible with current TSs, do not result in any substantive change in operating requirements, and are consistent with the Commission's regulations. The following is the NRC staff's detailed evaluation of the licensee's proposed changes to the TSs.

3.1 Removal of References to Part-Length Control Element Assemblies

APS has completed removal of all part-length control element assemblies (CEAs) at PVNGS, Units 1, 2, and 3 as approved by the NRC in Amendment Nos. 152, 152, and 152, respectively, issued on March 23, 2004 (ADAMS Accession No. ML040860597). APS is proposing to remove the phrase "part length or" from the PVNGS TSs as follows:

- The title of TS 3.1.8, "Part Length or Part Strength CEA Insertion Limits," in the TS table of contents;
- The "Shutdown Margin (SDM)" definition in TS 1.1, "Definitions";

- LCO 3.1.5 and SR 3.1.5.1 in TS 3.1.5, "Control Element Assembly (CEA) Alignment";
- The title, LCO 3.1.8, Conditions A and B, Required Actions A.1 and B.1, and SR 3.1.8.1 of TS 3.1.8, "Part Length or Part Strength CEA Insertion Limits";
- The references to the title of LCO 3.1.8 in TS 3.1.10, "Special Test Exceptions (STE) – MODES 1 and 2," TS 3.1.11, "Special Test Exceptions (STE) – Reactivity Coefficient Testing," and TS 5.6.5, "Core Operating Limits Report (COLR)"; and
- Required Action B.2.2 in TS 3.3.3, "Control Element Assembly Calculators (CEACs)."

This change removes information that is no longer applicable at PVNGS, Units 1, 2, and 3. This is an administrative change that is compatible with current TSs, does not result in any substantive change in operating requirements, and is consistent with the Commission's regulations. Therefore, this administrative change is acceptable.

3.2 Removal of References to Former Rated Thermal Power

The PVNGS, Units 1, 2, and 3 power uprates were evaluated by the NRC and approved in Amendment No. 149 for Unit 2, dated September 29, 2003 (ADAMS Accession No. ML032731029), and Amendment Nos. 157 and 157, respectively, for Units 1 and 3, dated November 16, 2005 (ADAMS Accession No. ML053130286). The rated thermal power for all three units is 3990 megawatts thermal (MWt). The licensee proposes to remove the references to 3876 MWt and information related to the value from:

- TS 1.1, "Definitions";
- TS Table 3.3.1-1, Functions 6 and 7 in TS 3.3.1, "Reactor Protective System (RPS) Instrumentation – Operating";
- TS Table 3.3.2-1, Functions 2 and 3 in TS 3.3.2, "Reactor Protective System (RPS) Instrumentation – Shutdown";
- TS Table 3.3.5-1, Function 4, in TS 3.3.5, "Engineered Safety Features Actuation System (ESFAS) Instrumentation";
- Figure 3.4.1-1 in TS 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits"; and
- Table 3.7.1-1 in TS 3.7.1, "Main Steam Safety Valves (MSSVs)."

In current TS 3.4.1 and TS 3.7.1, there are labels for a graph and a table column, respectively, that distinguish that graph and column from information relating to 3876 MWt. The licensee also requests that those labels be removed since there will no longer be any information relating to 3876 MWt. Furthermore, the licensee proposes to remove the reference to 3876 MWt in

License Condition 2.C(1) for the Facility Operating Licenses for Units 1 and 3 and replace it with 3990 MWt. These are administrative changes that are compatible with current TSs, do not result in any substantive change in operating requirements, and are consistent with the Commission's regulations. Therefore, these administrative changes are acceptable.

3.3 Removal of Reference to Alloy 600 MA Tube Steam Generators

The Alloy 600 MA tube steam generators were replaced with Alloy 690 tube steam generators, as approved by the NRC in Amendment No. 149 for Unit 2, and Amendment Nos. 157 and 157, respectively, for Units 1 and 3. TS 5.5.9, "Steam Generator (TS) Program," paragraph d.2a specifies the inspection requirements for Alloy 600 MA tube steam generators. The proposed amendment removes TS 5.5.9 paragraph d.2a, and renumbers paragraph d.2b as paragraph d.2, because the previous paragraph is no longer applicable. In addition, the phrase "Replacement SGs with Alloy 690 TT tubes:" was removed from paragraph d.2, since its purpose was to distinguish the inspection requirements for the two types of steam generator tubes. With only one type remaining, it is not necessary to distinguish that the requirements are for Alloy 690 TT tubes. These are administrative changes that are compatible with current TSs, do not result in any substantive change in operating requirements, and are consistent with the Commission's regulations. Therefore, these administrative changes are acceptable.

