Palo Verde Nuclear
Generating Station

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U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
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Subject: **Palo Verde Nuclear Generating Station (PVNGS)**
Units 1, 2 and 3
Docket Nos. STN 50-528/529/530
License Amendment Request to Various Technical Specifications Associated with Replacement of Part Length Control Element Assemblies (CEAs)

Dear Sirs:

In accordance with 10 CFR 50.90, Arizona Public Service Company (APS) hereby requests an amendment to Facility Operating License Nos. NPF-41, NPF-51, and NPF-74 for Palo Verde Nuclear Generating Station (PVNGS) Units 1, 2, and 3. This License Amendment Request (LAR) revises the following sections of the Technical Specifications:

- Table of Contents
- 1.1 "Definitions"
- 3.1.5 "Control Element Assembly (CEA) Alignment"
- 3.1.8 "Part Length Control Element Assembly (CEA) Insertion Limits"
- 3.1.9 "Special Test Exception (STE) – Shutdown Margin (SDM)"
- 3.1.10 "Special Test Exception (STE) – MODES 1 and 2"
- 3.1.11 "Special Test Exception (STE) – Reactivity Coefficient Testing"
- 3.3.3 "Control Element Assembly Calculators (CEACs)"
- 4.2.2 "Design Features - Control Element Assemblies"
- 5.6.5 "Reporting Requirements - Core Operating Limits Report (COLR)"

This LAR is necessary to support the replacement of Part Length Control Element Assemblies (PLCEAs) with a new design control element assembly described as Part Strength Control Element Assembly (PSCEA).

Additionally, TS 3.1.5 – "Control Element Assembly (CEA) Alignment," Condition B will be modified to prevent a potential unwarranted plant shutdown condition from occurring.

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The enclosure to this LAR provides a description and assessment of the proposed change. Attachment 1 provides the existing TS pages marked up to show the proposed changes. Attachment 2 provides the revised (retyped) TS pages. Additionally, the marked up and retyped Technical Specification pages for LCO 3.3.3 contained in Attachment 1 and 2 are the associated pages for a pending change with the NRC for the approval of the replacement of Core Protection Calculator Systems (CPCS), submitted on 11/07/02 (102-04864-CDM/TNW/DWG - Request for Amendment to Technical Specifications: 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)). Attachment 3 provides the existing TS Bases pages marked up to show the proposed changes (for information only).

Once the implementation of PSCEAs has been completed in all 3 PVNGS Units, APS will submit another LAR to remove from the Technical Specifications references to the PLCEAs.

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 10CFR 50.91(b)(1).

The anticipated use of the Part Strength Control Element Assemblies (PSCEAs) is scheduled for Unit 1, refueling outage 11 (U1R11). U1R11 is currently scheduled to start April 3, 2004. Approval of this amendment application is requested by February 17, 2004. APS requests to implement the proposed amendment within 60 days of its issuance.

No commitments are being made to the NRC by this letter.

Should you have any questions, please contact Thomas N. Weber at (623) 393-5764.

Sincerely,

David Mauldin

CDM/TNW/JAP

Enclosures:

- Notarized Affidavit
- APS' evaluation of proposed changes

Attachments:

1. Markup of Technical Specification Pages
2. Retyped Technical Specification Pages
3. Associated Changes to Technical Specification Bases (for information only)

cc: Regional Administrator, NRC Region IV
M. B. Fields
N. L. Salgado
A. V. Godwin

STATE OF ARIZONA)
) ss.
COUNTY OF MARICOPA)

I, David Mauldin, represent that I am Vice President Nuclear Engineering and Support, Arizona Public Service Company (APS), that the foregoing document has been signed by me on behalf of APS with full authority to do so, and that to the best of my knowledge and belief, the statements made therein are true and correct.

David Mauldin

David Mauldin

Sworn To Before Me This 17th Day Of September, 2003.

OFFICIAL SEAL
Susie Lynn Ergish
NOTARY PUBLIC - STATE of ARIZONA
MARICOPA COUNTY
MY COMM. EXPIRES July 14, 2007

Susie Lynn Ergish

Notary Public

Notary Commission Stamp

Enclosure

APS' Evaluation of Proposed LAR

Proposed Amendment for Replacement of PLCEAs with PSCEAs

1.0 DESCRIPTION

This license amendment request (LAR) will 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 would revise sections of the Technical Specifications (TS) to support replacement of the part length control element assemblies (PLCEAs) with a new design that contains neutron absorber over the entire control section of each control element assembly (CEA) finger. The replacements are referred to as part strength control element assemblies (PSCEAs). The current PLCEAs have been in use since the start of operation of each PVNGS unit and are planned to be replaced before reaching 15 effective full power years (EFPYs). This design life requires the replacement of PLCEAs in Units 1 & 3 to be at the end of Cycle 11 and at the end of Cycle 12 for Unit 2.

The expected installations of the part strength CEAs (PSCEAs) are planned to occur during upcoming refueling outages as listed below:

Unit 1, Refueling Outage 11 - Spring 2004

Unit 3, Refueling Outage 11 - Fall 2004

Unit 2, Refueling Outage 12 - Spring 2005

The proposed changes associated with this LAR are mainly changing the wording from "part length" to "part length or part strength" control element assemblies (CEAs). Along with this change will be the addition of the part strength CEAs description to Section 4.2.2 of the Technical Specifications. Even though there will be no changes or modifications to full length CEAs, for consistency and for ease of reading, the wording for "full length" CEAs will be changed to "full strength" CEAs.

Additionally, TS 3.1.5 – "Control Element Assembly (CEA) Alignment," Condition B, will be modified to eliminate a potential condition which could cause an unwarranted plant shutdown. This condition will be modified such that when more than one CEA in a group has only one operable position indication, a plant shutdown will not be required.

2.0 PROPOSED CHANGES

The following changes describe the modification to the wording and description associated with part length and part strength CEAs, along with modifying wording for full length to full strength CEAs. In the sections of the Technical Specifications that currently list "part length CEAs," this will be changed to "part length or part strength CEAs". The intent of this change is to accommodate the implementation of part strength CEAs during different time frames between the three Palo Verde units.

- In the "Table of Contents" on page "i", TS 3.1.8 currently is listed as:

  "Part Length CEA Insertion Limits"

  <u>The "Table of Contents", page "i", for TS 3.1.8 will be changed to read:</u>

  "Part Length or Part Strength CEA Insertion Limits

- In the "Definitions" section of TS on page 1.1-4, for "K_{n-1}", the definition currently reads:

  "K_{n-1} is the K effective calculated by considering the actual CEA configuration and assuming that the fully or partially inserted full-length CEA of highest worth is fully withdrawn."

  <u>This definition for "K_{n-1}" will be changed to read:</u>

  "K_{n-1} is the K effective calculated by considering the actual CEA configuration and assuming that the fully or partially inserted full strength CEA of highest worth is fully withdrawn."

- In the "Definitions" section of TS on page 1.1-6, for "Shutdown Margin (SDM)," the definition currently reads:

  "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 length 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 length 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 CEA position."

<u>This definition for "SDM" will be changed to read:</u>

"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."

- TS Limiting Condition For Operation (LCO) 3.1.5 "Control Element Assembly (CEA) Alignment," currently reads:

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

  <u>LCO 3.1.5 will be changed to read:</u>

  "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."

- TS LCO 3.1.5, Condition C, currently reads:

  "Required Action and associated Completion Time of Condition A or B not met

  <u>OR</u>

One or more full length CEAs untrippable."

LCO 3.1.5, Condition C, will be changed to read:

"Required Action and associated Completion Time of Condition A or B not met

OR

One or more full strength CEAs untrippable."

- TS Surveillance Requirement (SR) 3.1.5.1, currently reads:

  "Verify the indicated position of each full and part length CEA is within 6.6 inches of all other CEAs in its group."

  SR 3.1.5.1 will be changed to read:

  "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."

- TS SR 3.1.5.3, currently reads:

  "Verify full length CEA freedom of movement (trippability) by moving each individual full length CEA that is not fully inserted in the core at least 5 inches."

  SR 3.1.5.3 will be changed to read:

  "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."

- TS SR 3.1.5.5, currently reads:

  "Verify each full length CEA drop time ≤ 4.0 seconds."

  SR 3.1.5.5 will be changed to read:

  "Verify each full strength CEA drop time ≤ 4.0 seconds."

- The title for LCO 3.1.8, currently reads:

  "Part Length Control Element Assembly (CEA) Insertion Limits"

  <u>The title for LCO 3.1.8 will be changed to read:</u>

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

- TS LCO 3.1.8, currently reads:

  "The part length CEA groups shall be limited to the insertion limits specified in the COLR."

  <u>LCO 3.1.8 will be changed to read:</u>

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

- TS LCO 3.1.8, Condition A and Required Action A.1, currently read:

"Condition A.	Part length CEA groups inserted beyond the transient insertion limit.
Required Action A.1	Restore part length CEA groups to within the limit."

  <u>LCO 3.1.8, Condition A and Required Action A.1, will be changed to read:</u>

"Condition A.	Part length or part strength CEA groups inserted beyond the transient insertion limit.
Required Action A.1	Restore part length or part strength CEA groups to within the limit."

- TS LCO 3.1.8, Condition B and Required Action B.1, currently read:

"Condition B.	Part length 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.

Required Action B.1 — Restore part length CEA groups to within the long term steady state insertion limit."

LCO 3.1.8, Condition B and Required Action B.1, will be changed to read:

"Condition 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.

Required Action B.1 — Restore part length or part strength CEA groups to within the long term steady state insertion limit."

- TS SR 3.1.8.1, currently reads:

  "Verify part length CEA group position."

  SR 3.1.8.1 will be changed to read:

  "Verify part length or part strength CEA group position."

- TS LCO 3.1.9, Condition A, currently reads:

  "Any full length CEA not fully inserted and less than the required shutdown reactivity available for trip insertion.

  OR

  All full length CEAs inserted and the reactor subcritical by less than the above required shutdown reactivity equivalent."

  LCO 3.1.9, Condition A will be changed to read:

  "Any full strength CEA not fully inserted and less than the required shutdown reactivity available for trip insertion.

  OR

All full strength CEAs inserted and the reactor subcritical by less than the above required shutdown reactivity equivalent."

- TS SR 3.1.9.2, currently reads:

  "Verify each full length CEA not fully inserted is capable of full insertion when tripped from at least the 50% withdrawn position."

  SR 3.1.9.2 will be changed to read:

  "Verify each full strength CEA not fully inserted is capable of full insertion when tripped from at least the 50% withdrawn position."

- TS SR 3.1.9.3, currently reads:

  "Verify that with all full length CEAs fully inserted, the reactor is subcritical within the acceptance criteria."

  SR 3.1.9.3 will be changed to read:

  "Verify that with all full strength CEAs fully inserted, the reactor is subcritical within the acceptance criteria."

- TS LCO 3.1.10, currently reads, in part:

  "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 CEA Insertion Limits"; ..."

  LCO 3.1.10 will be changed to read:

  "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"..."

- TS LCO 3.1.11, currently reads, in part:

  "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 Control Element Assembly (CEA) Insertion Limits;" and ..."

  <u>LCO 3.1.11 will be changed to read:</u>

  "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 ..."

- TS LCO 3.3.3, Required Action B.2, currently reads:

  "Verify all full length and part length 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."

  <u>LCO 3.3.3, Required Action B.2, will be changed to read:</u>

  "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."

- TS Section 4.2.2, "Design Features - Control Element Assemblies," currently reads:

  "The reactor core shall contain 76 full length and 13 part length control element assemblies (CEAs). The control material shall be boron carbide with Inconel Alloy 625 used as a wear absorber over a portion of the part length control element assemblies as approved by the NRC."

  TS 4.2.2 will be changed to read:

  "The reactor core shall contain 76 full strength and either 13 part length or 13 part strength control element assemblies (CEAs).

  The control section for the full strength CEAs shall be boron carbide with Inconel Alloy 625 cladding.

  For units that have part length CEAs, the control section shall be Inconel Alloy 625 in the lower half, followed by perforated stainless steel tubing over the next 40%, and boron carbide pellets with Inconel 625 clad over the last 10% of the control section.

  For units that have part strength CEAs, the control section shall be solid Inconel Alloy 625 slugs with Inconel Alloy 625 cladding."

- TS 5.6.5.a.7. – "Core Operating Limits Report (COLR)," currently reads:

  "Part Length CEA Insertion Limits for Specification 3.1.8."

  TS 5.6.5.a.7. will be changed to read:

  "Part Length or Part Strength CEA Insertion Limits for Specification 3.1.8."

- TS 5.6.5.b.3. – "Core Operating Limits Report (COLR)," currently reads, in part:

  "Safety Evaluation Report related to the Final Design of the Standard Nuclear Steam Supply Reference Systems CESSAR system 80, ...; 3.1.8, Part Length CEA Insertion Limits and 3.2.3, Azimuthal Power Tilt - T_q]."

TS 5.6.5.b.3. will be changed to read (in part):

"Safety Evaluation Report related to the Final Design of the Standard Nuclear Steam Supply Reference Systems CESSAR system 80, ...; 3.1.8, Part Length or Part Strength CEA Insertion Limits and 3.2.3, Azimuthal Power Tilt - T_q]."

- TS 5.6.5.b.12. – "Core Operating Limits Report (COLR)," currently reads, in part:

  "Technical Manual for the CENTS code ....; 3.1.8, Part Length CEA Insertion Limits and 3.2.3, Azimuthal Power Tilt - T_q]."

  TS 5.6.5.b.12. will be changed to read (in part):

  "Technical Manual for the CENTS code ....; 3.1.8, Part Length or Part Strength CEA Insertion Limits and 3.2.3, Azimuthal Power Tilt - T_q]."

In addition to the above changes associated with the replacement of the part length CEAs, the following change associated with TS LCO 3.1.5 is being made:

- TS 3.1.5, Condition B, currently reads:

  "Only one CEA position indicator channel OPERABLE for one CEA per CEA Group."

  TS 3.1.5, Condition B will be changed to read:

  "Only one CEA position indicator channel OPERABLE for one or more CEA(s)."

3.0 BACKGROUND

The purpose of the control element assemblies (CEAs) are for reactivity control during operation and shutdown of the reactor. The shutdown and regulating groups are made up of 4- and 12-finger full length CEAs. The regulating CEA groups may be used to compensate for changes in reactivity associated with routine power level changes and to compensate for minor variations in moderator temperature and boron concentration changes during operation at power, and to dampen axial xenon oscillations. Thirteen part length CEAs are provided in the design to help control the core power distribution. This function includes the suppression of xenon-induced axial power oscillations.

A new design for the original equipment part length CEAs was developed by Combustion Engineering for use in the System-80 designed reactors. The originally supplied part length CEAs (PLCEAs) control section design consists of solid Inconel 625 over the bottom 50% of their length, a stainless steel tube open to the reactor coolant over the next 40%, and a sealed chamber containing 73% (theoretical density) boron carbide (B_4C) pellets in the top 10%. A holddown spring, similar to the spring in the full length rods, maintains the orientation of the B_4C. The new design maintains the same external dimensions as the original design, but with changes to the construction and internal components of the CEA finger. The new CEA is composed of an Inconel 625 tube filled with Inconel 625 slugs throughout the full length of the active region of the finger (nominally 150 inches). This new design is called part strength CEAs (PSCEAs). The perforated tube in the upper 40% section and sealed chamber of B_4C pellets at the top of the original PLCEA design is not present in the new PSCEA design. The PLCEAs are currently planned to be replaced with PSCEAs in each PVNGS reactor beginning with Unit 1, during U1R11 (Spring 2004).

There are 13 PLCEAs currently installed in each reactor. PVNGS intends to replace the 13 existing PLCEAs in each unit's reactor with PSCEAs, which are functionally equivalent except for the amount and geometry of neutron absorber inserted into the core.

