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March 21, 2014

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

Subject: **Docket Nos. 50-361 and 50-362**
Amendment Applications 266 and 251
Permanently Defueled Technical Specifications
San Onofre Nuclear Generating Station, Units 2 and 3

References:

1. Letter from P. T. Dietrich (SCE) to the U. S. Nuclear Regulatory Commission (NRC) dated June 12, 2013; Subject: Certification of Permanent Cessation of Power Operations San Onofre Nuclear Generating Station, Units 2 and 3

2. Letter from P. T. Dietrich (SCE) to the U. S. Nuclear Regulatory Commission (NRC) dated June 28, 2013; Subject: Permanent Removal of Fuel from the Reactor Vessel, San Onofre Nuclear Generating Station Unit 3

3. Letter from P. T. Dietrich (SCE) to the U. S. Nuclear Regulatory Commission (NRC) dated July 22, 2013; Subject: Permanent Removal of Fuel from the Reactor Vessel, San Onofre Nuclear Generating Station Unit 2

4. Letter from P. T. Dietrich (SCE) to the U. S. Nuclear Regulatory Commission (NRC) dated October 21, 2013; Subject: Amendment Application Numbers 265 and 250, Responsibility, Organization, and Qualifications, San Onofre Nuclear Generating Station, Units 2 and 3

Dear Sir or Madam:

Pursuant to 10 CFR 50.90, Southern California Edison (SCE) hereby submits License Amendment Applications 266 and 251, to Facility Operating License Numbers NPF-10 and NPR-15 for San Onofre Nuclear Generating Station (SONGS) Units 2 and 3, respectively. The proposed amendment would revise the Operating License and revise the associated Technical Specifications (TS) to Permanently Defueled Technical Specifications (PDTS) to reflect the permanent cessation of reactor operation.

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On June 12, 2013, SCE submitted a certification to the NRC indicating its intention to permanently cease power operations at SONGS Units 2 and 3 (Reference 1) pursuant to 10 CFR 50.82(a)(1)(i). The certification stated that SCE had decided to permanently cease power operation of SONGS on June 7, 2013. With the docketing of the certification for permanent removal of fuel from the reactor vessel pursuant to 10 CFR 50.82(a)(1)(ii) on June 28, 2013 and July 22, 2013 (References 2 and 3), the 10 CFR Part 50 licenses for SONGS Units 2 and 3 no longer authorize operation of the reactor or emplacement or retention of fuel into the reactor vessel, as specified in 10 CFR 50.82(a)(2). In support of this condition, the SONGS Units 2 and 3 licenses and associated TS are being proposed for revision to conform to this permanently shut down and defueled condition in accordance with 10 CFR 50.36(c)(6).

The existing SONGS TS contain Limiting Conditions for Operation (LCOs) that provide for appropriate functional capability of equipment required for safe operation of the facility, including the plant being in a defueled condition. Since the safety function related to safe storage and management of irradiated fuel at an operating plant is similar to the corresponding function at a permanently defueled facility, the existing TS provide an appropriate level of control. However, the majority of the existing TS are only applicable with the reactor in an operational Mode. Because the SONGS Units 2 and 3 Part 50 licenses no longer authorize emplacement or retention of fuel in the reactor vessel, the LCOs (and associated Surveillance Requirements (SRs)) that do not apply in a defueled condition are being proposed for deletion. The remaining portions of the TS are being proposed for revision and incorporation as the PDTS to provide a continuing acceptable level of safety, which addresses the reduced scope of postulated design basis accidents associated with a defueled plant, as described in the SONGS Units 2 and 3 safety analyses.

There are two license amendment applications docketed for SONGS Units 2 and 3 currently under NRC review. On October 21, 2013, SCE submitted license amendment applications 265 and 250 for SONGS Units 2 and 3, respectively (Reference 4). The proposed TS 5.1, Responsibility, TS 5.2, Organization, and TS 5.3, Staff Qualifications, changes contained in the October 2013 letter are reflected in the TS markups of this license amendment application.

The Enclosure to this letter provides the Description and No Significant Hazards Consideration for the proposed amendments. SCE has determined that there is no significant hazards consideration associated with the proposed change and that the change is exempt from environmental review pursuant to the provisions of 10 CFR 51.22(c)(9).

SCE requests approval of these proposed license amendments to be made effective upon issuance and to be implemented within 60 days of approval.

In accordance with 10 CFR 50.91, SCE is notifying the State of California of this License Amendment Request by transmitting a copy of this letter and its enclosure to the designated State Official.

If you have any questions or require any additional information, please contact Mr. Mark Morgan at (949) 368-6745.

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

Executed on 3/21/2014
(Date)

Sincerely,

Enclosures:
PCN-605 with Attachments

cc: M. L. Dapas, Regional Administrator, NRC Region IV
M. H. Chernoff, NRC Licensing Project Manager, SONGS Units 2 & 3
R. E. Lantz, NRC Region IV San Onofre Units 2 and 3
G. G. Warnick, NRC Senior Resident Inspector, SONGS Units 2 and 3
S. Y. Hsu, California Department of Public Health, Radiologic Health Branch

ENCLOSURE

EVALUATION OF THE PROPOSED AMENDMENT

PCN-605

Permanently Defueled Technical Specifications

ATTACHMENTS:

1. Proposed Technical Specifications - Markup - Unit 2
2. Proposed Technical Specifications - Markup - Unit 3
3. Proposed Technical Specifications Bases Markup Pages, Unit 2 (For Information Only)
4. Proposed Technical Specifications Bases Markup Pages, Unit 3 (For Information Only)
5. Proposed Technical Specifications - Clean - Units 2 and 3
6. Proposed Technical Specifications Bases Pages - Clean, Units 2 and 3 (For Information Only)

1.0 SUMMARY DESCRIPTION

Southern California Edison (SCE) is proposing to revise the Facility Operating Licenses and associated Technical Specifications (TS) to Permanently Defueled Technical Specifications (PDTS) to reflect the permanent cessation of reactor operation.

2.0 DETAILED DESCRIPTION

2.1 Technical Specifications

The following table provides a summary of which SONGS Unit 2 and Unit 3 TS are being deleted in their entirety and which TS are being retained into the PDTS. The details and justification for the proposed changes follow in subsequent sections (arranged by TS Section).

TS Being Deleted	TS Being Retained
1.0 USE AND APPLICATION	
	1.1 Definitions
	1.2 Logical Connectors
	1.3 Completion Times
	1.4 Frequency
2.0 SAFETY LIMITS (SLs)	
2.1 Safety Limits (SLs)	
2.2 SL Violations	
3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY	
	3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY
	3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY
3.1 REACTIVITY CONTROL SYSTEMS	
3.1.1 SHUTDOWN MARGIN (SDM) – T_{avg} > 200°F	
3.1.2 SHUTDOWN MARGIN (SDM) – T_{avg} ≤ 200°F	
3.1.3 Reactivity Balance	
3.1.4 Moderator Temperature Coefficient (MTC)	
3.1.5 Control Element Assembly (CEA) Alignment	

TS Being Deleted	TS Being Retained
3.1.6 Shutdown Control Element Assembly (CEA) Insertion Limits	
3.1.7 Regulating CEA Insertion Limits	
3.1.8 Part Length Control Element Assembly (CEA) Insertion Limits	
3.1.9 Boration Systems – Operating	
3.1.10 Boration Systems – Shutdown	
3.1.11 Not Used	
3.1.12 Special Test Exception (STE) – Low Power Physics Testing	
3.1.13 Special Test Exception (STE) – At Power Physics Testing	
3.1.14 Special Test Exceptions (STE) – Reactivity Coefficient Testing	
3.2 POWER DISTRIBUTION LIMITS	
3.2.1 Linear Heat Rate (LHR)	
3.2.2 Planar Radial Peaking Factors (F_{xy})	
3.2.3 AZIMUTHAL POWER TILT (T_q)	
3.2.4 Departure From Nucleate Boiling Ratio (DNBR)	
3.2.5 AXIAL SHAPE INDEX (ASI)	
3.3 INSTRUMENTATION	
3.3.1 Reactor Protective System (RPS) Instrumentation – Operating	
3.3.2 Reactor Protective System (RPS) Instrumentation – Shutdown	
3.3.3 Control Element Assembly Calculators (CEACs)	
3.3.4 Reactor Protective System (RPS) Logic and Trip Initiation	
3.3.5 Engineered Safety Features Actuation System (ESFAS) Instrumentation	
3.3.6 Engineered Safety Features Actuation System (ESFAS) Logic and Manual Trip	
3.3.7 Diesel Generator (DG) – Undervoltage Start	

TS Being Deleted	TS Being Retained
3.3.8 Containment Purge Isolation Signal (CPIS)	
3.3.9 Control Room Isolation Signal (CRIS)	
3.3.10 Not Used	
3.3.11 Post Accident Monitoring Instrumentation (PAMI)	
3.3.12 Remote Shutdown System	
3.3.13 Source Range Monitoring Channels	
3.4 REACTOR COOLANT SYSTEM (RCS)	
3.4.1 RCS DNB Pressure, Temperature, and Flow Limits	
3.4.2 RCS Minimum Temperature for Criticality	
3.4.3 RCS Pressure and Temperature (P/T) Limits	
3.4.3.1 Pressurizer Heatup/Cooldown Limits	
3.4.4 RCS Loops – MODES 1 and 2	
3.4.5 RCS Loops – MODE 3	
3.4.6 RCS Loops – MODE 4	
3.4.7 RCS Loops – MODE 5, Loops Filled	
3.4.8 RCS Loops – MODE 5, Loops Not Filled	
3.4.9 Pressurizer	
3.4.10 Pressurizer Safety Valves	
3.4.11 Not Used	
3.4.12.1 Low Temperature Overpressure Protection System, RCS Temperature ≤ PTLR Limit	
3.4.12.2 Low Temperature Overpressure Protection System, RCS Temperature > PTLR Limit	
3.4.13 RCS Operational LEAKAGE	
3.4.14 RCS Pressure Isolation Valve (PIV) Leakage	

TS Being Deleted	TS Being Retained
3.4.15 RCS Leakage Detection Instrumentation	
3.4.16 RCS Specific Activity	
3.4.17 RCS Steam Generator (SG) Tube Integrity	
3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)	
3.5.1 Safety Injection Tanks (SITs)	
3.5.2 ECCS – Operating	
3.5.3 ECCS – Shutdown	
3.5.4 Refueling Water Storage Tank (RWST)	
3.5.5 Trisodium Phosphate (TSP) Dodecahydrate	
3.6 CONTAINMENT SYSTEMS	
3.6.1 Containment	
3.6.2 Containment Air Locks	
3.6.3 Containment Isolation Valves	
3.6.4 Containment Pressure	
3.6.5 Containment Air Temperature	
3.6.6.1 Containment Spray and Cooling Systems	
3.6.6.2 Containment Cooling System	
3.6.7 Not Used	
3.6.8 Containment Dome Air Circulators	
3.7 PLANT SYSTEMS	
3.7.1 Main Steam Safety Valves (MSSVs)	
3.7.2 Main Steam Isolation Valves (MSIVs)	
3.7.3 Main Feedwater Isolation Valves (MFIVs)	
3.7.4 Atmospheric Dump Valves (ADVs)	
3.7.5 Auxiliary Feedwater (AFW) System	
3.7.6 Condensate Storage Tank (CST T-120 and T-121)	

TS Being Deleted	TS Being Retained
3.7.7 Component Cooling Water (CCW) System	
3.7.7.1 Component Cooling Water (CCW) Safety Related Makeup System	
3.7.8 Salt Water Cooling (SWC) System	
3.7.9 Not Used	
3.7.10 Emergency Chilled Water (ECW)	
3.7.11 Control Room Emergency Air Cleanup System (CREACUS)	
3.7.12 Not Used	
3.7.13 Not Used	
3.7.14 Not Used	
3.7.15 Not Used	
	3.7.16 Fuel Storage Pool Water Level
	3.7.17 Fuel Storage Pool Boron Concentration
	3.7.18 Spent Fuel Assembly Storage
3.7.19 Secondary Specific Activity	
3.8 ELECTRICAL POWER SYSTEMS	
3.8.1 AC Sources – Operating	
3.8.2 AC Sources – Shutdown	
3.8.3 Diesel Fuel Oil, Lube Oil , and Starting Air	
3.8.4 DC Sources – Operating	
3.8.5 DC Sources – Shutdown	
3.8.6 Battery Parameters	
3.8.7 Inverters – Operating	
3.8.8 Inverters – Shutdown	
3.8.9 Distribution Systems – Operating	
3.8.10 Distribution Systems – Shutdown	
3.9 REFUELING OPERATIONS	
3.9.1 Boron Concentration	

TS Being Deleted	TS Being Retained
3.9.2 Nuclear Instrumentation	
3.9.3 Containment Penetrations	
3.9.4 Shutdown Cooling (SDC) and Coolant Circulation – High Water Level	
3.9.5 Shutdown Cooling (SDC) and Coolant Circulation – Low Water Level	
3.9.6 Refueling Water Level	
4.0 DESIGN FEATURES	
	4.1 Site
4.2 Reactor Core	
	4.3 Fuel Storage
5.0 ADMINISTRATIVE CONTROLS	
	5.1 Responsibility
	5.2 Organization
	5.3 Facility Staff Qualifications
	5.4 Technical Specifications (TS) Bases Control
	5.5 Procedures, Programs, and Manuals
5.6 Safety Function Determination Program (SFDP)	
	5.7 Reporting Requirements
	5.8 High Radiation Area

The TS Table of Contents is being revised accordingly.

The corresponding TS Bases are also being either deleted or revised (as applicable) to reflect these changes.

2.2 Facility Operating License

This section describes the proposed changes to the SONGS Unit 2 and Unit 3 Facility Operating Licenses and the justification for each change. Proposed deletions are indicated with text strikethrough. Proposed additions are italicized and underlined.

2.2.1 SONGS Unit 2

License Condition 1.B

B. ~~Construction of the San Onofre Nuclear Generating Station, Unit 2 (the facility), has been substantially completed in conformity with Construction Permit No. CPPR-97 and the application as amended, the provisions of the Act, and the regulations of the Commission;~~

This section is proposed for deletion in its entirety. Decommissioning of SONGS Unit 2 is not dependent on the regulations that governed construction of the facility.

This paragraph will read as follows:

B. Deleted;

License Condition 1.J

J. ~~The receipt, possession, and use of source, byproduct, and special nuclear material as authorized by this license will be in accordance with the Commission's regulations in 10 CFR Parts 30, 40 and 70.~~

This section is proposed for deletion in its entirety. The Commission's finding regarding possession and use of byproduct, source, and special nuclear material for operating SONGS Unit 2 is not dependent on decommissioning of the facility. Additionally, possession and use of byproduct, source, and special nuclear material at SONGS Unit 2 during decommissioning activities is covered by License Condition 2.B, which will remain in effect. Therefore, License Condition 1.J is not needed.

This paragraph will read as follows:

J. Deleted.

License Condition 2.B.(2)

(2) Southern California Edison Company (SCE), pursuant to Section 103 of the Act and 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities", to possess~~, use, and operate~~ the facility at the designated location in San Diego County, California, in accordance with the procedures and limitations set forth in this license;

The language regarding use and operation is proposed for deletion. The license no longer authorizes use and operation of the facility.

This paragraph will read as follows:

(2) Southern California Edison Company (SCE), pursuant to Section 103 of the Act and 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities", to possess the facility at the designated location in San Diego County, California, in accordance with the procedures and limitations set forth in this license;

License Condition 2.B.(3)

(3) SCE, pursuant to the Act and 10 CFR Part 70, to ~~receive,~~ possess~~, and use~~ at any time special nuclear material *that was used* as reactor fuel, in accordance with the limitations for storage ~~and amounts required for reactor operation~~, as described in the Final Safety Analysis Report, as supplemented and amended;

The language regarding receipt and use of special nuclear material as reactor fuel is proposed for deletion (and referring to use of reactor fuel in the past tense). The license no longer authorizes use and operation of the facility and only authorizes possession of the existing fuel.

This paragraph will read as follows:

(3) SCE, pursuant to the Act and 10 CFR Part 70, to possess at any time special nuclear material that was used as reactor fuel, in accordance with the limitations for storage, as described in the Final Safety Analysis Report, as supplemented and amended;

License Condition 2.B.(4)

(4) SCE, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as ~~sealed neutron sources for reactor startup,~~ sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required; *and possess any byproduct, source and special material as sealed neutron sources that was used for reactor startup;*

The language regarding receipt and use of sealed neutron sources for reactor startup is proposed for deletion. This license condition is revised to reflect authorization only for continued possession of those sources that were used for reactor startups. The license no longer authorizes use and operation of the facility and this condition will no longer authorize receipt and use of sources used for reactor startup.

This paragraph will read as follows:

(4) SCE, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required; and possess any byproduct, source and special material as sealed neutron sources that was used for reactor startup;

License Condition 2.C.(1)

(1) ~~Maximum Power Level~~

~~Southern California Edison Company (SCE) is authorized to operate the facility at reactor core power levels not in excess of full power (3438 megawatts thermal).~~

This section is proposed for deletion in its entirety. SONGS Unit 2 has permanently ceased power operation. 10 CFR 50.82(a)(2) prohibits operation of the SONGS Unit 2 reactor since the certifications described therein have been submitted.

This paragraph will read as follows:

(1) Deleted.

License Condition 2.C.(14)

(14) ~~Fire Protection (Section 9.5.1, SER, SSER #4, SSER #5, Section 1.12, SSER #5; SE dated November 15, 1982; Revision 1 to Updated Fire Hazards Analysis Evaluation dated June 29, 1988)~~

~~SCE shall implement and maintain in effect all provisions of the approved fire protection program. This program shall be (1) as described in the Updated Fire Hazards Analysis through Revision 3 as revised by letters to the NRC dated May 31, July 22, and November 20, 1987 and January 21, February 22, and April 21, 1988; and (2) as approved in the NRC staff's Safety Evaluation Report (SER) (NUREG-0712) dated February 1981; Supplements 4 and 5 to the SER, dated January 1982 and February 1982, respectively: and the safety evaluation dated November 15, 1982: as supplemented and amended by the Updated Fire Hazards Analysis Evaluation for San Onofre 2 and 3, Revision 1 dated June 29, 1988. SCE may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.~~

This License Condition is proposed for elimination consistent with the restriction of 10 CFR 50.82(a)(2) that SONGS Unit 2 is no longer authorized to operate or place fuel in the reactor vessel. This condition for making changes to the Fire Protection Program is no longer required to assure fire safety by maintaining the ability to achieve and maintain safe shutdown in the event of a fire.

License Condition 2.C.(14), which is based on maintaining an operational fire protection program, in accordance with 10 CFR 50.48, with the ability to achieve and maintain safe shutdown of the reactor in the event of a fire, is no longer applicable for SONGS Unit 2. However, many of the elements that are applicable for the operating plant fire protection program continue to be applicable during plant decommissioning. During the decommissioning process, a fire protection program is required by 10 CFR 50.48(f) to address the potential for fires that could result in a radiological hazard. However, the regulation is applicable regardless of whether a requirement for a fire protection program is included in the facility license. Therefore, a license condition requiring such a program for a permanently shut down and defueled plant is not required.

This paragraph will read as follows:

(14) Deleted.

License Condition 2.C.(26)

(26) *Loss of Spent Fuel Pool Inventory* Mitigation Strategy ~~License Condition~~

While spent fuel is stored in the spent fuel pool, ~~D~~*d*evelop and maintain strategies for addressing ~~large fires and explosions and~~ *losses of spent fuel pool inventory* that

include the following key areas:

(a) ~~Fire fighting response strategy with the following elements:~~
~~1. Pre-defined coordinated fire response strategy and guidance~~
~~2. Assessment of mutual aid fire fighting assets~~
~~3. Designated staging areas for equipment and materials~~
~~4. Command and control~~
~~5. Training of response personnel~~

(b) ~~Operations to mitigate fuel damage considering the following:~~
1. Protection and use of personnel assets
2. Communications
~~3. Minimizing fire spread~~
~~4~~*3*. Procedures for implementing *mitigating strategies* ~~integrated fire response strategy~~
~~5~~*4*. Identification of readily-available pre-staged equipment
~~6. Training on integrated fire response strategy~~
~~7. Spent fuel pool mitigation measures~~

(c) ~~Actions to minimize release to include consideration of:~~
~~1. Water spray scrubbing~~
~~2. Dose to onsite responders~~

[SCE] may make changes to the mitigation strategies without prior approval of the Commission only if those changes would not have a significant adverse effect on the ability to respond to substantial losses in spent fuel pool inventory.

The NRC issued this license condition on July 26, 2007, to incorporate the requirements for the Interim Compensatory Measures (ICM) Order EA-02-026, Section B.5.b mitigation strategies (dated February 25, 2002). Subsequently, 10 CFR 50.54(hh)(2) became effective on May 26, 2009. This section provides mitigation strategies and response procedure requirements for loss of large areas of the plant due to explosions or fire. However, as stated in 10 CFR 50.54(hh)(3), this section does not apply to a defueled reactor that has submitted the certification for permanent removal of fuel under 10 CFR 50.82(a). On November 28, 2011, the NRC issued a letter to rescinded Item B.5.b of the ICM Order EA-02-26. Therefore, neither the ICM Order nor 10 CFR 50.54(hh) continue to apply to SONGS Unit 2.

Although the License Condition is no longer applicable to SONGS Unit 2, the language is proposed for revision to reflect mitigation strategies focused on maintaining spent fuel pool inventory.

This paragraph will read as follows:

(26) Loss of Spent Fuel Pool Inventory Mitigation Strategy

While spent fuel is stored in the spent fuel pool, develop and maintain strategies for addressing losses of spent fuel pool inventory that include the following key areas:

1. Protection and use of personnel assets
2. Communications

3. Procedures for implementing mitigating strategies
4. Identification of readily-available pre-staged equipment

[SCE] may make changes to the mitigation strategies without prior approval of the Commission only if those changes would not have a significant adverse effect on the ability to respond to substantial losses in spent fuel pool inventory.

License Condition 2.C.(27)

~~(27) Upon implementation of Amendment No. 214 adopting TSTF-448, Revision 3, the determination of control room envelope (CRE) unfiltered air inleakage as required by SR 3.7.11.4, in accordance with TS 5.5.2.16.c(i), the assessment of CRE habitability as required by Specification 5.5.2.16.c(ii), and the measurement of CRE pressure as required by Specification 5.5.2.16.d, shall be considered met. Following implementation:~~

~~(a) The first performance of SR 3.7.11.4, in accordance with Specification 5.5.2.16.c(i), shall be within the specified frequency of 6 years, plus the 18-month allowance of SR 3.0.2, as measured from May 18, 2004, the date of the most recent successful tracer gas test, as stated in the September 17, 2004 letter response to Generic Letter 2003-01, or within the next 18 months if the time period since the most recent successful trace gas test is greater than 6 years.~~

~~(b) The first performance of the periodic assessment of CRE habitability, Specification 5.5.2.16.c(ii), shall be within 3 years, plus the 9-month allowance of SR 3.0.2, as measured from May 18, 2004, the date of the most recent successful tracer gas test, as stated in the September 17, 2004, letter response to Generic Letter 2003-01, or within the next 9 month if the time period since the most recent successful tracer gas is greater than 3 years.~~

~~(c) The first performance of the periodic measurement of CRE pressure, Specification 5.5.2.16.d, shall be within 6 months.~~

This section is proposed for deletion in its entirety. SONGS Unit 2 has permanently ceased power operation. 10 CFR 50.82(a)(2) prohibits operation of the SONGS Unit 2 reactor since the certifications described therein have been submitted.

This paragraph will read as follows:

(27) Deleted.

License Condition 2.J

J. ~~This license is effective as of the date of issuance and shall expire at midnight on February 16, 2022.~~

This section is proposed for deletion in its entirety. SONGS Unit 2 has permanently ceased operation prior to the period of extended operation. 10 CFR 50.82(a)(2) prohibits operation of the SONGS Unit 2 reactor since the certifications described therein have been docketed. Therefore, SONGS Unit 2 will not operate during the period of extended

operation. This License Condition is being replaced by new License Condition 3, which conforms to 10 CFR 50.51 in that the possession-only license authorizes ownership and possession of SONGS Unit 2 until the Commission notifies the licensee in writing that the license is terminated.

This paragraph will read as follows:

J. Deleted.

License Condition 3

A new License Condition 3 is being proposed to address the permanently defueled possession-only status of the facility and replace existing License Condition 2.J so as to conform to 10 CFR 50.51 in that the possession-only license authorizes ownership and possession of SONGS until the Commission notifies the licensee in writing that the license is terminated.

This new License Condition will read as follows for SONGS Unit 2:

3. On June 12, 2013, Southern California Edison (SCE) certified that operations at San Onofre Nuclear Generating Station Unit 2 would permanently cease in accordance with 10 CFR 50.82(a)(1)(i). On July 22, 2013, SCE certified that the fuel had been permanently removed from the reactor vessel in accordance with 10 CFR 50.82(a)(1)(ii). As a result, the 10 CFR 50 license no longer authorizes operation of the reactor, or the emplacement or retention of fuel in the reactor vessel.

   This license is effective as of the date of issuance and authorizes ownership and possession of San Onofre Nuclear Generating Station Unit 2 until the Commission notifies the licensee in writing that the license is terminated. The licensee shall:

   A. Take actions necessary to decommission the plant and continue to maintain the facility, including, where applicable, the storage, control and maintenance of the spent fuel, in a safe condition; and

   B. Conduct activities in accordance with all other restrictions applicable to the facility in accordance with the NRC regulations and the applicable provisions of the 10 CFR 50 facility license as defined in Section 2 of this license.

2.2.2 SONGS Unit 3

License Condition 1.B

B. ~~Construction of the San Onofre Nuclear Generating Station, Unit 3 (the facility), has been substantially completed in conformity with Construction Permit No. CPPR-98 and the application as amended, the provisions of the Act, and the regulations of the Commission;~~

This section is proposed for deletion in its entirety. Decommissioning of SONGS Unit 3 is not dependent on the regulations that governed construction of the facility.

This paragraph will read as follows:

B. Deleted;

License Condition 1.I

I. ~~The receipt, possession, and use of source, byproduct, and special nuclear material as authorized by this license will be in accordance with the Commission's regulations in 10 CFR Parts 30, 40 and 70.~~

This section is proposed for deletion in its entirety. The Commission's finding regarding possession and use of byproduct, source, and special nuclear material for operating SONGS Unit 3 is not dependent on decommissioning of the facility. Additionally, possession and use of byproduct, source, and special nuclear material at SONGS Unit 3 during decommissioning activities is covered by License Condition 2.B, which will remain in effect. Therefore, License Condition 1.I is not needed.

This paragraph will read as follows:

I. Deleted.

License Condition 2.B.(2)

(2) Southern California Edison Company (SCE), pursuant to Section 103 of the Act and 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities", to possess~~, use, and operate~~ the facility at the designated location in San Diego County, California, in accordance with the procedures and limitations set forth in this license~~.~~;

The language regarding use and operation is proposed for deletion. The license no longer authorizes use and operation of the facility.

This paragraph will read as follows:

(2) Southern California Edison Company (SCE), pursuant to Section 103 of the Act and 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities", to possess the facility at the designated location in San Diego County, California, in accordance with the procedures and limitations set forth in this license;

License Condition 2.B.(3)

(3) SCE, pursuant to the Act and 10 CFR Part 70, to ~~receive,~~ possess~~, and use~~ at any time special nuclear material *that was used* as reactor fuel, in accordance with the limitations for storage ~~and amounts required for reactor operation~~, as described in the Final Safety Analysis Report, as supplemented and amended;

The language regarding receipt and use of special nuclear material as reactor fuel is proposed for deletion (and referring to use of reactor fuel in the past tense). The license no longer authorizes use and operation of the facility and only authorizes possession of the existing fuel.

This paragraph will read as follows:

(3) SCE, pursuant to the Act and 10 CFR Part 70, to possess at any time special nuclear material that was used as reactor fuel, in accordance with the limitations for storage, as described in the Final Safety Analysis Report, as supplemented and amended;

License Condition 2.B.(4)

(4) SCE, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as ~~sealed neutron sources for reactor startup,~~ sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required; *and possess any byproduct, source and special material as sealed neutron sources that was used for reactor startup;*

The language regarding receipt and use of sealed neutron sources for reactor startup is proposed for deletion. This license condition is revised to reflect authorization only for continued possession of those sources that were used for reactor startups. The license no longer authorizes use and operation of the facility and this condition will no longer authorize receipt and use of sources used for reactor startup.

This paragraph will read as follows:

(4) SCE, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required; and possess any byproduct, source and special material as sealed neutron sources that was used for reactor startup;

License Condition 2.C.(1)

(1) ~~Maximum Power Level~~

~~Southern California Edison Company (SCE) is authorized to operate the facility at reactor core power levels not in excess of full power (3438 megawatts thermal).~~

This section is proposed for deletion in its entirety. SONGS Unit 3 has permanently ceased power operation. 10 CFR 50.82(a)(2) prohibits operation of the SONGS Unit 3 reactor since the certifications described therein have been submitted.

This paragraph will read as follows:

(1) Deleted.

License Condition 2.C.(12)

(12) ~~Fire Protection (Section 9.5.1, SER, SSER #4, SSER #5, Section 1.12, SSER #5; SE dated November 15, 1982; Revision 1 to Updated Fire Hazards Analysis Evaluation dated June 29, 1988)~~

~~SCE shall implement and maintain in effect all provisions of the approved fire protection program. This program shall be (1) as described in the Updated Fire~~

~~Hazards Analysis through Revision 3 as revised by letters to the NRC dated May 31, July 22, and November 20, 1987 and January 21, February 22, and April 21, 1988; and (2) as approved in the NRC staff's Safety Evaluation Report (SER) (NUREG-0712) dated February 1981; Supplements 4 and 5 to the SER, dated January 1982 and February 1982, respectively: and the safety evaluation dated November 15, 1982: as supplemented and amended by the Updated Fire Hazards Analysis Evaluation for San Onofre 2 and 3, Revision 1 dated June 29, 1988. SCE may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.~~

This License Condition is proposed for elimination consistent with the restriction of 10 CFR 50.82(a)(2) that SONGS Unit 3 is no longer authorized to operate or place fuel in the reactor vessel. This condition for making changes to the Fire Protection Program is no longer required to assure fire safety by maintaining the ability to achieve and maintain safe shutdown in the event of a fire.

License Condition 2.C.(12), which is based on maintaining an operational fire protection program, in accordance with 10 CFR 50.48, with the ability to achieve and maintain safe shutdown of the reactor in the event of a fire, is no longer applicable for SONGS Unit 3. However, many of the elements that are applicable for the operating plant fire protection program continue to be applicable during plant decommissioning. During the decommissioning process, a fire protection program is required by 10 CFR 50.48(f) to address the potential for fires that could result in a radiological hazard. However, the regulation is applicable regardless of whether a requirement for a fire protection program is included in the facility license. Therefore, a license condition requiring such a program for a permanently shut down and defueled plant is not required.

This paragraph will read as follows:

(12) Deleted.

License Condition 2.C.(27)

(27) *Loss of Spent Fuel Pool Inventory* Mitigation Strategy ~~License Condition~~

While spent fuel is stored in the spent fuel pool, ~~D~~*d*evelop and maintain strategies for addressing ~~large fires and explosions and~~ *losses of spent fuel pool inventory* that include the following key areas:

~~(a) Fire fighting response strategy with the following elements:~~
~~1. Pre-defined coordinated fire response strategy and guidance~~
~~2. Assessment of mutual aid fire fighting assets~~
~~3. Designated staging areas for equipment and materials~~
~~4. Command and control~~
~~5. Training of response personnel~~

~~(b) Operations to mitigate fuel damage considering the following:~~
1. Protection and use of personnel assets
2. Communications
~~3. Minimizing fire spread~~

~~4~~*3*. Procedures for implementing *mitigating strategies* ~~integrated fire response strategy~~
~~5~~*4*. Identification of readily-available pre-staged equipment
~~6. Training on integrated fire response strategy~~
~~7. Spent fuel pool mitigation measures~~

~~(c) Actions to minimize release to include consideration of:~~
~~1. Water spray scrubbing~~
~~2. Dose to onsite responders~~

SCE may make changes to the mitigation strategies without prior approval of the Commission only if those changes would not have a significant adverse effect on the ability to respond to substantial losses in spent fuel pool inventory.

The NRC issued this license condition on July 26, 2007, to incorporate the requirements for the Interim Compensatory Measures (ICM) Order EA-02-026, Section B.5.b mitigation strategies (dated February 25, 2002). Subsequently, 10 CFR 50.54(hh)(2) became effective on May 26, 2009. This section provides mitigation strategies and response procedure requirements for loss of large areas of the plant due to explosions or fire. However, as stated in 10 CFR 50.54(hh)(3), this section does not apply to a defueled reactor that has submitted the certification for permanent removal of fuel under 10 CFR 50.82(a). On November 28, 2011, the NRC issued a letter to rescinded Item B.5.b of the ICM Order EA-02-26. Therefore, neither the ICM Order nor 10 CFR 50.54(hh) continue to apply to SONGS Unit 3.

Although the License Condition is no longer applicable to SONGS Unit 3, the language is proposed for revision to reflect mitigation strategies focused on maintaining spent fuel pool inventory.

This paragraph will read as follows:

(27) Loss of Spent Fuel Pool Inventory Mitigation Strategy

While spent fuel is stored in the spent fuel pool, develop and maintain strategies for addressing losses of spent fuel pool inventory that include the following key areas:

1. Protection and use of personnel assets
2. Communications
3. Procedures for implementing mitigating strategies
4. Identification of readily-available pre-staged equipment

SCE may make changes to the mitigation strategies without prior approval of the Commission only if those changes would not have a significant adverse effect on the ability to respond to substantial losses in spent fuel pool inventory.

License Condition 2.C.(28)

~~(28) Upon implementation of Amendment No. 206 adopting TSTF-448, Revision 3, the determination of control room envelope (CRE) unfiltered air inleakage as required by SR 3.7.11.4, in accordance with TS 5.5.2.16.c(i), the assessment of CRE habitability as required by Specification 5.5.2.16.c(ii), and the measurement of CRE pressure as~~

~~required by Specification 5.5.2.16.d, shall be considered met. Following implementation:~~

~~(a) The first performance of SR 3.7.11.4, in accordance with Specification 5.5.2.16.c(i), shall be within the specified frequency of 6 years, plus the 18-month allowance of SR 3.0.2, as measured from May 18, 2004, the date of the most recent successful tracer gas test, as stated in the September 17, 2004 letter response to Generic Letter 2003-01, or within the next 18 months if the time period since the most recent successful trace gas test is greater than 6 years.~~

~~(b) The first performance of the periodic assessment of CRE habitability, Specification 5.5.2.16.c(ii), shall be within 3 years, plus the 9-month allowance of SR 3.0.2, as measured from May 18, 2004, the date of the most recent successful tracer gas test, as stated in the September 17, 2004, letter response to Generic Letter 2003-01, or within the next 9 month if the time period since the most recent successful tracer gas is greater than 3 years.~~

~~(c) The first performance of the periodic measurement of CRE pressure, Specification 5.5.2.16.d, shall be within 6 months.~~

This section is proposed for deletion in its entirety. SONGS Unit 3 has permanently ceased power operation. 10 CFR 50.82(a)(2) prohibits operation of the SONGS Unit 3 reactor since the certifications described therein have been submitted.
This paragraph will read as follows:

(28) Deleted.

License Condition 2.J

J. ~~This license is effective as of the date of issuance and shall expire at midnight on November 15, 2022.~~

This section is proposed for deletion in its entirety. SONGS Unit 3 has permanently ceased operation prior to the period of extended operation. 10 CFR 50.82(a)(2) prohibits operation of the SONGS Unit 3 reactor since the certifications described therein have been docketed. Therefore, SONGS Unit 3 will not operate during the period of extended operation. This License Condition is being replaced by new License Condition 3, which conforms to 10 CFR 50.51 in that the possession-only license authorizes ownership and possession of SONGS Unit 3 until the Commission notifies the licensee in writing that the license is terminated.