3.4 Deletion and Updating of TSs Due to Amendment No. 150 (Core Protection Calculator (CPC) System Upgrade)

APS completed upgrading the Core Protection Calculator (CPC) System for PVNGS, Units 1, 2, and 3 as approved by the NRC in Amendment Nos. 150, 150, and 150, respectively, dated October 24, 2003 (ADAMS Accession No. ML033030618). Upon completion of the CPC upgrade, the TSs designated as "before CPC upgrade" are no longer applicable. Therefore, APS is deleting the pages designated in the page header as "before CPC upgrade" and renumbering the remaining pages from the following PVNGS TSs:

- TS 3.2.4, "Departure from Nucleate Boiling Ratio (DNBR)" – pages 3.2.4-1 and 3.2.4-2 are deleted;
- TS 3.3.1, "Reactor Protective System (RPS) Instrumentation" pages 3.3.1-1 through 3.3.1-10 are deleted;
- TS 3.3.3, "Control Element Assembly Calculators (CEACs)" – pages 3.3.3-1 through 3.3.3-4 are deleted; and
- TS 5.4.1, "Procedures" – page 5.4.1-1 is deleted.

The removal of these TS pages is an administrative change that is compatible with current TSs, does not result in any substantive change in operating requirements, and is consistent with the Commission's regulations. Therefore, this administrative change is acceptable.

PVNGS is proposing to remove the phrase "after CPC upgrade" from the page header of TS 3.2.4, TS 3.3.1, TS 3.3.3, and TS 5.4.1. This is an administrative change that is compatible with current TSs, does not result in any substantive change in operating requirements, and is consistent with the Commission's regulations. Therefore, this administrative change is acceptable.

APS proposes to move the logical connector 'AND' between Required Actions B.1 and B.2.1 in TS 3.2.4 one indent to the left and the logical connector 'OR' between Required Actions B.2.1 and B.2.2 one indent to the right. This is consistent with NUREG-1432, "Standard Technical Specifications Combustion Engineering Plants," and PVNGS TS 3.2.4, "Departure from Nucleate Boiling Ratio (before CPC upgrade)." This change corrects an editorial error introduced into the TSs for PVNGS, Units 1, 2, and 3 in Amendment Nos. 150, 150, and 150, respectively. This editorial change is compatible with current TSs, does not result in any substantive change in operating requirements, and is consistent with the Commission's regulations. Therefore, this administrative change is acceptable.

3.5 Proposed Change to TS 3.5.3, "ECCS – Operating," SR 3.5.3.8

APS is proposing to change the nomenclature in SR 3.5.3.8 of TS 3.5.3, "ECCS – Operating," from "trash racks and screens" to "strainers." This change does not change the intent of SR 3.5.3.8. APS is still required to verify every 18 months, by visual inspection, that each emergency core coolant system (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. This is an administrative change that will make PVNGS nomenclature consistent with the industry nomenclature. This is an administrative change that is compatible with current TSs, does not result in any substantive change in operating requirements, and is consistent with the Commission's regulations. Therefore, this administrative change is acceptable.

3.6 Summary of Changes

The proposed changes in Sections 3.1 through 3.5 of this safety evaluation are non-technical, administrative changes that are editorial in nature or involve the reorganization or reformatting of requirements without affecting technical content or operational requirements. These administrative changes are compatible with current TSs, do not result in any substantive change in operating requirements, are consistent with the Commission's regulations, and do not approve any design changes for PVNGS, Units 1, 2, and 3. The NRC staff concludes the proposed changes are acceptable.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendments made administrative changes to update the licenses and the technical specifications as a result of changes that were approved in previously issued amendments and make a minor administrative change to the nomenclature of the containment sump trash tacks and screens. These amendments therefore only make editorial, corrective or other minor revisions. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(10). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

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

Principal Contributor: K. Bucholtz

Date: September 10, 2010

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

Sincerely,

/RA/

James R. Hall, Senior Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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

Enclosures:
1. Amendment No. 179 to NPF-41
2. Amendment No. 179 to NPF-51
3. Amendment No. 179 to NPF-74
4. Safety Evaluation

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