The original design of the PLCEAs, which has Inconel 625 and B_4C used as neutron absorber located in the bottom 50% and top 10%, respectively, introduces an incident of moderate frequency which is addressed in the PVNGS Updated Final Safety Analysis Report (UFSAR) Section 7.2.2.1.1.C. A specific safety analysis that comes from this category of single PLCEA misoperation is the malpositioning of a PLCEA between 51% and 90% inserted into the core, resulting in flux peaking in the top of the core. The new design contains neutron absorber (Inconel 625) within 100% of the control section for each CEA finger, which will eliminate the possibility of this incident of moderate frequency. Therefore, this specific analysis will no longer be applicable to the PVNGS licensing bases.

The design of the full length CEAs is not changing, but their name will change to "full strength CEAs" (FSCEAs) so that terminology for CEAs will be consistent.

TS 3.1.5, Condition B, Modification

This change involves a revision to Condition 'B' of Technical Specification 3.1.5 such that it will apply if one or more CEA(s) have only one operable position indication channel. Currently, Condition 'B' of TS 3.1.5 applies if one CEA per group has only one operable position indication channel.

The change is necessary to provide a Technical Specification Condition that applies to the situation in which more than one CEA in a group has only one operable position indicator. This is being accomplished by simply rewording the existing Condition 'B' to expand the scope of applicability from "one CEA per CEA group", to "one or more CEA(s)."

The current alternative to not revising Condition 'B' as proposed is to require entry into TS LCO 3.0.3 when more than one CEA per group has only one operable position indicator. However the requirements of LCO 3.0.3 (i.e.. plant shutdown) are not appropriate for this situation. This position is based on the following:

1) The situation under review is when more than one CEA per group has only one operable position indication channel. Even in this situation, all CEAs still have at least one operable position indication. Entry into LCO 3.0.3 should not be required for situations involving only a loss of redundancy while maintaining operability of the required feature on one train/channel.

2) The existing PVNGS TS Bases for Condition 'B' of Tech Spec 3.1.5 does not address any limitation of applicability to only one CEA per group. The Bases justify continued operation with only one operable position indication channel provided that within 6 hours, either; 1) at least 2 channels are restored to operable status or 2) the affected group is positioned fully withdrawn or fully inserted (while maintaining compliance with the insertion limits).

4.0 <u>TECHNICAL ANALYSES</u>

Each reactor at Palo Verde contains 89 control element assemblies (CEAs). Seventy-six of the CEAs are referred to as full length CEAs (FLCEAs) and contain boron carbide (B_4C) neutron absorber pellets, which span the range of the entire height of the fuel core when the CEA is fully inserted. Forty-eight of the 76 FLCEAs are comprised of twelve poison fingers each and the remaining 28 FLCEAs each have four poison fingers. The remaining 13 CEAs are part length CEAs (PLCEAs) each with four fingers which contain B_4C neutron absorber sections in only the top 10% of their control section and solid Inconel 625 in the bottom 50%. The remaining 40% of the finger control section region between the B_4C and Inconel 625 is made of a perforated stainless steel tube, which allows the presence of primary coolant to act as a neutron moderator. The PLCEAs were intended to provide control of axial power distribution, particularly in the event of axial xenon oscillations. The CEAs are described in the PVNGS UFSAR Sections 4.2 and 4.3.

The FLCEAs are categorized by their intended function, i.e., shutdown or regulating. During reactor startup and operation, the shutdown FLCEAs are fully withdrawn followed by the withdrawal of the PLCEAS, before the regulating

FLCEAs can be withdrawn for controlling the approach to criticality. Only the regulating FLCEAs are then used in a predetermined sequence for reactivity control.

A series of testing, prior to plant operation, is performed to assure that each FLCEA will function as expected. Such testing includes a drop test to confirm that each FLCEA safely reaches 90% insertion in less than or equal to four seconds. Another test involves measurement of the reactivity worth of each FLCEA during startup testing for each reload cycle in order to verify the expected design values.

Even though the PLCEAs drop times are tested, they are not credited for shutdown margin (SDM) and their drop times and reactivity worth are not considered for accident mitigation in the safety analyses.

During plant startup and operation, changes in core reactivity are used to increase or decrease reactor power and can be accomplished using the regulating FLCEA groups to initiate changes in reactivity associated with the desired change in power level. The group sequence and overlap limits for regulating FLCEAs are specified in the Core Operating Limits Report (COLR) for each fuel cycle. Insertion of the PLCEAs during operation is restricted based on the reactor power level. The maximum insertion at or below 50% power is 50% insertion, which drops to 25% insertion for power levels above 50%. All CEA groups (full and part length) are dropped into the core to ensure a rapid shutdown of the reactor following a manual trip or an automatic reactor trip signal from the plant protection system. Although the PLCEAs are released for insertion along with the FLCEAs following a reactor trip signal, the reactivity insertion of the PLCEAs is not credited in the safety analyses. The PLCEAs are used only to adjust the neutron flux distribution within the reactor core during normal operations.

The replacements for the PLCEAs are referred to as part strength CEAs (PSCEAs). The PSCEAs also use Inconel 625 as a neutron absorber, but unlike the PLCEAs, Inconel 625 is used over the entire active length (approximately 150 inches) of each finger. The physical dimensions of the PSCEA fingers is essentially the same as used for the 4-finger FLCEAs since the design of the finger cladding and assembly structure (with the exception of the poison being used) will be the same. The FLCEAs will be referred to as full strength CEAs (FSCEAs) for consistency. There are no changes being performed to the full length CEAs.

A comparison of the significant design characteristic differences are summarized below:

PLCEAs vs. PSCEAs Design Criteria

Design Criteria	PLCEAs	PSCEAs	Comment
Clad Material	Inconel 625 & 304 Stainless Steel (SS)	Inconel 625	Center 61.5 inches of PLCEAs is made of 304 SS tubing with 0.25 inch diameter perforations.
Clad Outer Diameter	0.816 $\pm$ 0.002 in.	0.816 $\pm$ 0.002 in.	Includes the Inconel, 304 SS, and B_4C regions of the PLCEAs.
Clad Inner Diameter	0.746 $\pm$ 0.002 in.	0.746 $\pm$ 0.002 in.	The gap between the PSCEA clad and slug corresponds to <2% of the Inconel mass in the solid region of a PLCEA. PLCEA inner diameter is for B_4C region at top of each finger.
Inconel Slug Diameter	N/A	0.737 $\pm$ 0.001 in.	The lower 75 in. of PLCEA fingers is solid Inconel with the same outer diameter as PSCEA fingers.
Inconel Slug Length	N/A	7.450 $\pm$ 0.020 in. and 0.500 $\pm$ 0.062 in.	No more than four 0.500 in. Inconel slugs may be used in any one finger at the top of the stack. These slugs are used to adjust the total poison stack length.
Total Neutron Absorber Length	75 in. of Inconel in the bottom of each finger and 16 in. of B_4C at the top of each finger	149.000 $\pm$ 0.005 in. of Inconel	The bottom nose cap of PSCEAs adds 0.875 in. to the height of Inconel.
Total Length of the Control Rod Top Assembly	73.625 $\pm$ 0.005in.	73.625 $\pm$ 0.005in.	This section connects the control rod section containing neutron absorber to the CEA spider.

PLCEAs vs. PSCEAs Design Criteria (cont.)

Total Finger Length (including top assembly)	244.625 ± 0.005in.	244.625 ± 0.005in.	Fingers are the same length for both designs.
Total CEA Length	252.969 ± 0.005in.	252.969 ± 0.005in.	Total CEA length is the same for both designs.
Estimated Total CEA Weight	116.8 lbs	141.1 lbs	The weight of a 12-finger FSCEA (211.9 lbs from same ref.) is much more limiting than a PSCEA, which has only 4 fingers.

The design of the outer geometries of the PLCEAs and PSCEAs are very similar. The principal design differences between the PLCEA and PSCEA are associated with the cladding and the neutron absorber materials used throughout each finger. As mentioned previously, the PLCEAs are comprised of solid Inconel 625, hollow 304 SS perforated tubing, and B_4C pellets with Inconel cladding, within 50%, 40%, and 10% of the absorber volume, respectively. The neutron absorber in PSCEA is made up entirely of Inconel 625 slugs with a clad gap of 0.009 inches. The two designs are geometrically very similar and contain essentially the same amount of neutron absorber in the lower 50% of each finger. This region also corresponds to the limiting PLCEA power dependent insertion limit (PDIL) which will be applied to the PSCEAs. Each PSCEA contains substantially more Inconel 625 resulting in a weight increase. This weight increase is still within the capability and design of the control element assembly design and its associated control mechanism design. This resulting weight is still much less than the weight involved with a 12-finger FSCEA, but it is comparable to that of a 4-finger full length CEA. Therefore, operation of the CEA drive mechanism system with the PSCEAs installed will not be adversely affected.

The principal design function of the PLCEAs is to control axial power distribution. However, the current part length design can cause undesirable flux redistribution if inserted past the PDIL due to the lack of a neutron absorber in 40% of the upper region of each PLCEA finger. The design of the PSCEAs contains Inconel slugs over the entire control section of each CEA finger. As a result, the accident event of concern regarding the PLCEAs does not apply to the new design of the PSCEAs. This occurs due to the fact that the neutron absorber is present throughout the entire control section of each CEA finger and this will not promote the undesired neutron flux shift to the upper region of the core when inserted past 50%. The PDILs established for the PLCEAs minimize undesirable axial power redistributions since the maximum allowed insertion of 50% corresponds to the lower region containing Inconel as a neutron absorber. This same PDIL will be conservatively maintained for the PSCEAs.

As mentioned above, the PSCEA design will eliminate an accident scenario from PVNGS licensing bases. This event analysis involves the insertion of a PLCEA past the PDIL which results in an axial shift in power due to a portion of the upper region of the PLCEAs which does not have a neutron absorber. This condition will not occur with the PSCEAs because they are filled with neutron absorber over 100% of the control section of the CEA. Additionally, the following constraints will be maintained for the PSCEAs:

1. PSCEAs will be in the same location as the existing PLCEAs with no change in subgroup assignments.

2. The PSCEAs will consist of four axially uniform fingers constructed of materials that have the same nuclear properties as the active region (lowest 50%) of the current PLCEA design. In particular, the bounding reactivity worth per inch of insertion in the active region is not significantly different.

3. The Power Dependent Insertion Limit (PDIL) for the PSCEAs will be the same as the current PDIL for the PLCEAs, which limits insertion to less than the length of the current active region (50% insertion).

The name change from "part length CEA" to "part strength CEA" is the principle change being made to the affected Technical Specifications. Although this change is principally an editorial change, the name change also reflects the physical and geometrical changes associated with the replacement CEAs. This name change effectively represents the function of these replacement CEAs in comparison to that of the "full strength" CEAs. The following discusses each specific Technical Specification change:

TS Section 1.1, "Definitions"

The definition for "Shutdown Margin (SDM)" currently includes a discussion of how the full length CEAs are involved in the determination of SDM. The proposed change will replace "full length" with "full strength". Since there are no changes involving the design or operation of the existing full length CEAs, this change is strictly editorial. The definition also states that the SDM is accurately assessed by restricting the movement of the part length CEAs with insertion of the FSCEAs. This criterion shall remain applicable to the replacement PSCEAs that have reactivity worths essentially the same as the existing PLCEAs at or above their PDILs.

The definition for K_{n-1} also refers to the "full length" CEAs with regard to determining the value of K-effective. Referring to "full strength" CEAs represents an editorial change with no technical impact since the design of the existing FLCEAs will not change.

TS Section 3.1.5, "Control Element Assembly (CEA) Alignment"

This section refers to both the FLCEAs and PLCEAs. The terminology for FLCEAs and PLCEAs shall be changed to FSCEAs and PSCEAs or PLCEAs with no technical impact. LCO 3.1.5.C is related to untrippable FLCEAs and will also be revised. The PLCEAs are required to be aligned within 6.6 inches of all other CEAs in their respective groups. This requirement will remain unchanged for the PSCEAs.

The event of primary concern has been the misalignment of the FLCEAs or PLCEAs. The existing Technical Specification Basis describes the PLCEA drop and PLCEA subgroup drop events as resulting in changes to the core power distribution, departure from nucleate boiling ratio (DNBR), and fuel centerline temperature which could result in a reactor trip. The design of the PLCEAs introduces a slightly different response than the FLCEAs as a result of the flux redistribution toward the top of the core due to 40% of the upper control section of each finger containing no neutron absorber. Replacing the PLCEAs with the PSCEAs will eliminate the flux redistribution resulting from these events. In addition, the design of the new PSCEAs is similar to the FLCEAs except for a weaker neutron absorber, which effectively prevents the PSCEAs from being more limiting than the FLCEAs for any accident scenario currently analyzed in the UFSAR. The FLCEA drop event remains the bounding event.

The changes to this TS LCO, Condition, and Surveillance Requirements (SRs) will only consist of name changes from "full length CEAs" to "full strength CEAs" and "part length CEAs" to "part length or part strength CEAs".

TS Section 3.1.8, "Part Length Control Element Assembly (CEA) Insertion Limits"

The insertion limits developed for the PLCEAs represent initial assumptions used in the existing safety analysis for CEA misoperation events. They are intended to prevent undesired neutron flux redistribution toward the top of the core. The associated LCO refers to the COLR for the explicit PLCEA insertion limits. The maximum designated insertion is 50%, which corresponds to the solid Inconel region of the PLCEAs. The limitations for insertion between the long term (steady-state) and transient insertion limits provided in Fig. 3.1.8-1 of the COLR remain applicable due to the similarity in design between the PLCEAs and PSCEAs. However, the core response following insertion of the PSCEAs beyond the PDIL will not be as undesirable as it would be with the PLCEAs. The existing safety analysis does not credit any neutron absorber in the upper 50% of the PLCEAs, which can result in a core power increase or undesirable flux redistribution. This concern is not applicable to the PSCEAs since neutron absorber is present throughout the entire active region of each finger, which would prevent the core response exhibited by the PLCEAs. The effect of the PSCEAs exceeding the PDIL would be similar to the FLCEAs in that the fingers in both CEAs contain neutron poison throughout the entire control section of the CEA. The current limit for returning the FLCEAs and PLCEAs to within the PDIL is two hours. Although the flux redistribution resulting from the PLCEAs can be different from the PSCEAs due to the neutron absorber distribution, the insertion of PSCEAs result in a similar flux redistribution as that of the FLCEAs, although it is not as strong. Therefore, the two-hour limit remains conservative and is still applicable to the PSCEAs.

The changes to this TS LCO, Conditions, Required Actions, and SR will only consist of the name change from "part length CEAs" to "part length or part strength CEAs".

TS Section 3.1.9, "Special Test Exception (STE) - Shutdown Margin (SDM)"

This section addresses suspending FLCEA insertion requirements, to assure SDM, during approved physics tests. The insertion requirements are not being changed and only the CEA terminology will be changed with no technical impact.

The changes to this TS Condition and SRs will only consist of the name change from "full length CEAs" to "full strength CEAs".

TS Section 3.1.10, "Special Test Exception (STE) - MODES 1 and 2"

This section refers to LCO 3.1.8, "Part Length Control Element Assembly (CEA) Insertion Limits" which may be suspended during physics tests. As discussed above, LCO 3.1.8 will be renamed by replacing "part length" CEAs with "part length or part strength" to reflect the new design. The new design of the PSCEAs does not introduce any new technical or operational considerations and no changes to the insertion limits are required. Suspending these limits during testing will not introduce any new concerns. The neutron absorber in the PSCEAs is located throughout the entire control section of each finger and would not result in a positive addition to the reactivity of the upper core region, which could cause an undesired axial flux redistribution. Therefore, suspending the insertion limits of the PSCEAs as currently identified for the PLCEAs in LCO 3.1.10 will have no impact on safe operation.

The change to this TS LCO will only consist of the name change from "Part Length CEA" to "Part Length or Part Strength CEA".