This paragraph will read as follows:

J. Deleted.

License Condition 3

A new License Condition 3 is being proposed to address the permanently defueled possession-only status of the facility and replace existing License Condition 2.J so as to conform to 10 CFR 50.51 in that the possession-only license authorizes ownership and

possession of SONGS until the Commission notifies the licensee in writing that the license is terminated.

This new License Condition will read as follows for SONGS Unit 3:

3. On June 12, 2013, Southern California Edison (SCE) certified that operations at San Onofre Nuclear Generating Station Unit 3 would permanently cease in accordance with 10 CFR 50.82(a)(1)(i). On June 28, 2013, SCE certified that the fuel had been permanently removed from the reactor vessel in accordance with 10 CFR 50.82(a)(1)(ii). As a result, the 10 CFR 50 license no longer authorizes operation of the reactor, or the emplacement or retention of fuel in the reactor vessel.

   This license is effective as of the date of issuance and authorizes ownership and possession of San Onofre Nuclear Generating Station Unit 3 until the Commission notifies the licensee in writing that the license is terminated. The licensee shall:

   A. Take actions necessary to decommission the plant and continue to maintain the facility, including, where applicable, the storage, control and maintenance of the spent fuel, in a safe condition; and

   B. Conduct activities in accordance with all other restrictions applicable to the facility in accordance with the NRC regulations and the applicable provisions of the 10 CFR 50 facility license as defined in Section 2 of this license.

3.0 TECHNICAL EVALUATION

3.1 Background

On June 12, 2013, SCE submitted a certification to the NRC indicating its intention to permanently cease power operations at SONGS Units 2 and 3 (Reference 1) pursuant to 10 CFR 50.82(a)(1)(i). The certification stated that SCE had decided to permanently cease power operation of SONGS on June 7, 2013. With the docketing of the certification for permanent removal of fuel from the reactor vessel pursuant to 10 CFR 50.82(a)(1)(ii) on June 28, 2013 and July 22, 2013 (References 5 and 6), the 10 CFR Part 50 licenses for SONGS Units 2 and 3 no longer authorize operation of the reactor or emplacement or retention of fuel into the reactor vessel, as specified in 10 CFR 50.82(a)(2). In support of this condition, the SONGS Units 2 and 3 licenses and associated TS are being proposed for revision to conform to this permanently shut down and defueled condition in accordance with 10 CFR 50.36(c)(6).

The existing SONGS TS contain Limiting Conditions for Operation (LCOs) that provide for appropriate functional capability of equipment required for safe operation of the facility, including the plant being in a defueled condition. Since the safety function related to safe storage and management of irradiated fuel at an operating plant is similar to the corresponding function at a permanently defueled facility, the existing TS provide an appropriate level of control. However, the majority of the existing TS are only applicable with the reactor in an operational MODE. Because the SONGS Units 2 and 3 Part 50 licenses no longer authorize emplacement or retention of fuel in the reactor vessel, the LCOs (and associated Surveillance Requirements (SRs)) that do not apply in a defueled condition are being proposed for deletion. The remaining portions of the TS are being

proposed for revision and incorporation as the PDTS to provide a continuing acceptable level of safety which addresses the reduced scope of postulated design basis accidents associated with a defueled plant, as described in the SONGS Units 2 and 3 safety analyses.

There is one other license amendment docketed for SONGS Units 2 and 3 currently under NRC review. On October 21, 2013, SCE submitted license amendment applications 265 and 250 for SONGS Units 2 and 3, respectively. The proposed TS 5.1, Responsibility, TS 5.2, Organization, and TS 5.3, Staff Qualifications, changes contained in the October 2013 letter are reflected in the TS markups of this license amendment application.

The proposed amendment would modify the SONGS Units 2 and 3 licenses to conform to a permanently defueled condition and revise SONGS Unit 2 and Unit 3 Technical Specifications (TS) into a combined Permanently Defueled Technical Specifications (PDTS) that applies to both units.

3.1.1 General Analysis Applicable to Proposed Change

SONGS Units 2 and 3 have permanently ceased operation and removed all nuclear fuel from their reactor vessels. The irradiated fuel will be stored in the spent fuel pool (SFP) and in the Independent Spent Fuel Storage Installation (ISFSI) until it is shipped offsite. In this configuration, the SFP and its systems are dedicated only to spent fuel storage. In this condition, the number of credible accidents/transients is significantly smaller than for a plant authorized to operate the reactor or emplace or retain fuel in the reactor vessel.

With irradiated fuel being stored in the SFP and the ISFSI, the reactor, Reactor Coolant System (RCS) and secondary system are no longer in operation and have no function related to storage of irradiated fuel. With the permanent cessation of power operation and the permanent removal of the fuel from the reactor core, the accident/transient initial conditions/initial reactor power level of the reactor core cannot be achieved and, as such, most of the accident/transient scenarios are not possible. Therefore, the postulated UFSAR Chapter 15 accidents/transients involving failure or malfunction of the reactor, RCS or secondary system are no longer applicable.

The irradiated fuel will be stored in the fuel storage pool and in the Independent Spent Fuel Storage Installation (ISFSI) until it is shipped off site. During decommissioning, the fuel storage pool and its systems will be isolated and dedicated only to spent fuel storage. In this condition the spectrum of credible accidents is much smaller than for an operational plant.

The UFSAR Chapter 15 DBAs accident scenarios that apply to a permanently defueled facility that have the potential to result in a radiological release are a fuel handling accident (FHA) in the fuel handling building (FHB), a spent fuel pool boiling accident, a liquid Radioactive Waste System leak or failure, a radioactive release due to liquid tank failures, or an accidental release of waste gas. (Other UFSAR Chapter 15 postulated accidents with the potential to damage spent fuel are bounded by the postulated fuel handling accident inside the fuel handling building.) Since the waste gas decay tanks have been purged of their contents, a rupture of these components will no longer be an applicable initiator or source of such an accident. With regard to the postulated radioactive release due to liquid tank failures, UFSAR Section 15.7.3.3.5 states no credible accident exists that would result in liquid releases exceeding 10 CFR 20 limits.

The remaining accident analyses for SONGS show that the dose consequences are acceptable without relying on structures, systems, and components remaining functional for accident mitigation during and following the event. (The one exception to this is the continued function of the passive fuel storage pool structure).

The definition of safety-related structures, systems, and components (SSCs) in 10 CFR 50.2, "Definitions," states that safety-related SSCs are those relied on to remain functional during and following design basis events to assure:

1. The integrity of the reactor coolant boundary;
2. The capability to shut down the reactor and maintain it in a safe shutdown condition; or,
3. The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the applicable guideline exposures set forth in 10 CFR 50.43(a)(1) or 100.11.

The first two criteria (integrity of the reactor coolant pressure boundary and safe shutdown of the reactor) are not applicable to a plant in a permanently defueled condition. The third criterion is related to preventing or mitigating the consequences of accidents that could result in potential offsite exposures exceeding limits. However, after the termination of reactor operations at SONGS and the permanent removal of the fuel from the reactor vessel (following 17 months of decay time after shut down), none of the SSCs at SONGS are required to be relied upon for accident mitigation. Therefore, none of the SSCs at SONGS meet the definition of a safety-related SSC stated in 10 CFR 50.2 (with the exception of the passive fuel storage pool structure).

10 CFR 50.36, Technical specifications, provides the regulatory requirements related to the content of Technical Specifications. As detailed in subsequent sections of this proposed amendment, this regulation lists four criteria to define the scope of equipment and parameters that must be included in TS. In a permanently defueled condition, the scope of equipment and parameters that must be included in the SONGS TS is limited to those needed to address the remaining applicable design basis accidents, so that the consequences of the accident are maintained within acceptable limits.

3.1.2 Radioactive Waste System Leak or Failure (Release to Atmosphere)

UFSAR Section 15.7.3.2 discusses the radiological consequences for a liquid Radioactive Waste System leak or failure. Liquid releases considered include rupture of radwaste tanks, refueling water storage tanks, primary ion-exchangers, and the blowdown demineralizer neutralization sump line. The most limiting of these is defined as an unexpected and uncontrolled release of the radioactive liquid stored in a radwaste secondary tank. The radwaste secondary tanks are Seismic Category II, Quality Class III tanks at atmospheric pressure. Rupture of these tanks is considered a limiting fault. A radwaste secondary tank rupture would release the liquid contents in the auxiliary building (radwaste area). All of the radioactive fission gases and iodines are assumed to be released to the outside atmosphere in 2 hours. Offsite doses due to the rupture of a radwaste secondary tank are less than the 100 mRem TEDE offsite dose criterion per Regulatory Issue Summary 2006-04.

DOSE RECEPTOR	DOSE (mRem TEDE)	ACCEPTANCE CRITERION (mREM TEDE)
EAB (Maximum 2-hour dose -- 0.0 to 2.0 hours)	7.1	100
LPZ (30-day accident duration)	1.4	100

Table 1 – Radiological Exposures as a Result of Liquid Tank Rupture (Release to Atmosphere)

3.1.3 Spent Fuel Cask Drop Accident

UFSAR Section 15.7.3.5 analyzes spent fuel cask drop events. Of the three situations considered, a spent fuel transfer cask drop (due to a seismic event) from the upper shelf in the cask pool back into the lower portion of the cask pool is the only credible event with the potential for radiological release. Even though single-failure-proof cranes are used to lift a spent fuel transfer cask out of a cask pool, a drop can be postulated when the cask is placed on the upper shelf (i.e., step) of a cask pool for lifting yoke change-out, prior to the transfer cask being welded closed. During this evolution, the transfer cask is not restrained and could fall back into the lower portion of the cask pool if an earthquake occurs.

It is assumed that a minimum of 17 months have elapsed since permanent discharge from the core for Unit 2 or 3 fuel assemblies that are loaded into a transfer cask. The fuel rods from all 32 fuel assemblies that may be present in a transfer cask are conservatively assumed to rupture on impact with the bottom of the cask pool. All of the radioactive iodine and noble gases present in the gap volumes of the decayed fuel rods are assumed to be released from the unwelded transfer cask.

No engineered safety feature (ESF) system is used to mitigate the Control Room, Exclusion Area Boundary (EAB) or Low Population Zone (LPZ) dose consequences of the cask drop accident event. This includes no credit for the Fuel Handling Isolation Signal (FHIS), the fuel handling building post accident cleanup unit (PACU) filtration system, the Control Room Isolation Signal (CRIS) and the control room (CR) emergency air cleanup system (CREACUS). Doses are evaluated for various control room unfiltered intake plus unfiltered inleakage inflow rates.

The release of radioactive material to the atmosphere represents a potential exposure hazard to control room personnel and the general public at the EAB and LPZ. However, the offsite radiological doses for the postulated Spent Fuel Cask Drop accident do not exceed 25% of the 10 CFR Part 50.67 exposure guidelines.

DOSE RECEPTOR	CASK DROP DOSE (REM TEDE)	ACCEPTANCE CRITERION (REM TEDE)
Control Room (30-day accident duration)	0.89E-3 (0.89 mRem TEDE)	5
EAB (Maximum 2-hour dose -- 0.0 to 2.0 hours)	3.09E-3 (3.09 mRem TEDE)	6.3
LPZ (30-day accident duration)	0.09E-3 (0.09 mRem TEDE)	6.3

Table 2 – Cask Drop Accident Dose Consequences

3.1.4 Spent Fuel Pool Boiling Accident

The postulated loss of all spent fuel pool (SFP) cooling is assumed to result in SFP boiling and the release of a portion of the radionuclide inventory contained in the stored spent fuel assemblies and the SFP water. The evaluation of the radiological consequences for the SFP boiling event assumes a minimum of 17 months since the shutdown of Units 2 and 3. Following a loss of SFP cooling, activity releases from the spent fuel due to evaporation and boiling disperse to the Control Room, EAB and LPZ locations.

No credit is taken for activity retention within the fuel handling building air. No credit is taken for FHIS or filtration by the Fuel Handling Building PACUs. All activity escaping from the SFP is assumed to be instantaneously released to the environment and atmospherically dispersed to the control room and offsite dose receptors.

No credit is taken for CRIS or CREACUS. For conservatism the control room dose is calculated for an individual at the control room outside air intake location. The total effective dose equivalent (TEDE) dose at this location is conservatively greater than it would be inside the Control Room. The activity concentration inside the control room would be smaller since only a portion of the outside cloud would enter the control room envelope via ventilation system inflow or inleakage. The offsite radiological doses for the postulated SFP boiling accident do not exceed 25% of the 10 CFR Part 50.67 exposure guidelines.

DOSE RECEPTOR	SFP BOILING DOSE (REM TEDE)	ACCEPTANCE CRITERION (REM TEDE)
Control Room (30-day accident duration)	11.96E-3 (11.96 mRem TEDE)	5
EAB (Maximum 2-hour dose -- 0.0 to 2.0 hours)	0.08E-3 (0.08 mRem TEDE)	6.3
LPZ (30-day accident duration)	0.25E-3 (0.25 mRem TEDE)	6.3

Table 3 – Radiological Consequences of Spent Fuel Pool Boiling

3.1.5 Fuel Handling Accident Analysis for the Permanently Defueled Condition

A FHA was incorporated into the SONGS UFSAR under the provisions of 10 CFR 50.59 to address the permanently defueled condition. The analysis determined a reasonable time post-cessation of operations for movement of fuel from the fuel storage pool during which, if a fuel handling accident occurs, dose consequences would be within 10 CFR 50.67 and Regulatory Guide 1.183 dose limits. The analysis assumed fuel storage pool decontamination based on 23 feet of water over the failed fuel assembly, no credit for emergency ventilation or filtration (control room or otherwise) and no credit taken for radioactive decay of the isotopes during atmospheric dispersion transit to the control room or offsite dose locations.

The FHA inside the FHB involves the inadvertent dropping of a fuel assembly during fuel handling operations, and the subsequent rupture of fuel pins in the dropped assembly and/or the impacted assembly. A maximum of 472 fuel rods are assumed to fail as a result of the drop of a fuel assembly on to the fuel assemblies stored in fuel storage pool fuel racks. The FHA-FHB dose analysis models 17 months (12,240 hours) of radioactive decay prior to the event. All gap activity in the damaged rods is instantaneously released into the fuel storage pool. During the movement of fuel assemblies in the fuel storage pool, the fuel storage pool water level is assumed to be at least 23 feet over the top of the irradiated fuel assemblies seated in the storage racks.

The radioactive material that escapes from the fuel storage pool to the FHB is released to the environment over a 2-hour time period (i.e., FHB closure is not modeled during the FHA-FHB event). Consistent with the 2-hour release model requirement, the FHA-FHB alternate source term (AST) dose analysis does not model the generation of an Engineered Safety Feature Actuation System (ESFAS) fuel handling [building] isolation signal (FHIS). The FHB normal ventilation exhaust is assumed to remain operational throughout the FHA-FHB event. The FHB air volume dilutes the gaseous activity released from the damaged fuel rods.

The FHA-FHB AST dose analysis does not model a reduction in the amount of radioactive material available for release from the FHB by the fuel handling building Post-Accident Cleanup Unit (PACU) filter system.

Activity released during the FHA-FHB event is transported by atmospheric dispersion to the control room HVAC intake and to the offsite EAB and LPZ dose receptors. Activity may be released to the environment via the FHB normal ventilation exhaust system through the main plant vent, or as leakage through FHB penetrations (e.g., doors). No credit is taken for radioactive decay of the isotopes during atmospheric dispersion transit to the control room or offsite dose locations. Table 4 presents the San Onofre site-specific 95th percentile meteorology atmospheric dispersion factors for these release pathways for control room dose calculation.

FHA-FHB to CR 95th Percentile Atmospheric Dispersion Factors (seconds/m^3)			
Time Interval	FHB Release Point	Main Plant Vent Release Point	Modeled Value
0 to 2 hours	9.48E-04	1.15E-03	1.15E-03
2 to 8 hours	7.61E-04	6.23E-04	7.61E-04
8 to 24 hours	1.92E-04	2.14E-04	2.14E-04
1 to 4 days	2.65E-04	2.22E-04	2.65E-04
4 to 30 days	2.43E-04	2.02E-04	2.43E-04

Table 4 – FHA-FHB Control Room Atmospheric Dispersion Factors

The FHA-FHB dose analysis for persons located at or beyond the boundary of the exclusion area, including the outer boundary of the low population zone, considers the dose consequences of inhalation and immersion. Radioactive material in the fuel handling building is assumed to be a negligible radiation shine source to the offsite dose receptors relative to the dose associated with immersion in the radioactive plume released from the facility.

The Control Room (CR) dose during a design basis FHA-FHB following permanent shut down of SONGS Units 2 and 3 is based on:

(a) No credit for control room emergency air cleanup system (CREACUS) and Control Room Isolation Signal (CRIS) and no gamma radiation shine from CREACUS charcoal and HEPA filters.
(b) CR doses are evaluated at various CR unfiltered inflow (including inleakage) flow rates. The flow rates were varied from 500 cfm to 15,000 cfm, but only the bounding CR dose is reported.

FHA-FHB dose analysis for persons located in the control room considers the dose consequences of inhalation, immersion, and radiation shine from the environmental (or outside) cloud. Radiation shine from contaminated air in the fuel handling building is considered negligible due to the presence of numerous intervening concrete walls and the geometric attenuation due to the distance between the FHB and the control room.

The resulting FHA-FHB offsite and control room operator doses are listed in Table 5. The analysis demonstrates that the FHA-FHB event criteria are met.

DOSE RECEPTOR	FHA-FHB DOSE (REM TEDE)	ACCEPTANCE CRITERION (REM TEDE)
Control Room (30-day accident duration)	0.06E-3 (0.06 mRem TEDE)	5
EAB (Maximum 2-hour dose -- 0.0 to 2.0 hours)	0.20E-3 (0.20 mRem TEDE)	6.3
LPZ (30-day accident duration)	0.01E-3 (0.01 mRem TEDE)	6.3

Table 5 – FHA-FHB Dose Consequences

3.2 Discussion

The following portion of this license amendment request contains the technical analysis for justifying the proposed change, and a summary of the change, for Technical Specifications (TS) Chapters 1.0, 2.0, and 3.0, Sections 3.1 through 3.9, and Chapters 4.0 and 5.0.

A separate description, the proposed change, technical analysis, and summary of the change are provided for each TS Chapter/Section.

3.2.1 TS Chapter 1.0, Use and Application

The existing TS Chapter 1.0, Use and Application, contains terms and guidance used to clarify the TS. This chapter is divided into the following four sections.

TS Being Deleted	TS Being Retained
1.0 USE AND APPLICATION	
	1.1 Definitions
	1.2 Logical Connectors
	1.3 Completion Times
	1.4 Frequency

All TS in Chapter 1.0 are being retained, as identified in the table above. Proposed revisions to these TS are as described below and shown in Attachments 1 and 2.

3.2.1.1 TS Section 1.1, Definitions

3.2.1.1.1 Description

TS Section 1.1, Definitions – Defines terms that are used and are applicable throughout the TS and Bases.

3.2.1.1.2 Proposed Changes

Definitions Being Deleted

Term	Definition
AXIAL SHAPE INDEX (ASI)	ASI shall be the power generated in the lower half of the core less the power generated in the upper half of the core, divided by the sum of the power generated in the lower and upper halves of the core. $$ASI = \frac{\text{lower - upper}}{\text{lower + upper}}$$
AZIMUTHAL POWER TILT (T_q)	AZIMUTHAL POWER TILT shall be the power asymmetry between azimuthally symmetric fuel assemblies.
CHANNEL CALIBRATION	A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds within the necessary range and accuracy to known values of the parameter that the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel, including the required sensor, alarm, display, and trip functions, and shall include the CHANNEL FUNCTIONAL TEST. Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an inplace cross calibration of the sensing elements and normal calibration of the remaining adjustable devices in the channel. Whenever a sensing element is replaced, the next required inplace cross calibration consists of comparing the other sensing elements with the recently installed sensing element. The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping, or total channel steps so that the entire channel is calibrated.
CHANNEL CHECK	A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from

independent instrument channels measuring the same parameter.

CHANNEL FUNCTIONAL TEST A CHANNEL FUNCTIONAL TEST shall be:

a. Analog channels – the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY, including required alarms, interlocks, display and trip functions;

b. Bistable channels (e.g., pressure switches and switch contacts) – the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY, including required alarm and trip functions; or

c. Digital computer channels – the use of diagnostic programs to test digital computer hardware and the injection of simulated process data into the channel to verify OPERABILITY, including alarm and trip functions.

The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps so that the entire channel is tested.

CORE ALTERATION

CORE ALTERATION shall be the movement or manipulation of any fuel, sources, reactivity control components, or other components, excluding control element assemblies (CEAs) withdrawn into the upper guide structure, affecting reactivity, within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.

CORE OPERATING LIMITS REPORT (COLR)

The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 5.7.1.5. Plant operation within these limits is addressed in individual Specifications.

DOSE EQUIVALENT I-131

DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in ICRP 30, Supplement to Part 1, page 192-212, Table titled, "Committed Dose

	Equivalent in Target Organs or Tissues per Intake of Unit Activity."
$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY	$\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives > 15 minutes, making up at least 95% of the total noniodine activity in the coolant.
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.
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) 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 to the Secondary System (primary to secondary LEAKAGE).

b. <u>Unidentified LEAKAGE</u>

All LEAKAGE that is not identified LEAKAGE.

	c. <u>Pressure Boundary LEAKAGE</u> LEAKAGE (except primary to secondary LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.
MODE	A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.
PHYSICS TESTS	PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are: a. Described in Chapter 14, Initial Test Program of the SONGS Units 2 and 3 UFSAR; b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission.
PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.7.1.6.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3438 MWt.
REACTOR PROTECTION 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 CEAs (shutdown and regulating) are fully inserted except for the single CEA of highest reactivity worth, which is assumed to be fully withdrawn. However, with all CEAs verified fully inserted by two independent means, it is not necessary to account for a stuck CEA in the SDM calculation. With any CEAs not capable of being fully inserted, the 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.

STAGGERED TEST BASIS	A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function.
THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

In conjunction with deletion of the term "MODE," TS Table 1.1-1, MODES, is also being deleted.

Definition Being Added

The following definition is being added.

CERTIFIED FUEL HANDLER	A CERTIFIED FUEL HANDLER is an individual who complies with provisions of the CERTIFIED FUEL HANDLER training program required by TS 5.3.2.

There are no corresponding TS Bases sections associated with this TS section.

3.2.1.1.3 Technical Analysis

TS Section 1.1 provides defined terms that are applicable throughout the TS and TS Bases. The SONGS Unit 2 and Unit 3 Part 50 licenses no longer authorize emplacement or retention of fuel in their respective reactor vessel. The definitions proposed for deletion are no longer used and no longer apply to the permanently defueled plant condition.

The definition of the term Certified Fuel Handler is being added to ensure consistent understanding and application. Further discussion regarding Certified Fuel Handlers is included in the Administrative Controls Section of the proposed TS.

3.2.1.2 TS Section 1.2, Logical Connectors

3.2.1.2.1 Description

Logical Connectors – An explanation of the logical connectors used to discriminate between, and yet connect, discrete Conditions, Required Actions, Completion Times, Surveillances, and Frequencies.

3.2.1.2.2 Proposed Changes

Example 1.2-2 is being deleted because the logical connectors that remain throughout the PDTS are limited and are adequately explained by Example 1.2-1.

3.2.1.2.3 Technical Analysis

Because 10 CFR 50.82(a)(2) prohibits operation of the plant or placing fuel in the reactor vessel, certain terms currently provided in the TS no longer apply. Therefore, TS Section 1.2 is being revised to be consistent with the permanently defueled condition and the use of logical connectors within the PDTS.

3.2.1.3 TS Section 1.3, Completion Times

3.2.1.3.1 Description

Completion Times – Establishes the Completion Time convention and provides guidance for its use.

3.2.1.3.2 Proposed Changes

Statements referring to "operation of the unit" are replaced with "storage of irradiated fuel," since operation of the units is no longer permitted and safe storage of irradiated fuel is the primary objective of the permanently defueled TS.

References to the term "unit" are replaced with the term "facility," because the term "unit" generally refers to the reactor, which can no longer be operated, whereas the term "facility" refers to the overall site, including the fuel storage facility.

References to the term "MODE" are deleted, as this term is no longer applicable to a permanently defueled facility.

A portion of the Description section, Examples 1.3-2 through 1.3-7 are deleted because they reference activities that no longer pertain to a permanently defueled condition. Example 1.3-1 has been modified to reflect Required Actions typical of the permanently defueled condition.

3.2.1.3.3 Technical Analysis

Because 10 CFR 50.82(a)(2) prohibits operation of the plant or placing fuel in the reactor vessel, certain terms currently provided in the TS no longer apply. Therefore, TS Section 1.3 is being revised to be consistent with the permanently defueled condition.

3.2.1.4 TS Section 1.4, Frequency

3.2.1.4.1 Description

Frequency – Defines the proper use and application of Frequency requirements throughout the TS.

3.2.1.4.2 Proposed Changes

References to the terms "MODE" and "reactor power" are either deleted or replaced with terms such as "specified condition." These two former terms ("MODE" and "reactor power") are no longer applicable to a permanently defueled facility.

References to the term "unit" are replaced with the term "facility," because the term "unit" generally refers to the reactor, which can no longer be operated, whereas the term "facility" refers to the overall site, including the fuel storage facility.

The final paragraph of the TS 1.4 Description section, regarding Notes that modify the Frequency of performance of some surveillances and the applicability of entry restrictions of SR 3.0.4, is being deleted in its entirety. This is consistent with the fact that none of the surveillances in the proposed TS contain Notes that modify the Frequency of performance or the conditions during which the acceptance criteria must be satisfied. Therefore, this paragraph is not applicable to the proposed TS LCOs or Surveillance Requirements and may be deleted.

Example 1.4-2 is being modified to make it more applicable to the permanently defueled condition and the use of a one time performance Frequency in the proposed TS.

Example 1.4-3 is being deleted because it references an operating reactor. This example is not needed in a permanently defueled condition, as the revised TS will no longer include a surveillance that would benefit from the clarification that it provides. The remaining examples are sufficient to explain application of the TS Frequency requirements.

3.2.1.4.3 Technical Analysis

Because 10 CFR 50.82(a)(2) prohibits operation of the plant or placing fuel in the reactor vessel, certain terms currently provided in the TS no longer apply. Therefore, TS Section 1.4 is being revised to be consistent with the permanently defueled condition.

3.2.1.5 Summary

The permanently defueled TS, Chapter 1.0, will continue to remain applicable with the reactor permanently defueled. As such, it is being retained and revised to reflect a permanently defueled condition. Therefore, retaining TS Chapter 1.0, as revised, provides appropriate control over use and application of the SONGS Units 2 and 3 TS.

3.2.2 TS Chapter 2.0, Safety Limits

The existing TS Chapter 2.0, Safety Limits, contains limits upon important process variables that are necessary to reasonably protect the integrity of certain physical barriers that guard against the uncontrolled release of radioactivity from the reactor core and the reactor coolant system. This chapter is divided into the following Sections.

TS Being Deleted	TS Being Retained
2.0 SAFETY LIMITS (SLs)	
2.1 Safety Limits (SLs)	
2.2 SL Violations	

All TS in Chapter 2.0 are being proposed for deletion, as identified in the table above. The corresponding TS Bases are also being deleted to reflect this change.

3.2.2.1 TS 2.1, Safety Limits (SLs)

3.2.2.1.1 Description

TS 2.1, Safety Limits, contains two Specifications:

- TS 2.1.1, Reactor Core SLs; and
- TS 2.1.2, Reactor Coolant System (RCS) SL.

The restrictions of the Reactor Core SL in TS 2.1.1 prevent overheating of the fuel and cladding, as well as possible cladding perforation that would result in the release of fission products to the reactor coolant. TS 2.1.1 is applicable in MODES 1 and 2.

The SL on RCS pressure in TS 2.1.2 protects the integrity of the RCS against overpressurization. TS 2.1.2 is applicable in MODES 1, 2, 3, 4, and 5.

3.2.2.1.2 Proposed Changes

The SONGS Unit 2 and Unit 3 Part 50 licenses no longer authorize emplacement or retention of fuel in their respective reactor vessel. Therefore, these safety limits do not apply and are being proposed for deletion.

3.2.2.1.3 Technical Analysis

The above TS contain limits upon important process variables that are necessary to reasonably protect the integrity of certain physical barriers required for safe operation of the facility only when the reactor is in MODES 1 through 5. However, 10 CFR 50.82(a)(2) prohibits operation of the plant or placing fuel in the reactor vessel. Therefore, the TS listed in the previous paragraphs, which only address these specific process variables, are no longer applicable. Based on the above, the proposed deletion of TS related to these safety limits is acceptable.

3.2.2.2 TS 2.2, SL Violation

3.2.2.2.1 Description

TS 2.2, Safety Limit Violations, directs actions to be taken if a safety limit specified in TS 2.1 is violated. TS 2.2 is applicable commensurate with the applicable MODES of the respective safety limits specified in TS 2.1.

3.2.2.2.2 Proposed Changes

The SONGS Unit 2 and Unit 3 Part 50 licenses no longer authorize emplacement or retention of fuel in their respective reactor vessel. Therefore, these safety limits do not apply and are being proposed for deletion.

3.2.2.2.3 Technical Analysis

The above TS contains actions to be taken if a safety limit specified in TS 2.1 is violated when the reactor is in MODES 1 through 5. However, 10 CFR 50.82(a)(2) prohibits operation of the plant or placing fuel in the reactor vessel. Therefore, the TS listed in the previous paragraphs, which only address these specific process variables, are no longer applicable. Based on the above, the proposed deletion of TS related to these safety limits is acceptable.

3.2.2.3 Summary

TS Chapter 2.0 does not apply with the reactor defueled. Therefore, the individual TS contained therein are not needed for a permanently defueled condition. As such, they may be deleted with no impact on continued safe operation of the facility.

3.2.3 TS Chapter 3.0, Limiting Condition for Operation (LCO) Applicability and Surveillance Requirement (SR) Applicability

The existing TS Chapter 3.0, Limiting Condition for Operation (LCO) Applicability and Surveillance Requirement (SR) Applicability, contains general requirements applicable to all Specifications. This chapter is divided into the following Sections.

TS Being Deleted	TS Being Retained
3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY	
	3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY
	3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

Both TS in Chapter 3.0 are being retained, as identified in the table above. Certain Specifications within Chapter 3.0 are being proposed for deletion. Proposed revisions to these Specifications (including those being deleted) are as described below and shown in Attachments 1 and 2. The corresponding TS Bases are also being revised to reflect this change.

3.2.3.1 TS Section 3.0, Limiting Condition for Operation (LCO) Applicability

The existing TS Section 3.0, Limiting Condition for Operation (LCO) Applicability, consists of LCO 3.0.1 through LCO 3.0.7. LCO 3.0.1 through LCO 3.0.7 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.

3.2.3.1.1 Description

LCO 3.0.1 establishes the Applicability statement within each individual Specification as the requirement for when the LCO is required to be met (i.e., when the unit is in the MODES or other specified conditions of the Applicability statement of each Specification).

LCO 3.0.2 establishes that upon discovery of a failure to meet an LCO, the associated ACTIONS shall be met.

LCO 3.0.3 establishes the actions that must be implemented when an LCO is not met. LCO 3.0.3 is only applicable in MODES 1, 2, 3, and 4.

LCO 3.0.4 establishes limitations on changes in MODES or other specified conditions in the Applicability when an LCO is not met.

LCO 3.0.5 establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service or declared inoperable to comply with ACTIONS.

LCO 3.0.6 establishes an exception to LCO 3.0.2 for supported systems that have a support system LCO specified in the TS.

LCO 3.0.7 pertains to certain special tests and operations required to be performed at various times over the life of the unit.

3.2.3.1.2 Proposed Changes

LCO 3.0.1 and LCO 3.0.2 are being retained in the permanently defueled TS with the proposed revisions shown in Attachments 1 and 2.

LCO 3.0.3 through LCO 3.0.7 are being proposed for deletion in their entirety.

3.2.3.1.3 Technical Analysis

Because 10 CFR 50.82(a)(2) prohibits operation of the plant or placing fuel in the reactor vessel, reference to operating MODES is no longer relevant. Therefore, LCO 3.0.3 and LCO 3.0.4 are no longer applicable and are being deleted.

The allowance of LCO 3.0.5 to not comply with the requirements of LCO 3.0.2 (i.e., to not comply with the Required Actions) to allow the performance of SRs on equipment declared inoperable or removed from service is no longer required. The remaining permanently defueled TS ACTIONS do not include requirements to declare equipment inoperable or to remove it from service.

The requirements of the Safety Function Determination Program (SFDP), contained in TS 5.6.3, Safety Function Determination Program (SFDP), which directs cross train checks of multiple and redundant safety systems, no longer apply. Therefore, LCO 3.0.6, which provides an exception to LCO 3.0.2 to allow the use of the SFDP is also being deleted.

LCO 3.0.7 is associated with test exceptions in TS Section 3.1, which are being deleted as described within this document. Therefore, references to LCO 3.0.7 are also being deleted.

LCO 3.0.1 and LCO 3.0.2 are revised to remove references to "MODES" and to reflect the deletion of the above exceptions to their requirements.

3.2.3.2 TS Section 3.0, Surveillance Requirement (SR) Applicability

TS 3.0, "Surveillance Requirement (SR) Applicability," consists of SR 3.0.1 through SR 3.0.4. SR 3.0.1 through SR 3.0.4 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.

3.2.3.2.1 Description

SR 3.0.1 establishes the requirement that SRs must be met during the MODES or other specified conditions in the Applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This Specification ensures that Surveillances are performed to verify the OPERABILITY of systems and components, and that variables are within specified limits.

SR 3.0.2 permits a 25% extension of the interval specified in the Frequency, and establishes the requirements for meeting the specified Frequency for Surveillances and

any Required Action with a Completion Time that requires the periodic performance of the Required Action on a "once per. .. "interval.

SR 3.0.3 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a surveillance requirement has not been completed within the specified Frequency.

SR 3.0.4 establishes the requirement that all applicable SRs must be met before entry into a MODE or other specified condition in the Applicability.

3.2.3.2.2 Proposed Changes

SR 3.0.1, SR 3.0.2, and SR 3.0.4 are being retained in the permanently defueled TS with the proposed revisions shown in Attachments 1 and 2.

SR 3.0.3 is being retained in the permanently defueled TS unchanged.

3.2.3.2.3 Technical Analysis

Because 10 CFR 50.82(a)(2) prohibits operation of the plant or placing fuel in the reactor vessel, the reference to operating MODES is no longer relevant and is therefore being deleted. In addition, SR 3.0.2 discussions pertaining to the applicability of the Frequencies specified as "once," and Completion Times that require periodic performance on a "once per. . ." basis, are deleted as the proposed TS no longer include these types of Frequencies or Completion Times.

3.2.3.3 Summary

The permanently defueled TS, Chapter 3.0, LCO 3.0.1, LCO 3.0.2, SR 3.0.1, SR 3.0.2, SR 3.0.3, and SR 3.0.4, will continue to remain applicable with the reactor permanently defueled. As such, they are being retained and revised, as necessary, to reflect a permanently defueled condition. Therefore, retaining TS Chapter 3.0, as revised, provides appropriate control over use and application of the SONGS Units 2 and 3 TS.

3.2.4 TS Section 3.1, Reactivity Control Systems

The existing TS Section 3.1, Reactivity Control Systems, contains LCOs that provide for appropriate control of process variables, design features, or operating restrictions that are required to protect the integrity of a fission product barrier. This Section is divided into the following Specifications.

TS Being Deleted	TS Being Retained
3.1 REACTIVITY CONTROL SYSTEMS	
3.1.1 SHUTDOWN MARGIN (SDM) – T_{avg} > 200°F	
3.1.2 SHUTDOWN MARGIN (SDM) – T_{avg} ≤ 200°F	
3.1.3 Reactivity Balance	
3.1.4 Moderator Temperature Coefficient (MTC)	
3.1.5 Control Element Assembly (CEA) Alignment	
3.1.6 Shutdown Control Element Assembly (CEA) Insertion Limits	
3.1.7 Regulating CEA Insertion Limits	
3.1.8 Part Length Control Element Assembly (CEA) Insertion Limits	
3.1.9 Boration Systems – Operating	
3.1.10 Boration Systems – Shutdown	
3.1.11 Not Used	
3.1.12 Special Test Exception (STE) – Low Power Physics Testing	
3.1.13 Special Test Exception (STE) – At Power Physics Testing	
3.1.14 Special Test Exceptions (STE) – Reactivity Coefficient Testing	

All TS in Section 3.1 are being proposed for deletion, as identified in the table above. The corresponding TS Bases are also being deleted to reflect this change.