TS Section 3.1.11, "Special Test Exception (STE) - Reactivity Coefficient Testing"

This TS section refers to suspending LCO 3.1.8, "Part Length Control Element Assembly (CEA) Insertion Limits" for reactivity coefficient testing. As discussed above, LCO 3.1.8 will be renamed by replacing "part length" with "part length or part strength" to reflect the new design. LCO 3.1.11 refers to the COLR for the PLCEA positioning requirements. Suspending these limits during testing will not introduce any new considerations since the design of the PSCEAs which use the neutron absorber throughout the entire control section of each finger effectively eliminates the concern associated with the axial flux redistribution to the top of the core. Therefore, suspending the insertion limits of the PSCEAs as currently identified for the PLCEAs in LCO 3.1.11 will have no impact on safe operation.

The change to this TS LCO will only consist of the name change from "Part Length CEA" to "Part Length or Part Strength CEA".

TS Section 3.3.3, "Control Element Assembly Calculators (CEACs)"

The CEACs are used by the Core Protection Calculator System (CPCS) to assure the position of the CEAs in each subgroup is within acceptable limits. LCO 3.3.3 requires that full length and part length CEAs be fully withdrawn in the event of certain conditions for CEAC(s) inoperability. The operation of the new PSCEAs is equivalent to the current PLCEAs (above the PDILs). In addition, the PSCEAs will not functionally impact operation of the CEACs since the same CEA extension shafts, control element drive mechanisms (CEDMs), and rod position indicators are used and will continue to provide position indication for the CEACs. This section also refers to the "full length" CEAs but the change has no technical impact on their design or operation. However, they will be renamed to "full strength CEAs" to be consistent with the new naming convention.

The DNBR-Low trip will provide protection against core damage in the event of PLCEA subgroup drop based on the expected impact on core conditions resulting from no neutron absorber in 40% of the upper control section of each PLCEA finger. Dropping a PLCEA subgroup can result in an increase in core power, in the upper region of the core, due to the lack of neutron absorber in the top half of each finger which results in a core flux redistribution to the top of the core. However, the design of the PSCEAs with neutron absorber covering 100% of the control section will not cause a similar shift in core flux redistribution if accidentally lowered or dropped within the core. Accident events applicable to the PSCEAs (e.g., dropped CEAs) are bounded by the existing safety analyses for the FSCEAs. Therefore, the requirements specified in LCO 3.3.3 will not be impacted by the design of the PSCEAs.

The change to this TS Required Action will only consist of the name changes from "full length CEAs" to "full strength CEAs" and "part length CEAs" to "part length or part strength CEAs".

TS Section 4.2.2, "Control Element Assemblies"

This section provides a summary description of the CEAs used at PVNGS. This section will be revised to provide a description of the PSCEAs along with maintaining a description of the PLCEAs to accommodate staggered installation of PSCEAs in each Unit. In addition, the name used for the FLCEAs will be changed from "full length CEAs" to "full strength CEAs".

TS Section 5.6.5, "Core Operating Limits Report (COLR)"

This section identifies the core operating limits required to be identified in the COLR along with their technical basis (i.e., Technical Specification referenced topicals).

Item 5.6.5.a.7 of the TS identifies the insertion limits of the part length CEAs to be included in the COLR. This section will be reworded to specify the "Part Length or Part Strength CEA Insertion Limits for Specification 3.1.8". The same technical information provided will apply to the new PSCEAs.

Item 5.6.5.b.3 of the TS identifies the reference for the analytical methodology used for specifying limiting data to be included in the COLR. This item includes reference to TS Section 3.1.8 relating to the PLCEAs. The operating limitations for the replacement PSCEAs (i.e., the PDILs) will not change and their effective reactivity worth, when inserted up to the limits of the PDIL, is essentially the same as the current PLCEAs. Therefore, the same analytical methodology will apply to the proposed change for TS 3.1.8 to address the PSCEAs. This section will be reworded to specify the "Part Length or Part Strength CEA Insertion Limits for Specification 3.1.8".

Item 5.6.5.b.12 of the TS refers to the technical basis documentation for the computer code CENTS as being applicable to TS 3.1.8 for the part length insertion limits. CENTS is used for transient accident analysis required in support of the plants operating license. Since the physical design characteristics of the new PSCEAs are similar to the PLCEAs, the analytical modeling of the PSCEAs can be implemented into the CENTS based analyses, which currently model the PLCEAs. Consequently, the methodology referring to the CENTS code can apply to the proposed change for TS 3.1.8 to address the PSCEAs. This section will be reworded to specify the "Part Length or Part Strength CEA Insertion Limits for Specification 3.1.8".

Summary of Changes for Part Length CEA Replacements

The design of the PLCEAs utilizes a solid region of Inconel in the bottom 50% and B_4C in the top 10% of each finger with no neutron absorber located in the middle region. The design of the PSCEAs contains solid Inconel slugs inside an Inconel tube throughout the entire control section of each finger. The outer geometry of the PLCEA fingers is similar to the PSCEAs and the gap between the slugs and the cladding in the PSCEAs is very small. As a result, the effective neutron absorption of the PSCEAs is equivalent to the solid region of the PLCEA fingers. Other general design issues (e.g., weight difference, vibrational difference, seismic, assembly specifications, etc...) have been evaluated and determined to be within analyses parameters. With the installation of the

PSCEAs, a part strength rod drop will not cause the addition of positive reactivity from any initial position in the core because the PSCEAs are entirely made of neutron absorber.

The design of the "spider" assembly, which holds the CEA fingers, is unchanged. The PSCEA fingers are heavier than the PLCEAs. Current analyses have been evaluated and are bounding for this additional weight. Consequently, the only design difference, which presents any technical significance, is the extension of the neutron absorber region from the lower 50% to 100% of the control section of each finger. The new neutron absorber distribution extending through the entire control section for the PSCEAs will result in similar and less severe core power and neutron flux distributions following anticipated operational occurrences (AOOs) for that of FLCEAs. The failure mode associated with aging for the PSCEA fingers is different than that of the FSCEAs (i.e., B_4C pellet swelling causing clad cracking). This is primarily due to the same material (Inconel) being used for both the cladding and neutron absorber slugs within the cladding of the PSCEAs. Due to the neutron absorber slugs and the cladding being made of the same material, no significant strain on the clad which could cause cracking, is expected from swelling of the neutron absorber slugs due to neutron irradiation.

All mechanical design aspects of the PSCEA meet applicable mechanical design criteria. Principal results and conclusions of this evaluation are summarized below:

Topic	Objective	Method	Results	Conclusion
Seismic / LOCA	Confirm stresses meet design criteria.	Evaluated existing analyses and accounted for design changes.	Design Allowables are satisfied for all conditions.	Acceptable
Threaded Joints	Confirm joint preload induced stresses are acceptably low, and that preload is sufficient to keep connections tight. Consider relaxation.	Evaluated existing analyses and accounted for design changes.	Preload stresses are within allowable values. Preloads exceed operating loads even with relaxation considered.	Acceptable
Fatigue	Confirm Utilization Factor for areas susceptible to fatigue are less than the maximum criterion.	Evaluated existing analyses and accounted for design changes.	All factors essentially zero, including effects of heavier weight and potentially longer operating time.	Acceptable
Control Rod Stress	Confirm PSCEA meets all acceptance criteria with added weight.	Evaluated existing analyses and accounted for added weight.	Confirmed that all design allowables are satisfied at all conditions.	Acceptable

Topic	Objective	Method	Results	Conclusion
Clad Welds	Confirm PSCEA meets all acceptance criteria with added weight.	Accounted for added weight and added vent holes in top assembly.	Confirmed that all design allowables are satisfied at all conditions.	Acceptable
CEA Scram	Show displacement vs. time behavior.	Results of existing analysis for other System 80 Units that use PSCEAs are applicable.	Required displacement vs. time is met by the PSCEA design.	Drop Times Acceptable
Spider Structure	Confirm stresses are less than design allowables under all conditions.	Added weight and referred to existing analysis of Spider structural integrity to make assessment.	All design allowables are satisfied with wide margins.	Acceptable
Spider Spring	Confirm arresting spring is sufficient to absorb energy of falling PSCEA without hard impact.	Existing CEA SCRAM analysis for other System 80 Units that use PSCEAs is applicable.	Spring absorbs remaining energy (after dashpot deceleration) without impacting on Upper Guide Structure.	Acceptable
Plenum Spring	Confirm criteria for stack preloads are met.	Previous analysis of other PSCEA designs is applicable.	Meets appropriate shipping and handling requirement and also BOL hot operational requirement. Never reaches solid height	Acceptable
Collapse Resistance	Confirm minimum required margin against collapse considering ovality.	Previous analysis is applicable.	Margin against collapse at max. ovality is greater than the required margin.	Acceptable
Clad Strain (CEA Life)	Typically for CEAs, define and communicate the life limiting parameters.	No IASCC limits are identified for PSCEAs; however, an active wear mechanism was considered.	The projected time to reach a wear limit criterion is provided.	Acceptable
Heating and Cooling (T&H)	Confirm that the Available guide tube flow is greater than the Required flow to suppress bulk boiling in the annulus.	Existing analyses for future FSCEAs that will contain AgInCd, which has a much higher heating rate, are bounding.	Existing analyses bound the PSCEA for the same plant conditions. No bulk boiling.	Acceptable

Topic	Objective	Method	Results	Conclusion
RSGs & Power Uprate	Confirm that RSG and Power Uprate do not adversely affect the PSCEAs.	Documentation on record demonstrates no impact on CEAs.	Recent fuel and CEA evaluations show no impact on CEAs due to RSGs and Power Uprate. This result is considered applicable to PSCEAs as well.	Acceptable

TS 3.1.5, Condition B, Modification

This LAR is also modifying the words associated with LCO 3.1.5, Condition B. This Condition is currently written to address what actions are required when one CEA, per CEA group, has only one operable position indicator available. As currently written, if more than one CEA, per CEA group, had only one operable CEA position indicator, the required action would be to enter LCO 3.0.3. Entering LCO 3.0.3 would require a plant shutdown in a very short period of time. Entry into LCO 3.0.3 should not be required for situations involving only a loss of redundancy while maintaining operability of the required feature on one train/channel.

There are three position indication channels for each individual CEA:

1) Reed Switch Position Transmitter (RSPT) #1
2) Reed Switch Position Transmitter (RSPT) #2
3) Pulse Counter indication

Although it is possible to have independent malfunctions that affect 2 indicator channels for more than one CEA, the most likely cause would be a loss of either the Channel 'C' or Channel 'D' 120 VAC Vital Instrument Bus. If either the Channel 'C' or 'D' Vital Instrument Bus is de-energized, both RSPT #2, and Pulse Counter indication channels are lost on multiple CEAs such that more than one CEA per group would have only one operable indication channel (RSPT #1).

The table below provides the effects on CEA indication channels due to loss of each Vital Instrument Bus.

Vital Bus De-energized	Number of CEAs with lost indication		
	RSPT #I	RSPT #2	Pulse Counter
PNA-D25 (Channel A)	22	Not Affected	Not Affected
PNB-D26 (Channel B)	67	Not Affected	Not Affected
PNC-D27 (Channel C)	Not Affected	67	67
PND-D28 (Channel D)	Not Affected	22	22

As shown above, a loss of Channel 'C' or Channel 'D' results in a loss of CEA position indication that is beyond the scope currently addressed by Condition 'B' of Tech Spec 3.1.5, since more than one CEA per group will have only one operable position indication channel. This problem is more significant for Unit 1 since unlike Units 2 and 3, there is no automatic transfer of the power source for the Vital Instrument Busses. Units 2 and 3 have static transfer switches which will maintain the Vital Instrument Bus energized on a loss of the normal (inverter) power supply by automatically transferring to the backup (voltage regulator) power supply. In Unit 1, there is no static transfer switch so the Vital Instrument Bus will be de-energized on a loss of the associated inverter. In addition, on a planned transfer between the inverter and the voltage regulator, the Vital Instrument Bus must be de-energized (for Unit 1 only) prior to powering from the alternate source.

Upon loss of any of the above Vital Instrument Buses, entry into Abnormal Operating Procedure 40AO-9ZZ13, 'Loss of Class Instrument or Control Power', is warranted. The applicable section directs declaring CEAC #1 inoperable for loss of either Channel 'A' or Channel 'B'; CEAC #2 inoperable for loss of either Channel 'C' or Channel 'D'. Condition 'A' of Tech Spec 3.3.3 [Control Element Assembly Calculators (CEAC)] will be entered and 40ST-92Z23, CEA Position Data Log, will be performed every 4 hours to verify the indicated position of each full and part length or part strength CEA is within 6.6 inches of all other CEAs in its group. This action is performed to comply with the 'Required Action/Completion Time' of LCO 3.3.3. In addition, the loss of any Vital Instrument Bus requires entry into Condition 'B' of TS 3.8.9 (Distribution Systems – Operating) which provides 2 hours to restore the bus operable. In the event the bus cannot be restored to operable, then the unit must be in Mode 3 within 6 hours. Loss of two Vital Instrument Buses (Condition E) will require entry into LCO 3.0.3.

Current TS Bases states that, "At least two of the following three CEA position indicator channels shall be OPERABLE for each CEA." Additionally the Bases states, "If only one CEA position indicator channel is OPERABLE, continued operation in MODES 1 and 2 may continue, provided, within 6 hours, at least two position indicator channels are returned to OPERABLE status; or within 6 hours and once per 12 hours, verify that the CEA group with the inoperable position indicators are either fully withdrawn or fully inserted while maintaining the insertion limits of LCO 3.1.6, LCO 3.1.7 and LCO 3.1.8." Current analyses already assumes that more than one CEA in a subgroup could have only one position indicator OPERABLE. This change will still require at least one position indication channel be available for each CEA.

The intent of this change is not to permit operation with less than 2 operable CEA position indication channels, per CEA. The operability requirements for CEA position indication will remain unchanged (at least 2 position indication channels for each CEA). This change is needed solely to address the lack of any existing Condition/Required Actions for situations in which more than one CEA per group has only one operable position indication channel. With no applicable Condition/Required Action, LCO 3.0.3 is required. However, providing a 6-hour completion time to restore the CEA indication is preferable to entering LCO 3.0.3 which would require shutdown to Mode 3 within 7 hours (and require a considerable amount of CEA manipulations during the power reduction). Also, the only credible single failure that would result in more than one CEA per group having only one operable position indication channel is the failure of Channel 'C' or Channel 'D', as discussed above. However in this case, the most limiting Tech Spec requirement would not be for CEA position indication. Required Action B.1 of Tech Spec 3.8.9 provides a 2-hour completion time to restore a de-energized Vital Instrument Bus. Thus, the 6-hour completion time associated with the proposed Condition 'B' of Tech Spec 3.1.5 would not be available for use if the vital instrument bus (Channel 'C' or Channel 'D') was not restored within 2 hours. When the vital instrument bus is restored, then the CEA position indication would also be restored. These time constraints serve as a limit to unit operation with only 1 CEA position indication for one or more CEA(s).

5.0 <u>REGULATORY ANALYSIS</u>

5.1 No Significant Hazards Consideration

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

This license amendment request (LAR) is 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 would revise sections of the Technical Specifications (TS) to support replacement of the part length control element assemblies (PLCEAs) with a new design that contains neutron absorber over the entire control section of the CEA. The replacements are referred to as part strength control element assemblies (PSCEAs). The proposed changes associated with this LAR are mainly changing the wording from "part length" to "part length or part strength" control element assemblies (CEAs) in several sections of TS. Included with this change will be the addition of the part strength CEAs description to Section 4.2.2 of the Technical Specifications. Even though there will be no changes or modifications to full length CEAs, for consistency and for ease of reading, the wording for "full length" CEAs will be changed to "full strength" CEAs. Additionally, TS 3.1.5 – "Control Element Assembly (CEA) Alignment," Condition B, will be modified to eliminate a potential condition which could cause an unwarranted plant shutdown.

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

   Response: No.

The physical difference between the 4-finger full strength control element assemblies (FSCEAs) and the PSCEAs involves using Inconel rather than B_4C (boron carbide) over 100% of the active control section of each CEA finger. In addition, the PSCEAs use Inconel tubing to encase solid Inconel slugs, which cover the entire control section of the control element assembly (CEA). The current PLCEAs (also have only 4-fingers) use solid Inconel rods for only the lower half of each finger and B_4C pellets in the top 15 inches (10%) of the control section of the CEA. Although failure of the solid Inconel region due to neutron fluence would be less likely than a typical clad design, the differences in swelling between the Inconel slugs encased by Inconel clad for the PSCEAs will be minor and result in a minimal impact on clad integrity. With the exception of the neutron absorber, the cladding design used for the PSCEAs is similar to the cladding of the full strength CEAs (FSCEAs). The geometry, cladding materials, and the spider assembly that supports the CEA fingers are essentially the same for the 4-finger FSCEAs and the PSCEAs. The principal difference results from the Inconel slugs contained in the PSCEAs being heavier than the B_4C pellets used in the FSCEAs. Even though the weight of a 4-finger PSCEA is greater than the weight of a 4-finger PLCEA or a 4-finger FSCEA, this weight difference is bounded by the 12-finger FSCEAs which are operated by the same CEA drive mechanism system.

The PSCEAs use Inconel as a neutron absorber in the entire control section of each CEA finger and will be operationally used the same way as the PLCEAs. In particular, the insertion restraints that are defined by the power dependent insertion limits (PDILs) for the PLCEAs will remain the same for the PSCEAs. This existing requirement will not result in any significant operational impact on the PSCEAs since the solid Inconel cylinder in the bottom 50% (operating range of the PDILs) of the PLCEAs has essentially the same reactivity worth as that of the PSCEAs.

In addition, renaming the full length CEAs and part length CEAs to full strength CEAs and part strength CEAs, respectively, and providing definition for the PSCEAs will not impact the safe operation of the plant. The terminology will be appropriately changed in any related document, equipment tag, or indication on a control panel.

The PLCEAs are not credited in the accident analyses for accident mitigation. The PSCEA design eliminates an accident scenario involving the insertion of a PLCEA past the PDIL, which results in an axial shift in power due to the upper region of the PLCEAs which has no neutron absorber. This condition will not occur with the PSCEAs because they are filled with neutron absorber over 100% of the control section of each finger.

Concerning TS Limiting Condition for Operation (LCO) 3.1.5, Condition B, proposed change; there are three position indicator channels available for each CEA. Current TS Bases state that, "At least two of the following three CEA position indicator channels shall be OPERABLE for each CEA." Additionally the TS Bases states, "If only one CEA position indicator channel is OPERABLE, continued operation in MODES 1 and 2 may continue, provided, within 6 hours, at least two position indicator channels are returned to OPERABLE status; or within 6 hours and once per 12 hours, verify that the CEA group with the inoperable position indicators are either fully withdrawn or fully inserted while maintaining the insertion limits of LCO 3.1.6, LCO 3.1.7 and LCO 3.1.8." The TS Bases make no restriction or condition limiting only one CEA within a subgroup to having only one CEA position indication channel. Current analyses already assume that more than one CEA in a subgroup could have only one position indicator OPERABLE. Modifying the wording for Condition B, of LCO 3.1.5, will not affect the likelihood or consequences of a CEA drop, slip, ejection, or misalignment. This change will still require at least one position indication channel be available for each CEA.

Consequently, the proposed change does not involve a significant increase in the probability or consequences of an accident.

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

   Response: No.

The proposed changes do not introduce any new mode of plant operation and the PSCEAs, like the PLCEAs, are not relied upon for accident mitigation. The PSCEAs will be operated in exactly the same manner in which the PLCEAs are operated. The existing operating restrictions for the PLCEAs will apply to the

PSCEAs. In particular, the power dependent insertion limit (PDIL) restrictions identified in the Core Operating Limits Report (COLR) will remain the same for the PSCEAs. The PSCEA design uses Inconel over the entire control section of each CEA finger, which will prevent the potential undesired flux redistribution currently associated with the misoperation of PLCEAs. Therefore, the analysis associated with the undesired flux redistribution misoperation for the PLCEAs will be eliminated from PVNGS safety analyses. PSCEA misoperation events are bounded by the existing PLCEA and FSCEA misoperation safety analyses.

In addition, renaming (within the Technical Specifications) the "full length CEAs" and "part length CEAs" to "full strength CEAs" and "part length or part strength CEAs," respectively, and providing a definition for the PSCEAs will not impact the safe operation of the plant. The terminology will be appropriately changed in any related document, equipment tag, or indication on a control panel.

Concerning TS LCO 3.1.5, Condition B proposed change, CEA position indication channels have no control function and provide input to the CEA Calculators (CEACs) and Core Protection Calculators (CPCs) for generation of a penalty factor. This change will still require at least one position indication channel be available for each CEA. Allowing Condition 'B' of LCO 3.1.5 to apply to more than one CEA per group does not create the possibility of a different type of malfunction than previously evaluated in the UFSAR.

Therefore, the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   Response: No.

The design of the PSCEAs is very similar to the FSCEAs except for the neutron absorber within each finger of a PSCEA. The PSCEAs do not have as strong of a neutron absorber (Inconel) as that which is contained in the FSCEAs (B_4C). There is a weight difference which results from the Inconel slugs contained in the PSCEAs being heavier than the B_4C pellets used in the FSCEAs. Even though the weight of the 4-finger PSCEAs is greater than the weight of the 4-finger PLCEAs, the CEA drive mechanism and support components shall operate within their design bases. Therefore, the PSCEAs can be considered adequate for safety-related applications. Consequently, the differences in design between the current PLCEAs and the PSCEAs do not adversely impact safe operation.

The PLCEAs are not relied upon for shutdown margin or accident mitigation and no new requirements will apply to the PSCEAs. However, the design of the PSCEAs is effectively eliminating the concern associated with the insertion of the PLCEAs past the PDILs which could result in an undesirable shift in neutron flux to the top of the core due to the region within the PLCEAs that do not have neutron absorber. The PSCEAs have neutron absorber throughout their entire control section, which prevents a neutron flux shift to the top of the core if inserted past the PDIL, when compared to that of the PLCEAs.

In addition, renaming the "full length CEAs" and "part length CEAs" to "full strength CEAs" and "part length or part strength CEAs," respectively, and providing definition for the PSCEAs will not impact the safe operation of the plant. The terminology will be appropriately changed in any related document, equipment tag, or indication on a control panel.

Concerning TS LCO 3.1.5, Condition B, proposed change, the current licensing bases already considers having more than one CEA in a CEA group with only one available position indication. The TS Bases for LCO 3.1.5, Condition B states that, "At least two of the following three CEA position indicator channels shall be OPERABLE for each CEA." Additionally the Bases states, "If only one CEA position indicator channel is OPERABLE, continued operation in MODES 1 and 2 may continue, provided, within 6 hours, at least two position indicator channels are returned to OPERABLE status; or within 6 hours and once per 12 hours, verify that the CEA group with the inoperable position indicators are either fully withdrawn or fully inserted while maintaining the insertion limits of LCO 3.1.6, LCO 3.1.7 and LCO 3.1.8." The TS Bases make no restriction or condition limiting only one CEA within a subgroup, to having only one CEA position indication channel OPERABLE. Therefore, modifying the wording for LCO 3.1.5, Condition B, does not involve a significant reduction in the margin of safety since loss of indication to more than one CEA is already considered in the licensing bases.

Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

Based on the above, APS concludes that the activities associated with 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.

5.2 Applicable Regulatory Requirements/Criteria

The proposed changes have been evaluated to determine whether applicable regulations and requirements continue to be met.

In the application for a license to operate a facility, 10 CFR 50.34(b)(6)(ii) requires that the following shall be part of the Updated Final Safety Analysis Report (UFSAR):

> "Managerial and administrative controls to be used to assure safe operation. Appendix B, "Quality Assurance Criteria for

> Nuclear Power Plants and Fuel Reprocessing Plants," sets forth the requirements for such controls for nuclear power plants and fuel reprocessing plants. The information on the controls to be used for a nuclear power plant or a fuel reprocessing plant shall include a discussion of how the applicable requirements of Appendix B will be satisfied."

In accordance with 10 CFR 50 Appendix B, a Quality Assurance Program, as outlined in Chapter 17.2 of the Palo Verde UFSAR, is utilized by APS for designing, purchasing, fabricating, handling, shipping, storing, cleaning, erecting, installing, inspecting, testing, operating, maintaining, repairing, and modifying activities that affect the safety-related functions of structures, systems, and components. As stated in the PVNGS Equipment Qualification Program,

> "The design, specification and procurement of new, replacement, or reworked equipment and parts shall consider the specific requirements necessary to maintain the continued qualification of installed equipment and environmental performance requirements of any "new" equipment."

Also, it states,

> "The qualification of new equipment and designs shall be verified prior to their installation in the plant."

In accordance with the Palo Verde Quality Assurance Program, the qualification requirements involving the PSCEAs such as suitability, functionality, environmental, seismic, electromagnetic and radio interference, human factors, software life cycle failure mode analysis, defense in depth and diversity analysis, and TMI action items were evaluated to ensure that the PSCEAs meet or exceed the original PLCEA requirements.

10 CFR Appendix A to Part 50, "General Design Criteria for Nuclear Power Plants," related to the design and operability requirements of the CEAs has been assessed to assure that the PSCEAs will satisfy regulatory design requirements. The criteria associated with the CEAs are summarized below.

> <u>Criterion 10 – Reactor design</u> – The principle difference associated with the PSCEA is the total mass and distribution of neutron absorber. However, PSCEAs are not subject to the potential anticipated operational occurrences (AOOs) currently seen by the PLCEAs due to the uniform distribution of the neutron absorber over the entire control section of each CEA finger for the PSCEAs. The only other significant difference is the weight of a PSCEA which is greater that a PLCEA. However, this difference has been analyzed for, as has the performance capability of the

CEA drive mechanisms, and found to be within design capabilities and design analyses.

Criterion 12 - Suppression of reactor power oscillations - Axial power oscillations are controlled using the PLCEAs and/or FLCEAs. The PSCEAs will be equally effective since their reactivity worth within the PDILs is essentially the same. The ability to reliably detect and suppress power oscillations is unaffected by the proposed changes.

Criterion 13 - Instrumentation and control - The existing systems and components used for monitoring and control of CEA positions are unaffected by the proposed changes and will be equally effective and relied upon for the control of the PSCEAs. The change for LCO 3.1.5, Condition B only addresses a more appropriate action to be taken given that the number of operable CEA position indications are less than that which is required for more than one CEA within a subgroup.

Criterion 26 - Reactivity control system redundancy and capability - The operational reactivity control characteristic of the PSCEAs is nearly identical to the PLCEAs. Redundancy and capability for the PSCEAs to control reactivity is not impacted and remains bounded by maintaining the operational restrictions required by the PDILs.

Criterion 27 - Combined reactivity control systems capability - The current design of the Reactor Control System includes a more than adequate capability for reactivity control using only the FLCEAs. As a result, neither the PLCEAs nor the PSCEAs are considered for shutdown margin and are not relied upon for accident mitigation. The design of the PSCEAs will not introduce any new effect which could impact the performance of the FLCEAs. Therefore, the reactivity control systems remain capable of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods, the capability to cool the core is maintained.

Criterion 28 - Reactivity limits - Operation of each reactor may require partial insertion of the PSCEAs in order to reconfigure the neutron flux distribution within the core. The ability of the PSCEAs to control reactivity for this effect is not impacted and remains bounded by maintaining the operational restrictions required by the PDILs. The PLCEAs and PSCEAs have nearly identical reactivity worth above the PDILs. Once inserted past the PDILs, the PSCEAs will add more negative reactivity than that of the PLCEAs when fully inserted into the core. Therefore, this proposed change would not cause a change in the amount or rate of reactivity increase different than what is already assumed in accident analyses.

Criterion 29 - Protection against anticipated operational occurrences - The PLCEAs are not relied upon for accident mitigation and provide no safety function; however, insertion of the PLCEAs past the PDIL could result in an event which is qualified as an AOO. A potential problem results due to 40% of the upper control section of each finger containing no neutron absorber. Significant insertion past the PDIL could result in undesirable core power redistribution. Since the design of the PSCEAs provides neutron absorber through the entire control section of each CEA finger, violation of the PDIL will be bounded by the AOOs involving the FLCEAs, which use a more reactive neutron absorber than present in the PSCEAs. Additionally, changing the name of the FLCEAs to FSCEAs does not affect safety function.

The requirements for Limiting Conditions for Operation (LCO) and Surveillance Requirements (SRs) to be included in the Technical Specifications (TS) are found in 10 CFR 50.36. As stated previously, the replacement PSCEAs (as they will be used with the existing PDILs) are functionally equivalent to the existing PLCEAs. Similarly, the proposed TS revisions are written to meet the same intent as the previous. Therefore, the lowest functional capability or performance levels of equipment required for safe operation of the facility will be retained in the proposed amendment. Likewise, Surveillance Requirements in the proposed amendment will continue to assure that the necessary quality of systems and components are maintained that facility operation will be within safety limits, and that limiting conditions for operation will be met.

6.0 ENVIRONMENTAL CONSIDERATIONS

Arizona Public Service Company has evaluated the proposed changes and has determined that the changes do not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amount of effluent that may be released offsite, or (iii) a significant increase in the individual or cumulative occupational radiation exposure. Accordingly, the proposed change meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), an environmental assessment of the proposed change is not required.

ATTACHMENT 1

MARKUP OF TECHNICAL SPECIFICATION PAGES

NOTE:

The attached marked up Technical Specification pages for LCO 3.3.3 (only) are the associated pages for a pending change with the NRC for the approval of the replacement of Core Protection Calculator Systems (CPCS), submitted on 11/07/02 (102-04864-CDM/TNW/DWG - Request for Amendment to Technical Specifications: 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))

PALO VERDE NUCLEAR GENERATING STATION
IMPROVED TECHNICAL SPECIFICATIONS
TABLE OF CONTENTS

1.1 Definitions (continued)

ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME

The ESF RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. 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.

K_{n-1}

K_{n-1} is the K effective calculated by considering the actual CEA configuration and assuming that the fully or partially inserted full ~~length~~ strength CEA of highest worth is fully withdrawn.

LEAKAGE

LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank;

2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE; or

3. Reactor Coolant System (RCS) LEAKAGE through a steam generator (SG) to the Secondary System.

(continued)

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.
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 ~~length~~ 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 ~~length~~ 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.

(continued)

3.1 REACTIVITY CONTROL SYSTEMS

3.1.5 Control Element Assembly (CEA) Alignment

LCO 3.1.5 All full ~~length~~ 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)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. Only one CEA position indicator channel OPERABLE for one ~~CEA per CEA Group.~~ or more CEA(s).	B.1	Restore at least two position indicator channels to OPERABLE status.	6 hours
	<u>OR</u>		
	B.2	Verify the CEA Group(s) with the inoperable position indicators are fully withdrawn or fully inserted while maintaining the insertion limits of LCO 3.1.6, LCO 3.1.7 and LCO 3.1.8.	6 hours <u>AND</u> Once per 12 hours thereafter.
C. Required Action and associated Completion Time of Condition A or B not met <u>OR</u> One or more full ~~length~~ strength CEAs untrippable.	C.1	Be in MODE 3.	6 hours
D. Two or more CEAs trippable and misaligned from their group by > 9.9 inches.	D.1	Open the reactor trip breakers.	Immediately

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 ~~length~~ strength CEA freedom of movement (trippability) by moving each individual full ~~length~~ 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 ~~length~~ strength CEA drop time $\leq$ 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

or part strength

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 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 CEA groups to within the long term steady state insertion limit.	2 hours

(continued)

or Part Strength

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 CEA group position.	12 hours

or part strength

3.1 REACTIVITY CONTROL SYSTEMS

3.1.9 Special Test Exception (STE) - SHUTDOWN MARGIN (SDM)

LCO 3.1.9 During performance of PHYSICS TESTS, the requirements of:

LCO 3.1.2, "SHUTDOWN MARGIN (SDM)-Reactor Trip Breakers Closed";
LCO 3.1.6, "Shutdown Control Element Assembly (CEA) Insertion Limits", and
LCO 3.1.7 "Regulating Control Element Assembly (CEA) Insertion Limits"

may be suspended for measurement of CEA worth, provided shutdown reactivity equivalent to at least the highest estimated CEA worth (of those CEAs actually withdrawn) is available for trip insertion or the reactor is subcritical by at least the reactivity equivalent of the highest CEA worth.