3.2.4.1 Description

TS 3.1.1, SHUTDOWN MARGIN (SDM) – T_{avg} > 200°F, specifies requirements to provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for normal shut down and anticipated operational occurrences (AOOs) assuming the highest reactivity worth control element assembly (CEA) remains fully withdrawn. TS 3.1.1 is applicable in MODES 3 and 4.

TS 3.1.2, SHUTDOWN MARGIN (SDM) – $T_{avg} \leq 200°F$, specifies requirements to provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for normal shut down and AOOs assuming the highest reactivity worth CEA remains fully withdrawn. TS 3.1.1 is applicable in MODE 5.

TS 3.1.3, Reactivity Balance, specifies requirements for the comparison of the predicted versus measured core reactivity during power operation. The periodic confirmation of core reactivity is necessary to ensure that Design Basis Accident (DBA) and transient safety analyses remain valid. TS 3.1.3 is applicable in MODES 1 and 2.

TS 3.1.4, Moderator Temperature Coefficient (MTC), specifies requirements to ensure that core overheating and overcooling accidents will not violate the accident analysis assumptions. TS 3.1.4 is applicable in MODE 1 and MODE 2 with $k_{eff} \geq 1.0$.

TS 3.1.5, Control Element Assembly (CEA) Alignment, specifies limits on shutdown and regulating CEA alignments to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved. TS 3.1.5 is applicable in MODES 1 and 2.

TS 3.1.6, Shutdown Control Element Assembly (CEA) Insertion Limits, specifies limits on shutdown CEA insertion to ensure that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required shutdown margin (SDM) following a reactor trip. TS 3.1.6 is applicable in MODE 1 and MODE 2 with any regulating CEA not fully inserted.

TS 3.1.7, Regulating CEA Insertion Limits, specifies limits on regulating CEA sequence and physical insertion for the function of preserving power distribution, ensuring that the SDM is maintained, ensuring that ejected CEA worth is maintained, and ensuring adequate negative reactivity insertion is available on trip. The overlap between regulating banks provides more uniform rates of reactivity insertion. TS 3.1.7 is applicable in MODES 1 and 2.

TS 3.1.8, Part Length Control Element Assembly (CEA) Insertion Limits, specifies limits on part length CEA insertion for the function of preserving power distribution and ensuring that ejected CEA worth is maintained within limits. TS 3.1.8 is applicable in MODE 1 > 20% RTP.

TS 3.1.9, Boration Systems – Operating, establishes the requirements for borated water sources and flow paths to the RCS is to ensure that sufficient borated water is available to maintain the reactor subcritical and provide makeup water to account for RCS shrinkage during cool down to cold shutdown conditions. TS 3.1.9 is applicable in MODES 1, 2, 3, and 4.

TS 3.1.10, Boration Systems – Shutdown, establishes the requirements for borated water sources and flow paths to the RCS is to ensure that sufficient borated water is available to maintain the reactor subcritical. TS 3.1.10 is applicable in MODES 5 and 6.

TS 3.1.12, Special Test Exception (STE) – Low Power Physics Testing, permits relaxation of existing TS Limiting Conditions for Operation (LCOs) to allow the performance of PHYSICS TESTS. TS 3.1.12 is applicable in MODES 2 and 3 during PHYSICS TESTS.

TS 3.1.13, Special Test Exception (STE) – At Power Physics Testing, permits relaxation of existing TS LCOs to allow the performance of PHYSICS TESTS. TS 3.1.13 is applicable in MODE 1 during PHYSICS TESTS.

TS 3.1.14, Special Test Exceptions (STE) – Reactivity Coefficient Testing, permits relaxation of existing TS LCOs to allow the performance of PHYSICS TESTS. TS 3.1.14 is applicable in MODE 1.

3.2.4.2 Proposed Changes

The SONGS Unit 2 and Unit 3 Part 50 licenses no longer authorize emplacement or retention of fuel in their respective reactor vessel. Therefore, these reactivity control systems do not apply and are being proposed for deletion.

3.2.4.3 Technical Analysis

The above TS are related to assuring the appropriate functional capability of plant equipment, and control of process variables, design features, or operating restrictions required for safe operation of the facility only when the reactor is in MODES 1 through 6. However, 10 CFR 50.82(a)(2) prohibits operation of the plant or placing fuel in the reactor vessel. Therefore, the TS listed in the previous paragraphs, which only address their associated specific plant equipment, control of process variables, design features, or operating restrictions are no longer applicable. Based on the above, the proposed deletion of all TS in Section 3.1 is acceptable.

3.2.4.4 Summary

Since these TS do not apply with the reactor defueled, they are not needed for a permanently defueled condition. As such, they may be deleted with no impact on continued safe operation of the facility.

3.2.5 TS Section 3.2, Power Distribution Limits

The existing TS Section 3.2, Power Distribution Limits, contains LCOs that provide for appropriate control of process variables, design features, or operating restrictions that are required to protect the integrity of a fission product barrier. This Section is divided into the following Specifications.

TS Being Deleted	TS Being Retained
3.2 POWER DISTRIBUTION LIMITS	
3.2.1 Linear Heat Rate (LHR)	
3.2.2 Planar Radial Peaking Factors (F_{xy})	
3.2.3 AZIMUTHAL POWER TILT (T_q)	
3.2.4 Departure From Nucleate Boiling Ratio (DNBR)	
3.2.5 AXIAL SHAPE INDEX (ASI)	

All TS in Section 3.2 are being proposed for deletion, as identified in the table above. The corresponding TS Bases are also being deleted to reflect this change.

3.2.5.1 Description

TS 3.2.1, Linear Heat Rate (LHR), specifies limits based on correlations between power peaking and certain measured variables used as inputs to the LHR and DNBR operating limits. The limitation on LHR ensures that in the event of a loss of coolant accident (LOCA) the peak temperature of the fuel cladding does not exceed 2200°F. TS 3.2.1 is applicable in MODE 1 with THERMAL POWER > 20% RTP.

TS 3.2.2, Planar Radial Peaking Factors (F_{xy}), specifies limits based on correlations between power peaking and certain measured variables used as inputs to the LHR and DNBR operating limits. Limiting of the calculated Planar Radial Peaking Factors to values equal to or greater than the measured Planar Radial Peaking Factors ensures that the calculated limits remain valid. TS 3.2.2 is applicable in MODE 1 with THERMAL POWER > 20% RTP.

TS 3.2.3, AZIMUTHAL POWER TILT (T_q), specifies limits based on correlations between power peaking and certain measured variables used as inputs to the LHR and DNBR operating limits. The limitations on the T_q are provided to ensure that design operating margins are maintained. TS 3.2.3 is applicable in MODE 1 with THERMAL POWER > 20% RTP.

TS 3.2.4, Departure From Nucleate Boiling Ratio (DNBR), specifies limits based on correlations between power peaking and certain measured variables used as inputs to the LHR and DNBR operating limits. Operation of the core with a DNBR at or above this limit ensures that an acceptable minimum DNBR is maintained in the event of a loss of flow transient. TS 3.2.4 is applicable in MODE 1 with THERMAL POWER > 20% RTP.

TS 3.2.5, AXIAL SHAPE INDEX (ASI), specifies limits based on correlations between power peaking and certain measured variables used as inputs to the LHR and DNBR operating limits. The limitation on ASI ensures that the actual ASI value is maintained within the range of values used in the accident analysis. The ASI limits ensure that with T_q at its maximum upper limit, the DNBR does not drop below the DNBR Safety Limit for AOOs. TS 3.2.5 is applicable in MODE 1 with THERMAL POWER > 20% RTP.

3.2.5.2 Proposed Changes

The SONGS Unit 2 and Unit 3 Part 50 licenses no longer authorize emplacement or retention of fuel in their respective reactor vessel. Therefore, these power distribution limits do not apply and are being proposed for deletion.

3.2.5.3 Technical Analysis

The above TS are related to assuring the appropriate functional capability of plant equipment, and control of process variables, design features, or operating restrictions required for safe operation of the facility only when the reactor is in MODE 1. However, 10 CFR 50.82(a)(2) prohibits operation of the plant or placing fuel in the reactor vessel. Therefore, the TS listed in the previous paragraphs, which only address their associated specific plant equipment, control of process variables, design features, or operating restrictions are no longer applicable. Based on the above, the proposed deletion of all TS in Section 3.2 is acceptable.

3.2.5.4 Summary

Since these TS do not apply with the reactor defueled, they are not needed for a permanently defueled condition. As such, they may be deleted with no impact on continued safe operation of the facility.

3.2.6 TS Section 3.3, Instrumentation

The existing TS Section 3.3, Instrumentation, contains LCOs that provide for appropriate functional capability of sensing and control instrumentation required for safe operation of the facility, including the plant being in a defueled condition. This Section is divided into the following Specifications.

TS Being Deleted	TS Being Retained
3.3 INSTRUMENTATION	
3.3.1 Reactor Protective System (RPS) Instrumentation – Operating	
3.3.2 Reactor Protective System (RPS) Instrumentation – Shutdown	
3.3.3 Control Element Assembly Calculators (CEACs)	
3.3.4 Reactor Protective System (RPS) Logic and Trip Initiation	
3.3.5 Engineered Safety Features Actuation System (ESFAS) Instrumentation	
3.3.6 Engineered Safety Features Actuation System (ESFAS) Logic and Manual Trip	
3.3.7 Diesel Generator (DG) – Undervoltage Start	
3.3.8 Containment Purge Isolation Signal (CPIS)	
3.3.9 Control Room Isolation Signal (CRIS)	
3.3.10 Not Used	
3.3.11 Post Accident Monitoring Instrumentation (PAMI)	
3.3.12 Remote Shutdown System	
3.3.13 Source Range Monitoring Channels	

All TS in Section 3.3 are being proposed for deletion, as identified in the table above. The corresponding TS Bases are also being deleted to reflect this change.

3.2.6.1 Section 3.3 TS That Are Not Applicable When Defueled

3.2.6.1.1 Description

TS 3.3.1, Reactor Protective System (RPS) Instrumentation – Operating, specifies requirements for the RPS instrumentation system to maintain the safety limits during all

anticipated operational occurrences, and mitigates the consequences of design basis accidents in MODES 1 and 2. TS 3.3.2, Reactor Protective System (RPS) Instrumentation – Shutdown, specifies requirements for the RPS instrumentation system to maintain the safety limits during all anticipated operational occurrences, and mitigate the consequences of design basis accidents in MODES 3, 4, and 5 when the reactor trip circuit breakers (RTCBs) are closed and the CEA Drive System is capable of CEA withdrawal. TS 3.3.3, Control Element Assembly Calculators (CEACs), specifies requirements that ensure the core protection calculators (CPCs) are either informed of individual CEA position within each subgroup, using one or both CEACs, or that appropriate conservatism is included in the CPC calculations to account for anticipated CEA deviations in MODES 1 and 2. TS 3.3.4, Reactor Protective System (RPS) Logic and Trip Initiation, specifies requirements for RPS Matrix Logic, RPS Initiation Logic, RTCBs, and manual trip channels to effect automatic trip signals received from RPS instruments and to provide a means to manually trip the reactor in MODES 1 and 2, and in MODES 3, 4, and 5 when the RTCBs are closed and the CEA Drive System is capable of CEA withdrawal. The RPS initiates a reactor trip, based on the values of selected unit parameters, to protect against violating the core fuel design limits and reactor coolant pressure boundary (RCPB) during AOOs and to assist the Engineered Safety Features systems in mitigating accidents. The protection and monitoring systems have been designed to assure safe operation of the reactor. This is achieved by specifying limiting safety system settings in terms of parameters directly monitored by the RPS, as well as specifying LCOs on other reactor system parameters and equipment performance. Although the CEACs do not provide a direct reactor trip Function, their input to the CPCs is taken credit for in the CEA misoperation analysis. TS 3.3.1 is applicable in MODES 1 and 2. TS 3.3.2 is applicable in MODES 3, 4, and 5 when the RTCBs are closed and the CEA Drive System is capable of CEA withdrawal. TS 3.3.3 is applicable in MODES 1 and 2. TS 3.3.4 is applicable in MODES 1 and 2, and in MODES 3, 4, and 5 when the RTCBs are closed and the CEA Drive System is capable of CEA withdrawal.

TS 3.3.5, Engineered Safety Features Actuation System (ESFAS) Instrumentation, specifies requirements for the ESFAS instrumentation to ensure that ESFAS initiates necessary safety systems, based on the values of selected unit parameters, to protect against violating core design limits and the RCPB during AOOs and ensures acceptable consequences during accidents. TS 3.3.6, Engineered Safety Features Actuation System (ESFAS) Logic and Manual Trip, specifies requirements for ESFAS Matrix Logic, ESFAS Initiation Logic, and manual trip channels to effect automatic ESFAS initiation received from ESFAS instruments and to provide a means to manually actuate an ESF system in MODES 1, 2, 3, and 4. This is achieved by monitoring variables that are indicative of conditions requiring protective action. ESFAS contains devices and circuitry that generate signals to actuate one or more ESF Systems when the monitored variables reach specified levels. Accidents are events that are analyzed even though they are not expected to occur during the unit life. The acceptable limit during accidents is that offsite dose shall be maintained within an acceptable fraction of 10 CFR 50.67 limits. TS 3.3.5 requires all instrumentation performing an ESFAS Function, listed in TS Table 3.3.5-1, to be OPERABLE. TS 3.3.5 applicability is as stated for each ESFAS function listed in TS Table 3.3.5-1).

TS 3.3.7, Diesel Generator (DG) – Undervoltage Start, specifies that the Loss of Voltage Start (LOVS) instrumentation functions be OPERABLE in MODES 1, 2, 3, and 4, and when the associated DG is required to be OPERABLE by TS 3.8.2, AC Sources – Shutdown. The DG – LOVS instrumentation is required for the ESF Systems to function during any

accident with a loss of offsite power. Its design basis is that of ESFAS. The required channels of DG – LOVS instrumentation, in conjunction with the ESF systems powered from the DGs, provide unit protection in the event of any of the analyzed accidents discussed in the UFSAR in which a loss of offsite power is assumed. The DG – LOVS instrumentation supports DG Operability. TS 3.3.7 is applicable in MODES 1, 2, 3, and 4 because ESF Functions are designed to provide protection in these MODES. TS 3.3.7 is also applicable whenever the associated DG is required to be OPERABLE by LCO 3.8.2 to ensure that the automatic start of the DG is available when needed.

TS 3.3.11, Post Accident Monitoring Instrumentation (PAMI), provides OPERABILITY requirements for Regulatory Guide 1.97 Type A monitors, which provide information required by the control room operators to perform certain manual actions specified in the unit Emergency Operating Procedures. These manual actions ensure that a system can accomplish its safety function, and are credited in the safety analyses. Additionally, TS 3.3.11 addresses Regulatory Guide 1.97 instruments that have been designated Category 1, non-Type A. The OPERABILITY of the PAMI ensures there is sufficient information available on selected unit parameters to monitor and assess unit status following an accident. The specific instrument Functions are listed in TS Table 3.3.11-1. TS 3.3.11 is applicable in MODES 1, 2, and 3.

TS 3.3.12, Remote Shutdown System, provides the OPERABILITY requirements for the instrumentation and controls necessary to place and maintain the unit in MODE 3 from a location other than the control room. The instrumentation and controls required are listed in TS Table 3.3.12-1. TS 3.3.12 is applicable in MODES 1, 2, and 3.

3.2.6.1.2 Proposed Changes

TS 3.3.1, TS 3.3.2, TS 3.3.3, TS 3.3.4, TS 3.3.5, TS 3.3.6, TS 3.3.7, TS 3.3.11, TS 3.3.12, and TS 3.3.13 do not currently apply with the reactor defueled. The SONGS Unit 2 and Unit 3 Part 50 licenses no longer authorize emplacement or retention of fuel in their respective reactor vessel. Therefore, these instrumentation requirements do not apply and are being proposed for deletion.

3.2.6.1.3 Technical Analysis

TS 3.3.1, TS 3.3.2, TS 3.3.3, TS 3.3.4, TS 3.3.5, TS 3.3.6, TS 3.3.11, and TS 3.3.12 are related to assuring the appropriate functional capability of plant equipment, and control of process variables, design features, or operating restrictions required for safe operation of the facility only when the reactor is in MODES 1 through 5. However, 10 CFR 50.82(a)(2) prohibits operation of the plant or placing fuel in the reactor vessel. Therefore, TS 3.3.1, TS 3.3.2, TS 3.3.3, TS 3.3.4, TS 3.3.5, TS 3.3.6, TS 3.3.11, and TS 3.3.12, which only address these specific plant systems, control of process variables, design features, or operating restrictions are no longer applicable and may be deleted.

As discussed in the justification for deleting TS 3.8.2, the requirement for DGs is being deleted from the TS because, after the plant is permanently shut down and defueled, there are no design basis events that rely on the DGs for mitigation. Since TS 3.3.7 exists solely to support DG OPERABILITY, the elimination of the need for DGs also obviates the need for their support systems. Since DG – LOVS instrumentation is no longer needed, TS 3.3.7 may be deleted.

3.2.6.2 TS 3.3.8, Containment Purge Isolation Signal

3.2.6.2.1 Description

TS 3.3.8, Containment Purge Isolation Signal, specifies requirements for instrumentation designed to close the containment purge isolation valves upon a detection of high gaseous radiation in containment. This action isolates the containment atmosphere from the environment to minimize releases of radioactivity in the event of an accident. TS 3.3.8 is applicable in MODES 1, 2, 3, and 4, during CORE ALTERATIONS, and during movement of fuel assemblies within containment.

3.2.6.2.2 Proposed Changes

The SONGS Unit 2 and Unit 3 Part 50 licenses no longer authorize emplacement or retention of fuel in their respective reactor vessel. Therefore, these instrumentation requirements do not apply and are being proposed for deletion.

3.2.6.2.3 Technical Analysis

TS 3.3.8 is related to assuring the appropriate functional capability of plant equipment, and control of process variables, design features, or operating restrictions required for safe operation of the facility only when the reactor is in MODES 1 through 4, during CORE ALTERATIONS, and during movement of fuel assemblies within containment. However, 10 CFR 50.82(a)(2) prohibits operation of the plant or placing fuel in the reactor vessel, which thereby precludes entry into MODES 1 through 4. The prohibition on placing fuel in the reactor vessel also precludes CORE ALTERATIONS and the movement of fuel assemblies within containment. Therefore, TS 3.3.8, which only address specific plant systems, control of process variables, design features, or operating restrictions associated with the containment is no longer applicable and may be deleted.

3.2.6.3 TS 3.3.9, Control Room Isolation Signal (CRIS)

3.2.6.3.1 Description

TS 3.3.9, Control Room Isolation Signal (CRIS), specifies requirements to ensure that instrumentation necessary to initiate Control Room Emergency Air Cleanup System (CREACUS) is OPERABLE. TS 3.3.9 requires one channel of CRIS to be OPERABLE. The required channel consists of Actuation Logic, Manual Trip, and gaseous radiation monitors. The CRIS terminates the normal supply of outside air to the control room and initiates actuation of the CREACUS to minimize operator radiation exposure. The CRIS includes two independent, redundant trains. Each train consists of a gaseous radiation monitor, manual trip function and actuation logic. If the bistable monitoring either sensor indicates an unsafe condition, that train will be actuated (one-out-of-two logic). Each train related actuation signal operates the same train isolation equipment. Actuating either train will perform the intended function. The radiation monitor actuation of the CREACUS is a backup for the Safety Injection Actuation Signal (SIAS). This ensures initiation of the CREACUS during a DBA when an initiation of SIAS is anticipated. The radiation monitor actuation of the CREACUS in MODES 5 and 6 and during movement of fuel assemblies is the primary means to ensure control room habitability in the event of a fuel handling accident. TS 3.3.9 is applicable in MODES 1, 2, 3, 4, 5, 6, and during movement of fuel assemblies within containment and in the fuel storage pool.

3.2.6.3.2 Proposed Changes

Because 10 CFR 50.82(a)(2) prohibits operation of the plant or placing fuel in the reactor vessel, TS 3.7.11 is no longer needed for assuring the appropriate functional capability of the CREACUS, including the CRIS, for safe operation of the facility when the reactor is in MODES 1 through 6.

3.2.6.3.3 Technical Analysis

As discussed in the justification for deleting TS 3.7.11, CREACUS, the requirement for the CRIS is being deleted from the TS because it is not required for providing airborne radiological protection for the control room operators in the event of a design basis event (fuel handling accident). Since TS 3.3.9 exists solely to support CREACUS OPERABILITY, the elimination of the need for the CREACUS also obviates the need for its support systems. Since CRIS is no longer needed, TS 3.3.9 may be deleted.

3.2.6.4 Summary

TS 3.3.1, TS 3.3.2, TS 3.3.3, TS 3.3.4, TS 3.3.5, TS 3.3.6, TS 3.3.11, and TS 3.3.12 do not apply with the reactor defueled. Therefore, they are not needed for a permanently defueled condition. As such, they may be deleted with no impact on continued safe operation of the facility.

TS 3.3.8 and TS 3.3.9 are not needed for accident mitigation in the permanently defueled condition. As such, they may be deleted with no impact on continued safe operation of the facility.

3.2.7 TS Section 3.4, Reactor Coolant Systems

The existing TS Section 3.4, Reactor Coolant Systems, contains LCOs that provide for appropriate control of process variables, design features, or operating restrictions needed for appropriate functional capability of RCS equipment required for safe operation of the facility. This Section is divided into the following Specifications.

TS Being Deleted	TS Being Retained
3.4 REACTOR COOLANT SYSTEM (RCS)	
3.4.1 RCS DNB Pressure, Temperature, and Flow Limits	
3.4.2 RCS Minimum Temperature for Criticality	
3.4.3 RCS Pressure and Temperature (P/T) Limits	
3.4.3.1 Pressurizer Heatup/Cooldown Limits	
3.4.4 RCS Loops – MODES 1 and 2	
3.4.5 RCS Loops – MODE 3	
3.4.6 RCS Loops – MODE 4	
3.4.7 RCS Loops – MODE 5, Loops Filled	
3.4.8 RCS Loops – MODE 5, Loops Not Filled	
3.4.9 Pressurizer	
3.4.10 Pressurizer Safety Valves	
3.4.11 Not Used	
3.4.12.1 Low Temperature Overpressure Protection System, RCS Temperature ≤ PTLR Limit	
3.4.12.2 Low Temperature Overpressure Protection System, RCS Temperature > PTLR Limit	
3.4.13 RCS Operational LEAKAGE	
3.4.14 RCS Pressure Isolation Valve (PIV) Leakage	
3.4.15 RCS Leakage Detection Instrumentation	
3.4.16 RCS Specific Activity	
3.4.17 RCS Steam Generator (SG) Tube Integrity	

All TS in Section 3.4 are being proposed for deletion, as identified in the table above. The corresponding TS Bases are also being deleted to reflect this change.

3.2.7.1 Section 3.4 TS That Are Not Applicable When Defueled

None of the TS in Section 3.4, except for TS 3.4.3 and TS 3.4.3.1, currently apply with the reactor defueled.

3.2.7.1.1 Description

TS 3.4.1, RCS DNB (Pressure, Temperature, and Flow) Limits, specifies process variables requirements for maintaining RCS pressure, temperature, and flow rate within limits assumed in the safety analyses. The limits placed on RCS pressure, temperature, and flow rate ensure that the minimum departure from nucleate boiling ratio (DNBR) will be met for each of the analyzed transients. TS 3.4.1 is applicable in MODE 1.

TS 3.4.2, RCS Minimum Temperature for Criticality, specifies requirements for RCS loop cold leg temperature (T_c) before the reactor can be made critical and while the reactor is critical. Compliance with the LCO ensures that the reactor will not be made or maintained critical ($k_{eff} \geq 1.0$) outside a temperature operating range of 522°F to 558°F, and to prevent operation in an unanalyzed condition. TS 3.4.2 is applicable in MODE 1, THERMAL POWER ≤ 30% RTP and $T_c < 535$°F, and in MODE 2 with $k_{eff} \geq 1.0$ and $T_c < 535$°F.

TS 3.4.4, RCS Loops – MODES 1 and 2, specifies requirements to ensure heat removal capability of the RCS loops with the reactor in MODES 1 and 2. The primary function of the RCS is removal of the heat generated in the fuel due to the fission process, and transfer of this heat, via the steam generators (SGs), to the secondary plant. The important aspect for this LCO is the reactor coolant forced flow rate, which is represented by the number of RCS loops in service. TS 3.4.4 is applicable in MODES 1 and 2.

TS 3.4.5, RCS Loops – MODE 3, specifies requirements to ensure heat removal capability of the RCS loops with the reactor in MODE 3. In MODE 3, the primary function of the reactor coolant is removal of decay heat and transfer of this heat, via the SGs, to the secondary plant fluid. TS 3.4.5 is applicable in MODE 3.

TS 3.4.6, RCS Loops – MODE 4, specifies requirements to ensure heat removal capability of the RCS loops with the reactor in MODE 4. In MODE 4, the primary function of the reactor coolant is the removal of decay heat and the transfer of this heat to the SGs or shutdown cooling (SDC) heat exchangers. TS 3.4.6 is applicable in MODE 4.

TS 3.4.7, RCS Loops – MODE 5, Loops Filled, specifies requirements to ensure heat removal capability of the RCS loops with the reactor in MODE 5 with the RCS loops filled with coolant. In MODE 5 with the RCS loops filled, the primary function of the reactor coolant is the removal of decay heat and transfer of this heat to the SGs or SDC heat exchangers. While the principal means for decay heat removal is via the SDC heat exchangers, the SGs are specified as a backup means for redundancy. TS 3.4.7 is applicable in MODE 5 with the RCS loops filled.

TS 3.4.8, RCS Loops – MODE 5, Loops Not Filled, specifies requirements to ensure heat removal capability of the RCS loops with the reactor in MODE 5 with the RCS loops not filled with coolant. In MODE 5 with the RCS loops not filled, the primary function of the

reactor coolant is the removal of decay heat and transfer of this heat to the SDC heat exchangers. The SGs are not available as a heat sink when the loops are not filled. TS 3.4.8 is applicable in MODE 5 with the RCS loops not filled.

TS 3.4.9, Pressurizer, specifies OPERABILITY requirements for the RCS pressurizer. The pressurizer provides a point in the RCS where liquid and vapor are maintained in equilibrium under saturated conditions for pressure control purposes to prevent bulk boiling in the remainder of the RCS. In MODES 1, 2, and 3, the LCO requirement for a steam bubble is reflected implicitly in the accident analyses. All analyses performed from a critical reactor condition assume the existence of a steam bubble and saturated conditions in the pressurizer. TS 3.4.9 is applicable in MODES 1, 2, and 3.

TS 3.4.10, Pressurizer Safety Valves, specifies OPERABILITY and lift setpoint parameters for the pressurizer safety valves. The pressurizer safety valves provide, in conjunction with the reactor protection system, overpressure protection for the RCS. The pressurizer safety valves are designed to prevent the system pressure from exceeding the system Safety Limit of 2750 psia in MODES 1, 2, and 3. In MODES 4, 5, and MODE 6 with the reactor vessel head on, overpressure protection is provided by operating procedures and by meeting the requirements of LCO 3.4.12, Low Temperature Overpressure Protection (LTOP) System. TS 3.4.10 is applicable in MODES 1, 2, and 3.

TS 3.4.12.1, Low Temperature Overpressure Protection (LTOP) System, RCS Temperature ≤ PTLR Limit, specifies requirements for controlling RCS pressure at low temperatures so the integrity of the RCPB is not compromised by violating the pressure and temperature (P/T) limits of 10 CFR 50, Appendix G. TS LCO 3.4.12.1 provides RCS overpressure protection by minimizing coolant input capability and having adequate pressure relief capacity. In MODES 1, 2, and in MODE 3, the pressurizer safety valves will prevent RCS pressure from exceeding limits. In MODE 4 when the temperature of any RCS cold leg is less than or equal to the enable temperature specified in the PTLR, MODE 5, and MODE 6 when the reactor vessel head is on and the RCS is not vented, overpressure prevention falls to the OPERABLE SDC System relief valve or to a depressurized RCS and a sufficient sized RCS vent. When the reactor vessel head is off, overpressurization cannot occur.

TS 3.4.12.2, Low Temperature Overpressure Protection (LTOP) System, RCS Temperature > PTLR Limit, specifies requirements for controlling RCS pressure at low temperatures so the integrity of the RCPB is not compromised by violating the pressure and temperature (P/T) limits of 10 CFR 50, Appendix G. TS LCO 3.4.12.2 provides RCS overpressure protection by having adequate pressure relief capacity. In MODES 1, 2, and in MODE 3, the pressurizer safety valves will prevent RCS pressure from exceeding limits. In MODE 4 when the temperature of all RCS cold legs are greater than the enable temperature specified in the PTLR, overpressure prevention falls to the OPERABLE SDC System relief valve or to an OPERABLE pressurizer code safety valve.

TS 3.4.13, RCS Operational LEAKAGE, specifies process variable limits and operating restrictions for RCS pressure boundary leakage, unidentified RCS leakage, identified RCS leakage, and primary to secondary leakage. RCS leakage is indicative of material deterioration, possibly of the RCS pressure boundary, which can affect the probability of a design basis event. The primary to secondary leakage limit is based on operating experience with SG tube degradation mechanisms that result in tube leakage. The operational leakage rate criterion in conjunction with the implementation of the Steam

Generator Program is an effective measure for minimizing the frequency of steam generator tube ruptures. TS 3.4.13 is applicable in MODES 1, 2, 3, and 4. In MODES 5 and 6, leakage limits are not required because the reactor coolant pressure is far lower, resulting in lower stresses and reduced potential for leakage.

TS 3.4.14, RCS Pressure Isolation Valve (PIV) Leakage, specifies process variable limits and operating restrictions for RCS PIV leakage. 10 CFR 50.2, 10 CFR 50.55a(c), and 10 CFR 50, Appendix A, GDC 55, discuss reactor coolant pressure boundary valves, which are normally closed valves in series within the RCPB that separate the high pressure RCS from an attached low pressure system. Although this specification provides a limit on allowable PIV leakage rate, its main purpose is to prevent overpressure failure of the low pressure portions of connecting systems (intersystem LOCA). PIVs are provided to isolate the RCS from the following typically connected systems: SDC System; Safety Injection System; and the Chemical and Volume Control System. The leakage limit is an indication that the PIVs between the RCS and the connecting systems are degraded or degrading. TS 3.4.14 is applicable in MODES 1, 2, 3, and 4. In MODES 5 and 6, leakage limits are not provided because the lower reactor coolant pressure results in a reduced potential for leakage and for a LOCA outside the containment.

TS 3.4.15, RCS Leakage Detection Instrumentation, specifies OPERABILITY requirements for RCS leakage detection instrumentation. Leakage detection systems are provided to detect significant RCPB degradation as soon after occurrence as practical to minimize the potential for propagation to a gross failure. Thus, they provide an early indication or warning signal to permit proper evaluation of RCS leakage into the containment area. TS LCO 3.4.15 requires instruments of diverse monitoring principles to be OPERABLE to provide a high degree of confidence that extremely small leaks are detected in time to allow actions to place the plant in a safe condition, when RCS leakage indicates possible RCPB degradation. TS 3.4.15 is applicable in MODES 1, 2, 3, and 4.

TS 3.4.16, RCS Specific Activity, specifies process variable limits and operating restrictions for Dose Equivalent I-131 and gross specific activity. The TS LCO limits on the specific activity of the reactor coolant ensure that the resulting offsite doses meet the appropriate RG 1.183 acceptance criteria following a steam generator tube rupture (SGTR) accident. TS 3.4.16 is applicable in MODES 1, 2, and MODE 3 with RCS average temperature ≥ 500°F.

TS 3.4.17, Steam Generator (SG) Tube Integrity, specifies requirements to ensure the RCPB integrity function of the SG. The SGTR accident is the limiting design basis event for SG tubes and avoiding an SGTR is the basis for this Specification. TS 3.4.17 is applicable in MODES 1, 2, 3, and 4.

3.2.7.1.2 Proposed Changes

The TS above do not currently apply with the reactor defueled. The SONGS Unit 2 and Unit 3 Part 50 licenses no longer authorize emplacement or retention of fuel in their respective reactor vessel. Therefore, these reactor coolant system requirements do not apply and are being proposed for deletion.

3.2.7.1.3 Technical Analysis

The above TS are related to assuring the appropriate functional capability of plant

equipment, and control of process variables, design features, or operating restrictions required for safe operation of the facility only when the reactor is in MODES 1 through 6. However, 10 CFR 50.82(a)(2) prohibits operation of the plant or placing fuel in the reactor vessel. Therefore, the TS listed in the previous paragraphs, which address plant equipment associated with the reactor coolant system, are no longer applicable.

3.2.7.2 TS 3.4.3, RCS Pressure and Temperature (P/T) Limits, and TS 3.4.3.1, Pressurizer Heatup/Cooldown Limits

3.2.7.2.1 Description

TS 3.4.3, RCS Pressure and Temperature (P/T) Limits, specifies that the RCS pressure, RCS temperature and RCS heatup and cooldown rates shall be maintained within the limits as specified in the Pressure – Temperature Limits Report (PTLR). The LCO establishes operating limits that provide a margin to brittle failure of the reactor vessel and piping of the RCPB. TS 3.4.3 is applicable at all times.

The purpose for TS LCO 3.4.3 during normal operation of the RCS is to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the reactor coolant pressure boundary, an unanalyzed condition. The RCS P/T limits in LCO 3.4.3 provide a definition of acceptable operation for prevention of nonductile failure in accordance with 10 CFR 50, Appendix G. Although the P/T limits were developed to provide guidance for operation during heatup, or cooldown, or ISLH testing, their Applicability is at all times in keeping with the concern for nonductile failure.

TS 3.4.3.1, Pressurizer Heatup/Cooldown Limits, requires that the pressurizer heatup and cooldown rates shall be maintained within the specified limits. The pressurizer is designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation. Therefore, TS 3.4.3.1 is applicable at all times.

3.2.7.2.2 Proposed Changes

The SONGS Unit 2 and Unit 3 Part 50 licenses no longer authorize emplacement or retention of fuel in their respective reactor vessel. Therefore, these pressure and temperature limit requirements do not apply and are being proposed for deletion.

3.2.7.2.3 Technical Analysis

10 CFR 50.82, Termination of license, states that, "upon docketing of the certifications for permanent cessation of operations and permanent removal of fuel from the reactor vessel..., the 10 CFR part 50 license no longer authorizes operation of the reactor or emplacement or retention of fuel into the reactor vessel." As such, the requirements of 10 CFR 50, Appendix G, no longer apply in such a condition because the RCPB will no longer be used as a fission product barrier when the reactor vessel is permanently defueled. Therefore, TS 3.4.3 is no longer needed and may be deleted. Similarly, operating the unit within the fatigue analysis performed in accordance with the ASME

Code Section III requirements no longer applies. Therefore, TS 3.4.3.1 is no longer needed and may be deleted.

3.2.7.3 Summary

TS 3.4.3 and TS 3.4.3.1 do not apply with the reactor defueled. Therefore, they are not needed for a permanently defueled condition. As such, they may be deleted with no impact on continued safe operation of the facility.

3.2.8 TS Section 3.5, Emergency Core Cooling Systems (ECCS)

The existing TS Section 3.5, Emergency Core Cooling Systems (ECCS), contains LCOs that provide for appropriate functional capability of ECCS equipment required for mitigation of design basis accidents or transients so as to protect the integrity of a fission product barrier. This Section is divided into the following Specifications.

TS Being Deleted	TS Being Retained
3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)	
3.5.1 Safety Injection Tanks (SITs)	
3.5.2 ECCS – Operating	
3.5.3 ECCS – Shutdown	
3.5.4 Refueling Water Storage Tank (RWST)	
3.5.5 Trisodium Phosphate (TSP) Dodecahydrate	

All TS in Section 3.5 are being proposed for deletion, as identified in the table above. The corresponding TS Bases are also being deleted to reflect this change.

None of the TS in Section 3.5 currently apply with the reactor defueled.

3.2.8.1 Description

TS 3.5.1, Safety Injection Tanks (SITs), specifies requirements for the safety injection tanks to ensure they are capable of supplying water to the reactor vessel during the blowdown phase of a LOCA, to provide inventory to help accomplish the refill phase that follows thereafter, and to provide reactor coolant system (RCS) makeup for a small break LOCA. TS 3.5.1 is applicable in MODES 1 and 2, and in MODE 3 with pressurizer pressure ≥ 715 psia.

TS 3.5.2, ECCS – Operating, specifies requirements for the emergency core cooling system (ECCS) trains so as to provide core cooling and negative reactivity to ensure that the reactor core is protected after a LOCA, CEA ejection accident, loss of secondary coolant accident (including uncontrolled steam release), and SGTR. The ECCS consists of the high pressure safety injection (HPSI) and the low pressure safety injection (LPSI) subsystems. TS 3.5.2 is applicable in MODES 1 and 2, and MODE 3 with pressurizer pressure ≥ 400 psia.

TS 3.5.3, ECCS – Shutdown, specifies requirements for ECCS with the reactor in MODE 3 with pressurizer pressure < 400 psia, and in MODE 4. In these MODES, an ECCS train is composed of a single HPSI subsystem. One OPERABLE ECCS train is acceptable without single failure consideration, based on the stable reactivity condition of the reactor and the limited core cooling requirements.

TS 3.5.4, Refueling Water Storage Tank (RWST), specifies requirements for RWST

OPERABILITY. During accident conditions, the RWST provides a source of borated water to the ECCS and Containment Spray System pumps. As such, it provides containment cooling and depressurization, core cooling, and replacement inventory and is a source of negative reactivity for reactor shutdown. TS 3.5.4 is applicable in MODES 1, 2, 3, and 4 because RWST OPERABILITY requirements are dictated by ECCS and Containment Spray System OPERABILITY requirements. Since both the ECCS and the Containment Spray System must be OPERABLE in MODES 1, 2, 3, and 4, the RWST must also be OPERABLE to support their operation.

TS 3.5.5, Trisodium Phosphate (TSP) Dodecahydrate, specifies the requirements for TSP crystals to be placed in baskets on the floor of the containment building to ensure that iodine, which may be dissolved in the recirculated reactor cooling water following a LOCA, remains in solution. TSP also helps inhibit stress corrosion cracking (SCC) of austenitic stainless steel components in containment during the recirculation phase following an accident. TS 3.5.5 is applicable in MODES 1, 2, and 3, when the RCS is at elevated temperature and pressure, providing an energy potential for a LOCA.

3.2.8.2 Proposed Changes

All TS in Section 3.5 do not currently apply with the reactor defueled. The SONGS Unit 2 and Unit 3 Part 50 licenses no longer authorize emplacement or retention of fuel in their respective reactor vessel. Therefore, these reactor coolant system requirements do not apply and are being proposed for deletion.

3.2.8.3 Technical Analysis

All TS in Section 3.5 are related to assuring the appropriate functional capability of ECCS required for mitigation of design basis accidents only when the reactor is in MODES 1 through 4. However, 10 CFR 50.82(a)(2) prohibits operation of the plant or placing fuel in the reactor vessel. Therefore, the TS listed in the previous paragraphs, which address the ECCS, are no longer applicable.

3.2.8.4 Summary

All TS in Section 3.5 do not apply with the reactor defueled. Therefore, it is not needed for a permanently defueled condition. As such, they may be deleted with no impact on continued safe operation of the facility.

3.2.9 TS Section 3.6, Containment Systems

The existing TS Section 3.6, Containment Systems, contains LCOs that provide for appropriate control of process variables, design features, or operating restrictions required to protect the integrity of a fission product barrier; and appropriate functional capability of engineered safety features (ESF) equipment required for mitigation of design basis accidents or transients so as to protect the integrity of a fission product barrier. This Section is divided into the following Specifications.

TS Being Deleted	TS Being Retained
3.6 CONTAINMENT SYSTEMS	
3.6.1 Containment	
3.6.2 Containment Air Locks	
3.6.3 Containment Isolation Valves	
3.6.4 Containment Pressure	
3.6.5 Containment Air Temperature	
3.6.6.1 Containment Spray and Cooling Systems	
3.6.6.2 Containment Cooling System	
3.6.7 Not Used	
3.6.8 Containment Dome Air Circulators	

All TS in Section 3.6 are being proposed for deletion, as identified in the table above. The corresponding TS Bases are also being deleted to reflect this change.

None of the TS in Section 3.6 currently apply with the reactor defueled.

3.2.9.1 Description

TS 3.6.1, Containment, specifies requirements for the containment to ensure it is capable of withstanding the pressures and temperatures of the limiting Design Basis Accident (DBA) without exceeding the design leakage rate. The containment is a reinforced concrete structure with a cylindrical wall, a flat foundation mat, and a shallow dome roof. The cylinder wall is prestressed with a post tensioning system in the vertical and horizontal directions, and the dome roof is prestressed utilizing a three way post tensioning system. The inside surface of the containment is lined with a carbon steel liner to ensure a high degree of leak tightness during operating and accident conditions. The concrete reactor building is required for structural integrity of the containment under DBA conditions. The steel liner and its penetrations establish the leakage limiting boundary of the containment. Maintaining the containment OPERABLE limits the leakage of fission product radioactivity from the containment to the environment. TS 3.6.1 is applicable in MODES 1, 2, 3, and 4. In MODE 5 and 6, the probability and consequences of a release are reduced due to the pressure and temperature limitations of these MODES.

TS 3.6.2, Containment Air Locks, specifies requirements for the structural integrity and leak tightness of the containment air locks. As part of the containment pressure boundary, the air lock safety function is related to control of the containment leakage rate resulting from a DBA. Thus, each containment air lock's structural integrity and leak tightness is essential to the successful mitigation of such an event. TS 3.6.2 is applicable in MODES 1, 2, 3, and 4, consistent with the applicability requirement of TS 3.6.1, Containment.

TS 3.6.3, Containment Isolation Valves, specifies requirements for the isolation capability of the containment via the containment isolation valves. Containment isolation valves form a part of the containment boundary and their OPERABILITY supports leak tightness of the containment. TS 3.6.3 is applicable in MODES 1, 2, 3, and 4, consistent with the applicability requirement of TS 3.6.1, Containment.

TS 3.6.4, Containment Pressure, specifies limitations on internal containment pressure. Containment internal pressure is an initial condition used in the design basis accident (DBA) analyses to establish the maximum peak containment internal pressure. Maintaining containment pressure at less than or equal to the LCO upper pressure limit ensures that, in the event of a DBA, the resultant peak containment accident pressure will remain below the maximum allowed containment internal pressure. Maintaining containment pressure at greater than or equal to the LCO lower pressure limit ensures that the containment will not exceed the design negative differential pressure following the accidental actuation of the Containment Spray System. TS 3.6.4 is applicable in MODES 1, 2, 3, and 4.

TS 3.6.5, Containment Air Temperature, specifies limitations on containment average air temperature. Containment average air temperature is an initial condition used in the DBA analyses that establishes the containment environmental qualification operating envelope for both pressure and temperature. During a DBA, with an initial containment average air temperature less than or equal to the LCO temperature limit, the resultant accident temperature profile assures that the containment structural temperature is maintained below its design temperature and that required safety related equipment will continue to perform its function. TS 3.6.5 is applicable in MODES 1, 2, 3, and 4.

TS 3.6.6.1, Containment Spray and Cooling Systems, specifies OPERABILITY requirements for containment atmosphere cooling to limit post accident pressure and temperature in containment to less than the design values. Reduction of containment pressure and the iodine removal capability of the spray reduce the release of fission product radioactivity from containment to the environment, in the event of a DBA, to within limits. The Containment Spray System consists of two separate trains. Each train includes a containment spray pump, spray headers, valves and piping. The refueling water storage tank (RWST) supplies borated water to the Containment Spray System during the injection phase of operation. In the recirculation mode of operation, containment spray pump suction is transferred from the RWST to the containment sump. Two trains of containment cooling, each of sufficient capacity to supply 50% of the design cooling requirement, are provided. Two trains with two fan units each are supplied with cooling water from the Component Cooling Water System. All four fans are required to furnish the design cooling capacity. Air is drawn into the coolers through the fans and discharged to the steam generator compartments and pressurizer compartment. TS 3.6.6.1 is applicable in MODES 1, 2, and 3. In MODES 5 and 6, the probability and

consequences of a release are reduced due to the pressure and temperature limitations of these MODES. Thus, the Containment Spray System and the Containment Cooling System are not required to be OPERABLE in MODES 5 and 6.

TS 3.6.6.2, Containment Cooling Systems, specifies OPERABILITY requirements for containment atmosphere cooling to limit post accident pressure and temperature in containment to less than the design values. Reduction of containment pressure reduces the release of fission product radioactivity from containment to the environment, in the event of a DBA, to within limits. Two trains of containment cooling, each of sufficient capacity to supply 50% of the design cooling requirement, are provided. Two trains with two fan units each are supplied with cooling water from the Component Cooling Water System. All four fans are required to furnish the design cooling capacity. Air is drawn into the coolers through the fans and discharged to the steam generator compartments and pressurizer compartment. TS 3.6.6.2 is applicable in MODE 4, when a DBA could cause a release of radioactive material to containment and an increase in containment pressure and temperature, requiring the operation of the containment cooling trains.

TS 3.6.8, Containment Dome Air Circulators, specifies requirements for the containment dome air circulators to reduce the potential for breach of the containment due to a hydrogen oxygen reaction. The dome air circulators accelerate the air mixing process between the upper dome space of the containment atmosphere during LOCA operations. They also prevent any hot spot air pockets during the containment cooling mode and avoid any hydrogen concentration in pocket areas. Two dome air circulator trains are required to be OPERABLE. Each train consists of two fans with their own motors and controls and is automatically initiated by a containment cooling actuation signal (CCAS). While each train has two fans, only one OPERABLE fan is required for the train to be OPERABLE, since each fan can provide the necessary flow rate to adequately mix the containment atmosphere. TS 3.6.8 is applicable in MODES 1 and 2.

3.2.9.2 Proposed Changes

All TS in Section 3.6 do not currently apply with the reactor defueled. The SONGS Unit 2 and Unit 3 Part 50 licenses no longer authorize emplacement or retention of fuel in their respective reactor vessel. Therefore, these containment system requirements do not apply and are being proposed for deletion.

3.2.9.3 Technical Analysis

All TS in Section 3.6 are related to assuring the appropriate functional capability of plant equipment associated with containment systems required for safe operation of the facility and accident mitigation only when the reactor is in MODES 1 through 4. However, 10 CFR 50.82(a)(2) prohibits operation of the plant or placing fuel in the reactor vessel. Therefore, the TS listed in the previous paragraphs, which only address containment systems, are no longer applicable.

3.2.9.4 Summary

All TS in Section 3.6 do not apply with the reactor defueled. Therefore, it is not needed for a permanently defueled condition. As such, they may be deleted with no impact on continued safe operation of the facility.

3.2.10 TS Section 3.7, Plant Systems

The existing TS Section 3.7, "Plant Systems," contains LCOs that provide for appropriate functional capability of plant equipment required for safe operation of the facility, including the plant being in a defueled condition. This Section is divided into the following Specifications.

TS Being Deleted	TS Being Retained
3.7 PLANT SYSTEMS	
3.7.1 Main Steam Safety Valves (MSSVs)	
3.7.2 Main Steam Isolation Valves (MSIVs)	
3.7.3 Main Feedwater Isolation Valves (MFIVs)	
3.7.4 Atmospheric Dump Valves (ADVs)	
3.7.5 Auxiliary Feedwater (AFW) System	
3.7.6 Condensate Storage Tank (CST T-120 and T-121)	
3.7.7 Component Cooling Water (CCW) System	
3.7.7.1 Component Cooling Water (CCW) Safety Related Makeup System	
3.7.8 Salt Water Cooling (SWC) System	
3.7.9 Not Used	
3.7.10 Emergency Chilled Water (ECW)	
3.7.11 Control Room Emergency Air Cleanup System (CREACUS)	
3.7.12 Not Used	
3.7.13 Not Used	
3.7.14 Not Used	
3.7.15 Not Used	
	3.7.16 Fuel Storage Pool Water Level
	3.7.17 Fuel Storage Pool Boron Concentration
	3.7.18 Spent Fuel Assembly Storage
3.7.19 Secondary Specific Activity	

The following TS in Section 3.7 are being proposed for deletion, as identified in the table above: TS 3.7.1, TS 3.7.2, TS 3.7.3, TS 3.7.4, TS 3.7.5, TS 3.7.6, TS 3.7.7, TS 3.7.7.1,

TS 3.7.8, TS 3.7.10, TS 3.7.11, and TS 3.7.19. The corresponding TS Bases Sections are also being deleted to reflect this change.

TS being retained and revised are TS 3.7.16, TS 3.7.17, and TS 3.7.18 as further described below and Shown in Attachments 1 and 2. The corresponding TS Bases Sections are also being revised to reflect this change.

3.2.10.1 Section 3.7 TS That Are Not Applicable When Defueled

TS 3.7.1, TS 3.7.2, TS 3.7.3, TS 3.7.4, TS 3.7.5, TS 3.7.6, TS 3.7.7, TS 3.7.7.1, TS 3.7.8, TS 3.7.10, and TS 3.7.19 currently do not apply with the reactor defueled.

3.2.10.1.1 Description

TS 3.7.1, Main Steam Safety Valves (MSSVs), specifies requirements for the MSSVs to ensure they are capable of providing overpressure protection for the secondary system. The MSSVs also provide protection against overpressurizing the RCPB by providing a heat sink for the removal of energy from the RCS, if the preferred heat sink, provided by the Condenser and Circulating Water System, is not available. TS 3.7.1 is applicable in MODES 1, 2, and 3.

TS 3.7.2, Main Steam Isolation Valves (MSIVs), specifies requirements for the MSIVs to ensure that they are capable of isolating steam flow from the secondary side of the steam generators following a high energy line break (HELB). MSIV closure terminates flow from the unaffected (intact) steam generator. One MSIV is located in each main steam line outside, but close to, containment. The MSIVs are downstream from the main steam safety valves (MSSVs), atmospheric dump valves, and auxiliary feedwater pump turbine steam supplies to prevent them from being isolated from the steam generators by MSIV closure. Closing the MSIVs isolates each steam generator from the other, and isolates the turbine, Steam Bypass System, and other auxiliary steam supplies from the steam generators. TS 3.7.2 is applicable in MODE 1, and in MODES 2 and 3 except when all MSIVs are closed and deactivated.

TS 3.7.3, Main Feedwater Isolation Valves (MFIVs), specifies requirements for these valves. The MFIVs isolate main feedwater (MFW) flow to the secondary side of the steam generators following a HELB. Closure of the MFIVs terminates flow to the steam generators, terminating the event for feedwater line breaks (FWLBs) occurring upstream of the MFIVs. The consequences of events occurring in the main steam lines or in the MFW lines downstream of the MFIVs will be mitigated by their closure. Closure of the MFIVs effectively terminates the addition of feedwater to an affected steam generator, limiting the mass and energy release for steam line breaks (SLBs) or FWLBs inside containment, and reducing the cooldown effects for SLBs. TS 3.7.3 is applicable in MODES 1, 2, and 3 except when MFIV is closed and deactivated.

TS 3.7.4, Atmospheric Dump Valves (ADVs), specifies requirements for providing a method for cooling the unit to SDC System entry conditions, should the preferred heat sink via the Steam Bypass System to the condenser not be available. This is done in conjunction with the Auxiliary Feedwater (AFW) System providing cooling water from the condensate storage tank (CST). TS 3.7.4 is applicable in MODES 1, 2, and 3, and in MODE 4 when steam generator is relied upon for heat removal.

TS 3.7.5, Auxiliary Feedwater (AFW) System, specifies requirements to ensure that the AFW System automatically supplies feedwater to the steam generators to remove decay heat from the RCS upon the loss of normal feedwater supply. TS 3.7.5 is applicable in MODES 1, 2, and 3, and in MODE 4 when steam generator is relied upon for heat removal.

TS 3.7.6, Condensate Storage Tank (CST T-120 and T-121), specifies requirements to ensure a safety grade source of water to the steam generators for removing decay and sensible heat from the RCS. The CSTs provide a passive flow of water, by gravity, to the AFW System (TS 3.7.5). TS 3.7.6 is applicable in MODES 1, 2, and 3, and in MODE 4 when steam generator is relied upon for heat removal.

TS 3.7.7, Component Cooling Water (CCW) System, specifies requirements to ensure that the CCW System provides a heat sink for the removal of process and operating heat from safety related components during a DBA or transient. During normal operation, the CCW System also provides this function for various nonessential components. The CCW System serves as a barrier to the release of radioactive byproducts between potentially radioactive systems and the Salt Water Cooling System, and thus to the environment. TS 3.7.7 is applicable in MODES 1, 2, 3, and 4.

TS 3.7.7.1, Component Cooling Water (CCW) Safety Related Makeup System, specifies requirements to ensure a safety related CCW makeup system is available to maintain the water inventory in the CCW trains during a 7-day post-accident period. The safety related makeup system is designed to supply water to the CCW trains following loss of normal CCW makeup from the nuclear service water system. For this purpose, sufficient water inventory is contained in the single primary plant makeup (PPMU) storage tank for both CCW trains. From the PPMU tank, water is transferred to the CCW return heads by two safety related pumps. TS 3.7.7.1 is applicable in MODES 1, 2, 3, and 4.

TS 3.7.8, Salt Water Cooling (SWC) System, specifies requirements to ensure that the SWC System provides a heat sink for the removal of process and operating heat from safety related components during a DBA or transient. During normal operation, and a normal shut down, the SWC System also provides this function for various safety related and nonsafety related components. The safety related function is covered by this TS. TS 3.7.8 is applicable in MODES 1, 2, 3, and 4.

TS 3.7.10, Emergency Chilled Water (ECW) System, specifies requirements to ensure that the ECW System provides a heat sink for the removal of process and operating heat from selected safety related air handling systems during a DBA or transient. The design basis of the ECW System is to remove the post accident heat load from ESF spaces following a DBA coincident with a loss of offsite power. Each train provides chilled water to the HVAC units at the design temperature and flow rate. TS 3.7.10 is applicable in MODES 1, 2, 3, and 4.

TS 3.7.19, Secondary Specific Activity, specifies a limit on secondary coolant specific activity during power operation, to minimize releases to the environment because of normal operation, AOOs, and accidents. The accident analysis of the main steam line break (MSLB) assumes an initial secondary coolant specific activity used for determining the radiological consequences of the postulated accident. The accident analysis, based on this and other assumptions, shows that the radiological consequences of an MSLB do not exceed the total effective dose equivalent (TEDE) limit. TS 3.7.19 is applicable in MODES 1, 2, 3, and 4.

3.2.10.1.2 Proposed Changes

TS 3.7.1, TS 3.7.2, TS 3.7.3, TS 3.7.4, TS 3.7.5, TS 3.7.6, TS 3.7.7, TS 3.7.7.1, TS 3.7.8, TS 3.7.10, and TS 3.7.19 do not currently apply with the reactor defueled. The SONGS Unit 2 and Unit 3 Part 50 licenses no longer authorize emplacement or retention of fuel in their respective reactor vessel. Therefore, these plant system requirements do not apply and are being proposed for deletion.

3.2.10.1.3 Technical Analysis

The above TS are related to assuring the appropriate functional capability of plant equipment required for safe operation of the facility only when the reactor is in MODES 1 through 4. However, 10 CFR 50.82(a)(2) prohibits operation of the plant or placing fuel in the reactor vessel. Therefore, TS 3.7.1, TS 3.7.2, TS 3.7.3, TS 3.7.4, TS 3.7.5, TS 3.7.6, TS 3.7.7, TS 3.7.7.1, TS 3.7.8, TS 3.7.10, and TS 3.7.19 do not apply and are being proposed for deletion.

3.2.10.2 TS 3.7.11, Control Room Emergency Air Cleanup System (CREACUS)

3.2.10.2.1 Description

TS 3.7.11, Control Room Emergency Air Cleanup System (CREACUS), specifies requirements to ensure that the CREACUS provides a protected environment from which operators can control the unit following an uncontrolled release of radioactivity, hazardous chemicals, or smoke.

The CREACUS consists of two independent, redundant trains that recirculate and filter the air in the control room envelop (CRE) and a CRE boundary that limits the inleakage of unfiltered air. Each CREACUS train consists of an emergency air conditioning unit, emergency ventilation air supply unit, emergency isolation dampers, and cooling coils and two cabinet coolers. Each emergency air conditioning unit includes a prefilter, a high efficiency particulate air (HEPA) filter, an activated carbon adsorber section for removal of gaseous activity (principally iodines), and a fan. A second bank of HEPA filters follows the adsorber section to collect carbon fines. Each emergency ventilation air supply unit includes prefilter, HEPA filter, carbon adsorber and fan. Ductwork, motor-operated dampers, doors, barriers, and instrumentation also form part of the system.

Upon receipt of the actuating signal, normal air supply to the CRE is isolated and the stream of ventilation air is recirculated through the system's filter trains. The prefilters remove any large particles in the air to prevent excessive loading of the HEPA filters and charcoal adsorbers. There are two CREACUS operational modes. Emergency mode is an operational mode when the control room is isolated to protect operational personnel from radioactive exposure through the duration of a DBA. Isolation mode is an operational mode when the CRE is isolated to protect operational personnel from toxic gases and smoke.

Actuation of the CREACUS places the system into either of two separate states of operation, depending on the initiation signal. Actuation of the system to either the emergency mode or isolation mode of CREACUS operation closes the unfiltered-outside-air intake and unfiltered exhaust dampers and aligns the system for

recirculation of air within the CRE through the redundant trains of HEPA and charcoal filters.

The emergency mode also initiates pressurization of the CRE. Outside air is added to the air being recirculated from the CRE. Pressurization of the CRE minimizes infiltration of unfiltered air through the CRE boundary from all the surrounding areas adjacent to the CRE boundary. The CRE supply and the outside air supply of the normal control room HVAC are monitored by radiation and toxic-gas detectors, respectively. One detector output above the setpoint will cause actuation of the emergency mode or isolation mode as required. The actions of the isolation mode are more restrictive, and will override the actions of the emergency mode of operation.

In MODES 1, 2, 3, 4, 5, and 6 and during movement of fuel assemblies in the containment or fuel storage pool the CREACUS must be OPERABLE to ensure that the CRE will remain habitable during and following a DBA. In MODES 5 and 6, the CREACUS is required to cope with the release from a rupture of a waste gas tank. During movement of fuel assemblies, the CREACUS must be OPERABLE to cope with the release from a fuel handling accident involving handling fuel.

3.2.10.2.2 Proposed Changes

The SONGS Unit 2 and Unit 3 Part 50 licenses no longer authorize emplacement or retention of fuel in their respective reactor vessel. Furthermore, the waste gas decay tanks have been purged of their contents. Therefore, a rupture of these tanks is no longer an applicable accident. Lastly, the remaining applicable accidents show that the dose consequences are acceptable without relying on any SSCs to remain functional (including the CREACUS) during and following the event. Therefore, TS 3.7.11 no longer applies and is being proposed for deletion.

3.2.10.2.3 Technical Analysis

As stated in the NRC Safety Evaluation associated with the issuance of Amendments 127 and 116 for SONGS Units 2 and 3, respectively (Reference 7), the toxic gas isolation of CREACUS is not relied on to prevent or mitigate a design basis accident or transient because the plant design includes other means to safely shut down the plant if the control room becomes uninhabitable. As such, the toxic gas isolation instrumentation was relocated from the TS and placed in the Licensee Controlled Specifications with an applicability of MODES 1, 2, 3, 4, 5, and 6. However, 10 CFR 50.82(a)(2) prohibits operation of the plant or placing fuel in the reactor vessel. Therefore, the toxic gas isolation of CREACUS is no longer applicable.

The remaining applicable design basis accidents and transients analyzed in UFSAR Chapter 15 are listed in Section 4.1 of this proposed amendment (Applicable Regulatory Requirements/Criteria). A description of each accident with the potential to result in a radiological release is provided in Section 3.1 of this submittal. These analyses show that the dose consequences are acceptable without relying on any SSCs to remain functional (including the CREACUS) during and following the event. As such, the CREACUS is not required for providing airborne radiological protection for the control room operators. Consequently, the CREACUS is not needed during movement of fuel assemblies in the fuel storage pool for mitigation of a potential accident.

The requirement for the CREACUS was previously included in the TS for power operation of the reactor based on Criterion 3 of 10 CFR 50.36(c)(2)(ii)(C), which states that TS limiting conditions for operation must be established for structures, systems, or components (SSCs) that are part of the primary success path and which function or actuate to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Because the remaining applicable accident analyses do not rely on the CREACUS for accident mitigation (including any need for providing airborne radiological protection), the CREACUS is not required during the movement of fuel assemblies in the fuel storage pool. There are no active systems credited as part of the initial conditions of an analysis or as part of the primary success path for mitigation of the DBA that is credible with the units permanently defueled. As such, the requirement for the CREACUS is being deleted because there are no design basis events that rely on the CREACUS for mitigation and the requirements of Criterion 3 of 10 CFR 50.36(c)(2)(ii)(C) no longer apply.

Although the CREACUS also provides the primary means to ensure control room habitability in the event of a waste gas decay tank rupture accident in MODES 5 and 6, generation of radioactive waste gases has ceased since the permanent cessation of reactor operation. In addition, the waste gas decay tanks have since been purged of their contents. Therefore, a rupture of these tanks is no longer an applicable accident.

Based on the above, the proposed deletion of TS 3.7.11 for the CREACUS is acceptable.

3.2.10.3 Section 3.7 TS That Are Being Retained and Revised

TS 3.7.16, TS 3.7.17, and TS 3.7.18 are being retained and revised as shown in Attachments 1 and 2. The corresponding TS Bases Sections are also being revised to reflect this change.

3.2.10.3.1 TS 3.7.16, Fuel Storage Pool Water Level

3.2.10.3.1.1 Description

TS 3.7.16, Fuel Storage Pool Water Level, specifies requirements to ensure that the minimum water level in the fuel storage pool meets the assumptions of iodine decontamination factors following a FHA. The specified water level shields and minimizes the general area dose when the storage racks are filled to their maximum capacity. The water also provides shielding during the movement of spent fuel. TS 3.7.16 is applicable during movement of fuel assemblies in the fuel storage pool.

3.2.10.3.1.2 Proposed Changes

TS 3.7.16 is being retained in the permanently defueled TS with essentially no change. The Note in Required Action A.1 (LCO 3.0.3 is not applicable) is being deleted to conform to the deletion of TS 3.0.3 described in the TS Chapter 3.0 justification. With the deletion of TS 3.0.3, this note is no longer required. As the result of the proposed deletion of all TS Section 3.1 through 3.6 Specifications, and as indicated in Attachments 1 and 2, TS 3.7.16 is being renumbered as TS 3.1.1.

3.2.10.3.1.3 Technical Analysis

The minimum water level in the fuel storage pool meets the assumptions of the FHA analysis described in Regulatory Guide (RG) 1.183. The resultant dose to a person at the exclusion area boundary or low population zone is a small fraction of the 10 CFR 50.67 limits. (A description of the FHA analysis for the permanently defueled condition is provided in Section 3.1.5 of this submittal.)

According to the UFSAR 15.7.3.4, there is 23 ft of water between the top of the damaged fuel bundle and the fuel pool surface for a fuel handling accident. With a 23 ft water level, the assumptions of RG 1.183 can be used directly.

In the FHA analysis, the drop orientations considered were a drop of an assembly onto the top of the racks with the assembly in a vertical position, drop of an assembly onto the top of the racks with the assembly in an inclined position, and a drop of a fuel assembly through an empty cell to the bottom of the pool, such that the cladding of all the fuel rods in the affected assemblies rupture. The gap activity in the damaged rods is instantaneously released into the fuel storage pool. The release occurs under 23 ft of water, which acts as a filter. The activity released from the fuel storage pool then mixes with the fuel building atmosphere before being exhausted to the environment. The fuel building exhaust rate is established to complete the release in 2 hours as required by RG 1.183. Therefore, the existing 23 ft water level requirement for the fuel storage pool remains appropriate based on the FHA analysis.

Retaining TS 3.7.16, with the proposed change, continues to ensure appropriate requirements for fuel storage pool water level.

3.2.10.3.2 TS 3.7.17, Fuel Storage Pool Boron Concentration

3.2.10.3.2.1 Description

TS 3.7.17, Fuel Storage Pool Boron Concentration, specifies requirements to ensure that the fuel storage pool boron concentration is ≥ 2000 ppm. The specified concentration of dissolved boron in the fuel storage pool preserves the assumptions used in the analyses of the potential critical accident scenarios. This concentration of dissolved boron is the minimum required concentration for fuel assembly storage and movement within the fuel storage pool. TS 3.7.17 applies whenever any fuel assembly is stored in the fuel storage pool.

3.2.10.3.2.2 Proposed Changes

TS 3.7.17 is being retained in the permanently defueled TS with essentially no change. The Note in Required Action A.1 (LCO 3.0.3 is not applicable) is being deleted to conform to the deletion of TS 3.0.3 described in the TS Chapter 3.0 justification. With the deletion of TS 3.0.3, this note is no longer required. As the result of the proposed deletion of all TS Section 3.1 through 3.6 Specifications, and as indicated in Attachments 1 and 2, TS 3.7.17 is being renumbered as TS 3.1.2.

3.2.10.3.2.3 Technical Analysis

As described in TS 3.7.18, Spent Fuel Assembly Storage, fuel assemblies are stored in the

spent fuel racks in accordance with criteria based on initial enrichment, discharge burnup, and cooling time (plutonium decay). Although the water in the spent fuel pool is normally borated to > 2000 ppm, the criteria that limit the storage of a fuel assembly to specific rack locations is conservatively developed without taking credit for boron while maintaining K_{eff} < 1.0.

Under normal, non-accident conditions, the soluble boron needed to maintain K_{eff} less than or equal to 0.95, including uncertainties, is 970 ppm. Under accident conditions, the soluble boron needed to maintain K_{eff} less than or equal to 0.95, including uncertainties, is 1700 ppm. A fuel storage pool boron dilution analysis shows that dilution from 2000 ppm to below 1700 is not credible. Therefore, the minimum required soluble boron concentration is 2000 ppm.

Retaining TS 3.7.17, with the proposed change, continues to ensure appropriate requirements for fuel storage pool boron concentration.

3.2.10.3.3 TS 3.7.18, Spent Fuel Assembly Storage

3.2.10.3.3.1 Description

TS 3.7.18, Spent Fuel Assembly Storage, specifies restrictions on the placement of fuel assemblies within the fuel storage pool as follows:

- The combination of initial enrichment and burnup of each SONGS 2 and 3 spent fuel assembly stored in Region I shall be within the acceptable burnup domain of Figure 3.7.18-1 or Figure 3.7.18-2, or the fuel assembly shall be stored in accordance with TS 4.3.1.1;

- The combination of initial enrichment and burnup of each SONGS 2 and 3 spent fuel assembly stored in Region II shall be within the acceptable burnup domain of Figure 3.7.18-3 or Figure 3.7.18-4, or the fuel assembly shall be stored in accordance with TS 4.3.1.1; and

- Each SONGS 1 uranium dioxide spent fuel assembly stored in Region II shall be stored in accordance with TS 4.3.1.1.

TS 3.7.18 applies whenever any fuel assembly is stored in the fuel storage pool.

3.2.10.3.3.2 Proposed Changes

TS 3.7.18 is being retained in the permanently defueled TS with essentially no change. The Note in Required Action A.1 (LCO 3.0.3 is not applicable) is being deleted to conform to the deletion of TS 3.0.3 described in the TS Chapter 3.0 justification. With the deletion of TS 3.0.3, this note is no longer required. As the result of the proposed deletion of all TS Section 3.1 through 3.6 Specifications, and as indicated in Attachments 1 and 2, TS 3.7.18 is being renumbered as TS 3.1.3.

3.2.10.3.3.3 Technical Analysis

The spent fuel storage facility is designed for noncriticality by use of adequate spacing, neutron absorbing stainless steel cans, borated water with a minimum soluble boron

concentration of 970 ppm, and storage of fuel assemblies in accordance with the administrative controls in TS 3.7.18 and LCS 4.0.100, Fuel Storage Patterns.

Retaining TS 3.7.18, with the proposed change, continues to ensure appropriate requirements for storing fuel assemblies in the fuel storage pool.

3.2.10.4 Summary

Since TS 3.7.1, TS 3.7.2, TS 3.7.3, TS 3.7.4, TS 3.7.5, TS 3.7.6, TS 3.7.7, TS 3.7.7.1, TS 3.7.8, TS 3.7.10, and TS 3.7.19 do not apply with the reactor defueled, they are not needed for a permanently defueled condition. As such, they may be deleted with no impact on continued safe operation of the facility.

TS 3.7.11 is not needed for accident mitigation in the permanently defueled condition. As such, TS 3.7.11 may be deleted with no impact on continued safe operation of the facility.

TS 3.7.16, TS 3.7.17, and TS 3.7.18 will remain applicable with the reactor permanently defueled. As such, they are being retained and revised to reflect a permanently defueled condition. The revised Specifications will be renumbered as TS 3.1.1, TS 3.1.2, and TS 3.1.3, respectively.

3.2.11 TS Section 3.8, Electrical Power Systems

The existing TS Section 3.8, Electrical Power Systems, contains LCOs that provide for appropriate functional capability of plant electrical equipment required for safe operation of the facility, including the plant being in a defueled condition. This Section is divided into the following Specifications.

TS Being Deleted	TS Being Retained
3.8 ELECTRICAL POWER SYSTEMS	
3.8.1 AC Sources – Operating	
3.8.2 AC Sources – Shutdown	
3.8.3 Diesel Fuel Oil, Lube Oil , and Starting Air	
3.8.4 DC Sources – Operating	
3.8.5 DC Sources – Shutdown	
3.8.6 Battery Parameters	
3.8.7 Inverters – Operating	
3.8.8 Inverters – Shutdown	
3.8.9 Distribution Systems – Operating	
3.8.10 Distribution Systems – Shutdown	

All TS in Section 3.8 are being proposed for deletion, as identified in the table above. The corresponding TS Bases are also being deleted to reflect this change.

3.2.11.1 Section 3.8 TS That Are Not Applicable When Defueled

TS 3.8.1, TS 3.8.4, TS 3.8.7, and TS 3.8.9 currently do not apply with the reactor defueled.

3.2.11.1.1 Description

TS 3.8.1, AC Sources – Operating, specifies requirements to ensure that the offsite power sources (normal preferred and alternate preferred power sources), and the standby power sources (Train A and Train B diesel generators (DGs)), provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to Engineered Safety Feature (ESF) systems so that the fuel, RCS, and containment design limits are not exceeded. TS 3.8.1 is applicable in MODES 1, 2, 3, and 4.

TS 3.8.4, DC Sources – Operating, specifies requirements to ensure that the DC electrical power subsystems (with each subsystem consisting of one battery, the required battery charger, and the corresponding control equipment and interconnecting cabling supplying power to the associated bus within the subsystem) are required to be OPERABLE to ensure the availability of the required power to shut down the reactor and maintain it in a safe condition after an AOO or a postulated DBA. TS 3.8.4 is applicable in MODES 1, 2,

3, and 4.

TS 3.8.7, Inverters – Operating, specifies requirements to ensure that required inverters are OPERABLE such that the redundancy incorporated into the design of the reactor protection system (RPS) and Engineered Safety Feature Actuation System (ESFAS) instrumentation and controls is maintained. These requirements include the required capacity, capability, redundancy, and reliability to ensure the availability of necessary power to the RPS and ESFAS instrumentation and controls so that the fuel, RCS, and containment design limits are not exceeded. TS 3.8.7 is applicable in MODES 1, 2, 3, and 4.