APPLICABILITY: MODES 2 and 3 during PHYSICS TESTS.

------------------------------NOTE------------------------------
Operation in MODE 3 shall be limited to 6 consecutive hours.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Any full ~~length~~ CEA not fully inserted and less than the required shutdown reactivity available for trip insertion. strength <u>OR</u> All full ~~length~~ CEAs inserted and the reactor subcritical by less than the above required shutdown reactivity equivalent.	A.1 Initiate boration to restore required shutdown reactivity.	15 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.9.1	Verify that the position of each CEA not fully inserted is within the acceptance criteria for available negative reactivity addition.	2 hours
SR 3.1.9.2	Verify each full ~~length~~ CEA not fully inserted is capable of full insertion when tripped from at least the 50% withdrawn position. strength	Within 7 days prior to reducing SDM requirements to less than the limits of LCO 3.1.2
SR 3.1.9.3	-------------------NOTE------------------- Only required to be performed in Mode 3. -- Verify that with all full ~~length~~ CEAs fully inserted, the reactor is subcritical within the acceptance criteria.	2 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"; or Part Strength
LCO 3.1.7, "Regulating Control Element Assembly (CEA) Insertion Limits";
LCO 3.1.8, "Part Length 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 THERMAL POWER is restricted to the test power plateau, which shall not exceed 85% RTP.

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"; or Part Strength
LCO 3.1.8, "Part Length ~~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

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. (continued)	B.2	Verify all full ~~length~~ 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)

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 ~~length~~ 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.	4 hours
	AND	
		(continued)

4.0 DESIGN FEATURES

4.1 Site Location

The Palo Verde Nuclear Generating Station is located in Maricopa County, Arizona, approximately 50 miles west of the Phoenix metropolitan area. The site is comprised of approximately 4,050 acres. Site elevations range from 890 feet above mean sea level at the southern boundary to 1,030 feet above mean sea level at the northern boundary. The minimum distance from a containment building to the exclusion area boundary is 871 meters.

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 241 fuel assemblies. Each assembly shall consist of a matrix of Zircaloy or ZIRLO fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions. Other cladding material may be used with an approved exemption.

4.2.2 Control Element Assemblies

~~The reactor core shall contain 76 full length and 13 part length control element assemblies (CEAs). The control material shall be boron carbide with Inconel Alloy 625 used as a wear absorber over a portion of the part length control element assemblies as approved by the NRC.~~

The reactor core shall contain 76 full strength and either 13 part length or 13 part strength control element assemblies (CEAs).

The control section for the full strength CEAs shall be boron carbide with Inconel Alloy 625 cladding.

For units that have part length CEAs, the control section shall be Inconel Alloy 625 in the lower half, followed by perforated stainless steel tubing over the next 40%, and boron carbide pellets with Inconel Alloy 625 clad over the last 10% of the control section.

For units that have part strength CEAs, the control section shall be solid Inconel Alloy 625 slugs with Inconel Alloy 625 cladding.

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.

or Part Strength

7. Part Length 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 CEA Insertion Limits and 3.2.3, Azimuthal Power Tilt - T_q]. or Part Strength

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 (continued)

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)].

or Part Strength

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 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.

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any mid cycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

(continued)

ATTACHMENT 2

RETYPED TECHNICAL SPECIFICATION PAGES

NOTE:

The attached retyped Technical Specification pages for LCO 3.3.3 (only) are the associated pages for a pending change with the NRC for the approval of the replacement of Core Protection Calculator Systems (CPCS), submitted on 11/07/02 (102-04864-CDM/TNW/DWG - Request for Amendment to Technical Specifications: 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))

PALO VERDE NUCLEAR GENERATING STATION
IMPROVED TECHNICAL SPECIFICATIONS
TABLE OF CONTENTS

1.1 Definitions (continued)

ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME	The ESF RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. 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.
K_{n-1}	K_{n-1} is the K effective calculated by considering the actual CEA configuration and assuming that the fully or partially inserted full strength CEA of highest worth is fully withdrawn.
LEAKAGE	LEAKAGE shall be: a. <u>Identified LEAKAGE</u> 1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank; 2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE; or 3. Reactor Coolant System (RCS) LEAKAGE through a steam generator (SG) to the Secondary System.

(continued)

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.
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.

(continued)

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)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. Only one CEA position indicator channel OPERABLE for one or more CEA(s).	B.1	Restore at least two position indicator channels to OPERABLE status.	6 hours
	<u>OR</u>		
	B.2	Verify the CEA Group(s) with the inoperable position indicators are fully withdrawn or fully inserted while maintaining the insertion limits of LCO 3.1.6, LCO 3.1.7 and LCO 3.1.8.	6 hours <u>AND</u> Once per 12 hours thereafter.
C. Required Action and associated Completion Time of Condition A or B not met <u>OR</u> One or more full strength CEAs untrippable.	C.1	Be in MODE 3.	6 hours
D. Two or more CEAs trippable and misaligned from their group by > 9.9 inches.	D.1	Open the reactor trip breakers.	Immediately

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 $\leq$ 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. OR A.2 Reduce THERMAL POWER to less than or equal to that fraction of RTP specified in the COLR.	2 hours 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

3.1.9 Special Test Exception (STE) – SHUTDOWN MARGIN (SDM)

LCO 3.1.9 During performance of PHYSICS TESTS, the requirements of:

LCO 3.1.2, "SHUTDOWN MARGIN (SDM)-Reactor Trip Breakers Closed";
LCO 3.1.6, "Shutdown Control Element Assembly (CEA) Insertion Limits", and
LCO 3.1.7 "Regulating Control Element Assembly (CEA) Insertion Limits"

may be suspended for measurement of CEA worth, provided shutdown reactivity equivalent to at least the highest estimated CEA worth (of those CEAs actually withdrawn) is available for trip insertion or the reactor is subcritical by at least the reactivity equivalent of the highest CEA worth.

APPLICABILITY: MODES 2 and 3 during PHYSICS TESTS.

---------------------------NOTE---------------------------
Operation in MODE 3 shall be limited to 6 consecutive hours.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Any full strength CEA not fully inserted and less than the required shutdown reactivity available for trip insertion. OR All full strength CEAs inserted and the reactor subcritical by less than the above required shutdown reactivity equivalent.	A.1	Initiate boration to restore required shutdown reactivity.	15 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.9.1	Verify that the position of each CEA not fully inserted is within the acceptance criteria for available negative reactivity addition.	2 hours
SR 3.1.9.2	Verify each full strength CEA not fully inserted is capable of full insertion when tripped from at least the 50% withdrawn position.	Within 7 days prior to reducing SDM requirements to less than the limits of LCO 3.1.2
SR 3.1.9.3	-------------------NOTE------------------- Only required to be performed in Mode 3. --	
	Verify that with all full strength CEAs fully inserted, the reactor is subcritical within the acceptance criteria.	2 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 THERMAL POWER is restricted to the test power plateau, which shall not exceed 85% RTP.

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

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
	<u>AND</u>		
	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
	<u>AND</u>		
	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
	<u>AND</u>		
	B.5	Perform SR 3.1.5.1.	Once per 4 hours
	<u>AND</u>		(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 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	4 hours
	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)

4.0 DESIGN FEATURES

4.1 Site Location

The Palo Verde Nuclear Generating Station is located in Maricopa County, Arizona, approximately 50 miles west of the Phoenix metropolitan area. The site is comprised of approximately 4,050 acres. Site elevations range from 890 feet above mean sea level at the southern boundary to 1,030 feet above mean sea level at the northern boundary. The minimum distance from a containment building to the exclusion area boundary is 871 meters.

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 241 fuel assemblies. Each assembly shall consist of a matrix of Zircaloy or ZIRLO fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions. Other cladding material may be used with an approved exemption.

4.2.2 Control Element Assemblies

The reactor core shall contain 76 full strength and either 13 part length or 13 part strength control element assemblies (CEAs).

The control section for the full strength CEAs shall be boron carbide with Inconel Alloy 625 cladding.

For units that have part length CEAs, the control section shall be Inconel Alloy 625 in the lower half, followed by perforated stainless steel tubing over the next 40%, and boron carbide pellets with Inconel Alloy 625 clad over the last 10% of the control section.

For units that have part strength CEAs, the control section shall be solid Inconel Alloy 625 slugs with Inconel Alloy 625 cladding.

(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 (continued)

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.

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any mid cycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

(continued)

ATTACHMENT 3

ASSOCIATED CHANGES TO TECHNICAL SPECIFICATION BASES

(for information only)

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.1 SHUTDOWN MARGIN (SDM) - Reactor Trip Breakers Open

BASES

BACKGROUND

The reactivity control systems must be redundant and capable of holding the reactor core subcritical when shutdown under cold conditions, in accordance with GDC 26 (Ref. 1). Maintenance of the SDM ensures that postulated reactivity events will not damage the fuel. SDM requirements provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for normal shutdown and anticipated operational occurrences (AOOs). As such, the SDM defines the degree of subcriticality that would be obtained immediately following the insertion of all full ~~length~~ strength control element assemblies (CEAs), assuming the single CEA of highest reactivity worth is fully withdrawn with Reactor Trip Breakers open. This reactivity worth is credited in establishing the required SDM.

The system design requires that two independent reactivity control systems be provided, and that one of these systems be capable of maintaining the core subcritical under cold conditions. These requirements are provided by the use of movable CEAs and soluble boric acid in the Reactor Coolant System (RCS). The CEA System provides the SDM during power operation and is capable of making the core subcritical rapidly enough to prevent exceeding acceptable fuel design limits, assuming that the CEA of highest reactivity worth remains fully withdrawn.

The soluble boron system can compensate for fuel depletion during operation and all xenon burnout reactivity changes, and maintain the reactor subcritical under cold conditions.

During power operation, SDM control is ensured by operating with the shutdown CEAs fully withdrawn and the regulating CEAs within the limits of LCO 3.1.7, "Regulating Control Element Assembly (CEA) Insertion Limits." When the unit is in the shutdown and refueling modes, the SDM requirements are met by means of adjustments to the RCS boron concentration.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

occurs as a result of the post trip return to power, and THERMAL POWER does not violate the Safety Limit (SL) requirement of SL 2.1.1.

In addition to the limiting MSLB transient, the SDM requirement for MODES 3, 4, and 5 must also protect against:

a. Inadvertent boron dilution;

b. Startup of an inactive reactor coolant pump (RCP); and

c. CEA ejection.

Each of these is discussed below.

In the inadvertent boron dilution analysis, the amount of reactivity by which the reactor is subcritical is determined by the reactivity difference between an initial subcritical boron concentration and the corresponding critical boron concentration. The initial subcritical boron concentration assumed in the analysis corresponds to the minimum SDM requirements. These two values (initial and critical boron concentrations), in conjunction with the configuration of the Reactor Coolant System (RCS) and the assumed dilution flow rate, directly affect the results of the analysis. For this reason the event is most limiting at the beginning of core life when critical boron concentrations are highest.

The startup of an inactive RCP will not result in a "cold water" criticality, even if the maximum difference in temperature exists between the SG and the core. Although this event was considered in establishing the requirements for SDM, it is not the limiting event with respect to the specification limits.

In the analysis of the CEA ejection event, maintaining SDM ensures the reactor remains subcritical following a CEA ejection and, therefore, satisfies the radially averaged enthalpy acceptance criterion considering power redistribution effects.

SHUTDOWN MARGIN is the amount by which the core is subcritical, or would be subcritical immediately following a reactor trip, considering a single malfunction resulting in the highest worth CEA failing to insert. With any full ~~length~~ (strength) CEAs not capable of being fully inserted, the withdrawn reactivity worth of these CEAs must be accounted for in the determination of SDM.

(continued)

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.2 SHUTDOWN MARGIN (SDM) - Reactor Trip Breakers Closed

BASES

BACKGROUND

The reactivity control systems must be redundant and capable of holding the reactor core subcritical when shut down under cold conditions, in accordance with GDC 26 (Ref. 1). Maintenance of the SDM ensures that postulated reactivity events will not damage the fuel. SDM requirements provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for normal shutdown and anticipated operational occurrences (AOOs). As such, SDM defines the degree of subcriticality that would be obtained immediately following the insertion of all full ~~length~~ strength control element assemblies (CEAs), assuming the single CEA of highest reactivity worth is fully withdrawn.

The system design requires that two independent reactivity control systems be provided, and that one of these systems be capable of maintaining the core subcritical under cold conditions. These requirements are provided by the use of movable CEAs and soluble boric acid in the Reactor Coolant System (RCS). The CEA System provides the SDM during power operation and is capable of making the core subcritical rapidly enough to prevent exceeding the acceptable fuel design limits, assuming that the CEA of highest reactivity worth remains fully withdrawn.

The soluble boron system can compensate for fuel depletion during operation and all xenon burnout reactivity changes, and maintain the reactor subcritical under cold conditions.

During power operation, SDM control is ensured by operating with the shutdown CEAs fully withdrawn and the regulating CEAs within the limits of LCO 3.1.7, "Regulating Control Element Assembly (CEA) Insertion Limits." When the unit is in the shutdown and refueling modes, the SDM requirements are met by means of adjustments to the RCS boron concentration.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

The startup of an inactive RCP will not result in a "cold water" criticality, even if the maximum difference in temperature exists between the SG and the core. Although this event was considered in establishing the requirements for SDM, it is not the limiting event with respect to the specification limits.

In the analysis of the CEA ejection event, SDM alone cannot prevent reactor criticality following a CEA ejection. At temperatures less than 500 F, the K_{N-1} requirement ensures the reactor remains subcritical and, therefore, satisfies the radially averaged enthalpy acceptance criterion considering power redistribution effects. Above 500 F, Doppler reactivity feedback is sufficient to preclude the need for a specific K_{N-1} requirement.

The function of SHUTDOWN MARGIN is to ensure that the reactor remains subcritical following a design basis accident or anticipated operational occurrence. During operation in MODES 1 and 2, with k_{eff} greater than or equal to 1.0, the transient insertion limits of Specification 3.1.3.6 ensure that sufficient SHUTDOWN MARGIN is available.

SHUTDOWN MARGIN is the amount by which the core is subcritical, or would be subcritical immediately following a reactor trip, considering a single malfunction resulting in the highest worth CEA failing to insert. With any full ~~length~~ strength CEAs not capable of being fully inserted, the withdrawn reactivity worth of the CEAs must be accounted for in the determination of SDM.

SHUTDOWN MARGIN requirements vary throughout the core life as a function of fuel depletion and reactor coolant system (RCS) cold leg temperature (T_{cold}). The most restrictive condition occurs at EOL, with T_{cold} at no-load operating temperature, and is associated with a postulated steam line break accident and the resulting uncontrolled RCS cooldown. In the analysis of this accident, the specified SHUTDOWN MARGIN is required to control the reactivity transient and ensure that the fuel performance and offsite dose criteria are satisfied.