TS 3.8.9, Distribution Systems – Operating, specifies requirements to ensure availability of AC, DC, and AC instrument bus electrical power for the systems required to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence or a postulated DBA. The AC, DC, and AC vital electrical power distribution systems are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, RCS, and containment design limits are not exceeded. TS 3.8.9 is applicable in MODES 1, 2, 3, and 4.

3.2.11.1.2 Proposed Changes

TS 3.8.1, TS 3.8.4, TS 3.8.7, and TS 3.8.9 do not currently apply with the reactor defueled. The SONGS Unit 2 and Unit 3 Part 50 licenses no longer authorize emplacement or retention of fuel in their respective reactor vessel. Therefore, these electrical power system requirements do not apply and are being proposed for deletion.

3.2.11.1.3 Technical Analysis

TS 3.8.1, TS 3.8.4, TS 3.8.7, and TS 3.8.9 are related to assuring the appropriate functional capability of plant equipment required for safe operation of the facility only when the reactor is in MODES 1 through 4. However, 10 CFR 50.82(a)(2) prohibits operation of the plant or placing fuel in the reactor vessel. Therefore, TS 3.8.1, TS 3.8.4, TS 3.8.7, and TS 3.8.9 do not apply and are being proposed for deletion.

3.2.11.2 TS 3.8.2, Control AC Sources – Shutdown

3.2.11.2.1 Description

TS 3.8.2, AC Sources – Shutdown, specifies requirements to ensure that the offsite power sources (normal preferred and alternate preferred power sources), and the standby power sources (Train A and Train B DGs), provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, RCS, and containment design limits are not exceeded.

The OPERABILITY of the minimum AC sources during MODES 5 and 6 and during movement of fuel assemblies in containment or in the fuel storage pool ensures that:

a. The unit can be maintained in the shutdown or refueling condition for extended periods;
b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate AC electrical power is provided to mitigate events postulated during shut down, such as a fuel handling accident.

In general, when the unit is shut down, the TS requirements ensure that the unit has the capability to mitigate the consequences of postulated accidents. However, assuming a single failure and concurrent loss of all offsite or all onsite power is not required. The rationale for this is based on the fact that many DBAs have no specific analyses in MODES 5 and 6. Worst case bounding events are deemed not credible in MODES 5 and 6 because the energy contained within the reactor pressure boundary, reactor coolant temperature and pressure, and the corresponding stresses result in the probabilities of occurrence being significantly reduced or eliminated, and in minimal consequences. These deviations from analysis assumptions and design requirements during shutdown conditions are allowed by the TS for required systems. TS 3.8.2 is applicable in MODES 5 and 6, and during movement of fuel assemblies within containment or in the fuel storage pool.

3.2.11.2.2 Proposed Changes

Because 10 CFR 50.82(a)(2) prohibits operation of the plant or placing fuel in the reactor vessel, TS 3.8.2 is no longer needed for assuring the appropriate functional capability of the AC sources for safe operation of the facility when the reactor is in MODES 5 and 6. The only remaining TS 3.8.2 Applicability requirement for AC sources is during movement of fuel assemblies. However, the remaining applicable accident analyses show that the dose consequences are acceptable without relying on any SSCs to remain functional (including the AC sources) during and following the event. Therefore, TS 3.8.2 no longer applies and is being proposed for deletion.

3.2.11.2.3 Technical Analysis

The remaining applicable design basis accidents and transients analyzed in UFSAR Chapter 15 are discussed in Section 4.1 of this proposed amendment. A description of each accident with the potential to result in a radiological release is provided in Section 3.1 of this submittal. The accident analyses show that the dose consequences are acceptable without relying on any SSCs to remain functional during and following the event.

The requirement for AC sources was previously included in the TS for power operation of the reactor based on Criterion 3 of 10 CFR 50.36(c)(2)(ii)(C), which states that TS limiting conditions for operation must be established for structures, systems, or components (SSCs) that are part of the primary success path and which function or actuate to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

The FHA is the applicable design basis accident related to the TS requirement for functional capability of AC sources (offsite power and DGs) during the TS specified condition of "During movement of fuel assemblies in the fuel storage pool." Because the FHA analysis does not rely on normal or emergency power for accident mitigation (including any need for providing airborne radiological protection), the AC sources are not required during movement of fuel assemblies in the fuel storage pool for mitigation of a potential FHA. Therefore, during movement of fuel assemblies in the fuel storage pool, there are no active systems credited as part of the initial conditions of an analysis or as part

of the primary success path for mitigation of the design basis accident that is credible with the unit permanently defueled. As such, the requirement for AC sources is being deleted because there are no design basis events that rely on AC sources for mitigation and the requirements of Criterion 3 of 10 CFR 50.36(c)(2)(ii)(C) no longer apply.

With the reactor permanently defueled, irradiated fuel is stored either in the independent spent fuel storage facility (ISFSI) or in the fuel storage pool. The ISFSI is a passive system that does not rely on electrical power for heat transfer. Since there is a large capacity for heat absorption in the fuel storage pool, active system components are not redundant. Alternate cooling capability can be made available under anticipated malfunctions or failures without reliance on DGs.

The existing requirement for a qualified offsite circuit is based on its need to be capable of maintaining rated frequency and voltage, accepting required loads during an accident, and capable of supplying the onsite Class 1E power distribution subsystem(s) of LCO 3.8.10, Distribution Systems – Shutdown. Because the requirement for ESF equipment no longer exists upon permanent defueling (as justified in the associated sections of this proposed amendment), there is no longer a need for ESF buses. Since the AC electrical power distribution subsystems are no longer required to power ESF equipment, TS 3.8.10 is being deleted, as described in the corresponding section below.

With no need for a qualified offsite circuit to be capable of supplying loads during an accident, while connected to the ESF buses; and no need for a DG, there is no longer a need for TS 3.8.2. Therefore, TS 3.8.2 is being deleted in its entirety.

This change is consistent with the associated change approved by NRC for Millstone Power Station Unit 1 in License Amendment 106, dated November 9, 1999 (Reference 3); and for Zion Nuclear Station in License Amendments 180 and 167 (for Units 1 and 2 respectively), dated December 30, 1999 (Reference 4). The license amendments for both Millstone and Zion relaxed all TS requirements for AC sources.

3.2.11.3 TS 3.8.3, Control Diesel Fuel Oil, Lube Oil, and Starting Air

3.2.11.3.1 Description

For proper operation of the DGs, it is necessary to ensure sufficient quantity and proper quality of the fuel oil as well as sufficient quantity of lube oil. TS 3.8.3, Diesel Fuel Oil, Lube Oil, and Starting Air, specifies these parameters for stored diesel fuel oil, lube oil, and starting air systems. Stored diesel fuel oil is required to have sufficient supply for 7 days of rated load operation for each DG. It is also required to meet specific standards for quality. Additionally, sufficient lubricating oil supply must be available to ensure the capability to operate each DG at rated load for 7 days. Lastly, each DG is equipped with two air start systems which are independent and redundant. Each air start system has adequate capacity for five successive start attempts on the DG without recharging the air start receivers. These requirements, in conjunction with an ability to obtain replacement supplies within 7 days, supports the availability of DGs required to shut down the reactor and to maintain it in a safe condition for an AOO or a postulated DBA with loss of offsite power.

The AC sources (TS 3.8.1 and TS 3.8.2) are required to ensure the availability of the required power to shut down the reactor and maintain it in a safe shutdown condition after

an AOO or a postulated DBA. Since stored diesel fuel oil, lube oil, and starting air support TS 3.8.1 and TS 3.8.2, stored diesel fuel oil, lube oil, and starting air are required to be within limits when the associated DG is required to be OPERABLE. As such TS 3.8.3 is applicable when the associated DG is required to be OPERABLE.

3.2.11.3.2 Proposed Changes

TS 3.8.3 is required to support the DG requirements of TS 3.8.1 and TS 3.8.2. With the deletion of TS 3.8.1 and TS 3.8.2, the requirements of TS 3.8.3 are no longer applicable. Therefore, TS 3.8.3 is being proposed for deletion.

3.2.11.3.3 Technical Analysis

As discussed in the justification for deleting TS 3.8.2 above, the requirement for DGs is being deleted from the TS because there are no design basis accidents or transients analyzed in UFSAR Chapter 15 that rely on the DGs for mitigation. Since TS 3.8.3 exists solely to support the DG requirements of TS 3.8.1 and TS 3.8.2, the elimination of the need for DGs also obviates the need for their support systems. As such, TS 3.8.3 may be deleted. Based on the above, the proposed deletion of TS 3.8.3 for fuel oil, lube oil, and starting air parameters is acceptable.

3.2.11.4 TS 3.8.5, DC Sources – Shutdown

3.2.11.4.1 Description

The DC electrical power system provides normal and emergency DC electrical power for the DGs, emergency auxiliaries, and control and switching during all MODES of operation. TS 3.8.5, DC Sources – Shutdown, specifies requirements to ensure that the DC electrical power subsystems (with each subsystem consisting of one battery, required battery charger, and the corresponding control equipment and interconnecting cabling supplying power to the associated bus within the subsystem) are required to be OPERABLE to support the DC electrical power distribution subsystem(s) required OPERABLE by LCO 3.8.10, Distribution Systems – Shutdown. This ensures the availability of sufficient DC electrical power sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shut down (i.e., fuel handling accidents).

TS 3.8.5 is applicable in MODES 5 and 6. It is also applicable during movement of fuel assemblies within containment and in the fuel storage pool. The OPERABILITY of the minimum DC electrical power sources during MODES 5 and 6 and during movement of fuel assemblies within containment or in the fuel storage pool ensures that:

a. The unit can be maintained in the shutdown or refueling condition for extended periods;
b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and
c. Adequate DC electrical power is provided to mitigate events postulated during shut down, such as a fuel handling accident.

3.2.11.4.2 Proposed Changes

Because 10 CFR 50.82(a)(2) prohibits operation of the plant or placing fuel in the reactor

vessel, TS 3.8.5 is no longer needed for assuring the appropriate functional capability of the DC sources for safe operation of the facility when the reactor is in MODES 5 or 6 or during the movement of fuel assemblies within containment. Also, the remaining applicable accident analyses do not rely on DC sources for accident mitigation. Consequently, DC sources are not needed during movement of fuel assemblies in the fuel storage pool for mitigation of a potential accident. Therefore, TS 3.8.5 is being proposed for deletion.

3.2.11.4.3 Technical Analysis

The remaining applicable design basis accidents and transients analyzed in UFSAR Chapter 15 are discussed in Section 4.1 of this proposed amendment. A description of each accident with the potential to result in a radiological release is provided in Section 3.1 of this submittal. The accident analyses show that the dose consequences are acceptable without relying on any SSCs to remain functional during and following the event. Because the accident analyses do not rely on DC sources for accident mitigation (dose consequences are acceptable without relying on any SSCs to remain functional during and following the event), DC sources are therefore not required for accident mitigation. Consequently, DC sources are not needed during movement of fuel assemblies in the fuel storage pool for mitigation of a potential FHA. Thus, the requirement for DC sources is being deleted.

The requirement for DC sources was previously included in the TS for power operation of the reactor based on Criterion 3 of 10 CFR 50.36(c)(2)(ii)(C), which states that TS limiting conditions for operation must be established for structures, systems, or components (SSCs) that are part of the primary success path and which function or actuate to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Because the FHA analysis does not rely on DC sources for accident mitigation, DC sources are therefore not required during movement of fuel assemblies in the fuel storage pool for mitigation of a potential FHA. There are no active systems credited as part of the initial conditions of an analysis or as part of the primary success path for mitigation of the DBA that is credible with the unit permanently defueled. As such, the requirement for DC sources is being deleted because there are no design basis accidents or transients analyzed in UFSAR Chapter 15 that rely on DC sources for mitigation and the requirements of Criterion 3 of 10 CFR 50.36(c)(2)(ii)(C) no longer apply.

Based on the above, the proposed deletion of TS 3.8.5 for DC sources is acceptable.

3.2.11.5 TS 3.8.6, Battery Parameters

3.2.11.5.1 Description

TS 3.8.6, Battery Parameters, delineates the limits on battery float current as well as electrolyte temperature, level, and float voltage for the DC power subsystem batteries. In addition to the limitations of this Specification, the Battery Monitoring and Maintenance Program also implements a program specified in Specification 5.5.2.17 for monitoring various battery parameters. Battery parameters must remain within acceptable limits to ensure availability of the required DC power to shut down the reactor and maintain it in a safe condition after an AOO or a postulated DBA.

3.2.11.5.2 Proposed Changes

Battery parameters are required solely for the support of the associated DC electrical power subsystems (per TS 3.8.4 and TS 3.8.5). Therefore, battery parameter limits are only required (and TS 3.8.6 is only applicable) when the DC electrical power source is required to be OPERABLE. As TS 3.8.4 and TS 3.8.5 are being proposed for deletion, TS 3.8.6 is also being proposed for deletion.

3.2.11.5.3 Technical Analysis

As discussed in the justification for deleting TS 3.8.5 above, the requirement for DC sources is being deleted from the TS because there are no design basis accidents and transients analyzed in UFSAR Chapter 15 that rely on the DC sources for mitigation. Since TS 3.8.6 exists solely to support the DC source requirements of TS 3.8.4 and TS 3.8.5, the elimination of the need for DC sources also obviates the need for their support systems. As such, TS 3.8.6 may be deleted.

Based on the above, the proposed deletion of TS 3.8.6 for battery parameters is acceptable.

3.2.11.6 TS 3.8.8, Inverters – Shutdown

3.2.11.6.1 Description

The inverters are the preferred source of power for the AC vital buses because of the stability and reliability they achieve. The function of the inverter is to provide AC electrical power to the AC vital buses. The inverters can be powered from an internal AC source/rectifier or from the station battery. The inverter provides an uninterruptible power source for the instrumentation and controls for the RPS and the ESFAS so that the fuel, RCS, and containment design limits are not exceeded.

The OPERABILITY of the inverters to a required 120 VAC vital bus during MODES 5 and 6 and during movement of fuel assemblies within containment and in the fuel storage pool ensures that:

a. The unit can be maintained in the shutdown or refueling condition for extended periods;
b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and
c. Adequate power is available to mitigate events postulated during shut down, such as a fuel handling accident.

3.2.11.6.2 Proposed Changes

Because 10 CFR 50.82(a)(2) prohibits operation of the plant or placing fuel in the reactor vessel, TS 3.8.8 is no longer needed for assuring the appropriate functional capability of the inverters for safe operation of the facility when the reactor is in MODES 5 or 6 or during the movement of fuel assemblies within containment. Also, the remaining applicable accident analyses do not rely on the inverters for accident mitigation. Consequently, the inverters are not needed during movement of fuel assemblies in the fuel storage pool for mitigation of a potential accident. Therefore, TS 3.8.8 is being proposed for deletion.

3.2.11.6.3 Technical Analysis

The requirement for inverters was previously included in the TS for power operation of the reactor based on Criterion 3 of 10 CFR 50.36(c)(2)(ii)(C), which states that TS limiting conditions for operation must be established for structures, systems, or components (SSCs) that are part of the primary success path and which function or actuate to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Because the FHA analysis does not rely on inverters for accident mitigation, the inverters are therefore not required during movement of fuel assemblies in the fuel storage pool for mitigation of a potential FHA. There are no active systems credited as part of the initial conditions of an analysis or as part of the primary success path for mitigation of the design basis accident that is credible with the unit permanently defueled. As such, the requirement for inverters is being deleted because there are no design basis accidents or transients analyzed in UFSAR Chapter 15 that rely on the inverters for mitigation and the requirements of Criterion 3 of 10 CFR 50.36(c)(2)(ii)(C) no longer apply.

Based on the above, the proposed deletion of TS 3.8.8 for inverters is acceptable.

3.2.11.7 TS 3.8.10, Distributions Systems – Shutdown

3.2.11.7.1 Description

TS 3.8.10, Distribution Systems – Shutdown, specifies requirements to ensure that the onsite AC, DC, and AC instrument bus electrical power distribution systems provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, Reactor Coolant System, and containment design limits are not exceeded.

The OPERABILITY of the minimum AC, DC, and AC vital bus electrical power distribution subsystems during MODES 5 and 6, and during movement of fuel assemblies within containment or in the fuel storage pool ensures that:

a. The unit can be maintained in the shutdown or refueling condition for extended periods;
b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and
c. Adequate power is available to mitigate events postulated during shut down, such as a fuel handling accident.

TS 3.8.10 explicitly requires energization of the portions of the electrical power distribution

system necessary to support OPERABILITY of required systems, equipment, and components. Maintaining these portions of the distribution system energized ensures the availability of sufficient power to operate the unit in a safe manner to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents). TS 3.8.10 is applicable in MODES 5 and 6, and during movement of fuel assemblies within containment and in the fuel storage pool.

3.2.11.7.2 Proposed Changes

Because 10 CFR 50.82(a)(2) prohibits operation of the plant or placing fuel in the reactor vessel, TS 3.8.10 is no longer needed for assuring the appropriate functional capability of the electrical distribution systems for safe operation of the facility when the reactor is in MODES 5 or 6 or during movement of fuel assemblies within containment. Also, the remaining applicable accident analyses do not rely on the electrical distribution systems for accident mitigation. Consequently, the electrical distribution systems are not needed during movement of fuel assemblies in the fuel storage pool for mitigation of a potential FHA. Therefore, TS 3.8.10 is being proposed for deletion.

3.2.11.7.3 Technical Analysis

The remaining applicable design basis accidents and transients analyzed in UFSAR Chapter 15 are discussed in Section 4.1 of this proposed amendment. A description of each accident with the potential to result in a radiological release is provided in Section 3.1 of this submittal. The accident analyses show that the dose consequences are acceptable without relying on any SSCs to remain functional during and following the event.

The requirement for electrical distribution systems was previously included in the TS for power operation of the reactor based on Criterion 3 of 10 CFR 50.36(c)(2)(ii)(C), which states that TS limiting conditions for operation must be established for SSCs that are part of the primary success path and which function or actuate to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Because the FHA analysis does not rely on electrical distribution systems for accident mitigation, electrical distribution systems are therefore not required during movement of fuel assemblies in the fuel storage pool for mitigation of a potential FHA. There are no active systems credited as part of the initial conditions of an analysis or as part of the primary success path for mitigation of the design basis accident that is credible with the unit permanently defueled. As such, the requirement for electrical distribution systems is being deleted because there are no design basis events that rely on electrical distribution systems for mitigation and the requirements of Criterion 3 of 10 CFR 50.36(c)(2)(ii)(C) no longer apply.

The existing requirement for onsite Class 1E electrical power distribution subsystems of LCO 3.8.10 is to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, RCS, and containment design limits are not exceeded. Because there is no longer a need for any ESF systems for accident mitigation, the requirements of TS 3.8.10 are no longer needed. Therefore, TS 3.8.10 is being deleted in its entirety.

This change is consistent with the associated change approved by NRC for Millstone Power Station Unit 1 in License Amendment 106, dated November 9, 1999 (Reference 3);

and for Zion Nuclear Station in License Amendments 180 and 167 (for Units 1 and 2 respectively), dated December 30, 1999 (Reference 4). The license amendments for both Millstone and Zion completely relaxed all TS requirements for onsite electrical distribution systems.

3.2.11.8 Summary

Since TS 3.8.1, TS 3.8.4, TS 3.8.7, and TS 3.8.9 do not apply with the reactor defueled, they are not needed for a permanently defueled condition. As such, they may be deleted with no impact on continued safe operation of the facility.

TS 3.8.2, TS 3.8.3, TS 3.8.5, TS 3.8.6, TS 3.8.8, and TS 3.8.10 are not needed for accident mitigation in the permanently defueled condition. As such, these Specifications may be deleted with no impact on continued safe operation of the facility.

The pertinent requirements of TS 3.8.2 and TS 3.8.10 that are relied on for functionality of irradiated fuel cooling and as defense in depth for mitigation of a postulated accident are being relocated to the Licensee Controlled Specifications.

3.2.12 TS Section 3.9, Refueling Operations

The existing TS Section 3.9, Refueling Operations, contains LCOs that provide for appropriate functional capability of parameters and equipment within containment that are required for mitigation of design basis accidents during refueling operations (moving fuel to or from the reactor core). This Section is divided into the following Specifications.

TS Being Deleted	TS Being Retained
3.9 REFUELING OPERATIONS	
3.9.1 Boron Concentration	
3.9.2 Nuclear Instrumentation	
3.9.3 Containment Penetrations	
3.9.4 Shutdown Cooling (SDC) and Coolant Circulation – High Water Level	
3.9.5 Shutdown Cooling (SDC) and Coolant Circulation – Low Water Level	
3.9.6 Refueling Water Level	

All TS in Section 3.9 are being proposed for deletion, as identified in the table above. The corresponding TS Bases are also being deleted to reflect this change.

None of the TS in Section 3.9 currently apply with the reactor permanently defueled and fuel being prohibited from being placed in the reactor vessel per 10 CFR 50.82(a)(2).

3.2.12.1 Description

TS 3.9.1, Boron Concentration, places limits on the boron concentrations of the RCS and the refueling canal to ensure that the reactor remains subcritical during MODE 6. Refueling boron concentration is the soluble boron concentration in the coolant in each of these volumes having direct access to the reactor core during refueling. The boron concentration limits required by TS LCO 3.9.1 are specified in the Core Operating Limits Report (COLR). The boron concentration limit specified in the COLR will maintain a k_{eff} of ≤ 0.95 during fuel handling operations with CEAs and fuel assemblies assumed to be in the most adverse configuration (least negative reactivity) allowed by unit procedures. TS 3.9.1 is applicable in MODE 6.

TS 3.9.2, Nuclear Instrumentation, requires that two source range monitors (SRM) to be OPERABLE to ensure that redundant monitoring capability is available to detect changes in core reactivity. The SRMs are required to provide a signal to alert the operator to unexpected changes in core reactivity such as by a boron dilution event or an improperly loaded fuel assembly. TS 3.9.2 is applicable in MODE 6.

TS 3.9.3, Containment Penetrations, specifies requirements for containment closure during the conduct of CORE ALTERATIONS and movement of fuel assemblies within containment. The containment penetrations included within TS 3.9.3 are the equipment hatch, personnel air lock doors, and penetrations that provide direct access from the

containment atmosphere to the outside atmosphere. TS 3.9.3 limits the consequences of a fuel handling accident involving handling fuel within containment by limiting the potential escape paths for fission product radioactivity released within containment. TS 3.9.3 is applicable during CORE ALTERATIONS and during the movement of fuel assemblies within containment.

TS 3.9.4, Shutdown Cooling (SDC) and Coolant Circulation – High Water Level, specifies requirements for the SDC System in MODE 6 to remove decay heat and sensible heat from the RCS, to provide mixing of borated coolant, to provide sufficient coolant circulation to minimize the effects of a boron dilution accident, and to prevent boron stratification. One loop of the SDC System is required to be OPERABLE and in operation in MODE 6, with the water level > 20 ft above the top of the reactor vessel flange. Only one SDC loop is required to be OPERABLE, because the volume of water above the reactor vessel flange provides backup decay heat removal capability. TS 3.9.4 is applicable in MODE 6, with the water level > 20 ft above the top of the reactor vessel flange.

TS 3.9.5, Shutdown Cooling (SDC) and Coolant Circulation – Low Water Level, also specifies requirements for the SDC System in MODE 6 to remove decay heat and sensible heat from the RCS, to provide mixing of borated coolant, to provide sufficient coolant circulation to minimize the effects of a boron dilution accident, and to prevent boron stratification. However, with the water level < 20 ft above the top of the reactor vessel flange, both SDC loops must be OPERABLE. Additionally, one loop of SDC must be in operation. TS 3.9.5 is applicable in MODE 6 with the water level < 20 ft above the top of the reactor vessel flange.

TS 3.9.6, Refueling Water Level, specifies a minimum water level of 23 ft above the top of the reactor vessel flange during movement of fuel assemblies or CEAs within the reactor pressure vessel, and during movement of fuel assemblies within containment. A minimum refueling cavity water level of 23 ft above the top of the reactor vessel flange is required to ensure that the radiological consequences of a postulated fuel handling accident inside containment are within acceptable limits. The requirements of TS LCO 3.9.6, in conjunction with a minimum decay time of 72 hours prior to fuel movement, ensures that the release of fission product radioactivity, subsequent to a fuel handling accident, results in doses that are well within the guideline values specified in Regulatory Guide 1.183. TS 3.9.6 is only applicable during movement of fuel assemblies or CEAs within the reactor pressure vessel, and during movement of fuel assemblies within containment.

3.2.12.2 Proposed Changes

All TS in Section 3.9 do not currently apply with the reactor defueled. The SONGS Unit 2 and Unit 3 Part 50 licenses no longer authorize emplacement or retention of fuel in their respective reactor vessel. Therefore, these refueling operation requirements do not apply and are being proposed for deletion.

3.2.12.3 Technical Analysis

The above TS are related to assuring the appropriate functional capability of plant equipment, and control of process variables, design features, or operating restrictions required for safe refueling operation of the facility only when the reactor is in MODE 6 or during movement of fuel assemblies within containment. However, 10 CFR 50.82(a)(2)

prohibits operation of the plant or placing fuel in the reactor vessel, which thereby precludes entry into MODE 6. The prohibition on placing fuel in the reactor vessel also precludes movement of fuel within containment. Therefore, the TS listed in the previous paragraphs, which only address these specific plant systems, control of process variables, design features, or operating restrictions are no longer applicable.

3.2.12.4 Summary

All TS in Section 3.9 do not apply with the reactor defueled. Therefore, it is not needed for a permanently defueled condition. As such, they may be deleted with no impact on continued safe operation of the facility.

3.2.13 TS Chapter 4.0, Design Features

Chapter 4.0, Design Features, is divided into the following Specifications.

TS Being Deleted	TS Being Retained
4.0 DESIGN FEATURES	
	4.1 Site
4.2 Reactor Core	
	4.3 Fuel Storage

The TS 4.2 is being proposed for deletion, as identified in the table above. TS 4.1 and TS 4.3 are being retained and revised, as further described below and shown in Attachments 1 and 2. There are no TS Bases associated with these Specifications.

3.2.13.1 Description

TS Chapter 4.0, Design Features, contains descriptions and requirements for those features of the facility such as materials of construction and geometric arrangements which, if altered or modified, would have a significant effect on safety and are not covered in the previous sections of the TS.

TS Chapter 4.0, Design Features, does not contain applicability requirements. As such, all parts of this chapter can be conservatively defined as being applicable at all times.

3.2.13.2 Proposed Changes

TS 4.1, Site Location, provides a description regarding the location of SONGS Unit 2 and Unit 3. This TS section is being retained in the permanently defueled TS with no changes.

TS 4.2, Reactor Core, provides a description and requirements regarding the reactor core fuel assemblies and control rod assemblies. Because the SONGS Part 50 license no longer authorizes emplacement or retention of fuel in the reactor vessel, this TS section does not apply in a defueled condition and is being proposed for deletion. This TS section will read as follows:

4.2 Deleted.

TS 4.3, Fuel Storage, provides a description and requirements regarding prevention of criticality of spent fuel, prevention of fuel storage pool drainage, and spent fuel capacity limitations. This TS section is being retained as-is in the permanently defueled TS, with the exception of TS 4.3.1.2. TS 4.3.1.2 will read as follows:

4.3.1.2 Deleted.

3.2.13.3 Technical Analysis

TS 4.1 will remain applicable with the reactor permanently defueled. As such, TS 4.1 is

being retained to reflect a permanently defueled condition.

TS 4.2 contains requirements only associated with the reactor core, which can no longer be used following submittal of the certifications required by 10 CFR 50.82(a). Therefore, TS 4.2 is not needed for a permanently defueled condition. 10 CFR 50.82(a)(2) prohibits SCE from operating the plant or placing fuel in the reactor vessel. Therefore, TS 4.2 is no longer applicable. As such, TS 4.2 may be deleted with no impact on continued safe operation of the facility.

TS 4.3 (with the exception of TS 4.3.1.2) will remain applicable with the reactor permanently defueled. As such, this TS section (with the exception of TS 4.3.1.2) is being retained to reflect a permanently defueled condition. Because License Condition 2.B.(3) is being revised to no longer allow receipt of new fuel, TS 4.3.1.2 may be deleted with no impact on continued safe operation of the facility.

3.2.13.4 Summary

Retaining TS 4.1 and TS 4.3, as revised, ensures appropriate requirements for the associated design features.

TS 4.3 (with the exception of TS 4.3.1.2) will remain applicable with the reactor permanently defueled. As such, this TS section (with the exception of TS 4.3.1.2) is being retained to reflect a permanently defueled condition.

3.2.14 TS Chapter 5.0, Administrative Controls

On October 21, 2013, SCE submitted license amendment applications 265 and 250 for SONGS Units 2 and 3, respectively. The proposed TS 5.1, Responsibility, TS 5.2, Organization, and TS 5.3, Staff Qualifications, changes contained in the October 2013 letter are reflected in the following discussions and the TS markups contained in Attachments 1 and 2 of this license amendment application.

Chapter 5.0, Administrative Controls, is divided into the following Specifications.

TS Being Deleted	TS Being Retained
5.0 ADMINISTRATIVE CONTROLS	
	5.1 Responsibility
	5.2 Organization
	5.3 Facility Staff Qualifications
	5.4 Technical Specifications (TS) Bases Control
	5.5 Procedures, Programs, and Manuals
5.6 Safety Function Determination Program (SFDP)	
	5.7 Reporting Requirements
	5.8 High Radiation Area

The TS 5.6 is being proposed for deletion, as identified in the table above. TS 5.1, TS 5.2, TS 5.3, TS 5.4, TS 5.5, TS 5.7, and TS 5.8 are being retained and revised, as further described below and shown in Attachments 1 and 2. There are no TS Bases associated with these Specifications.

3.2.14.1 Description

The existing TS Chapter 5.0, Administrative Controls, contains provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner.

TS Chapter 5.0, Administrative Controls, does not contain applicability requirements. As such, all parts of this chapter can be conservatively defined as being applicable at all times.

3.2.14.2 Proposed Changes

TS 5.1, Responsibility

TS 5.1, Responsibility, provides a description and requirements regarding certain key operational management responsibilities. TS 5.1 will remain applicable with the reactor

permanently defueled. As such, it is being retained as revised in license amendment applications 265 and 250 to reflect a permanently defueled condition.

TS 5.2, Organization

TS 5.2, Organization, provides a description and requirements regarding the facility organization. TS 5.2 will remain applicable with the reactor permanently defueled. As such, it is being retained as revised in license amendment applications 265 and 250 (with one exception) to reflect a permanently defueled condition. However, three changes are being made to TS 5.2, as provided in license amendment applications 265 and 250. The first change is to capitalize the position of CERTIFIED FUEL HANDLER, consistent with its use as a defined term in TS 1.0. The second change is to TS 5.2.2.c, whereby a radiation protection technician is required to be on site during fuel handling operations and during movement of heavy loads over storage racks containing fuel. The third change is to TS 5.2.2.b, to clarify that during unexpected absences of on-duty shift crew members, no fuel movement or movement of heavy loads over storage racks containing fuel is permitted.

TS 5.3, Facility Staff Qualifications

TS 5.3, Facility Staff Qualifications, provides a description and requirements regarding qualifications of the facility staff. TS 5.3 will remain applicable with the reactor permanently defueled. As such, it is being retained as revised in license amendment applications 265 and 250 to reflect a permanently defueled condition with the following exceptions. As provided in license amendment applications 265 and 250, the position of CERTIFIED FUEL HANDLER is capitalized, consistent with its use as a defined term in TS 1.0. Additionally, the qualification requirements for multi-discipline supervisors are being deleted. The multi-discipline supervisor qualification requirements were added to SONGS Unit 2 and Unit 3 TS 5.3 with Amendments 168 and 159. Going forward, SONGS will not be utilizing the position of Multi-discipline Supervisor. The remaining facility staff (with the exception of the radiation protection manager) will meet or exceed the minimum qualifications of ANSI N18-1971. The radiation protection manager shall meet or exceed the qualifications of Regulatory Guide 1.5, September 1975.

TS 5.4, Technical Specifications (TS) Bases Control

TS 5.4, Technical Specifications (TS) Bases Control, provides the requirements regarding licensee changes to the TS Bases that can be made without prior NRC approval. TS 5.4 will remain applicable with the reactor permanently defueled. As such, it is being retained and revised to reflect a permanently defueled condition.

TS 5.4.4 is being revised to indicate the requirements of 10 CFR 50.71(e) for providing TS Bases changes implemented without prior NRC approval, as they apply to a licensee of a permanently defueled unit, i.e., every 24 months.

TS 5.5, Procedures, Programs and Manuals

TS 5.5, Procedures, Program and Manuals, provides a description and requirements regarding procedures, programs and manuals that are to be established, implemented, and maintained. TS 5.5 will remain applicable with the reactor permanently defueled. As such, it is being retained and revised to reflect a permanently defueled condition.

TS 5.5.1.1.b is being deleted because the emergency operating procedures discussed therein are no longer required. The emergency operating procedures pertained only to events resulting from reactor operation. Therefore, they are not needed with the reactor in the permanently defueled condition. TS 5.5.1.1.b will read as follows:

5.5.1.1.b Deleted.

TS 5.5.1.1.f is being deleted because the core protection calculator (CPC) is no longer required. The core protection calculators (CPCs) are one of two systems that monitor core power distribution online and derive the linear heat rate (LHR) and departure from nucleate boiling ratio (DNBR) parameters and associated RPS trips. The TS 3.3.1 RPS trips are applicable in MODES 1 and 2. Therefore, the CPCs are no longer applicable in the permanently defueled condition.

TS 5.5.2.4, Component Cyclic or Transient Limit Program, is being deleted because the Component Cyclic or Transient Limit Program pertains only to reactor support systems that do not apply in a defueled condition.

TS 5.5.2.5, Reactor Coolant Pump Flywheel Inspection Program, is being deleted because the Reactor Coolant Pump Flywheel Inspection Program pertains only to reactor support systems that do not apply in a permanently defueled condition.

TS 5.5.2.6, Secondary Water Chemistry Program, is being deleted because the Secondary Water Chemistry Program pertains only to reactor support systems that do not apply in a permanently defueled condition.

TS 5.5.2.7, Explosive Gas and Storage Tank Radioactivity Monitoring Program, is being revised. Paragraphs a. and b. of the program are being deleted because these portions of the Explosive Gas and Storage Tank Radioactivity Monitoring Program pertain only to reactor support systems that do not apply in a permanently defueled condition. There will no longer be any source of explosive or radioactive gases generated from reactor operation. In addition, the Gaseous Radwaste System and the waste gas decay tank have been vented and removed from service. As such, references to potentially explosive gas mixtures and methods for determining gaseous radioactivity have been deleted. The removal of these aspects of the program resulted in the change of the program title to, "Storage Tank Radioactivity Monitoring Program."

TS 5.5.2.8, Primary Coolant Sources Outside Containment Program, is being deleted because the program pertains only to reactor support systems that do not apply in a permanently defueled condition.

TS 5.5.2.9, Pre-Stressed Concrete Containment Tendon Surveillance Program, is being deleted because the program pertains only to the containment structure that does not apply in a permanently defueled condition.

TS 5.5.2.10, Inservice Inspection and Testing Program, is being deleted because the Inservice Inspection and Testing Program is no longer required in a permanently defueled condition. There are no longer any ASME Code Class 1, 2, or 3 components or Code Class CC and MC components that are required to perform a specific function in mitigating the consequences of an accident when in a permanently defueled condition.

TS 5.5.2.11, Steam Generator (SG) Program, is being deleted because the Steam Generator Program pertains only to reactor support systems that do not apply in a permanently defueled condition.

TS 5.5.2.12, Ventilation Filter Testing Program (VFTP), is being deleted because the Ventilation Filter Testing Program pertains only to reactor support systems that do not apply in a permanently defueled condition. The accident analysis applicable to the permanently defueled condition does not rely on ventilation filters for accident mitigation.

TS 5.5.2.13, Diesel Fuel Oil Testing Program, is being deleted because the Diesel Fuel Oil Testing Program pertains only to reactor support systems that do not apply in a permanently defueled condition. The requirement for diesel generators, which are supported by the fuel oil being tested per this program, is being deleted as described in preceding sections. The accident analysis applicable to the permanently defueled condition does not rely on diesel generators for accident mitigation.

TS 5.5.2.15, Containment Leakage Rate Testing Program, is being deleted because the Containment Leakage Rate Testing Program pertains only to reactor support systems that do not apply in a permanently defueled condition. The requirement for containment systems is being deleted as described in preceding sections. Therefore, the need for leakage rate testing of containment is no longer applicable.

TS 5.5.2.16, Control Room Envelope Habitability Program, is being deleted because the control room envelope (CRFE) is not required for providing airborne radiological protection for the control room operators in the event of a postulated accident.

TS 5.5.2.17, Battery Monitoring and Maintenance Program, is being deleted because the Battery Monitoring and Maintenance Program pertains only to reactor support systems that do not apply in a permanently defueled condition. The requirement for station batteries is being deleted as described in preceding sections. The accident analysis applicable to the permanently defueled condition does not rely on batteries for accident mitigation.

TS 5.6, Safety Function Determination Program (SFDP)

TS 5.6, Safety Function Determination Program (SFDP), is being deleted because the Safety Function Determination Program (SFDP) is not needed in a permanently defueled condition. TS 5.6 will read as follows:

5.6 Deleted.

TS 5.7, Reporting Requirements

TS 5.7.1, Routine Reports, provides a description and requirements regarding reports that are to be submitted in accordance with 10 CFR 50.4. TS 5.7.1 will remain applicable with the reactor permanently defueled. As such, it is being retained and revised to reflect a permanently defueled condition.

TS 5.7.1.1, Annual Reports, is being deleted because the only remaining annual report within TS 5.7.1.1 is the Reactor Coolant System Specific Activity Report. The Reactor

Coolant System Specific Activity Report pertains only to an activity that does not apply in a permanently defueled condition. Therefore, the need for this report no longer exists. TS 5.7.1.1 will read as follows:

5.7.1.1 Deleted.

TS 5.7.1.2, Annual Radiological Environmental Operating Report, is being retained in the permanently defueled TS with essentially no change. References to the term "unit" are replaced with the term "facility," because the term "unit" generally refers to the reactor, which can no longer be operated, whereas the term "facility" refers to the overall site, including the fuel storage facility. Also, due to this change, the Note indicating a single submittal may be made for a multiple unit station is no longer necessary and is therefore deleted.

TS 5.7.1.3, Radioactive Effluent Release Report, is being retained in the permanently defueled TS with essentially no change. References to the term "unit" are replaced with the term "facility," because the term "unit" generally refers to the reactor, which can no longer be operated, whereas the term "facility" refers to the overall site, including the fuel storage facility. Also, due to this change, the Note indicating a single submittal may be made for a multiple unit station is no longer necessary and is therefore deleted.

TS 5.7.1.5, Core Operating Limits Report (COLR), is being deleted because the COLR pertains only to an activity that does not apply in a permanently defueled condition. The eleven TS for which limits must be established (listed in TS 5.7.1.5.a) are being deleted as described in preceding sections. Therefore, the need for this report no longer exists.

TS 5.7.1.6, Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR), is being deleted because the PTLR pertains only to an activity that does not apply in a permanently defueled condition. The five TS that reference the report (listed in TS 5.7.1.6.a) are being deleted as described in preceding sections. Therefore, the need for this report no longer exists.

TS 5.7.1.7, Hazardous Cargo Traffic Report, is being deleted because the report pertains only to an event postulated to occur in MODES 1 through 6. Therefore, the event does not apply in a permanently defueled condition, and the need for this report no longer exists.

TS 5.7.2, Special Reports, provides a description and requirements regarding reports that are to be submitted in accordance with 10 CFR 50.4. The listed Special Reports pertain to 1) a pre-planned alternate method of monitoring post-accident instrumentation functions, 2) abnormal degradation of the containment structure detected during tests required by the Pre-Stressed Concrete Containment Tendon Surveillance Program, and 3) a report following entry into MODE 4 following completion of an inspection performed in accordance with the Steam Generator Program. These reports are being deleted because they pertain to activities that do not apply in a permanently defueled condition.

TS 5.8, High Radiation Area

TS 5.8, High Radiation Area, provides a description and requirements regarding controls applied to high radiation areas in place of the controls required by paragraph 20.1601 (a) and (b) of 10 CFR 20 (as provided in paragraph 20.1601(c) of 10 CFR 20). TS 5.8 will remain applicable with the reactor permanently defueled. As such, it is being retained

as-is with no changes being proposed.

3.2.14.3 Technical Analysis

TS 5.1, TS 5.2, TS 5.3, TS 5.4, and TS 5.8 will remain applicable with the reactor permanently defueled. As such, TS 5.1, TS 5.2, TS 5.3, TS 5.4, and TS 5.8 are being retained, as revised in license amendment applications 265 and 250 and the included changes described above, to reflect a permanently defueled condition.

TS 5.5 will remain applicable with the reactor permanently defueled, although several of the procedures, programs, and manuals contained therein are no longer applicable with the reactor permanently defueled. TS 5.6, SFDP, is no longer applicable with the reactor permanently defueled, and is therefore proposed for deletion. TS 5.7 will remain applicable with the reactor permanently defueled, although some of the reports are no longer applicable with the reactor permanently defueled.

The existing TS Chapter 5.0, Administrative Controls, contains provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner. Because 10 CFR 50.82(a)(2) prohibits operation of the plant or placing fuel in the reactor vessel, several of the TS Chapter 5.0 administrative controls Specifications are no longer applicable. As discussed in Section 4.1 of this proposed amendment, most of the design basis accidents and transients analyzed in UFSAR Chapter 15 are no longer applicable in the permanently defueled condition. After the termination of reactor operations at SONGS Units 2 and 3, and the permanent removal of the fuel from the reactor vessel (following 17 months of decay time after shut down), none of the systems, structures, and components (SSCs) at SONGS Units 2 and 3 are required to be relied on for accident mitigation. Therefore, the administrative controls Specifications that provide procedural requirements, programs, and reporting requirements related to safety-related SSCs are no longer applicable.

3.2.14.4 Summary

TS Section 5.0, Administrative Controls, does not contain applicability requirements. As such, all parts of this section can be conservatively defined as being applicable at all times. Portions of all TS in Section 5.0 will remain applicable with the reactor permanently defueled. As such, they are being retained and revised to reflect a permanently defueled condition. Retaining selected TS in Section 5.0, as revised, ensures appropriate requirements for administrative controls.

4.0 REGULATORY ANALYSIS

4.1 Applicable Regulatory Requirements/Criteria

Appendix A to Part 50 - General Design Criteria for Nuclear Power Plants

VI. Fuel and Radioactivity Control

Criterion 61 – Fuel storage and handling and radioactivity control. The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions.

Criterion 62 – Prevention of criticality in fuel storage and handling. Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

The proposed change does not affect any design features or processes related to fuel storage.

Design Basis Accidents (DBAs)

Per UFSAR Section 15.0.1.1, Safety Analyses Applicable after Permanent Cessation of Power Operation, SONGS has permanently ceased operation and removed all nuclear fuel from both units' reactor vessels. The irradiated fuel will be stored in the spent fuel pool (SFP) and in the Independent Spent Fuel Storage Installation (ISFSI) until it is shipped offsite. In this configuration, the SFP and its systems are dedicated only to spent fuel storage. In this condition, the number of credible accidents/transients is significantly smaller than for a plant authorized to operate the reactor or emplace or retain fuel in the reactor vessel.

With irradiated fuel being stored in the SFP and the ISFSI, the reactor, Reactor Coolant System (RCS) and secondary system are no longer in operation and have no function related to storage of irradiated fuel. With the permanent cessation of power operation and the permanent removal of the fuel from the reactor core, the accident/transient initial conditions/initial reactor power level of the reactor core cannot be achieved and, as such, most of the accident/transient scenarios are not possible. Therefore, the postulated UFSAR Chapter 15 accidents/transients involving failure or malfunction of the reactor, RCS or secondary system are no longer applicable. UFSAR Chapter 15 accidents/transients that are applicable are:

- Radioactive Waste Gas System Leak or Failure.
- Radioactive Waste System Leak or Failure (Release to Atmosphere).
- Postulated Radioactive Release Due to Liquid Tank Failures.
- Design Basis Fuel Handling Accident Inside Fuel Building.
- Spent Fuel Cask Drop Accidents.

- Spent Fuel Pool Gate Drop Accident.
- Test Equipment Drop.
- Spent Fuel Pool Boiling Accident.
- Spent Fuel Assembly Drop.
- Use of Miscellaneous Equipment Under 2000 lbs.

As discussed in UFSAR Section 15.7.3.1, the potential hazard associated with accidental waste gas decay tank releases is derived only from the release of stored gaseous activity. With the associated tanks purged, this hazard no longer exists.

UFSAR Section 15.7.3.2 discusses the radiological consequences for a liquid Radioactive Waste System leak or failure. A radwaste secondary tank is assumed to rupture, releasing the contents of the tank to the auxiliary building. Offsite doses due to the rupture of a radwaste secondary tank are less than the 100 mRem TEDE offsite dose criterion per Regulatory Issue Summary 2006-04.

With regard to the postulated radioactive release due to liquid tank failures, UFSAR Section 15.7.3.3.5 states, no credible accident exists, based on the scenarios, showing that there would be no liquid release exceeding 10 CFR 20 limits.

UFSAR Section 15.7.3.4 describes the fuel handling accident. Since SCE is no longer authorized to operate or to place or retain fuel in the reactors (following docketing of the certification specified in 10 CFR 50.82(a)(2)), a fuel handling accident onto the top of the core (or elsewhere within containment) is no longer possible and therefore no longer part of the licensing basis. However, a fuel handling accident in the fuel handling building (including the fuel storage pool) is still possible and is analyzed.

UFSAR Section 15.7.3.5 analyzes spent fuel cask drop events. Of the three situations considered, a spent fuel transfer cask drop (due to a seismic event) from the upper shelf in the cask pool back into the lower portion of the cask pool is the only credible event with the potential for radiological release. Other than the number of fuel assemblies considered to fail, the cask drop accident is modeled identically to that of the fuel handling accident in the fuel handling building, as addressed in UFSAR Section 15.7.3.4.

UFSAR Section 15.7.3.6 analyzes a postulated gate drop accident where a gate is accidentally dropped while being carried over the racks. Administrative controls provide assurance that the cask pool and transfer pool gates will not impact fuel assemblies stored in the impact zone or while reconstitution activities are in progress, thereby eliminating the potential for the event.

UFSAR Section 15.7.3.7 analyzes the drop of a test equipment skid drop above the spent fuel pool (SFP). Administrative controls ensure that the fuel assemblies are not damaged, since the depth of penetration will not impact the racks at the level where the fuel assemblies are located. Since no fuel assemblies are damaged, there are no radiological consequences for the test equipment drop.

As discussed in UFSAR Section 15.7.3.8, the postulated loss of all SFP cooling is assumed to result in SFP boiling and the release of a portion of the radionuclide inventory contained in the stored spent fuel assemblies and the SFP water. The offsite radiological doses for the postulated SFP boiling accident do not exceed 25% of the 10 CFR Part 50.67 exposure guidelines.

UFSAR Section 15.7.3.10 discusses spent fuel assembly drop onto the reconstitution station. Administrative controls preclude damage to spent fuel assemblies during reconstitution. Damage to the dropped fuel assembly is addressed in UFSAR Section 15.7.3.4. UFSAR Section 15.7.3.10 also discusses the drop of a spent fuel assembly onto control element assemblies (CEAs) bearing spent fuel assemblies. The radiological consequences for the failure of the fuel assemblies is addressed by the postulated fuel handling accident inside the fuel handling building in UFSAR Section 15.7.3.4.

UFSAR Section 15.7.3.11 analyzes the potential for damaging fuel assemblies, if it is postulated that equipment which does not exceed the weight of a fuel assembly, CEA, and associated handling equipment (i.e., less than 2000 lbs) is dropped onto other spent fuel assemblies. The activity release would be less than a fuel handling accident inside the fuel handling building, which analyzes a drop weight of 2065 pounds. Therefore, these events are bounded by the postulated fuel handling accident inside the fuel handling building discussed in UFSAR Section 15.7.3.4.

10 CFR 50.2, Definitions, Safety-Related Structures, Systems and Components

10 CFR 50.2 defines safety-related structures, systems, and components (SSCs) as those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:

1. The integrity of the reactor coolant pressure boundary;
2. The capability to shut down the reactor and maintain it in a safe shutdown condition; or
3. The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the applicable guideline exposures set forth in § 50.34(a)(1) or § 100.11 of this chapter, as applicable.

10 CFR 50.36, Technical Specifications

In 10 CFR 50.36, the Commission established its regulatory requirements related to the content of Technical Specifications (TS). In doing so, the Commission placed emphasis on those matters related to the prevention of accidents and mitigation of accident consequences; the Commission noted that applicants were expected to incorporate into their TS "those items that are directly related to maintaining the integrity of the physical barriers designed to contain radioactivity." (Statement of Consideration, "Technical Specification for Facility Licenses; Safety Analysis Reports," 33 FR 18610 (December 17, 1968).) Pursuant to 10 CFR 50.36, TS are required to include items in the following five categories: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation (LCOs); (3) surveillance requirements (SRs); (4) design features; and (5) administrative controls. However, the rule does not specify the particular requirements to be included in a plant's TS.

In September 1992, the Commission issued NUREG-1432, which was developed using the guidance and criteria contained in the Commission's Interim Policy Statement. Standard Technical Specifications (STS) were established as a model for developing improved TS for Combustion Engineering plants in general. STS reflect the results of a detailed review of the application of the interim policy statement criteria to generic system functions, which was published in a "Split Report" issued to the Nuclear Steam System Supplier (NSSS) Owners Groups in May 1988. STS also reflect the results of extensive

discussions concerning various drafts of STS, so that the application of the TS criteria and the Writer's Guide would consistently reflect detailed system configurations and operating characteristics for all NSSS designs. As such, the generic Bases presented in NUREG-1432 provide an abundance of information regarding the extent to which the STS present requirements that are necessary to protect public health and safety.

On July 22, 1993, the Commission issued its Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors, indicating that satisfying the guidance in the policy statement also satisfies Section 182a of the Atomic Energy Act of 1954, as amended (the Act), and 10 CFR 50.36 (58 FR 39132). The Final Policy Statement described the safety benefits of the improved STS, and encouraged licensees to use the improved STS as the basis for plant-specific TS amendments, and for complete conversions to improved STS. Further, the Final Policy Statement gave guidance for evaluating the required scope of the TS and defined the guidance criteria to be used in determining which of the LCOs and associated surveillances should remain in the TS.

The final Commission Policy Statement established four criteria to define the scope of equipment and parameters to be included in the improved Standard Technical Specifications. These criteria were developed for licenses authorizing operation (i.e., operating reactors) and focused on instrumentation to detect degradation of the reactor coolant system pressure boundary, process variables and equipment, design features, or operating restrictions that affect the integrity of fission product barriers during design bases accidents or transients. A fourth criterion refers to the use of operating experience and probabilistic risk assessment to identify and include in the Technical Specifications structures, systems, and components (SSCs) shown to be significant to public health and safety. These criteria, which were subsequently codified in changes to Section 36 of Part 50 of Title 10 of the *Code of Federal Regulations* (10 CFR 50.36) (60 FR 36953), also pertain to the Technical Specification requirements for safe storage of spent fuel. A general discussion of these considerations is provided below.

Criterion 1 of 10 CFR 50.36(c)(2)(ii)(A) states that Technical Specification limiting conditions for operation must be established for "installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary." Since no fuel is present in the reactor or reactor coolant system at the SONGS facility following permanent defueling, this criterion is not applicable.

Criterion 2 of 10 CFR 50.36(c)(2)(ii)(B) states that Technical Specification limiting conditions for operation must be established for a "process variable, design feature, or operating restriction that is an initial condition of a design basis accident [DBA] or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier." The purpose of this criterion is to capture those process variables that have initial values assumed in the design basis accident and transient analyses, and which are monitored and controlled during power operation. While this criterion was developed for operating reactors, there are some design basis accidents which continue to apply to a plant authorized only to handle, store, and possess nuclear fuel. The scope of DBAs applicable to a plant with a reactor that is permanently shut down and defueled is markedly reduced from those postulated for an operating plant. The DBAs for SONGS Units 2 and 3 are discussed within this proposed amendment.

Criterion 3 of 10 CFR 50.36(c)(2)(ii)(C) states that Technical Specification limiting conditions for operation must be established for structures, systems, or components

(SSCs) that are part of the primary success path and which function or actuate to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. The intent of this criterion is to capture into Technical Specifications only those SSCs that are part of the primary success path of a safety sequence analysis. Also captured by this criterion are those support and actuation systems that are necessary for items in the primary success path to successfully function. The primary success path of a safety sequence analysis consists of the combination and sequences of equipment needed to operate (including consideration of the single failure criterion), so that the plant response to design basis accidents and transients limits the consequences of these events to within the appropriate acceptance criteria. While there are no transients that continue to apply to SONGS Units 2 and 3, there is still a design basis accident that continues to apply to a plant authorized only to handle, store, and possess nuclear fuel. The scope of DBAs applicable to a plant with a reactor that is permanently shut down and defueled is markedly reduced from those postulated for an operating plant. The scope of DBAs applicable to SONGS Units 2 and 3 is discussed in more detail within this proposed amendment.

Criterion 4 of 10 CFR 50.36(c)(2)(ii)(D) states that Technical Specification limiting conditions for operation must be established for SSCs that operating experience or probabilistic risk assessment has shown to be significant to public health and safety. The intent of this criterion is that risk insights and operating experience be factored into the establishment of Technical Specification limiting conditions for operation. All of the accident sequences that previously dominated risk at SONGS Units 2 and 3 are no longer applicable with the reactor in the permanently shut down and defueled condition.

Addressing administrative controls, 10 CFR 50.36(c)(5) states that they "...are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner." The particular administrative controls to be included in the TS, therefore, are the provisions that the Commission deems essential for the safe operation of the facility that are not already covered by other regulations. Accordingly, the NRC staff determined that administrative control requirements that are not specifically required under Section 50.36(c)(5), and which are not otherwise necessary to obviate the possibility of an abnormal situation or an event giving rise to an immediate threat to the public health and safety, may be relocated to more appropriate documents (e.g., Quality Assurance Program, Security Plan, or Emergency Plan), which are subject to regulatory controls. Similarly, while the required content of TS administrative controls is specified in 10 CFR 50.36(c)(5), particular details may be relocated to licensee-controlled documents, where other regulations provide adequate regulatory control.

10 CFR 50.36(c)(6), "Decommissioning," applies only to nuclear power reactor facilities that have submitted the certifications required by § 50.82(a)(1). For such facilities, Technical Specifications involving safety limits, limiting safety system settings, and limiting control system settings; limiting conditions for operation; surveillance requirements; design features; and administrative controls will be developed on a case-by-case basis.

This proposed amendment deletes the portions of the previous SONGS TS that are no longer applicable to a permanently defueled facility while modifying the remaining portions to correspond to the permanently shut down condition, consistent with STS.

10 CFR 50.51, Continuation of License

10 CFR 50.51(b) states "Each license for a facility that has permanently ceased operations, continues in effect beyond the expiration date to authorize ownership and possession of the production or utilization facility, until the Commission notifies the licensee in writing that the license is terminated. During such period of continued effectiveness the licensee shall:

(1) Take actions necessary to decommission and decontaminate the facility and continue to maintain the facility, including, where applicable, the storage, control and maintenance of the spent fuel, in a safe condition, and

(2) Conduct activities in accordance with all other restrictions applicable to the facility in accordance with the NRC regulations and the provisions of the specific 10 CFR part 50 license for the facility."

10 CFR 50.82, Termination of License

10 CFR 50.82(a)(2) states "Upon docketing of the certifications for permanent cessation of operations and permanent removal of fuel from the reactor vessel, or when a final legally effective order to permanently cease operations has come into effect, the 10 CFR part 50 license no longer authorizes operation of the reactor or emplacement or retention of fuel into the reactor vessel."

4.2 Precedent

This proposed amendment is consistent with the existing license currently in effect for Millstone Nuclear Power Station (DPR-21), which was last substantively revised on March 31, 2001 (Reference 2). The Millstone license amendment that was issued to reflect the permanently shutdown status of the plant on November 9, 1999 (Reference 3), contains license conditions and Technical Specifications (TS) similar to those being proposed herein.

This proposed amendment is also consistent with the license, and accompanying TS, issued to Zion Nuclear Power Station on December 30, 1999 (Reference 4), which was issued to reflect the permanently shutdown status of the plant.

4.3 No Significant Hazards Consideration

Pursuant to 10 CFR 50.90, Southern California Edison (SCE) requests an amendment to Facility Operating License Numbers NPF-10 and NPR-15 for San Onofre Nuclear Generating Station (SONGS) Units 2 and 3, respectively. The proposed amendment would revise the Operating License and revise the associated Technical Specifications (TS) to Permanently Defueled Technical Specifications (PDTS) to reflect the permanent cessation of reactor operation.

On June 12, 2013, SCE submitted a certification to the NRC indicating its intention to permanently cease power operations at SONGS Units 2 and 3 (Reference 1) pursuant to 10 CFR 50.82(a)(1)(i). The certification stated that SCE had decided to permanently cease power operation of SONGS on June 7, 2013. With the docketing of the certification for permanent removal of fuel from the reactor vessels pursuant to 10 CFR 50.82(a)(1)(ii)

on June 28, 2013 and July 22, 2013, the 10 CFR Part 50 licenses for SONGS Units 2 and 3 no longer authorize operation of the reactors or emplacement or retention of fuel into the reactor vessels, as specified in 10 CFR 50.82(a)(2). In support of this condition, the SONGS Units 2 and 3 licenses and associated TS are being proposed for revision to conform to this permanently shut down and defueled condition in accordance with 10 CFR 50.36(c)(6).

The existing SONGS TS contain Limiting Conditions for Operation (LCOs) that provide for appropriate functional capability of equipment required for safe operation of the facility, including the plant being in a defueled condition. Because the SONGS Units 2 and 3 Part 50 licenses no longer authorize emplacement or retention of fuel in the reactor vessels, the LCOs (and associated Surveillance Requirements (SRs)) that do not apply in a defueled condition are being proposed for deletion. The remaining portions of the TS are being proposed for revision and incorporation as the PDTS to provide a continuing acceptable level of safety which addresses the reduced scope of postulated design basis accidents associated with a defueled plant, as described in the SONGS Units 2 and 3 safety analyses.

SCE has evaluated the proposed amendment to determine if a significant hazards consideration is involved by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

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

   Response: No

   SONGS Units 2 and 3 have permanently ceased operation. The proposed amendment would modify the SONGS Units 2 and 3 facility operating licenses and TS by deleting the portions of the licenses and TS that are no longer applicable to a permanently defueled facility, while modifying the remaining portions to correspond to the permanently shutdown condition. This change is consistent with the criteria set forth in 10 CFR 50.36 for the contents of TS.

   Section 15 of the SONGS Updated Final Safety Analysis Report (UFSAR) described the design basis accident (DBA) and transient scenarios applicable to SONGS Units 2 and 3 during power operations. With the reactors in a permanently defueled condition, the fuel storage pools and their systems have been isolated and are dedicated only to spent fuel storage. In this condition, the spectrum of credible accidents is much smaller than for an operational plant. As a result of the certifications submitted by SCE in accordance with 10 CFR 50.82(a)(1), and the consequent removal of authorization to operate the reactors or to place or retain fuel in the reactors in accordance with 10 CFR 50.82(a)(2), most of the accident scenarios postulated in the UFSAR are no longer possible.

   The definition of safety-related structures, systems, and components (SSCs) in 10 CFR 50.2 states that safety-related SSCs are those relied on to remain functional during and following design basis events to assure:

   1. The integrity of the reactor coolant boundary;
   2. The capability to shut down the reactor and maintain it in a safe shutdown

condition; or

3. The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the applicable guideline exposures set forth in 10 CFR 50.43(a)(1) or 100.11.

The first two criteria (integrity of the reactor coolant pressure boundary and safe shut down of the reactor) are not applicable to a plant in a permanently defueled condition. The third criterion is related to preventing or mitigating the consequences of accidents that could result in potential offsite exposures exceeding limits. However, after the termination of reactor operations at SONGS Units 2 and 3 and the permanent removal of the fuel from the reactor vessels (following 17 months of decay time after shut down) and purging of the contents of the waste gas decay tanks, none of the SSCs at SONGS Units 2 and 3 are required to be relied on for accident mitigation. Therefore, none of the SSCs at SONGS Units 2 and 3 meet the definition of a safety-related SSC stated in 10 CFR 50.2 (with the exception of the passive fuel storage pool structure).

The deletion of TS definitions and rules of usage and application, that are currently not applicable in a defueled condition, has no impact on facility SSCs or the methods of operation of such SSCs. The deletion of design features and safety limits not applicable to the permanently shut down and defueled status of SONGS Units 2 and 3 has no impact on the remaining DBA. The removal of limiting conditions for operation (LCOs) or surveillance requirements (SRs) that are related only to the operation of the nuclear reactors or only to the prevention, diagnosis, or mitigation of reactor-related transients or accidents do not affect the applicable DBAs previously evaluated since these DBAs are no longer applicable in the defueled mode. The safety functions involving core reactivity control, reactor heat removal, reactor coolant system inventory control, and containment integrity are no longer applicable at SONGS Units 2 and 3 as a permanently defueled plant. The analyzed accidents involving damage to the reactor coolant system, main steam lines, reactor core, and the subsequent release of radioactive material are no longer possible at SONGS Units 2 and 3.

Since SONGS Units 2 and 3 has permanently ceased operation, the future generation of fission products has ceased and the remaining source term will decay. The radioactive decay of the irradiated fuel since shut down of the reactor will have reduced the consequences of the FHA to levels well below those previously analyzed. The relevant parameter (water level) associated with the fuel pool provides an initial condition for the FHA analysis and is included in the permanently defueled TS.

The fuel storage pool water level, fuel storage pool boron concentration, and spent fuel assembly storage TS are retained to preserve the current requirements for safe storage of irradiated fuel.

Fuel pool cooling and makeup related equipment and support equipment (e.g., electrical power systems) are not required to be continuously available since there is sufficient time to effect repairs, establish alternate sources of makeup flow, or establish alternate sources of cooling in the event of a loss of cooling and makeup flow to the fuel storage pool.

The deletion and modification of provisions of the administrative controls does not directly affect the design of SSCs necessary for safe storage of irradiated fuel or the methods used for handling and storage of such fuel in the fuel pool. The changes to

the administrative controls are administrative in nature and do not affect any accidents applicable to the safe management of irradiated fuel or the permanently shut down and defueled condition of the reactors.

The probability of occurrence of previously evaluated accidents is not increased, since extended operation in a defueled condition is the only operation currently allowed, and therefore bounded by the existing analyses. Additionally, the occurrence of postulated accidents associated with reactor operation is no longer credible in a permanently defueled reactor. This significantly reduces the scope of applicable accidents.

Therefore, the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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 have no impact on facility SSCs affecting the safe storage of irradiated fuel, or on the methods of operation of such SSCs, or on the handling and storage of irradiated fuel itself. The removal of TS that are related only to the operation of the nuclear reactor or only to the prevention, diagnosis, or mitigation of reactor-related transients or accidents cannot result in different or more adverse failure MODES or accidents than previously evaluated because the reactor is permanently shut down and defueled and SCE is no longer authorized to operate the reactors.

   The proposed deletion of requirements of the SONGS Unit 2 and Unit 3 TS do not affect systems credited in the accident analysis. The proposed permanently defueled TS (PDTS) continue to require proper control and monitoring of safety significant parameters and activities.

   The proposed restriction on the fuel pool level is fulfilled by normal operating conditions and preserves initial conditions assumed in the analyses of the postulated DBA. The fuel storage pool water level, fuel storage pool boron concentration, and spent fuel assembly storage TS are retained to preserve the current requirements for safe storage of irradiated fuel.

   The proposed amendment does not result in any new mechanisms that could initiate damage to the remaining relevant safety barriers for defueled plants (i.e., fuel cladding and spent fuel cooling). Since extended operation in a defueled condition is the only operation currently allowed, and therefore bounded by the existing analyses, such a condition does not create the possibility of a new or different kind of accident.

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

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

   Response: No

Because the 10 CFR Part 50 licenses for SONGS Units 2 and 3 no longer authorize operation of the reactors or emplacement or retention of fuel into the reactor vessels, as specified in 10 CFR 50.82(a)(2), the occurrence of postulated accidents associated with reactor operation is no longer credible. The remaining credible accidents do not credit SSCs for mitigation. The proposed amendment does not adversely affect the inputs or assumptions of any of the design basis analyses that impact an accident.

The proposed changes are limited to those portions of TS and license that are not related to the safe storage of irradiated fuel. The requirements for SSCs that have been deleted from the SONGS TS Units 2 and 3 are not credited in the existing accident analysis for the remaining applicable postulated accident; and as such, do not contribute to the margin of safety associated with the accident analysis. Postulated DBAs involving the reactors are no longer possible because the reactors are permanently shut down and defueled and SCE is no longer authorized to operate the reactors.

Therefore, the proposed amendment does not involve a significant reduction in a margin of safety because the current design limits continue to be met for the accidents of concern.

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

4.4 Conclusion

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

5.0 ENVIRONMENTAL CONSIDERATION

SCE has evaluated this license amendment against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21. SCE has determined that this license amendment meets the criteria for a categorical exclusion set forth in 10 CFR 51.22(c)(9). This determination is based on the fact that this change is being proposed as an amendment to a license issued pursuant to 10 CFR 50, that changes a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or that changes an inspection or a surveillance requirement.

However, (i) the proposed amendment involves no significant hazards consideration, (ii) there is no significant change in the types or significant increase in the amounts of any effluent that may be released offsite, and (iii) there is no significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9).

Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 REFERENCES

1. Letter from Peter T. Dietrich to U.S. Nuclear Regulatory Commission, "Docket Nos. 50-361 and 50-362 Certification of Permanent Cessation of Power Operations San Onofre Nuclear Generating Station Units 2 and 3," dated June 12, 2013.

2. Millstone Nuclear Power Station, Unit 1, Amendment No.109, License No. DPR-21, Date of Issuance March 31, 2001.

3. NRC Safety Evaluation for Millstone Power Station Unit 1 in License Amendment 106 to DPR-21, dated November 9, 1999.

4. NRC Safety Evaluation for Zion Nuclear Station in License Amendments 180 and 167 (for Units 1 and 2 respectively (License Nos. DPR-39 and DPR-48)), dated December 30, 1999.

5. Letter from Peter T. Dietrich to U.S. Nuclear Regulatory Commission, "Docket No. 50-361 Permanent Removal of Fuel from the Reactor Vessel San Onofre Nuclear Generating Station Unit 2", dated July 22, 2013.

6. Letter from Peter T. Dietrich to U.S. Nuclear Regulatory Commission, "Docket No. 50-362 Permanent Removal of Fuel from the Reactor Vessel San Onofre Nuclear Generating Station Unit 3", dated June 28, 2013.

7. NRC Safety Evaluation for San Onofre Nuclear Generating Station in License Amendments 127 and 116 (for Units 2 and 3 respectively (License Nos. NPF-10 and NPF-15)), dated February 9, 1996.

Attachment 1

Proposed Technical Specifications - Markup - Unit 2

1.0 USE AND APPLICATION

1.1 Definitions

---------------------------------NOTE---------------------------------
The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications and Bases.

Term	Definition
ACTIONS	ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.

INSERT 1

~~AXIAL SHAPE INDEX (ASI)~~	~~ASI shall be the power generated in the lower half of the core less the power generated in the upper half of the core, divided by the sum of the power generated in the lower and upper halves of the core.~~

$$\cancel{ASI = \frac{lower - upper}{lower + upper}}$$

~~AZIMUTHAL POWER TILT (T_q)~~	~~AZIMUTHAL POWER TILT shall be the power asymmetry between azimuthally symmetric fuel assemblies.~~
~~CHANNEL CALIBRATION~~	~~A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds within the necessary range and accuracy to known values of the parameter that the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel, including the required sensor, alarm, display, and trip functions, and shall include the CHANNEL FUNCTIONAL TEST. Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an inplace cross calibration of the sensing elements and normal calibration of the remaining adjustable devices in the channel. Whenever a sensing element is replaced, the next required inplace cross calibration consists of comparing the other sensing elements with the recently installed sensing element.~~

(continued)

INSERT 1

CERTIFIED FUEL HANDLER	A CERTIFIED FUEL HANDLER is an individual who complies with provisions of the CERTIFIED FUEL HANDLER training program required by TS 5.3.2.

1.1 Definitions

~~CHANNEL CALIBRATION (continued)~~

~~The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping, or total channel steps so that the entire channel is calibrated.~~

~~CHANNEL CHECK~~

~~A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter.~~

~~CHANNEL FUNCTIONAL TEST~~

~~A CHANNEL FUNCTIONAL TEST shall be:~~

~~a.~~ ~~Analog channels—the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY, including required alarms, interlocks, display and trip functions;~~

~~b.~~ ~~Bistable channels (e.g., pressure switches and switch contacts)—the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY, including required alarm and trip functions; or~~

~~c.~~ ~~Digital computer channels—the use of diagnostic programs to test digital computer hardware and the injection of simulated process data into the channel to verify OPERABILITY, including alarm and trip functions.~~

~~The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps so that the entire channel is tested.~~

~~CORE ALTERATION~~

~~CORE ALTERATION shall be the movement or manipulation of any fuel, sources, reactivity control components, or other components, excluding control element assemblies (CEAs) withdrawn into the upper guide structure, affecting reactivity,~~

(continued)

1.1 Definitions

~~CORE ALTERATION (continued)~~	~~within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.~~
~~CORE OPERATING LIMITS REPORT (COLR)~~	~~The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 5.7.1.5. Plant operation within these limits is addressed in individual Specifications.~~
~~DOSE EQUIVALENT I-131~~	~~DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in ICRP-30, Supplement to Part 1, pages 192-212, Tables titled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity."~~
~~$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY~~	~~$\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives > 15 minutes, making up at least 95% of the total noniodine activity in the coolant.~~
~~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~~

(continued)

1.1 Definitions

~~ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME (Continued)~~	~~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.~~
~~LEAKAGE~~	~~LEAKAGE shall be:~~ ~~a. Identified LEAKAGE~~ ~~1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) 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 to the Secondary System (primary to secondary LEAKAGE).~~ ~~b. Unidentified LEAKAGE~~ ~~All LEAKAGE that is not identified LEAKAGE.~~ ~~c. Pressure Boundary LEAKAGE~~ ~~LEAKAGE (except primary to secondary LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.~~
~~MODE~~	~~A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.~~

~~(continued)~~

1.1 Definitions

OPERABLE — OPERABILITY	A system, subsystem, train, component, or device shall be OPERABLE when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
~~PHYSICS TESTS~~	~~PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are:~~ ~~a. Described in Chapter 14, Initial Test Program of the SONGS Units 2 and 3 UFSAR;~~ ~~b. Authorized under the provisions of 10 CFR 50.59; or~~ ~~c. Otherwise approved by the Nuclear Regulatory Commission.~~
~~PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)~~	~~The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.7.1.6.~~
~~RATED THERMAL POWER (RTP)~~	~~RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3438 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~~

(continued)

1.1 Definitions

~~REACTOR PROTECTIVE SYSTEM (RPS) RESPONSE TIME (continued)~~	~~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 CEAs (shutdown and regulating) are fully inserted except for the single CEA of highest reactivity worth, which is assumed to be fully withdrawn. However, with all CEAs verified fully inserted by two independent means, it is not necessary to account for a stuck CEA in the SDM calculation. With any CEAs not capable of being fully inserted, the 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.~~
~~STAGGERED TEST BASIS~~	~~A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function.~~
~~THERMAL POWER~~	~~THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.~~

(continued)

~~Table 1.1-1 (page 1 of 1)~~
~~MODES~~

~~MODE~~	~~TITLE~~	~~REACTIVITY CONDITION (k_{eff})~~	~~% RATED THERMAL POWER[(a)]~~	~~AVERAGE REACTOR COOLANT TEMPERATURE (°F)~~
~~1~~	~~Power Operation~~	~~≥ 0.99~~	~~> 5~~	~~NA~~
~~2~~	~~Startup~~	~~≥ 0.99~~	~~≤ 5~~	~~NA~~
~~3~~	~~Hot Standby~~	~~< 0.99~~	~~NA~~	~~≥ 350~~
~~4~~	~~Hot Shutdown~~	~~< 0.99~~	~~NA~~	~~350 > T_{avg} > 200~~
~~5~~	~~Cold Shutdown[(b)]~~	~~< 0.99~~	~~NA~~	~~≤ 200~~
~~6~~	~~Refueling[(c)]~~	~~NA~~	~~NA~~	~~NA~~

~~(a) Excluding decay heat.~~

~~(b) All reactor vessel head closure bolts fully tensioned.~~

~~(C) One or more reactor vessel head closure bolts less than fully tensioned.~~

1.0 USE AND APPLICATION

1.2 Logical Connectors

PURPOSE

The purpose of this section is to explain the meaning of logical connectors.