(continued)

BASES

BACKGROUND (continued)

The CEAs are arranged into groups that are radially symmetric. Therefore, movement of the CEAs does not introduce radial asymmetries in the core power distribution. The shutdown and regulating CEAs provide the required reactivity worth for immediate reactor shutdown upon a reactor trip. The regulating CEAs also provide reactivity (power level) control during normal operation and transients. Their movement may be automatically controlled by the Reactor Regulating System. Part length (or part strength) CEAs are not credited in the safety analyses for shutting down the reactor, as are the regulating and shutdown groups. The part length (or part strength) CEAs are used solely for ASI control.

The axial position of shutdown and regulating CEAs is indicated by two separate and independent systems, which are the Pulse Counting CEA Position Indication System (described in Ref. 4) and the Reed Switch CEA Position Indication System (described in Ref. 5).

The Pulse Counting CEA Position Indicating System indicates CEA position to the actual step, if each CEA moves one step for each command signal. However, if each CEA does not follow the commands, the system will incorrectly reflect the position of the affected CEA(s). This condition may affect the operability of COLSS (refer to Section 3.2, Power Distribution Limits for the applicable actions) and should be detected by the Reed Switch Position Indication System through surveillance or alarm. Although the Reed Switch Position Indication System is less precise that the Pulse Counting CEA Position Indicating System, it is not subject to the same error mechanisms.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

CEA misalignment accidents are analyzed in the safety analysis (Ref. 3). The accident analysis defines CEA misoperation as any event, with the exception of sequential group withdrawals, which could result from a single malfunction in the reactivity control systems. For example, CEA misalignment may be caused by a malfunction of the CEDM, CEDMCS, or by operator error. A stuck CEA may be caused by mechanical jamming of the CEA fingers or of the gripper.

Inadvertent withdrawal of a single CEA may be caused by an electrical failure in the CEA coil power programmers. A dropped CEA could be caused by an opening of the electrical circuit of the CEDM holding coil for a full ~~length~~ strength, part length or part strength CEA.

The acceptance criteria for addressing CEA inoperability or misalignment are that:

a. There shall be no violations of:

   1. specified acceptable fuel design limits, or

   2. Reactor Coolant System (RCS) pressure boundary integrity; and

b. The core must remain subcritical after accident transients.

Three types of misalignment are distinguished. During movement of a group, one CEA may stop moving while the other CEAs in the group continue. This condition may cause excessive power peaking. The second type of misalignment occurs if one CEA fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition requires an evaluation to determine that sufficient reactivity worth is held in the remaining CEAs to meet the SDM requirement with the maximum worth CEA stuck fully withdrawn. If a CEA is stuck in the fully withdrawn position, its worth is added to the SDM requirement, since the safety analysis does not take two stuck CEAs into account. The third type of misalignment occurs when one CEA drops partially or fully into the reactor core. This event causes an initial power reduction followed by a return towards the original power due to positive reactivity feedback from the negative moderator temperature coefficient. Increased peaking during the power increase may result in erosion of DNB margin.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Analysis considers the case of a single CEA withdrawn approximately 10 inches from a bank inserted to its insertion limit. Satisfying limits on departure from nucleate boiling ratio (DNBR) bounds the situation when a CEA is misaligned from its group by 6.6 inches.

The effect of any misoperated CEA on the core power distribution will be assessed by the CEA calculators, and an appropriately augmented power distribution penalty factor will be supplied as input to the core protection calculators (CPCs). As the reactor core responds to the reactivity changes caused by the misoperated CEA and the ensuing reactor coolant and Doppler feedback effects, the CPCs will initiate a low DNBR or high local power density trip signal if specified acceptable fuel design limits (SAFDLs) are approached.

part strength

~~Since~~ The CEA drop incidents result in the most rapid approach to SAFDLs caused by a CEA misoperation. The accident analysis analyzed a single full ~~length~~ strength CEA drop, a single part length CEA drop, and a part length CEA subgroup drop. The most rapid approach to the DNBR SAFDL or the fuel centerline melt SAFDL is caused by a single full ~~length~~ strength CEA drop.

A part strength CEA drop would cause a similiar reactivity response although with less of a magnitude due to the full strength CEAs having a more significant reactivity worth.

In the case of the full ~~length~~ strength CEA drop, a prompt decrease in core average power and a distortion in radial power are initially produced, which when conservatively coupled result in local power and heat flux increases, and a decrease in DNBR. As the dropped CEA is detected, core power and an appropriately augmented power distribution penalty factor are supplied to the CPCs. For plant operation within the DNBR and local power density (LPD) LCOs, DNBR and LPD trips can normally be avoided on a dropped 4-finger CEA.

For a part length or part strength CEA subgroup drop, a distortion in power distribution, and a decrease in core power are produced. As the position of the dropped part length or part strength CEA subgroup is detected, an appropriate power distribution penalty factor is supplied to the CPCs, and a reactor trip signal on low DNBR is generated. For the part length CEA drop, both core average power and three dimensional peak to average power density increase promptly. As the dropped part length CEA is detected, core power and an appropriately augmented power distribution penalty factor are supplied to the CPCs.

start new paragraph

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES	CEA alignment satisfies Criteria 2 and 3 of 10 CFR 50.36 (c)(2)(ii).

LCO

The limits on part length or part strength, shutdown, and regulating CEA alignments ensure that ~~the assumptions~~ in the safety analysis will remain valid. The requirements on OPERABILITY ensure that upon reactor trip, the CEAs will be available and will be inserted to provide enough negative reactivity to shut down the reactor. The OPERABILITY requirements also ensure that the CEA banks maintain the correct power distribution and CEA alignment.

The requirement is to maintain the CEA alignment to within 6.6 inches between any CEA and all other CEAs in its group.

Failure to meet the requirements of this LCO may produce unacceptable power peaking factors, DNBR, and LHRs, or unacceptable SDMs, all of which may constitute initial conditions inconsistent with the safety analysis.

APPLICABILITY

The requirements on CEA OPERABILITY and alignment are applicable in MODES 1 and 2 because these are the only MODES in which neutron (or fission) power is generated, and the OPERABILITY (e.g., trippability) and alignment of CEAs have the potential to affect the safety of the plant. In MODES 3, 4, 5, and 6, the alignment limits do not apply because the reactor is shut down and not producing fission power. In the shutdown modes, the OPERABILITY of the shutdown and regulating CEAs has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the RCS. See LCO 3.1.2, "SHUTDOWN MARGIN (SDM) - Reactor Trip Breakers Closed," for SDM in MODES 3, 4, and 5, and LCO 3.9.1, "Boron Concentration," for boron concentration requirements during refueling.

(continued)

BASES (continued)

ACTIONS

A.1 and A.2

A CEA may become misaligned, yet remain trippable. In this condition, the CEA can still perform its required function of adding negative reactivity should a reactor trip be necessary.

If one or more CEAs (regulating, shutdown, ~~or~~ part length, or part strength) are misaligned by 6.6 inches and ≤ 9.9 inches but trippable, or one CEA misaligned by > 9.9 inches but trippable, continued operation in MODES 1 and 2 may continue, provided, within 1 hour, the power is reduced in accordance with the limits in the COLR, and within 2 hours CEA alignment is restored. Regulating and part length or part strength CEA alignment can be restored by either aligning the misaligned CEA(s) to within 6.6 inches of its group or aligning the misaligned CEA's group to within 6.6 inches of the misaligned CEA(s). Shutdown CEA alignment can be restored by aligning the misaligned CEA(s) to within 6.6 inches of its group.

Xenon redistribution in the core starts to occur as soon as a CEA becomes misaligned. Reducing THERMAL POWER in accordance with the limits in the COLR ensures acceptable power distributions are maintained (Ref. 3). For small misalignments (< 9.9 inches) of the CEAs, there is:

a. A small effect on the time dependent long term power distributions relative to those used in generating LCOs and limiting safety system settings (LSSS) setpoints;

b. A negligible effect on the available SDM; and

c. A small effect on the ejected CEA worth used in the accident analysis.

With a large CEA misalignment (≥ 9.9 inches), however, this misalignment would cause distortion of the core power distribution. This distortion may, in turn, have a significant effect on the time dependent, long term power distributions relative to those used in generating LCOs and LSSS setpoints. The effect on the available SDM and the ejected CEA worth used in the accident analysis remain small.

Therefore, this condition is limited to the single CEA misalignment, while still allowing 2 hours for recovery.

(continued)

BASES

ACTIONS

C.1 (continued)

known to be untrippable), the unit is required to be brought to MODE 3. By being brought to MODE 3, the unit is brought outside its MODE of applicability.

When a Required Action cannot be completed within the required Completion Time, a controlled shutdown should be commenced. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

If a full ~~length~~ strength CEA is untrippable, it is not available for reactivity insertion during a reactor trip. With an untrippable CEA, meeting the insertion limits of LCO 3.1.6, "Shutdown Control Element Assembly (CEA) Insertion Limits," and LCO 3.1.7, "Regulating Control Element Assembly (CEA)

Insertion Limits," does not ensure that adequate SDM exists. Therefore, the plant must be shut down in order to evaluate the SDM required boron concentration and power level for critical operation. Continued operation is allowed with untrippable part length or part strength CEAs if the alignment and insertion limits are met.

Continued operation is not allowed with one or more full length CEAs untrippable. This is because these cases are indicative of a loss of SDM and power distribution, and a loss of safety function, respectively.

D.1

Continued operation is not allowed in the case of more than one CEA misaligned from any other CEA in its group by > 9.9 inches. For example, two CEAs in a group misaligned from any other CEA in that group by > 9.9. inches, or more than one CEA group that has a least one CEA misaligned from any other CEA in that group by > 9.9 inches. This is indicative of a loss of power distribution and a loss of safety function, respectively. Multiple CEA misalignments should result in automatic protective action. Therefore, with two or more CEAs misaligned more than 9.9 inches, this could result in a situation outside the design basis and immediate action would be required to prevent any potential fuel damage. Immediately opening the reactor trip breakers minimizes these effects.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.5.1

Verification that individual CEA positions are within 6.6 inches (indicated reed switch positions) of all other CEAs in the group at a 12 hour Frequency allows the operator to detect a CEA that is beginning to deviate from its expected position. The specified Frequency takes into account other CEA position information that is continuously available to the operator in the control room, so that during actual CEA motion, deviations can immediately be detected.

SR 3.1.5.2

OPERABILITY of at least two CEA position indicator channels is required to determine CEA positions, and thereby ensure compliance with the CEA alignment and insertion limits. The CEA full in and full out limits provide an additional independent means for determining the CEA positions when the CEAs are at either their fully inserted or fully withdrawn positions.

SR 3.1.5.3

Verifying each full ~~length~~ strength CEA is trippable would require that each CEA be tripped. In MODES 1 and 2 tripping each full ~~length~~ strength CEA would result in radial or axial power tilts, or oscillations. Therefore individual full ~~length~~ strength CEAs are exercised every 92 days to provide increased confidence that all full ~~length~~ strength CEAs continue to be trippable, even if they are not regularly tripped. A movement of 5 inches is adequate to demonstrate motion without exceeding the alignment limit when only one full ~~length~~ strength CEA is being moved. The 92 day Frequency takes into consideration other information available to the operator in the control room and other surveillances being performed more frequently, which add to the determination of OPERABILITY of the CEAs (Ref. 3). Between required performances of SR 3.1.5.3, if a CEA(s) is discovered to be immovable but remains trippable and aligned, the CEA is considered to be OPERABLE. At anytime, if a CEA(s) is immovable, a determination of the trippability (OPERABILITY) of that CEA(s) must be made, and appropriate action taken.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.5.4

Performance of a CHANNEL FUNCTIONAL TEST of each reed switch position transmitter channel ensures the channel is OPERABLE and capable of indicating CEA position. Since this test must be performed when the reactor is shut down, an 18 month Frequency to be coincident with refueling outage was selected. Operating experience has shown that these components usually pass this Surveillance when performed at a Frequency of once every 18 months. Furthermore, the Frequency takes into account other factors, which determine the OPERABILITY of the CEA Reed Switch Indication System. These factors include:

a. Other, more frequently performed surveillances that help to verify OPERABILITY;

b. On-line diagnostics performed automatically by the CPCs, CEACs, and the Plant Computer which include CEA position comparisons and sensor validation; and

c. The CHANNEL CALIBRATIONs for the CPCs (SR 3.3.1.9) and CEACs (SR 3.3.3.4) input channels that are performed at 18 month intervals and is an overlapping test.

SR 3.1.5.5

Verification of full ~~length~~ strength CEA drop times determines that the maximum CEA drop time permitted is consistent with the assumed drop time used in the safety analysis (Ref. 3). Measuring drop times prior to reactor criticality, after reactor vessel head removal, ensures the reactor internals and CEDM will not interfere with CEA motion or drop time, and that no degradation in these systems has occurred that would adversely affect CEA motion or drop time. Individual CEAs whose drop times are greater than safety analysis assumptions are not OPERABLE. This SR is performed prior to criticality due to the plant conditions needed to perform the SR and the potential for an unplanned plant transient if the Surveillance were performed with the reactor at power.

The 4 second CEA drop time is the maximum time it takes for a fully withdrawn individual full ~~length~~ strength CEA to reach its 90% insertion position when electrical power is interrupted to the CEA drive mechanism with RCS T_{cold} greater than or equal to 550°F and all reactor coolant pumps operating.

(continued)

BASES

The CEA drop time of full ~~length~~ strength CEAs shall also be demonstrated through measurement prior to reactor criticality for specifically affected individual CEAs following any maintenance on or modification to the CEA drive system which could affect the drop time of those specific CEAs.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10 and GDC 26.
2. 10 CFR 50.46.
3. UFSAR, Section 15.4.
4. UFSAR, Section 7.7.1.3.2.3.
5. UFSAR, Section 7.5.1.1.4.

BASES

BACKGROUND (continued)

event of a CEA ejection accident, and the shutdown and regulating bank insertion limits ensure the required SDM is maintained.

Operation within the subject LCO limits will prevent fuel cladding failures that would breach the primary fission product barrier and release fission products to the reactor coolant in the event of a LOCA, loss of flow, ejected CEA, or other accident requiring termination by a Reactor Protection System trip function.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation (Condition I) and anticipated operational occurrences (Condition II). The acceptance criteria for the regulating CEA insertion, part length (or part strength) CEA insertion, ASI, and T_q LCOs preclude core power distributions from occurring that would violate the following fuel design criteria:

a. During a large break LOCA, the peak cladding temperature must not exceed a limit of 2200°F, 10 CFR 50.46 (Ref. 2);

b. During CEA misoperation events, there must be at least a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition;

c. During an ejected CEA accident, the fission energy input to the fuel must not exceed 280 cal/gm (Ref. 3); and

d. The CEAs must be capable of shutting down the reactor with a minimum required SDM, with the highest worth CEA stuck fully withdrawn, GDC 26 (Ref. 1).

Regulating CEA position, ASI, and T_q are process variables that together characterize and control the three dimensional power distribution of the reactor core.

Fuel cladding damage does not occur when the core is operated outside these LCOs during normal operation. However, fuel cladding damage could result, should an

(continued)

or Part Strength

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.8 Part Length Control Element Assembly (CEA) Insertion Limits

BASES

BACKGROUND

The insertion limits of the part length CEAs are initial assumptions in the safety analyses for CEA misoperation events. The insertion limits directly affect core power distributions. The applicable criteria for these power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Plants" (Ref. 2). Limits on part length CEA insertion have been established, and all CEA positions are monitored and controlled during power operation to ensure that the power distribution defined by the design power peaking limits is preserved.

or part strength

The part length CEAs are used for axial power shape control of the reactor. The positions of the part length CEAs are manually controlled. They are capable of changing reactivity very quickly (compared to borating or diluting).