Logical connectors are used in Technical Specifications (TS) to discriminate between, and yet connect, discrete Conditions, Required Actions, Completion Times, Surveillances, and Frequencies. The only logical connectors that appear in TS are <u>AND</u> and <u>OR</u>. The physical arrangement of these connectors constitutes logical conventions with specific meanings.

BACKGROUND

Several levels of logic may be used to state Required Actions. These levels are identified by the placement (or nesting) of the logical connectors and by the number assigned to each Required Action. The first level of logic is identified by the first digit of the number assigned to a Required Action and the placement of the logical connector in the first level of nesting (i.e., left justified with the number of the Required Action). The successive levels of logic are identified by additional digits of the Required Action number and by successive indentions of the logical connectors.

When logical connectors are used to state a Condition, Completion Time, Surveillance, or Frequency, only the first level of logic is used, and the logical connector is left justified with the statement of the Condition, Completion Time, Surveillance, or Frequency.

EXAMPLE~~S~~

The following example~~s~~ illustrate the use of logical connectors.

$\boxed{S}$

(continued)

1.2 Logical Connectors

EXAMPLES (continued)

EXAMPLE 1.2-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Verify . . . AND A.2 Restore . . .	

In this example the logical connector AND is used to indicate that when in Condition A, both Required Actions A.1 and A.2 must be completed.

(continued)

1.2 Logical Connectors

~~EXAMPLES (continued)~~

~~EXAMPLE 1.2-2~~

~~ACTIONS~~

~~CONDITION~~	~~REQUIRED ACTION~~	~~COMPLETION TIME~~
~~A. LCO not met.~~	~~A.1 Trip . . .~~ ~~OR~~ ~~A.2.1 Verify . . .~~ ~~AND~~ ~~A.2.2.1 Reduce . . .~~ ~~OR~~ ~~A.2.2.2 Perform . . .~~ ~~OR~~ ~~A.3 Align . . .~~	

~~This example represents a more complicated use of logical connectors. Required Actions A.1, A.2, and A.3 are alternative choices, only one of which must be performed as indicated by the use of the logical connector OR and the left justified placement. Any one of these three Actions may be chosen. If A.2 is chosen, then both A.2.1 and A.2.2 must be performed as indicated by the logical connector AND. Required Action A.2.2 is met by performing A.2.2.1 or A.2.2.2. The indented position of the logical connector OR indicates that A.2.2.1 and A.2.2.2 are alternative choices, only one of which must be performed.~~

1.0 USE AND APPLICATION

1.3 Completion Times

PURPOSE

The purpose of this section is to establish the Completion Time convention and to provide guidance for its use.

BACKGROUND

storage of irradiated fuel

Limiting Condition for Operation (LCOs) specify minimum requirements for ensuring safe ~~operation of the unit~~. The ACTIONS associated with an LCO state Conditions that typically describe the ways in which the requirements of the LCO can fail to be met. Specified with each stated Condition are Required Action(s) and Completion Time(s).

DESCRIPTION

The Completion Time is the amount of time allowed for completing a Required Action. It is referenced to the time of discovery of a situation (e.g., inoperable equipment or variable not within limits) that requires entering an ACTIONS Condition unless otherwise specified, providing the facility ~~unit~~ is in a ~~MODE or~~ specified condition stated in the Applicability of the LCO. Required Actions must be completed prior to the expiration of the specified Completion Time. An ACTIONS Condition remains in effect and the Required Actions apply until the Condition no longer exists or the facility ~~unit~~ is not within the LCO Applicability.

~~If situations are discovered that require entry into more than one Condition at a time within a single LCO (multiple Conditions), the Required Actions for each Condition must be performed within the associated Completion Time. When in multiple Conditions, separate Completion Times are tracked for each Condition starting from the time of discovery of the situation that required entry into the Condition.~~

~~Once a Condition has been entered, subsequent trains, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition, unless specifically stated. The Required Actions of the Condition continue to apply to each additional failure, with Completion Times based on initial entry into the Condition.~~

(continued)

1.3 Completion Times

~~DESCRIPTION (continued)~~

~~However, when a subsequent train, subsystem, component, or variable expressed in the Condition is discovered to be inoperable or not within limits, the Completion Time(s) may be extended. To apply this Completion Time extension, two criteria must first be met. The subsequent inoperability:~~

~~a. Must exist concurrent with the first inoperability; and~~

~~b. Must remain inoperable or not within limits after the first inoperability is resolved.~~

~~The total Completion Time allowed for completing a Required Action to address the subsequent inoperability shall be limited to the more restrictive of either:~~

~~a. The stated Completion Time, as measured from the initial entry into the Condition, plus an additional 24 hours; or~~

~~b. The stated Completion Time as measured from discovery of the subsequent inoperability.~~

~~The above Completion Time extensions do not apply to those Specifications that have exceptions that allow completely separate re-entry into the Condition (for each train, subsystem, component, or variable expressed in the Condition) and separate tracking of Completion Times based on this re-entry. These exceptions are stated in individual Specifications.~~

~~The above Completion Time extension does not apply to a Completion Time with a modified "time zero." This modified "time zero" may be expressed as a repetitive time (i.e., "once per 8 hours," where the Completion Time is referenced from a previous completion of the Required Action versus the time of Condition entry) or as a time modified by the phrase "from discovery . . ." Example 1.3-3 illustrates one use of this type of Completion Time. The 10 day Completion Time specified for Conditions A and B in Example 1.3-3 may not be extended.~~

(continued)

1.3 Completion Times (continued)

EXAMPLE~~S~~

The following example~~s~~ illustrate s the use of Completion Times ~~with different types of Conditions and changing Conditions~~.

EXAMPLE 1.3-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1 Verify . . . ~~Be in MODE 3~~. AND B.2 Restore . . . ~~Be in MODE 5~~.	6 hours 36 hours

Condition B has two Required Actions. Each Required Action has its own separate Completion Time. Each Completion Time is referenced to the time that Condition B is entered.

The Required Actions of Condition B are to perform the verification ~~Be in MODE 3~~ within 6 hours AND perform the restoration ~~in MODE 5~~ within 36 hours. A total of 6 hours is allowed for performing the verification ~~reaching MODE 3~~ and a total of 36 hours (not 42 hours) is allowed for performing the restoration ~~reaching MODE 5~~ from the time that Condition B was entered. If verification is performed ~~MODE 3 is reached~~ within 3 hours, the time allowed for performing the restoration ~~reaching MODE 5~~ is the next 33 hours because the total time allowed for performing the restoration ~~reaching MODE 5~~ is 36 hours.

~~If Condition B is entered while in MODE 3, the time allowed for reaching MODE 5 is the next 36 hours.~~

(continued)

1.3 Completion Times

~~EXAMPLES (continued)~~

~~EXAMPLE 1.3-2~~

~~ACTIONS~~

~~CONDITION~~	~~REQUIRED ACTION~~	~~COMPLETION TIME~~
~~A. One pump inoperable.~~	~~A.1 Restore pump to OPERABLE status.~~	~~7 days~~
~~B. Required Action and associated Completion Time not met.~~	~~B.1 Be in MODE 3.~~ ~~AND~~ ~~B.2 Be in MODE 5.~~	~~6 hours~~ ~~36 hours~~

~~When a pump is declared inoperable, Condition A is entered. If the pump is not restored to OPERABLE status within 7 days, Condition B is also entered and the Completion Time clocks for Required Actions B.1 and B.2 start. If the inoperable pump is restored to OPERABLE status after Condition B is entered, Condition A and B are exited, and therefore the Required Actions of Condition B may be terminated.~~

~~When a second pump is declared inoperable while the first pump is still inoperable, Condition A is not re-entered for the second pump. LCO 3.0.3 is entered, since the ACTIONS do not include a Condition for more than one inoperable pump. The Completion Time clock for Condition A does not stop after LCO 3.0.3 is entered, but continues to be tracked from the time Condition A was initially entered.~~

~~While in LCO 3.0.3, if one of the inoperable pumps is restored to OPERABLE status and the Completion Time for Condition A has not expired, LCO 3.0.3 may be exited and operation continued in accordance with Condition A.~~

(continued)

1.3 Completion Times

~~EXAMPLES~~

~~EXAMPLE 1.3-2 (continued)~~

~~While in LCO 3.0.3, if one of the inoperable pumps is restored to OPERABLE status and the Completion Time for Condition A has expired, LCO 3.0.3 may be exited and operation continued in accordance with Condition B. The Completion Time for Condition B is tracked from the time the Condition A Completion Time expired.~~

~~On restoring one of the pumps to OPERABLE status, the Condition A Completion Time is not reset, but continues from the time the first pump was declared inoperable. This Completion Time may be extended if the pump restored to OPERABLE status was the first inoperable pump. A 24 hour extension to the stated 7 days is allowed, provided this does not result in the second pump being inoperable for > 7 days.~~

(continued)

1.3 Completion Times

~~EXAMPLES (continued)~~

~~EXAMPLE 1.3-3~~

~~ACTIONS~~

~~CONDITION~~	~~REQUIRED ACTION~~	~~COMPLETION TIME~~
~~A. One Function X train inoperable.~~	~~A.1 Restore Function X train to OPERABLE status.~~	~~7 days~~ ~~AND~~ ~~10 days from discovery of failure to meet the LCO~~
~~B. One Function Y train inoperable.~~	~~B.1 Restore Function Y train to OPERABLE status.~~	~~72 hours~~ ~~AND~~ ~~10 days from discovery of failure to meet the LCO~~
~~C. One Function X train inoperable.~~ ~~AND~~ ~~One Function Y train inoperable.~~	~~C.1 Restore Function X train to OPERABLE status.~~ ~~OR~~ ~~C.2 Restore Function Y train to OPERABLE status.~~	~~72 hours~~ ~~72 hours~~

~~(~~continued)

1.3 Completion Times

~~EXAMPLES~~

~~Example 1.3-3 (continued)~~

~~When one Function X train and one Function Y train are inoperable, Condition A and Condition B are concurrently applicable. The Completion Times for Condition A and Condition B are tracked separately for each train starting from the time each train was declared inoperable and the Condition was entered. A separate Completion Time is established for condition C and tracked from the time the second train was declared inoperable (i.e., the time the situation described in Condition C was discovered).~~

~~If Required Action C.2 is completed within the specified Completion Time, Conditions B and C are exited. If the Completion Time for Required Action A.1 has not expired, operation may continue in accordance with Condition A. The remaining Completion Time in Condition A is measured from the time the affected train was declared inoperable (i.e., initial entry into Condition A).~~

~~The Completion Times of Conditions A and B are modified by a logical connector, with a separate 10 day Completion Time measured from the time it was discovered the LCO was not met. In this example, without the separate Completion Time, it would be possible to alternate between Conditions A, B, and C in such a manner that operation could continue indefinitely without ever restoring systems to meet the LCO. The separate Completion Time modified by the phrase "from discovery of failure to meet the LCO" is designed to prevent indefinite continued operation while not meeting the LCO. This Completion Time allows for an exception to the normal "time zero" for beginning the Completion Time "clock." In this instance, the Completion Time "time zero" is specified as commencing at the time the LCO was initially not met, instead of at the time the associated Condition was entered.~~

(continued)

1.3 Completion Times

~~EXAMPLES (continued)~~

~~EXAMPLE 1.3-4~~

~~ACTIONS~~

~~CONDITION~~	~~REQUIRED ACTION~~	~~COMPLETION TIME~~
~~A. One or more valves inoperable.~~	~~A.1 Restore valve(s) to OPERABLE status.~~	~~4 hours~~
~~B. Required Action and associated Completion Time not met.~~	~~B.1 Be in MODE 3.~~ ~~AND~~ ~~B.2 Be in MODE 4.~~	~~6 hours~~ ~~12 hours~~

~~A single Completion Time is used for any number of valves inoperable at the same time. The Completion Time associated with Condition A is based on the initial entry into Condition A and is not tracked on a per-valve basis. Declaring subsequent valves inoperable, while Condition A is still in effect, does not trigger the tracking of separate Completion Times.~~

~~Once one of the valves has been restored to OPERABLE status, the Condition A Completion Time is not reset, but continues from the time the first valve was declared inoperable. The Completion Time may be extended if the valve restored to OPERABLE status was the first inoperable valve. The Condition A Completion Time may be extended for up to 4 hours provided this does not result in any subsequent valve being inoperable for > 4 hours.~~

~~If the Completion Time of 4 hours (including any extensions) expires while one or more valves are still inoperable, Condition B is entered.~~

(continued)

1.3 Completion Times

~~EXAMPLES (continued)~~

~~EXAMPLE 1.3-5~~

~~ACTIONS~~

~~NOTE~~
~~Separate Condition entry is allowed for each inoperable valve.~~

~~CONDITION~~	~~REQUIRED ACTION~~	~~COMPLETION TIME~~
~~A. One or more valves inoperable.~~	~~A.1 Restore valve to OPERABLE status.~~	~~4 hours~~
~~B. Required Action and associated Completion Time not met.~~	~~B.1 Be in MODE 3.~~	~~6 hours~~
	~~AND~~	
	~~B.2 Be in MODE 4.~~	~~12 hours~~

~~The Note above the ACTIONS table is a method of modifying how the Completion Time is tracked. If this method of modifying how the Completion Time is tracked was applicable only to a specific Condition, the Note would appear in that Condition rather than at the top of the ACTIONS Table.~~

~~The Note allows Condition A to be entered separately for each inoperable valve, and Completion Times tracked on a per valve basis. When a valve is declared inoperable, Condition A is entered and its Completion Time starts. If subsequent valves are declared inoperable, Condition A is entered for each valve and separate Completion Times start and are tracked for each valve.~~

~~(~~continued)

1.3 Completion Times

~~EXAMPLES~~

~~EXAMPLE 1.3-5 (continued)~~

~~If the Completion Time associated with a valve in Condition A expires, Condition B is entered for that valve. If the Completion Times associated with subsequent valves in Condition A expire, Condition B is entered separately for each valve and separate Completion Times start and are tracked for each valve. If a valve that caused entry into Condition B is restored to OPERABLE status, Condition B is exited for that valve.~~

~~Since the Note in this example allows multiple Condition entry and tracking of separate Completion Times, Completion Time extensions do not apply.~~

~~EXAMPLE 1.3-6~~

~~ACTIONS~~

~~CONDITION~~	~~REQUIRED ACTION~~	~~COMPLETION TIME~~
~~A. One channel inoperable.~~	~~A.1 Perform SR 3.x.x.x.~~	~~Once per 8 hours~~
	~~OR~~	
	~~A.2 Reduce THERMAL POWER to ≤ 50% RTP.~~	~~8 hours~~
~~B. Required Action and associated Completion Time not met.~~	~~B.1 Be in MODE 3.~~	~~6 hours~~

(continued)

1.3 Completion Times

~~EXAMPLES~~

~~EXAMPLE 1.3-6 (continued)~~

~~Entry into Condition A offers a choice between Required Action A.1 or A.2. Required Action A.1 has a "Once per" Completion Time, which qualifies for the 25% extension, per SR 3.0.2, to each performance after the initial performance. If Required Action A.1 is followed and the Required Action is not met within the Completion Time (including the 25% extension allowed by SR 3.0.2), Condition B is entered. If Required Action A.2 is followed and the Completion Time of 8 hours is not met, Condition B is entered.~~

~~If after entry into Condition B, Required Action A.1 or A.2 is met, Condition B is exited and operation may then continue in Condition A.~~

(continued)

1.3 Completion Times

~~EXAMPLES (continued)~~

~~EXAMPLE 1.3-7~~

~~ACTIONS~~

~~CONDITION~~	~~REQUIRED ACTION~~	~~COMPLETION TIME~~
~~A. One subsystem inoperable.~~	~~A.1 Verify affected subsystem isolated.~~	~~1 hour~~ ~~AND~~ ~~Once per 8 hours thereafter~~
	~~AND~~	
	~~A.2 Restore subsystem to OPERABLE status.~~	~~72 hours~~
~~B. Required Action and associated Completion Time not met.~~	~~B.1 Be in MODE 3.~~	~~6 hours~~
	~~AND~~	
	~~B.2 Be in MODE 5.~~	~~36 hours~~

~~Required Action A.1 has two Completion Times. The 1 hour Completion Time begins at the time the Condition is entered and each "Once per 8 hours thereafter" interval begins upon performance of Required Action A.1.~~

~~If after Condition A is entered, Required Action A.1 is not met within either the initial 1 hour or any subsequent 8 hour interval from the previous performance (including the 25% extension allowed by SR 3.0.2), Condition B is entered.~~

(continued)

1.3 Completion Times

~~EXAMPLES~~

~~EXAMPLE 1.3-7 (continued)~~

~~The Completion Time clock for Condition A does not stop after Condition B is entered, but continues from the time Condition A was initially entered. If Required Action A.1 is met after Condition B is entered, Condition B is exited and operation may continue in accordance with Condition A, provided the Completion Time for Required Action A.2 has not expired.~~

IMMEDIATE COMPLETION TIME

When "Immediately" is used as a Completion Time, the Required Action should be pursued without delay and in a controlled manner.

1.0 USE AND APPLICATION

1.4 Frequency

PURPOSE

The purpose of this section is to define the proper use and application of Frequency requirements.

DESCRIPTION

Each Surveillance Requirement (SR) has a specified Frequency in which the Surveillance must be met in order to meet the associated LCO. An understanding of the correct application of the specified Frequency is necessary for compliance with the SR.

The "specified Frequency" is referred to throughout this section and each of the Specifications of Section 3.0, Surveillance Requirement (SR) Applicability. The "specified Frequency" consists of the requirements of the Frequency column of each SR, as well as certain Notes in the Surveillance column that modify performance requirements.

~~Situations where a Surveillance could be required (i.e., its Frequency could expire), but where it is not possible or not desired that it be performed until sometime after the associated LCO is within its Applicability, represent potential SR 3.0.4 conflicts. To avoid these conflicts, the SR (i.e., the Surveillance or the Frequency) is stated such that it is only "required" when it can be and should be performed. With an SR satisfied, SR 3.0.4 imposes no restriction.~~

EXAMPLES

The following examples illustrate the various ways that Frequencies are specified. In these examples, the Applicability of the LCO (LCO not shown) ~~is MODES 1, 2, and 3.~~

occurs whenever any fuel assembly is stored in the fuel storage pool

(continued)

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify . . . ~~Perform CHANNEL CHECK.~~	7 days ~~12 hours~~

Example 1.4-1 contains the type of SR most often encountered in the Technical Specifications (TS). The Frequency specifies an interval (7 days ~~12 hours~~) during which the associated Surveillance must be performed at least one time. Performance of the Surveillance initiates the subsequent interval. Although the Frequency is stated as 7 days ~~12 hours~~, an extension of the time interval to 1.25 times the stated Frequency is allowed by SR 3.0.2 for operational flexibility. The measurement of this interval continues at all times, even when the SR is not required to be met per SR 3.0.1 (such as when the equipment is inoperable, a variable is outside specified limits, or the facility ~~unit~~ is outside the Applicability of the LCO). If the interval specified by SR 3.0.2 is exceeded while the facility ~~unit~~ is in a ~~MODE or other~~ specified condition in the Applicability of the LCO~~, and the performance of the Surveillance is not otherwise modified (refer to Example 1.4-3)~~, then SR 3.0.3 becomes applicable.

If the interval as specified by SR 3.0.2 is exceeded while the facility ~~unit~~ is not in a ~~MODE or other~~ specified condition in the Applicability of the LCO for which performance of the SR is required, the Surveillance must be performed within the Frequency requirements of SR 3.0.2 prior to entry into the ~~MODE or other~~ specified condition. Failure to do so would result in a violation of SR 3.0.4.

(continued)

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
~~Verify flow is within limits.~~ Verify . . .	~~Once within 12 hours after ≥ 25% RTP~~ Prior to moving a fuel assembly . . . ~~AND~~ ~~24 hours thereafter~~

Example 1.4-2 illustrates ~~has two Frequencies.~~ ~~The first is~~ a one time performance Frequency~~, and the second is of the type shown in Example 1.4-1~~. ~~The logical connector "AND" indicates that both Frequency requirements must be met.~~ ~~Each time reactor power is increased from a power level < 25% RTP to ≥ 25% RTP, the Surveillance must be performed within 12 hours.~~

~~The use of "once" indicates a single performance will satisfy the specified Frequency (assuming no other Frequencies are connected by "AND").~~ This type of Frequency does not qualify for the 25% extension allowed by SR 3.0.2. ~~"Thereafter" indicates future performances must be established per SR 3.0.2, but only after a specified condition is first met (i.e., the "once" performance in this example).~~ ~~If reactor power decreases to < 25% RTP, the measurement of both intervals stops.~~ ~~New intervals start upon reactor power reaching 25% RTP.~~

(continued)

1.4 Frequency

~~EXAMPLES (continued)~~

~~EXAMPLE 1.4-3~~

~~SURVEILLANCE REQUIREMENTS~~

~~SURVEILLANCE~~	~~FREQUENCY~~
~~-----NOTE-----~~ ~~Not required to be performed until 12 hours after ≥ 25% RTP.~~ ~~----------~~ ~~Perform channel adjustment.~~	~~7 days~~

~~The interval continues, whether or not the unit operation is < 25% RTP between performances.~~

~~As the Note modifies the required performance of the Surveillance, it is construed to be part of the "specified Frequency." Should the 7 day interval be exceeded while operation is < 25% RTP, this Note allows 12 hours after power reaches ≥ 25% RTP to perform the Surveillance. The Surveillance is still considered to be performed within the "specified Frequency." Therefore, if the Surveillance were not performed within the 7 day (plus 25% per SR 3.0.2) interval, but operation was < 25% RTP, it would not constitute a failure of the SR or failure to meet the LCO. Also, no violation of SR 3.0.4 occurs when changing MODES, even with the 7 day Frequency not met, provided operation does not exceed 12 hours with power ≥ 25% RTP.~~

~~Once the unit reaches 25% RTP, 12 hours would be allowed for completing the Surveillance. If the Surveillance were not performed within this 12 hour interval, there would then be a failure to perform a Surveillance within the specified Frequency; MODE changes then would be restricted in accordance with SR 3.0.4 and the provisions of SR 3.0.3 would apply.~~

3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

LCO 3.0.1

LCOs shall be met during the ~~MODES or other~~ specified conditions in the Applicability, except as provided in LCO 3.0.2 ~~and LCO 3.0.7~~.

LCO 3.0.2

Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met~~, except as provided in LCO 3.0.5 and LCO 3.0.6~~.

If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required, unless otherwise stated.

~~LCO 3.0.3~~

~~When an LCO is not met and the associated ACTIONS are not met or an associated ACTION is not provided, the unit shall be placed in a MODE or other specified condition in which the LCO is not applicable. Action shall be initiated within 1 hour to place the unit, as applicable, in:~~

~~a. MODE 3 within 7 hours;~~

~~b. MODE 4 within 13 hours; and~~

~~c. MODE 5 within 37 hours.~~

~~Exceptions to this Specification are stated in the individual Specifications.~~

~~Where corrective measures are completed that permit operation in accordance with the LCO or ACTIONS, completion of the actions required by LCO 3.0.3 is not required.~~

~~LCO 3.0.3 is applicable in MODES 1, 2, 3, and 4.~~

~~LCO 3.0.4~~

~~When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall not be made except when the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time. This~~

(continued)

3.0 LCO APPLICABILITY (continued)

~~LCO 3.0.4 (continued)~~	~~Specification shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS.~~ ~~Exceptions to this Specification are stated in the individual Specifications. These exceptions allow entry into MODES or other specified conditions in the Applicability when the associated ACTIONS to be entered allow unit operation in the MODE or other specified condition in the Applicability only for a limited period of time.~~
~~LCO 3.0.5~~	~~Equipment removed from service or declared inoperable to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to LCO 3.0.2 for the system returned to service under administrative control to perform the testing required to demonstrate OPERABILITY.~~
~~LCO 3.0.6~~	~~When a supported system LCO is not met solely due to a support system LCO not being met, the Conditions and Required Actions associated with this supported system are not required to be entered. Only the support system LCO ACTIONS are required to be entered. This is an exception to LCO 3.0.2 for the supported system. In this event, additional evaluations and limitations may be required in accordance with Specification 5.6, "Safety Function Determination Program (SFDP)." If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.~~ ~~When a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.~~

(continued)

3.0 LCO APPLICABILITY (continued)

~~LCO 3.0.7~~ ~~Special test exception (STE) LCOs in each applicable LCO section allow specified Technical Specifications (TS) LCO 3.0.7 requirements to be changed to permit performance of special tests and operations. Unless otherwise specified, all other TS requirements remain unchanged. Compliance with STE LCOs is optional. When an STE LCO is desired to be met but is not met, the ACTIONS of the STE LCO shall be met. When an STE LCO is not desired to be met, entry into a MODE or other specified condition in the Applicability shall only be made in accordance with the other applicable Specifications.~~

3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 3.0.1

SRs shall be met during the ~~MODES or other~~ specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.

SR 3.0.2

The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met.

~~For Frequencies specified as "once," the above interval extension does not apply.~~

~~If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance.~~

~~Exceptions to this Specification are stated in the individual Specifications.~~

SR 3.0.3

If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is greater. This delay period is permitted to allow performance of the Surveillance. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.

If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered. The Completion Times of the Required Actions begin immediately upon expiration of the delay period.

(continued)

3.0 SR APPLICABILITY

SR 3.0.3 (continued)	When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered. The Completion Times of the Required Actions begin immediately upon failure to meet the Surveillance.
SR 3.0.4	Entry into a ~~MODE or other~~ specified condition in the Applicability of an LCO shall not be made unless the LCO's Surveillances have been met within their specified Frequency. This provision shall not prevent entry into ~~MODES or other~~ specified conditions in the Applicability that are required to comply with Actions.

~~3.7~~ 3.1 PLANT SYSTEMS

~~3.7.16~~ 3.1.1 Fuel Storage Pool Water Level

LCO ~~3.7.16~~ 3.1.1 The fuel storage pool water level shall be ≥ 23 ft over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: During movement of fuel assemblies in the fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Fuel storage pool water level not within limit.	A.1 ~~---------NOTE----------~~ ~~LCO 3.0.3 is not applicable.~~ ~~---------------------~~ Suspend movement of fuel assemblies in fuel storage pool.	 Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR ~~3.7.16.1~~ 3.1.1.1 Verify the fuel storage pool water level is ≥ 23 ft above the top of irradiated fuel assemblies seated in the storage racks.	7 days

~~3.7~~ 3.1 PLANT SYSTEMS

~~3.7.17~~ 3.1.2 Fuel Storage Pool Boron Concentration

LCO ~~3.7.17~~ 3.1.2 The fuel storage pool boron concentration shall be ≥ 2000 ppm.

APPLICABILITY: **Whenever any fuel assembly is stored in the fuel storage** pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Fuel storage pool boron concentration not within limit.	~~------------NOTE------------~~ ~~LCO 3.0.3 is not applicable.~~ ~~----------------------------~~	
	A.1 Suspend movement of fuel assemblies in the fuel storage pool.	Immediately
	AND	
	A.2 Initiate action to restore fuel storage pool boron concentration to within limit.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR ~~3.7.17.1~~ 3.1.2.1	Verify the fuel storage pool boron concentration is within limit.	7 days

~~3.7~~ 3.1 PLANT SYSTEMS

~~3.7.18~~ 3.1.3 Spent Fuel Assembly Storage

LCO ~~3.7.18~~ 3.1.3 The combination of initial enrichment and burnup of each SONGS 2 and 3 spent fuel assembly stored in Region I shall be within the acceptable burnup domain of Figure ~~3.7.18-1~~ 3.1.3-1 or Figure ~~3.7.18-2~~ 3.1.3-2, or the fuel assembly shall be stored in accordance with Technical Specification 4.3.1.1.

The combination of initial enrichment and burnup of each SONGS 2 and 3 spent fuel assembly stored in Region II shall be within the acceptable burnup domain of Figure ~~3.7.18-3~~ 3.1.3-3 or Figure ~~3.7.18-4~~ 3.1.3-4, or the fuel assembly shall be stored in accordance with Technical Specification 4.3.1.1.

Each SONGS 1 uranium dioxide spent fuel assembly stored in Region II shall be stored in accordance with Technical Specification 4.3.1.1.

APPLICABILITY: Whenever any fuel assembly is stored in the fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 ~~---------NOTE---------~~ ~~LCO 3.0.3 is not applicable.~~ ~~----------------------~~ Initiate action to bring the noncomplying fuel assembly into compliance.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR ~~3.7.18.1~~ 3.1.3.1 Verify by administrative means the initial enrichment, burnup, and cooling time of the fuel assembly are in accordance with LCO ~~3.7.18~~ 3.1.3, or Design Features 4.3.1.1, or LCS 4.0.100. Rev 2, dated 09/27/07.	Prior to moving a fuel assembly to any spent fuel pool storage location.

FIGURE ~~3.7.18-1~~ 3.1.3-1

MINIMUM BURNUP AND COOLING TIME VS. INITIAL ENRICHMENT
FOR
UNRESTRICTED PLACEMENT OF SONGS 2 AND 3 FUEL
IN
REGION I RACKS

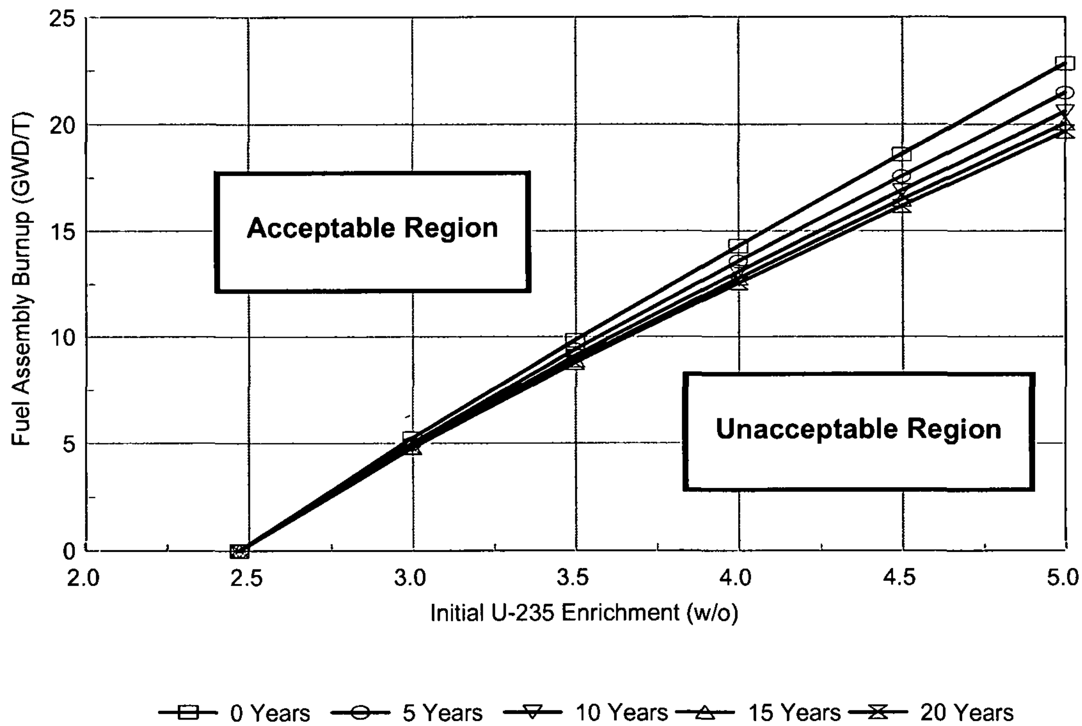


FIGURE ~~3.7.18-2~~ 3.1.3-2

MINIMUM BURNUP AND COOLING TIME VS. INITIAL ENRICHMENT
FOR
PLACEMENT OF SONGS 2 AND 3 FUEL IN PERIPHERAL POOL LOCATIONS
IN
REGION I RACKS

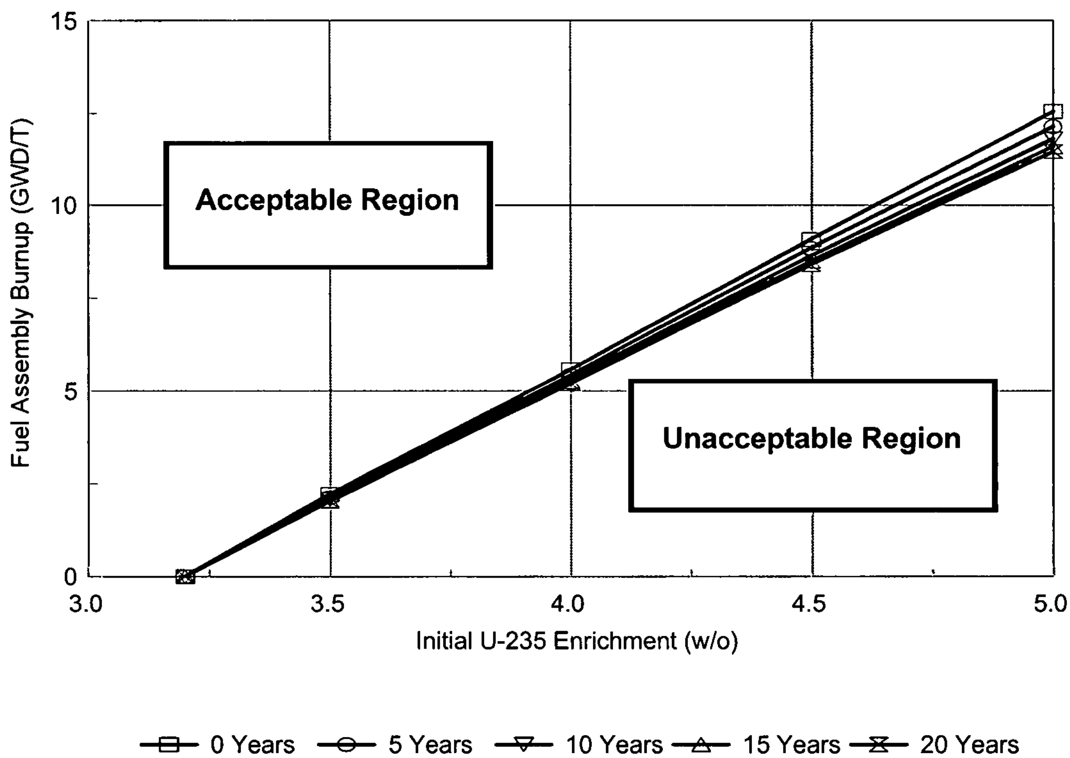


FIGURE ~~3.7.18-3~~ 3.1.3-3

MINIMUM BURNUP AND COOLING TIME VS. INITIAL ENRICHMENT
FOR
UNRESTRICTED PLACEMENT OF SONGS 2 AND 3 FUEL
IN
REGION II RACKS

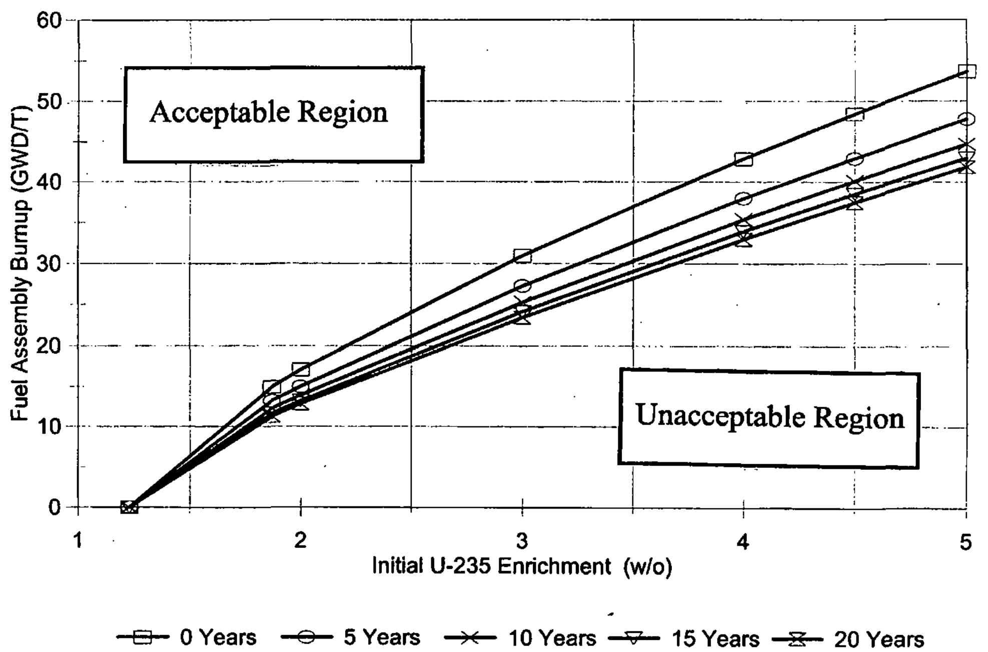
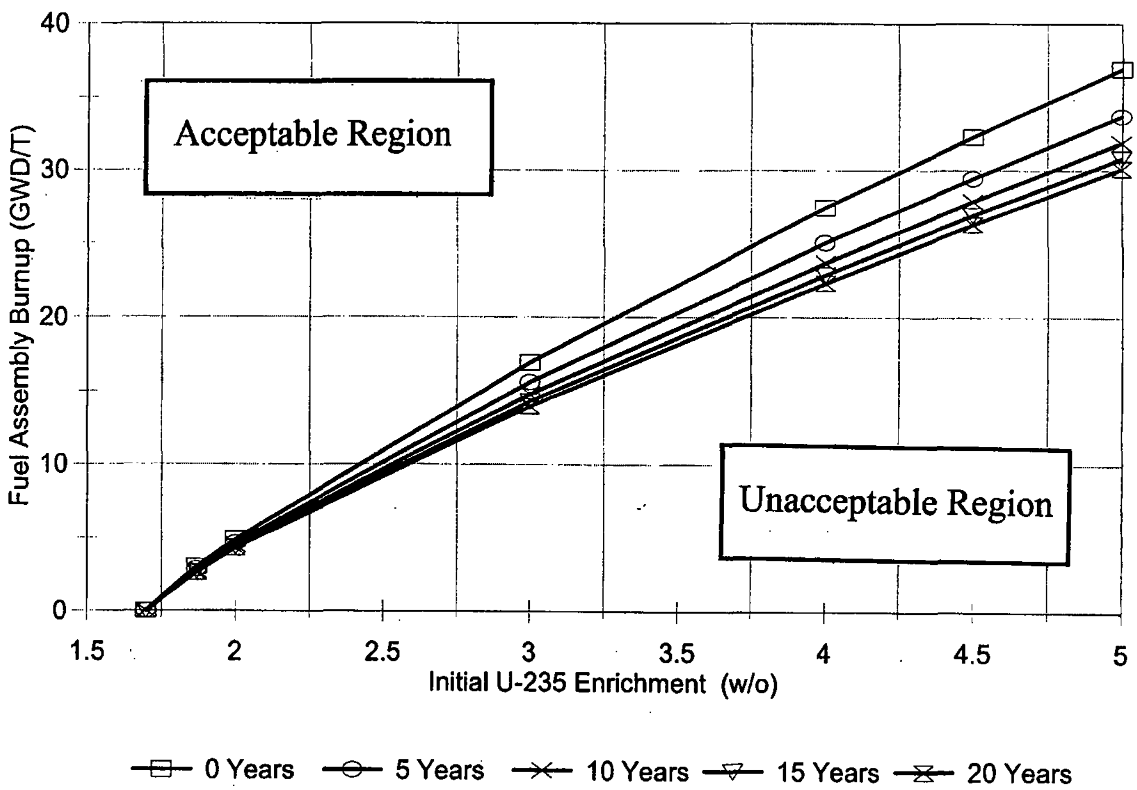


FIGURE ~~3.7.18-4~~
3.1.3-4

MINIMUM BURNUP AND COOLING TIME VS. INITIAL ENRICHMENT
FOR
PLACEMENT OF SONGS 2 AND 3 FUEL IN PERIPHERAL POOL LOCATIONS
IN
REGION II RACKS



4.0 DESIGN FEATURES

4.1 Site

4.1.1 Exclusion Area Boundary

The exclusion area boundary shall be as shown in Figure 4.1-1.