The power density at any point in the core must be limited to maintain specified acceptable fuel design limits, including limits that preserve the criteria specified in 10 CFR 50.46 (Ref. 2). Together, LCO 3.1.7, "Regulating Control Element Assembly (CEA) Insertion Limits"; LCO 3.1.8; LCO 3.2.4, "Departure From Nucleate Boiling Ratio (DNBR)"; and LCO 3.2.5, "AXIAL SHAPE INDEX (ASI)," provide limits on control component operation and on monitored process variables to ensure the core operates within the linear heat rate (LHR) (LCO 3.2.1, "Linear Heat Rate (LHR)"); planar peaking factor (F_{xy}) (LCO 3.2.2, "Planar Radial Peaking Factors (F_{xy})"); and LCO 3.2.4 limits in the COLR.

Operation within the limits given in the COLR prevents power peaks that would exceed the loss of coolant accident (LOCA) limits derived by the Emergency Core Cooling Systems analysis. Operation within the F_{xy} and departure from nucleate boiling (DNB) limits given in the COLR prevents DNB during a loss of forced reactor coolant flow accident.

(continued)

or Part Strength

BASES

BACKGROUND (continued)

The establishment of limiting safety system settings and LCOs requires that the expected long and short term behavior of the radial peaking factors be determined. The long term behavior relates to the variation of the steady state radial peaking factors with core burnup; it is affected by the amount of CEA insertion assumed, the portion of a burnup cycle over which such insertion is assumed, and the expected power level variation throughout the cycle. The short term behavior relates to transient perturbations to the steady state radial peaks due to radial xenon redistribution. The magnitudes of such perturbations depend upon the expected use of the CEAs during anticipated power reductions and load maneuvering. Analyses are performed, based on the expected mode of operation of the Nuclear Steam Supply System (base loaded, maneuvering, etc.). From these analyses, CEA insertions are determined, and a consistent set of radial peaking factors are defined. The long term (steady state) and short term insertion limits are determined, based upon the assumed mode of operation used in the analyses; they provide a means of preserving the assumptions on CEA insertions used. The long and short term insertion limits of LCO 3.1.8 are specified for the plant, which has been designed primarily for base loaded operation, but has the ability to accommodate a limited amount of load maneuvering.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation (Condition I) and anticipated operational occurrences (Condition II). The regulating CEA insertion, part length CEA insertion, ASI, and T_q LCOs preclude core power distributions from occurring that would violate the following fuel design criteria:

or part strength

a. During a large break LOCA, the peak cladding temperature must not exceed 2200°F (Ref. 2);

b. During a loss of forced reactor coolant flow accident, there must be at least a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition;

c. During an ejected CEA accident, the fission energy input to the fuel must not exceed 280 cal/gm (Ref. 3); and

(continued)

or Part Strength

BASES

APPLICABLE SAFETY ANALYSES (continued)

d. The CEAs must be capable of shutting down the reactor with a minimum required SDM, with the highest worth CEA stuck fully withdrawn, GDC 26 (Ref. 1).

Regulating CEA position, part length CEA position, ASI, and T_q are process variables that together characterize and control the three dimensional power distribution of the reactor core.

Fuel cladding damage does not occur when the core is operated outside these LCOs during normal operation. However, fuel cladding damage could result, should an accident occur with simultaneous violation of one or more of these LCOs. Changes in the power distribution can cause increased power peaking and corresponding increased local LHRs.

or part strength

The part length CEA insertion limits satisfy Criterion 2 of 10 CFR 50.36 (c)(2)(ii). The part length CEAs are required due to the potential peaking factor violations that could occur if part length CEAs exceed insertion limits.

LCO

The limits on part length CEA insertion, as defined in the COLR, must be maintained because they serve the function of preserving power distribution.

APPLICABILITY

The part length insertion limits shall be maintained with the reactor in MODES 1 and 2. These limits must be maintained, since they preserve the assumed power distribution. Applicability in lower MODES is not required, since the power distribution assumptions would not be exceeded in these MODES.

(continued)

or Part Strength

BASES (continued)

ACTIONS

A.1, A.2 and B.1

If the part length CEA groups are inserted beyond the following limits ~~transient insertion limit or between the long term (steady state) insertion limit and the transient limit for 7 or more effective full power days (EFPD) out of any 30 EFPD period, or for 14 EFPD or more out of any 365 EFPD period,~~ flux patterns begin to develop that are outside the range assumed for long term fuel burnup. If allowed to continue beyond this limit, the peaking factors assumed as initial conditions in the accident analysis may be invalidated (Ref. 4). Restoring the CEAs to within limits or reducing THERMAL POWER to that fraction of RTP that is allowed by CEA group position, using the limits specified in the COLR, ensures that acceptable peaking factors are maintained.

1) Transient insertion limits;

2) Between the long term (steady-state) insertion limit and the transient insertion limit for:
a) 7 or more effective full power days (EFPD) out of any 30 EFPD period;
b) 14 EFPD or more out of any 365 EFPD period.

Since these effects are cumulative, actions are provided to limit the total time the part length CEAs can be out of limits in any 30 EFPD or 365 EFPD period. Since the cumulative out of limit times are in days, an additional Completion Time of 2 hours is reasonable for restoring the part length CEAs to within the allowed limits.

or part strength

C.1

When a Required Action cannot be completed within the required Completion Time, a controlled shutdown should commence. A Completion Time of 6 hours is reasonable, based on operating experience, for reaching Mode 3 from full power conditions in an orderly manner and without challenging plant systems.

BASES (continued)

or Part Strength

SURVEILLANCE REQUIREMENTS

SR 3.1.8.1

Verification of each part length CEA group position every 12 hours is sufficient to detect CEA positions that may approach the limits, and provide the operator with time to undertake the Required Action(s), should insertion limits be found to be exceeded. The 12 hour frequency also takes into account the indication provided by the power dependent insertion limit alarm circuit and other information about CEA group positions available to the operator in the control room.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10 and GDC 26.

2. 10 CFR 50.46.

3. Regulatory Guide 1.77, Rev. 0, May 1974.

4. UFSAR, Section 15.4.

BASES (continued)

ACTIONS

A.1

With any CEA not fully inserted and less than the minimum required reactivity equivalent available for insertion, or with all CEAs inserted and the reactor subcritical by less than the reactivity equivalent of the highest worth withdrawn CEA, restoration of the minimum shutdown reactivity requirements must be accomplished by increasing the RCS boron concentration. The required Completion Time of 15 minutes for initiating boration allows the operator sufficient time to align the valves and start the boric acid pumps and is consistent with the Completion Time of LCO 3.1.2.

In the determination of the required combination of boration flow rate and boron concentration, there is no unique requirement that must be satisfied. Since it is imperative to raise the boron concentration of the RCS as soon as possible, the boron concentration should be a highly concentrated solution, such as that normally found in the refueling water tank. The operator should borate with the best source available for the plant conditions.

In determining the boration flow rate the time in core life must be considered. For instance, the most difficult time in core life to increase the RCS boron concentration is at the beginning of cycle, when boron concentration may approach or exceed 2000 ppm. Assuming that a value of 1% $\Delta k/k$ must be recovered and a boration flow rate of 26 gpm, it is possible to increase the boron concentration of the RCS by 100 ppm in approximately 35 minutes with a 4000 ppm source. If a boron worth of 10 pcm/ppm is assumed, this combination of parameters will increase the SDM by 1% $\Delta k/k$. These boration parameters of 26 gpm and 4000 ppm represent typical values and are provided for the purpose of offering a specific example.

SURVEILLANCE REQUIREMENTS

SR 3.1.9.1

Verification of the position of each partially or fully withdrawn full ~~length~~ strength ~~or~~ part length or part strength CEA is necessary to ensure that the minimum negative reactivity requirements for insertion on a trip are preserved. A 2 hour Frequency is sufficient for the operator to verify that each CEA position is within the acceptance criteria.

(continued)

BASES

BACKGROUND (continued)

PHYSICS TESTS procedures are written and approved in accordance with established formats. The procedures include all information necessary to permit a detailed execution of testing required to ensure that design intent is met. PHYSICS TESTS are performed in accordance with these procedures and test results are approved prior to continued power escalation and long term power operation.

Examples of PHYSICS TESTS include determination of critical boron concentration, CEA group worth, reactivity coefficients, flux symmetry, and core power distribution.

APPLICABLE SAFETY ANALYSES

It is acceptable to suspend certain LCOs for PHYSICS TESTS because fuel damage criteria are not exceeded. Even if an accident occurs during PHYSICS TESTS with one or more LCOs suspended, fuel damage criteria are preserved because the limits on power distribution and shutdown capability are maintained during PHYSICS TESTS.

Reference 5 defines requirements for initial testing of the facility, including PHYSICS TESTS. Requirements for reload fuel cycle PHYSICS TESTS are defined in ANSI/ANS-19.6.1-1985 (Ref. 4). Although these PHYSICS TESTS are generally accomplished within the limits of all LCOs, conditions may occur when one or more LCOs must be suspended to make completion of PHYSICS TESTS possible or practical. This is acceptable as long as the fuel design criteria are not violated. As long as the linear heat rate (LHR) remains within its limit, fuel design criteria are preserved.

In this test, the following LCOs are suspended:

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 (F_{xy}^T)";
LCO 3.1.8, "Part Length or Part Strength Control Element Assembly (CEA) Insertion Limits";
LCO 3.2.2, "Planar Radial Peaking Factors";
LCO 3.2.3, "AZIMUTHAL POWER TILT (T_q)";
LCO 3.2.5, "AXIAL SHAPE INDEX (ASI)"; and
LCO 3.3.3, "Control Element Assembly Calculators (CEACs)".

(continued)

BASES

BACKGROUND (continued)

The PHYSICS TESTS requirements for reload fuel cycles ensure that the operating characteristics of the core are consistent with the design predictions and that the core can be operated as designed (Ref. 4).

PHYSICS TESTS procedures are written and approved in accordance with established formats. The procedures include all information necessary to permit a detailed execution of testing required to ensure that design intent is met. PHYSICS TESTS are performed in accordance with these procedures and test results are approved prior to continued power escalation and long term power operation.

Examples of PHYSICS TESTS include determination of critical boron concentration, CEA group worth, reactivity coefficients, flux symmetry, and core power distribution.

APPLICABLE SAFETY ANALYSES

It is acceptable to suspend certain LCOs for PHYSICS TESTS because fuel damage criteria are not exceeded. Even if an accident occurs during PHYSICS TESTS with one or more LCOs suspended, fuel damage criteria are preserved because the limits on power distribution and shutdown capability are maintained during PHYSICS TESTS.

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In this test, the following LCOs are suspended:

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).

(continued)

BASES

APPLICABLE SAFETY ANALYSIS (continued)

PHYSICS TESTS meet the criteria for inclusion in the Technical Specifications, since the component and process variable LCOs suspended during PHYSICS TESTS meet Criteria 1, 2, and 3 of 10 CFR 50.36 (c)(2)(ii).

or Part Strength

LCO

This LCO permits Part Length CEAs and Regulating CEAs to be positioned outside of their normal group heights and insertion limits, and RCS cold leg temperature to be outside its limits during the performance of PHYSICS TESTS. These PHYSICS TESTS are required to determine the isothermal temperature coefficient (ITC), MTC, and power coefficient.

The requirements of LCO 3.1.7, LCO 3.1.8, and LCO 3.4.1, (for RCS cold leg temperature only) may be suspended during the performance of PHYSICS TESTS provided COLSS is in service.

APPLICABILITY

This LCO is applicable in MODE 1 with THERMAL POWER > 20% RTP because the reactor must be critical at THERMAL POWER levels > 20% RTP to perform the PHYSICS TESTS described in the LCO section.

ACTIONS

A.1

With the LHR or DNBR outside the limits specified in the COLR, adequate safety margin is not assured and power must be reduced to restore LHR and DNBR to within limits. The required Completion Time of 15 minutes ensure prompt action is taken to restore LHR or DNBR to within limits.

(continued)

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 Linear Heat Rate (LHR)

BASES

BACKGROUND

The purpose of this LCO is to limit the core power distribution to the initial values assumed in the accident analyses. Operation within the limits imposed by this LCO limits or prevents potential fuel cladding failures that could breach the primary fission product barrier and release fission products to the reactor coolant in the event of a Loss Of Coolant Accident (LOCA), ejected Control Element Assembly (CEA) accident, or other postulated accident requiring termination by a Reactor Protection System (RPS) trip function. This LCO limits the damage to the fuel cladding during an accident by ensuring that the plant is operating within acceptable bounding conditions at the onset of a transient.

or part strength

Methods of controlling the power distribution include:

a. Using full strength ~~or~~ part length CEAs to alter the axial power distribution;

b. Decreasing CEA insertion by boration, thereby improving the radial power distribution; and

c. Correcting off optimum conditions (e.g., a CEA drop or misoperation of the unit) that cause margin degradations.

The core power distribution is controlled so that, in conjunction with other core operating parameters (e.g., CEA insertion and alignment limits), the power distribution does not result in violation of this LCO. The limiting safety system settings and this LCO are based on the accident analyses (Refs. 1 and 2), so that specified acceptable fuel design limits are not exceeded as a result of Anticipated Operational Occurrences (AOOs), and the limits of acceptable consequences are not exceeded for other postulated accidents.

Limiting power distribution skewing over time also minimizes xenon distribution skewing, which is a significant factor in controlling the axial power distribution.

(continued)

BACKGROUND (continued)

In addition to the monitoring performed by the COLSS, the RPS (via the CPCs) continually infers the core power distribution and thermal margins by processing reactor coolant data, signals from excore neutron flux detectors, and input from redundant reed switch assemblies that indicate CEA positions. In this case, the CPCs assume a minimum core power of 20% RTP because the power range excore neutron flux detecting system is inaccurate below this power level. If power distribution or other parameters are perturbed as a result of an AOO, the high LPD or low DNBR trips in the RPS initiate a reactor trip prior to exceeding fuel design limits.

The LHR and DNBR algorithms are valid within the limits on ASI, F_{xy} and T_q. These limits are obtained directly from initial core or reload analysis.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation or AOOs (Ref. 4).

The power distribution and CEA insertion and alignment LCOs prevent core power distributions from reaching levels that violate the following fuel design criteria:

a. During a LOCA, peak cladding temperature must not exceed 2200°F (Ref. 5);

b. During a loss of flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition (Ref. 4);

c. During an ejected CEA accident, the fission energy input to the fuel must not exceed 280 al/gm (Ref. 6); and

or part strength

d. The control rods (excluding part length rods) must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 7).

(continued)

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 Planar Radial Peaking Factors (F_{xy})

BASES

BACKGROUND

The purpose of this LCO is to limit the core power distribution to the initial values assumed in the accident analyses. Operation within the limits imposed by this LCO either limits or prevents potential fuel cladding failures that could breach the primary fission product barrier and release fission products to the reactor coolant in the event of a Loss Of Coolant Accident (LOCA), loss of flow accident, ejected Control Element Assembly (CEA) accident, or other postulated accident requiring termination by a Reactor Protection System (RPS) trip function. This LCO limits damage to the fuel cladding during an accident by ensuring that the plant is operating within acceptable conditions at the onset of a transient.

or part strength

Methods of controlling the power distribution include:

a. Using full strength ~~or~~ part length CEAs to alter the axial power distribution;

b. Decreasing CEA insertion by boration, thereby improving the radial power distribution; and

c. Correcting off optimum conditions (e.g., a CEA drop or misoperation of the unit) that cause margin degradations.

The core power distribution is controlled so that, in conjunction with other core operating parameters (CEA insertion and alignment limits), the power distribution does not result in violation of this LCO. Limiting safety system settings and this LCO are based on the accident analyses (Refs. 1 and 2), so that specified acceptable fuel design limits are not exceeded as a result of Anticipated Operational Occurrences (AOOs), and the limits of acceptable consequences are not exceeded for other postulated accidents.

Limiting power distribution skewing over time also minimizes xenon distribution skewing, which is a significant factor in controlling axial power distribution. Power distribution is a product of multiple parameters, various combinations of

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

b. During a loss of flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition (Ref. 4);

c. During an ejected CEA accident, the fission energy input to the fuel must not exceed 280 cal/gm (Ref. 6); and

or part strength

d. The control rods (excluding part length rods) must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 7).

The power density at any point in the core must be limited to maintain the fuel design criteria (Refs. 4 and 5). This result is accomplished by maintaining the power distribution and reactor coolant conditions so that the peak LHR and DNB parameters are within operating limits supported by the accident analyses (Ref. 1) with due regard for the correlations between measured quantities, the power distribution, and the uncertainties in the determination of power distribution.

Fuel cladding failure during a LOCA is limited by restricting the maximum Linear Heat Generation Rate (LHGR) so that the peak cladding temperature does not exceed 2200°F (Ref. 5). Peak cladding temperatures exceeding 2200°F cause severe cladding failure by oxidation due to a Zircaloy water reaction.

The LCOs governing LHR, ASI, CEAs, and RCS ensure that these criteria are met as long as the core is operated within the ASI and F_{xy} limits specified in the COLR, and within the T_q limits. The latter are process variables that characterize the three dimensional power distribution of the reactor core. Operation within the limits for these variables ensures that their actual values are within the ranges used in the accident analyses (Ref. 1).

Fuel cladding damage does not occur because of conditions outside the limits of these LCOs for ASI, F_{xy}, and T_q during normal operation. However, fuel cladding damage results if an accident occurs from initial conditions outside the limits of these LCOs. This potential for fuel cladding damage exists because changes in the power distribution can

(continued)

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 AZIMUTHAL POWER TILT (T_q)

BASES

BACKGROUND

The purpose of this LCO is to limit the core power distribution to the initial values assumed in the accident analyses. Operation within the limits imposed by this LCO either limits or prevents potential fuel cladding failures that could breach the primary fission product barrier and release fission products to the reactor coolant in the event of a Loss Of Coolant Accident (LOCA), loss of flow accident, ejected Control Element Assembly (CEA) accident, or other postulated accident requiring termination by a Reactor Protection System (RPS) trip function. This LCO limits the amount of damage to the fuel cladding during an accident by ensuring that the plant is operating within acceptable conditions at the onset of a transient.

or part strength

Methods of controlling the power distribution include:

a. Using full strength, ~~or~~ part length CEAs to alter the axial power distribution;

b. Decreasing CEA insertion by boration, thereby improving the radial power distribution; and

c. Correcting off optimum conditions, (e.g., a CEA drop or misoperation of the unit) that cause margin degradations.

The core power distribution is controlled so that, in conjunction with other core operating parameters (e.g., CEA insertion and alignment limits), the power distribution does not result in violation of this LCO. The limiting safety system settings and this LCO are based on the accident analyses (Refs. 1 and 2), so that specified acceptable fuel design limits are not exceeded as a result of Anticipated Operational Occurrences (AOOs) and the limits of acceptable consequences are not exceeded for other postulated accidents.

Limiting power distribution skewing over time also minimizes xenon distribution skewing, which is a significant factor in controlling axial power distribution.

(continued)

APPLICABLE SAFETY ANALYSES (continued)

b. During a loss of flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition (Ref. 4);

c. During a CEA ejection accident, the fission energy input to the fuel must not exceed 280 cal/gm (Ref. 6); and

or part strength

d. The control rods (excluding part length rods) must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 7).

The power density at any point in the core must be limited to maintain the fuel design criteria (Ref. 1). This result is accomplished by maintaining the power distribution and reactor coolant conditions so that the peak LHR and DNB parameters are within operating limits supported by the accident analysis (Ref. 2) with due regard for the correlations between measured quantities, the power distribution, and uncertainties in the determination of power distribution.

Fuel cladding failure during a LOCA is limited by restricting the maximum Linear Heat Generation Rate (LHGR) so that the peak cladding temperature does not exceed 2200°F (Ref. 1). Peak cladding temperatures exceeding 2200°F cause severe cladding failure by oxidation due to a Zircaloy water reaction.

The LCOs governing LHR, ASI, CEAs, and RCS ensure that these criteria are met as long as the core is operated within the ASI and F_{xy} limits specified in the COLR, and within the T_q limits. The latter are process variables that characterize the three dimensional power distribution of the reactor core. Operation within the limits of these variables ensures that their actual values are within the range used in the accident analyses (Ref. 1).

(continued)

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 Departure from Nucleate Boiling Ratio (DNBR)

BASES

BACKGROUND

The purpose of this LCO is to limit the core power distribution to the initial value assumed in the accident analyses. Specifically, operation within the limits imposed by this LCO either limits or prevents potential fuel cladding failures that could breach the primary fission product barrier and release fission products to the reactor coolant in the event of a Loss Of Coolant Accident (LOCA), loss of flow accident, ejected Control Element Assembly (CEA) accident, or other postulated accident requiring termination by a Reactor Protection System (RPS) trip function. This LCO limits the amount of damage to the fuel cladding during an accident by ensuring that the plant is operating within acceptable conditions at the onset of a transient.

Methods of controlling the power distribution include:

a. Using full strength, ~~or~~ part length or part strength CEAs to alter the axial power distribution;

b. Decreasing CEA insertion by boration, thereby improving the radial power distribution; and

c. Correcting off optimum conditions (e.g., a CEA drop or misoperation of the unit) that cause margin degradations.

The core power distribution is controlled so that, in conjunction with other core operating parameters (e.g., CEA insertion and alignment limits), the power distribution does not result in violation of this LCO. The limiting safety system settings and this LCO are based on the accident analysis (Refs. 1 and 2), so that specified acceptable fuel design limits are not exceeded as a result of Anticipated Operational Occurrences (AOOs) and the limits of acceptable consequences are not exceeded for other postulated accidents.

Limiting power distribution skewing over time also minimizes the xenon distribution skewing, which is a significant factor in controlling axial power distribution.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

b. During a loss of flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition (Ref. 4);

c. During an ejected CEA accident, the fission energy input to the fuel must not exceed 280 cal/gm (Ref. 6); and

or part strength

d. The control rods (excluding part length rods) must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 7).

The power density at any point in the core must be limited to maintain the fuel design criteria (Ref. 4). This is accomplished by maintaining the power distribution and reactor coolant conditions so that the peak LHR and DNB parameters are within operating limits supported by the accident analyses (Ref. 1) with due regard for the correlations between measured quantities, the power distribution, and uncertainties in the determination of power distribution.

Fuel cladding failure during a LOCA is limited by restricting the maximum Linear Heat Generation Rate (LHGR) so that the peak cladding temperature does not exceed 2200°F (Ref. 4). Peak cladding temperatures exceeding 2200°F may cause severe cladding failure by oxidation due to a Zircaloy water reaction.

The LCOs governing LHR, ASI, CEAs, and RCS ensure that these criteria are met as long as the core is operated within the ASI and F_{xy} limits specified in the COLR, and within the T_q limits. The latter are process variables that characterize the three dimensional power distribution of the reactor core. Operation within the limits for these variables ensures that their actual values are within the range used in the accident analyses (Ref. 1).

(continued)

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.5 AXIAL SHAPE INDEX (ASI)

BASES

BACKGROUND

The purpose of this LCO is to limit the core power distribution to the initial values assumed in the accident analysis. Operation within the limits imposed by this LCO either limits or prevents potential fuel cladding failures that could breach the primary fission product barrier and release fission products to the reactor coolant in the event of a Loss Of Coolant Accident (LOCA), loss of flow accident, ejected Control Element Assembly (CEA) accident, or other postulated accident requiring termination by a Reactor Protection System (RPS) trip function. This LCO limits the amount of damage to the fuel cladding during an accident by ensuring that the plant is operating within acceptable conditions at the onset of a transient.

or part strength

Methods of controlling the axial power distribution include:

a. Using full strength, ~~or~~ part length CEAs to alter the axial power distribution;

b. Decreasing CEA insertion by boration, thereby improving the axial power distribution; and

c. Correcting off optimum conditions (e.g., a CEA drop or misoperation of the unit) that cause margin degradations.

The core power distribution is controlled so that, in conjunction with other core operating parameters (CEA insertion and alignment limits), the power distribution does not result in violation of this LCO. The limiting safety system settings are based on the accident analyses (Refs. 1 and 2), so that specified acceptable fuel design limits are not exceeded as a result of Anticipated Operational Occurrences (AOOs) and the limits of acceptable consequences are not exceeded for other postulated accidents.

Limiting power distribution skewing over time also minimizes xenon distribution skewing, which is a significant factor in controlling axial power distribution.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

b. During a loss of flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition (Ref. 4);

c. During an ejected CEA accident, the fission energy input to the fuel must not exceed 280 cal/gm (Ref. 6);

d. The control rods (excluding part length or part strength rods) must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 7).

The power density at any point in the core must be limited to maintain the fuel design criteria (Refs. 4 and 5). This is accomplished by maintaining the power distribution and reactor coolant conditions so that the peak LHR and DNB parameters are within operating limits supported by the accident analyses (Ref. 1) with due regard for the correlations among measured quantities, the power distribution, and uncertainties in the determination of power distribution.

Fuel cladding failure during a LOCA is limited by restricting the maximum Linear Heat Generation Rate (LHGR) so that the peak cladding temperature does not exceed 2200°F (Ref. 5). Peak cladding temperatures exceeding 2200°F may cause severe cladding failure by oxidation due to a Zircaloy water reaction.

The LCOs governing LHR, ASI, and RCS ensure that these criteria are met as long as the core is operated within the ASI and F_{xy} limits specified in the COLR, and within the T_q limits. The latter are process variables that characterize the three dimensional power distribution of the reactor core. Operation within the limits for these variables ensures that their actual values are within the range used in the accident analysis (Ref. 1).

Fuel cladding damage does not occur from conditions outside these LCOs during normal operation. However, fuel cladding damage results when an accident occurs due to initial conditions outside the limits of these LCOs. This potential for fuel cladding damage exists because changes in the power distribution can cause increased power peaking and correspondingly increased local LHRs.

(continued)

BASES

ACTIONS

A.1

The ASI limits specified in the COLR ensure that the LOCA and loss of flow accident criteria assumed in the accident analyses remain valid. If the ASI exceeds its limit, a Completion Time of 2 hours is allowed to restore the ASI to within its specified limit. This duration gives the operator sufficient time to reposition the regulating or part length CEAs to reduce the axial power imbalance. The magnitude of any potential xenon oscillation is significantly reduced if the condition is not allowed to persist for more than 2 hours.

or part strength

B.1

If the ASI is not restored to within its specified limits within the required Completion Time, the reactor continues to operate with an axial power distribution mismatch. Continued operation in this configuration induces an axial xenon oscillation, and results in increased LHGRs when the xenon redistributes. Reducing thermal power to ≤ 20% RTP reduces the maximum LHR to a value that does not exceed the fuel design limits if a design basis event occurs. The allowed Completion Time of 4 hours is reasonable, based on operating experience, to reduce power in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.5.1

The ASI can be monitored by both the incore (COLSS) and excore (CPC) neutron detector systems. The COLSS provides the operator with an alarm if an ASI limit is approached.

Verification of the ASI every 12 hours ensures that the operator is aware of changes in the ASI as they develop. A 12 hour Frequency for this Surveillance is acceptable because the mechanisms that affect the ASI, such as xenon redistribution or CEA drive mechanism malfunctions, cause slow changes in the ASI, which can be discovered before the limits are exceeded.

(continued)

BASES

APPLICABLE SAFETY ANALYSES

Design Basis Definition (continued)

12, 13. Reactor Coolant Flow – Low

The Reactor Coolant Flow Steam Generator #1-Low and Reactor Coolant Flow Steam Generator #2-Low trips provide protection against an RCP Sheared Shaft Event. A trip is initiated when the pressure differential across the primary side of either steam generator decreases below a variable setpoint. This variable setpoint stays below the pressure differential by a preset value called the step function, unless limited by a preset maximum decreasing rate determined by the Ramp Function, or a set minimum value determined by the Floor Function. The setpoints ensure that a reactor trip occurs to limit fuel failure and ensure offsite doses are within 10 CFR 100 guidelines.

14. Local Power Density – High

The CPCs perform the calculations required to derive the DNBR and LPD parameters and their associated RPS trips. The DNBR – Low and LPD – High trips provide plant protection during the following AOOs and assist the ESF systems in the mitigation of the following accidents.

The LPD – High trip provides protection against fuel centerline melting due to the occurrence of excessive local power density peaks during the following AOOs:

- Decrease in Feedwater Temperature;
- Increase in Feedwater Flow;
- Increased Main Steam Flow (not due to the steam line rupture) Without Turbine Trip;
- Uncontrolled CEA Withdrawal From Low Power;
- Uncontrolled CEA Withdrawal at Power; and
- CEA Misoperation; Single Part Length CEA Drop. **(for Units that have part length CEAs)**

For the events listed above (except CEA Misoperation; Single Part Length CEA Drop), DNBR – Low will trip the reactor first, since DNB would occur before fuel centerline melting would occur.

(continued)

BASES

APPLICABLE SAFETY ANALYSES

Design Basis Definition (continued)

15. Departure from Nucleate Boiling Ratio (DNBR) – Low

The CPCs perform the calculations required to derive the DNBR and LPD parameters and their associated RPS trips. The DNBR – Low and LPD – High trips provide plant protection during the following AOOs and assist the ESF systems in the mitigation of the following accidents.

The DNBR – Low trip provides protection against core damage due to the occurrence of locally saturated conditions in the limiting (hot) channel during the following events and is the primary reactor trip (trips the reactor first) for these events:

- Decrease in Feedwater Temperature;
- Increase in Feedwater Flow;
- Increased Main Steam Flow (not due to steam line rupture) Without Turbine Trip;
- Increased Main Steam Flow (not due to steam line rupture) With a Concurrent Single Failure of an Active Component;
- Steam Line Break With Concurrent Loss of Offsite AC Power;
- Loss of Normal AC Power;
- Partial Loss of Forced Reactor Coolant Flow;
- Total Loss of Forced Reactor Coolant Flow;
- Single Reactor Coolant Pump (RCP) Shaft Seizure;
- Uncontrolled CEA Withdrawal From Low Power;
- Uncontrolled CEA Withdrawal at Power; or Part Strength
- CEA Misoperation; ~~Full Length~~ CEA ~~Drop~~;
- CEA Misoperation; Part Length CEA Subgroup Drop;
- Primary Sample or Instrument Line Break; and
- Steam Generator Tube Rupture.

In the above list, only the steam generator tube rupture, the RCP shaft seizure, and the sample or instrument line break are accidents. The rest are AOOs.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The transients analyzed for include loss of coolant flow events and dropped or stuck Control Element Assembly (CEA) events. A key assumption for the analysis of these events is that the core power distribution is within the limits of LCO 3.1.7, "Regulating CEA Insertion Limits"; LCO 3.1.8, "Part Length CEA Insertion Limits"; LCO 3.2.3, "AZIMUTHAL POWER TILT (T_q)"; and LCO 3.2.5, "AXIAL SHAPE INDEX (ASI).

or Part Strength

The RCS DNB limits satisfy Criterion 2 of 10 CFR 50.56(c)(2)(ii).

LCO

This LCO specifies limits on the monitored process variables – RCS pressurizer pressure, RCS cold leg temperature, and RCS total flow rate – to ensure that the core operates within the limits assumed for the plant safety analyses. Operating within these limits will result in meeting the DNBR criterion in the event of a DNB limited transient.

The LCO numerical value for minimum flow rate is given for the measurement location but has not been adjusted for instrument error. Plant specific limits of instrument error are established by the plant staff to meet the operational requirements of minimum flow rate.

APPLICABILITY

In MODE 1 for RCS flow rate, MODES 1 and 2 for RCS pressurizer pressure, Mode 1 for RCS cold leg temperature, and MODE 2 with $K_{eff} \geq 1$ for RCS cold leg temperature, the limits must be maintained during steady state operation in order to ensure that DNBR criteria will be met in the event of an unplanned loss of forced coolant flow or other DNB limited transient. In all other MODES, the power level is low enough so that DNBR is not a concern.

(continued)