4.1.2 Low Population Zone (LPZ)

The LPZ shall be as shown in Figure 4.1-2.

4.2 ~~Reactor Core~~ Deleted.

~~4.2.1 Fuel Assemblies~~

~~The reactor shall contain 217 fuel assemblies. Each assembly shall consist of a matrix of Zircaloy or ZIRLO™ clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Integral or Discrete Burnable Absorber Rods may be used. They may include: borosilicate glass - Na_2O-B_2O_3-SiO_2 components, boron carbide - B_4C, zirconium boride - ZrB_2, gadolinium oxide - Gd_2O_3, erbium oxide - Er_2O_3. Limited substitutions of zirconium alloy (such as ZIRLO™ or Zircaloy) 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.~~

~~4.2.2 Control Element Assemblies~~

~~The reactor core shall contain 83 full length and eight part length control element assemblies (CEAs). The control material shall be silver indium cadmium, boron carbide, and inconel as approved by the NRC.~~

(continued)

4.0 DESIGN FEATURES (continued)

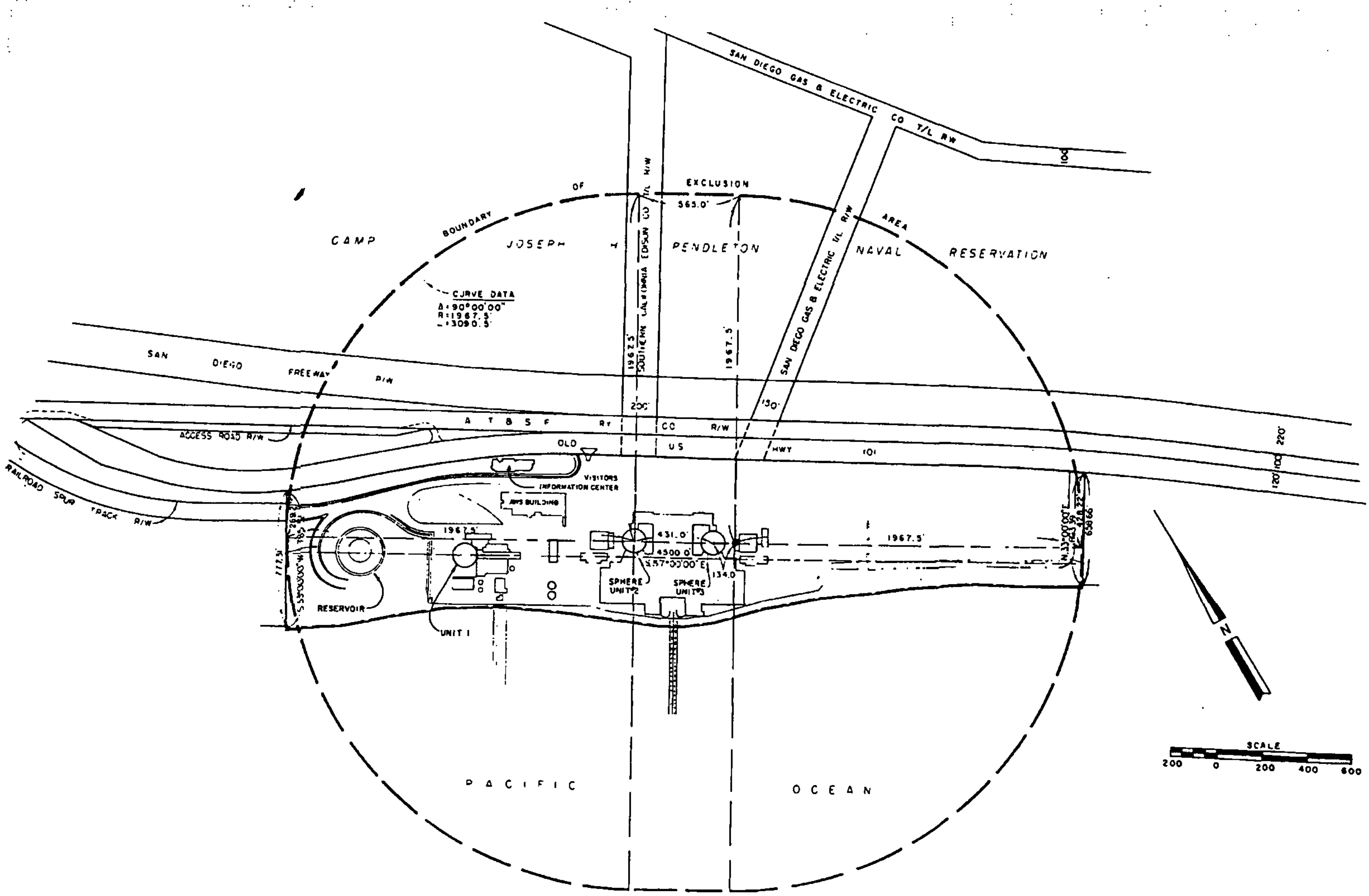


Figure 4.1-1 (page 1 of 1)
Exclusion Area Boundary

(continued)

4.0 DESIGN FEATURES (continued)

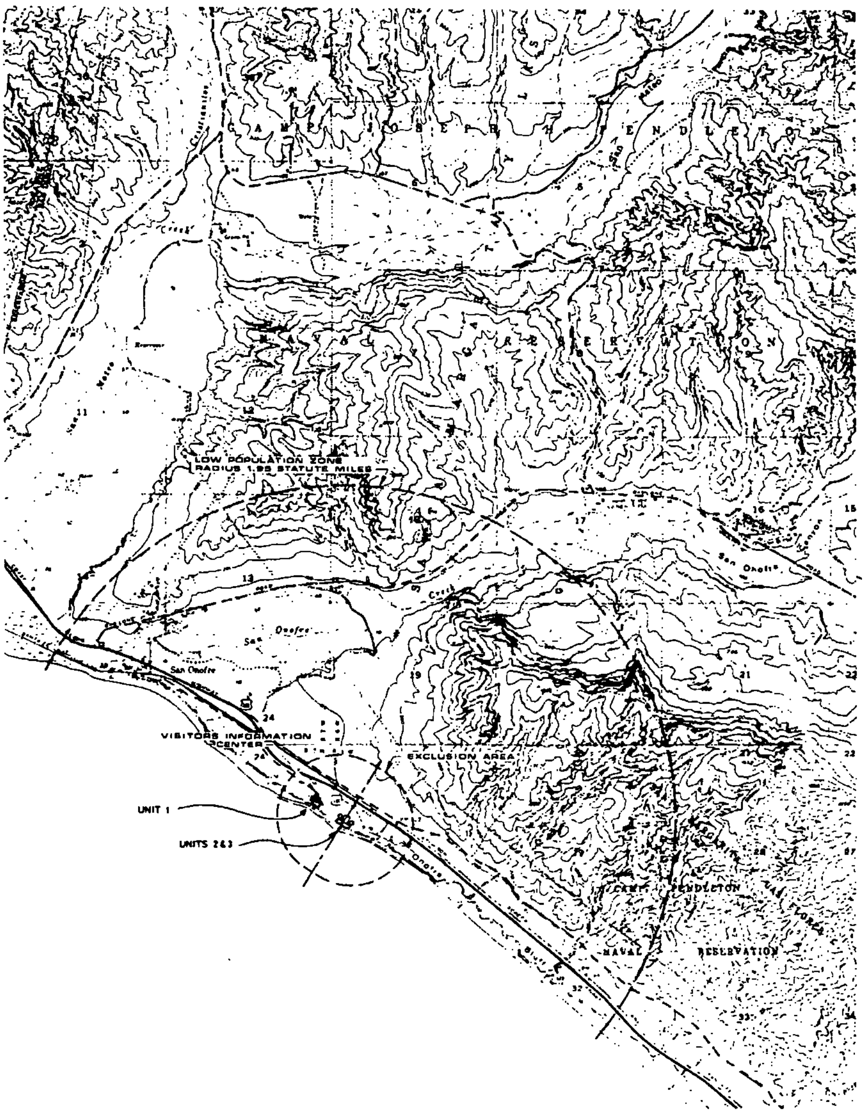


Figure 4.1-2 (page 1 of 1)
Low Population Zone

(continued)

4.0 DESIGN FEATURES (continued)

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 4.8 weight percent;

b. $K_{eff} < 1.0$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR;

c. $K_{eff} \leq 0.95$ if fully flooded with water borated to 1700 ppm, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR;

d. Three or five Borated stainless steel guide tube inserts (GT-Insert) may be used. When three Borated stainless steel guide tube inserts are used, they will be installed in an assembly's center guide tube, the guide tube associated with the serial number, and the diagonally opposite guide tube. Fuel containing GT-Inserts may be placed in either Region I or Region II. However, credit for GT-Inserts is only taken for Region II storage.

   A five-finger CEA may be installed in an assembly. Fuel containing a five-finger CEA may be placed in either Region I or Region II. Credit for inserted 5-finger CEAs is taken for both Region I and Region II.

e. A nominal 8.85 inch center to center distance between fuel assemblies placed in Region II;

f. A nominal 10.40 inch center to center distance between fuel assemblies placed in Region I;

(continued)

4.0 DESIGN FEATURES (continued)

4.3.1 Criticality (continued)

3.1.3

g. Prior to using the storage criteria of LCO ~~3.7.18~~ and LCS 4.0.100, the following uncertainties will be applied:

(1) The calculated discharge burnup of San Onofre Units 2 and 3 assemblies will be reduced by 6.6%.

(2) The calculated discharge burnup of San Onofre Unit 1 fuel assemblies will be reduced by 10.0%.

3.1.3-1 h. Units 2 and 3 fuel assemblies with a burnup in the "acceptable range" of Figure ~~3.7.18-1~~ are allowed unrestricted storage in Region I;

3.1.3-2 i. Units 2 and 3 fuel assemblies with a burnup in the "acceptable range" of Figure ~~3.7.18-2~~ are allowed unrestricted storage in the peripheral pool locations with 1 or 2 faces toward the spent fuel pool walls of Region I;

3.1.3-3 j. Units 2 and 3 fuel assemblies with a burnup in the "acceptable range" of Figure ~~3.7.18-3~~ are allowed unrestricted storage in Region II;

3.1.3-4 k. Units 2 and 3 fuel assemblies with a burnup in the "acceptable range" of Figure ~~3.7.18-4~~ are allowed unrestricted storage in the peripheral pool locations with 1 or 2 faces toward the spent fuel pool walls of Region II;

Figure 3.1.3-1, Figure 3.1.3-2, Figure 3.1.3-3, and Figure 3.1.3-4

l. Units 2 and 3 fuel assemblies with a burnup in the "unacceptable range" of ~~Figure 3.7.18-1, Figure 3.7.18-2, Figure 3.7.18-3, and Figure 3.7.18-4~~ will be stored in compliance with Licensee Controlled Specification 4.0.100 Rev. 2, dated 09/27/07; and

m. Each SONGS 1 uranium dioxide spent fuel assembly stored in Region II shall be stored in accordance with Licensee Controlled Specification 4.0.100 Rev. 2, dated 09/27/07.

(continued)

4.0 DESIGN FEATURES (continued)

4.3 Fuel Storage (continued)

~~4.3.1.2~~ ~~The new fuel storage racks are designed and shall be maintained with:~~

~~a.~~ ~~Fuel assemblies having a maximum U-235 enrichment of 4.8 weight percent;~~

~~b.~~ ~~$K_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR;~~

~~c.~~ ~~$K_{eff} \leq 0.98$ if moderated by aqueous foam, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR; and~~

~~d.~~ ~~A minimum 29 inch center to center distance between fuel assemblies placed in the storage racks.~~

4.3.2 Drainage

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below Technical Specification 3.1.1 ~~3.7.16~~ value (23 feet above the top of irradiated fuel assemblies seated in the storage racks).

4.3.3 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1542 fuel assemblies.

5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

5.1.1 The corporate officer with direct responsibility for the plant shall be responsible for overall ~~unit operation and maintenance of~~ management of the ~~Units 2 and 3 at~~ San Onofre Nuclear Generating Station, and all site support functions. He shall delegate in writing the succession to this responsibility during his absence.

5.1.2 The Shift Manager shall be responsible for the ultimate command decision authority for all unit activities and operations which affect the safety of the plant, site personnel, and/or the general public. ~~A management directive to this effect, signed by the corporate officer with direct responsibility for the plant shall be reissued to all site/station personnel on an annual basis.~~

~~5.1.3~~ ~~The Control Room Supervisor (CRS) shall be responsible for the Control Room command function. A management directive to this effect, signed by the corporate officer with direct responsibility for the plant, shall be issued annually to all site/station personnel. The confines of the Control Room Area shall be defined as depicted in the Licensee Controlled Specification (LCS). During any absence of the CRS from the Control Room Area while the Unit is in MODE 1, 2, 3, or 4, an individual with an active Senior Reactor Operator's (SRO) license shall be designated to assume the Control Room command function. During any absence of the CRS from the Control Room Area while the Unit is in MODE 5 or 6, an individual with an active SRO license or Reactor Operator's license shall be designated to assume the Control Room command function.~~

5.0 ADMINISTRATIVE CONTROLS

5.2 Organization

5.2.1 Onsite and Offsite Organizations

Onsite and offsite organizations shall be established for ~~unit~~ plant operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting the safety of the nuclear ~~power plant~~ fuel.

a. Lines of authority, responsibility, and communication shall be established and defined for the highest management levels through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These relationships, including the plant-specific titles of those personnel fulfilling the responsibilities for the positions delineated in these Technical Specifications, are documented in the UFSAR.

b. The corporate officer with direct responsibility for the plant shall be responsible for overall ~~unit~~ safe handling and storage ~~operation~~ of nuclear fuel and shall have control over those onsite activities necessary for safe ~~operation and maintenance~~ handling and storage of the nuclear fuel ~~plant~~.

c. A specified corporate officer (or officers) shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure safe management of nuclear fuel ~~safety~~.

CERTIFIED FUEL HANDLERS

d. The individuals who train ~~the operating staff~~ ~~Certified Fuel Handlers~~, and those who carry out ~~health physics~~ radiation protection and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their ~~independence from operating pressures.~~ ability to perform their assigned functions.

(continued)

5.2 Organization (continued)

5.2.2 ~~UNIT~~ FACILITY STAFF

The ~~unit~~ facility staff organization shall include the following:

a. ~~A non-Licensed Operator shall be assigned to each reactor containing fuel and an additional non-Licensed Operator shall be assigned for each unit when a reactor is operating in MODES 1, 2, 3, or 4.~~ Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 5.2.2-1.

~~With both units shutdown or defueled, a total of three non-Licensed operators are required for the two units.~~

~~b.~~ ~~At least one licensed Reactor Operator (RO) shall be in the Control Room when fuel is in the reactor. In addition, while the unit is in MODE 1, 2, 3 or 4, at least one licensed Senior Reactor Operator (SRO) shall be in the Control Room Area.~~

~~c.~~b. Shift crew composition may be less than the minimum requirement of ~~10 CFR 50.54(m)(2)(i) and 5.2.2.a~~ Table 5.2.2-1 for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements. INSERT 2

INSERT 3

~~d.~~c. ~~A radiation protection technician shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.~~ Oversight of fuel handling operations shall be provided by a ~~Certified Fuel Handler~~. CERTIFIED FUEL HANDLER

~~e.~~ ~~Deleted~~

~~f.~~d. The Shift Manager~~, Plant Operations (at time of appointment), Shift Managers, and Control Room Supervisors~~ shall ~~hold~~ be a ~~Senior Reactor Operator's license.~~ ~~Certified Fuel Handler~~. CERTIFIED FUEL HANDLER

~~g.~~ ~~The Shift Technical Advisor (STA) shall provide advisory technical support to the Shift Manager in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. The STA shall have a Bachelor's Degree or equivalent in a scientific or engineering discipline with specific training in plant design and in the response and analysis of the plant for transients and accidents.~~

(continued)

INSERT 2

During such absences, no fuel movement or movement of heavy loads over storage racks containing fuel is permitted.

INSERT 3

A radiation protection technician shall be on site during fuel handling operations and during movement of heavy loads over storage racks containing fuel.

5.2 Organization (continued)

Table 5.2.2-1
Minimum Shift Crew Composition

POSITION	MINIMUM STAFFING
~~Certified Fuel Handler~~ CERTIFIED FUEL HANDLER	1*
Certified Operator	1

Note: The Certified Operator position may be filled by a ~~Certified Fuel Handler~~ CERTIFIED FUEL HANDLER.
* May be shared between Units 2 and 3.

(continued)

5.0 ADMINISTRATIVE CONTROLS

5.3 ~~Unit~~ Facility Staff Qualifications

5.3.1 Each member of the ~~unit~~ facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except ~~a)~~ the radiation protection manager who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975~~, and b) multi-discipline supervisors who shall meet or exceed the qualifications listed below~~.

~~In addition, the Shift Technical Advisor shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift.~~

~~Multi-discipline supervisors shall meet or exceed the following qualifications:~~

~~a. Education: Minimum of a high school diploma or equivalent.~~

~~b. Experience: Minimum of four years of related technical experience which shall include three years power plant experience of which one year is at a nuclear plant.~~

~~c. Training: Complete the multi-discipline supervisor training program.~~

5.3.2 An NRC approved training and retraining program for the ~~Certified Fuel Handlers~~ shall be maintained.

CERTIFIED FUEL HANDLERS

5.0 ADMINISTRATIVE CONTROLS

5.4 Technical Specifications (TS) Bases Control

5.4.1 Changes to the Bases of the TS shall be made under appropriate administrative controls.

5.4.2 Changes to the Bases may be made without prior NRC approval provided the changes do not require either of the following:

a. A change in the TS incorporated in the license; or

b. A change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

5.4.3 The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the UFSAR.

5.4.4 Proposed changes that meet the criteria of (a) or (b) above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC every ~~within 6 months following every Unit 3 refueling, not to exceed~~ 24 months. ~~This schedule is consistent with SCE's submittal of UFSAR updates as allowed by the NRC approved exemption from 10 CFR 50.71(e) dated April 27, 1999.~~

5.0 ADMINISTRATIVE CONTROLS

5.5 Procedures, Programs, and Manuals

5.5.1 Procedures

5.5.1.1 Scope

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

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

Deleted

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

c. Quality assurance for effluent and environmental monitoring using the guidance in Regulatory Guide 4.15, Revision 1, 1979;

d. Fire Protection Program implementation; and

e. Programs, as specified in Specification 5.5.2.

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

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

(continued)

5.5 Procedures, Programs, and Manuals (continued)

5.5.2 Programs and Manuals

The following programs and manuals shall be established, implemented, and maintained.

5.5.2.1 Offsite Dose Calculation Manual (ODCM)

a. The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the Radiological Environmental Monitoring Program;

b. The ODCM shall also contain the Radioactive Effluent Controls required by Specification 5.5.2.3 and the Radiological Environmental Monitoring programs required by the LCS, and descriptions of the information that should be included in the Annual Radiological Environmental Operating Report and the Radioactive Effluent Release Report required by Specification 5.7.1.2 and Specification 5.7.1.3.

5.5.2.1.1 Licensee-initiated changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

   1. Sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s);

   2. A determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.106, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

   3. Documentation of the fact that the change has been reviewed and found acceptable.

b. Shall become effective upon review and approval by the corporate officer with direct responsibility for the plant or designee.

(continued)

5.5 Procedures, Programs, and Manuals (continued)

5.5.2.1.1 Licensee-initiated changes to the ODCM: (continued)

c. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

5.5.2.2 Deleted

5.5.2.3 Radioactive Effluent Controls Program

This program conforming to 10 CFR 50.36a provides for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by operating procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to 10 CFR 20, Appendix B, Table II, Column 2;

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.106 and with the methodology and parameters in the ODCM;

(continued)

5.5 Procedures, Programs, and Manuals (continued)

5.5.2.3 Radioactive Effluent Controls Program (continued)

d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each unit to unrestricted areas, conforming to 10 CFR 50, Appendix I;

e. Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days;

f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2 percent of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I;

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the site boundary conforming to the dose associated with 10 CFR 20, Appendix B, Table II, Column 1;

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and

j. Limitations on the annual dose or dose commitment to any member of the public due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

5.5.2.4 ~~Component Cyclic or Transient Limit Program~~

Deleted

~~This program provides controls to track the UFSAR Table 3.9-1 cyclic and transient occurrences to ensure that components are maintained within the design limits.~~

(continued)

5.5 Procedures, Programs, and Manuals (continued)

5.5.2.5 ~~Reactor Coolant Pump Flywheel Inspection Program~~

Deleted

~~Surveillance of the primary coolant pump flywheels shall consist of a 100% volumetric inspection of the flywheels each 10 years.~~

5.5.2.6 ~~Secondary Water Chemistry Program~~

Deleted

~~This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation and low pressure turbine disc stress corrosion cracking.~~ ~~The program shall include:~~

~~a.~~ ~~Identification of a sampling schedule for the critical parameters and control points for these parameters;~~

~~b.~~ ~~Identification of the procedures used to measure the values of the critical parameters;~~

~~c.~~ ~~Identification of process sampling points;~~

~~d.~~ ~~Procedures for the recording and management of data;~~

~~e.~~ ~~Procedures defining corrective actions for all off-control point chemistry conditions; and~~

~~f.~~ ~~A procedure identifying (a) the authority responsible for interpretation of data and (b) the sequence and timing of administrative events, required to initiate corrective action.~~

5.5.2.7 ~~Explosive Gas and~~ Storage Tank Radioactivity Monitoring Program

This program provides controls for ~~potentially explosive gas mixtures contained in the Gaseous Radwaste System, the quantity of radioactivity contained in gas storage tanks, and~~ the quantity of radioactivity contained in unprotected outdoor liquid storage tanks. ~~The gaseous radioactivity quantities shall be determined following methodology comparable with Branch Technical Position (BTP) ETSB 11-5, "Postulated Radioactive Release due to Waste Gas System Leak or Failure".~~ The liquid radwaste quantities shall be determined in accordance with Standard Review Plan, Section 15.7.3, "Postulated Radioactive Release due to Tank Failures".

(continued)

5.5 Procedures, Programs, and Manuals (continued)

5.5.2.7 ~~Explosive Gas and~~ Storage Tank Radioactivity Monitoring Program (continued)

The program shall include~~:~~

~~a.~~ ~~The limits for the concentrations of hydrogen and oxygen in the Gaseous Radwaste System and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion); and~~

~~b.~~ ~~A surveillance program to ensure that the quantity of radioactivity contained in each waste gas decay tank and fed into the gaseous radwaste vent system is less than the amount that would result in a whole body exposure of greater than or equal to 0.5 rem to any individual in the unrestricted area, in the event of an uncontrolled release of the tanks contents; and~~

~~c.~~ a ~~A~~ surveillance program to ensure that the quantity of radioactivity contained in all outdoor liquid radwaste tanks that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the Liquid Waste Management System is less than the amount that would result in concentrations less than the limits of 10 CFR Part 20, Appendix B, Table II, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tanks' contents.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the ~~Explosive Gas and~~ Storage Tank Radioactivity Monitoring Program surveillance frequencies.

~~5.5.2.8~~ ~~Primary Coolant Sources Outside Containment Program~~

~~This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The systems include high pressure safety injection recirculation, the shutdown cooling system, the reactor coolant sampling system (post-accident sampling piping only until such time as a modification eliminates the post-accident piping as a potential leakage path), the containment spray system, the radioactive waste gas system (post-accident sampling return piping only until such time as a modification eliminates the post-accident piping as a potential leakage path), and the liquid radwaste~~

(continued)

5.5 Procedures, Programs, and Manuals (continued)

~~5.5.2.8~~ ~~Primary Coolant Sources Outside Containment Program (continued)~~

~~system (post-accident sampling return piping only until such time as a modification eliminates the post-accident piping as a potential leakage path). The program shall include the following:~~

~~a.~~ ~~Preventive maintenance and periodic visual inspection requirements; and~~

~~b.~~ ~~Integrated leak test requirements for each system at refueling cycle intervals or less.~~

~~5.5.2.9~~ ~~Pre-Stressed Concrete Containment Tendon Surveillance Program~~

~~This program provides controls for monitoring any tendon degradation in pre-stressed concrete containment, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. Program itself is relocated to the LCS.~~

~~5.5.2.10~~ ~~Inservice Inspection and Testing Program~~

~~This program provides controls for inservice inspection of ASME Code Class 1, 2, and 3 components and Code Class CC and MC components including applicable supports. The program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components. The program itself is located in the LCS.~~

~~5.5.2.11~~ ~~Steam Generator (SG) Program~~

~~A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following provisions:~~

~~a.~~ ~~Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged, to confirm that the performance criteria are being met.~~

(continued)

5.5 Procedures, Programs, and Manuals (continued)

~~5.5.2.11 Steam Generator (SG) Program (continued)~~

~~b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.~~

~~1. Structural integrity performance criterion: All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.~~

~~2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 0.5 gpm per SG and 1 gpm through both SGs.~~

~~3. The operational LEAKAGE performance criterion is specified in LCO 3.4.13, "RCS Operational LEAKAGE."~~

(continued)

5.5 Procedures, Programs, and Manuals (continued)

~~5.5.2.11 Steam Generator (SG) Program (continued)~~

~~c. Provisions for SG tube repair criteria.~~

~~1. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 35% of the nominal tube wall thickness shall be plugged.~~

~~d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. The tube-to-tubesheet weld is not part of the tube.~~

~~In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.~~

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

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

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

~~e. Provisions for monitoring operational primary to secondary LEAKAGE.~~

5.5 Procedures, Programs, and Manuals (continued)

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5.5 Procedures, Programs, and Manuals (continued)

~~5.5.2.12~~ ~~Ventilation Filter Testing Program (VFTP)~~

~~This Program establishes the required testing of the Engineered Safety Feature filter ventilation system "Control Room Emergency Air Cleanup System." The frequency of testing shall be in accordance with Regulatory Guide 1.52, Revision 2. As a minimum the VFTP program shall include the following:~~

~~a. Inplace testing of the high efficiency particulate air (HEPA) filters to demonstrate acceptable penetration and system bypass when tested at the appropriate system flowrate in accordance with Regulatory Guide 1.52, Revision 2, and ANSI N510-1975 (see Note 1); and~~

~~b. Inplace testing of the charcoal adsorber to demonstrate acceptable penetration and system bypass when tested at the appropriate system flowrate in accordance with Regulatory Guide 1.52, Revision 2, and ANSI N510-1975 (see Note 1); and~~

~~c. Laboratory testing of charcoal adsorber samples obtained in accordance with Regulatory Guide 1.52, Revision 2 and tested per the methodology of ASTM D3803-1989 at 30°C and 70% relative humidity to show acceptable methyl iodide penetration; and~~

~~d. Testing to demonstrate the pressure drop across the combined HEPA filters, the prefilters, and the charcoal adsorbers, when tested at the appropriate system flowrate.~~

~~Note 1: Sample and injection points shall be qualified per ANSI N510-1975 unless manifolds have been qualified per ASME N510-1989. HEPA testing will be conducted with DOP aerosol or suitable alternate.~~

(continued)

5.5 Procedures, Programs, and Manuals (continued)

~~5.5.2.12~~ ~~Ventilation Filter Testing Program (VFTP) (continued)~~

~~The provisions of Technical Specification Surveillance Requirement 3.0.2 and Technical Specification Surveillance Requirement 3.0.3 are applicable to the VFTP test frequencies.~~

~~5.5.2.13~~ ~~Diesel Fuel Oil Testing Program~~

~~This program implements required testing of both new fuel oil and stored fuel oil. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM standards. The purpose of the program is to establish the following:~~

~~a. Acceptability of new fuel oil use prior to addition to storage tanks by determining that the fuel oil has:~~

~~1.~~ ~~an API gravity or an absolute specific gravity within limits,~~

~~2.~~ ~~a flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and~~

~~3.~~ ~~a water and sediment content within limits.~~

~~b. Other properties for ASTM 2D fuel oil are within limits within 31 days following sampling and addition to the storage tanks, with exceptions noted in the Bases for Surveillance Requirement 3.8.3.3; and,~~

~~c. Total particulate concentration of fuel oil is ≤ 10 mg/l when tested every 92 days in accordance with ASTM D-2276, Method A.~~

~~5.5.2.14~~ ~~Deleted~~

(continued)

5.5 Procedures, Programs, and Manuals (continued)

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(continued)

5.5 Procedures, Programs, and Manuals (continued)

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~~(~~continued)

5.5 Procedures, Programs, and Manuals (continued)

~~5.5.2.15 Containment Leakage Rate Testing Program~~

~~A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995 as modified by the following exception:~~

~~NEI 94-01 - 1995, Section 9.2.3: The first Type A Test performed after the March 31, 1995 Type A Test shall be performed no later than March 30, 2010.~~

~~The calculated peak containment internal pressure related to the design basis loss-of-coolant accident,~~ $\cancel{P_a}$, ~~is 48.0 psig (~~$\cancel{P_a}$ ~~will conservatively be assumed to be equal to the calculated peak containment internal pressure for the design basis Main Steam Line Break (51.5 psig) for the purpose of containment testing in accordance with this Technical Specification).~~

~~The maximum allowable containment leakage rate,~~ $\cancel{L_a}$, ~~at~~ $\cancel{P_a}$, ~~shall be 0.10% of containment air weight per day.~~

~~Leakage rate acceptance criteria are:~~

~~a. The Containment overall leakage rate acceptance criterion is~~ $\cancel{\le 1.0\ L_a}$. ~~During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are~~ $\cancel{\le 0.60\ L_a}$ ~~for the Type B and Type C tests and~~ $\cancel{\le 0.75\ L_a}$ ~~for the Type A tests;~~

~~b. Air lock testing acceptance criteria are:~~

~~1)~~ ~~Overall air lock leakage rate is~~ $\cancel{\le 0.05\ L_a}$ ~~when tested at~~ $\cancel{\ge P_a}$.

~~2)~~ ~~For each door, the leakage rate is~~ $\cancel{\le 0.01\ L_a}$ ~~when pressurized to~~ $\cancel{\ge 9.0}$ ~~psig.~~

5.5 Procedures, Programs, and Manuals (continued)

~~5.5.2.15~~ ~~Containment Leakage Rate Testing Program (Continued)~~

~~The provisions of Surveillance Requirement 3.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program. However, test frequencies specified in this Program may be extended consistent with the guidance provided in NEI 94-01, "Industry Guideline For Implementing Performance-Based Option Of 10CFR 50, Appendix J," as endorsed by Regulatory Guide 1.163. Specifically, NEI 94-01 has these provisions for test frequencies extension:~~

~~1. Consistent with standard scheduling practices for Technical Specifications Required Surveillances, intervals for recommended Type A testing may be extended by up to 15 months. This option should be used only in cases where refueling schedules have been changed to accommodate other factors.~~

~~2. Consistent with standard scheduling practices for Technical Specifications Required Surveillances, intervals for the recommended surveillance frequency for Type B and Type C testing may be extended by up to 25 percent of the test interval, not to exceed 15 months.~~

~~The provisions of Surveillance Requirement 3.0.3 are applicable to the Containment Leakage Rate Testing Program.~~

~~5.5.2.16~~ ~~Control Room Envelope Habitability Program~~

~~A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Emergency Air Cleanup System (CREACUS), CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem total effective dose equivalent (TEDE) for the duration of the accident. The program shall include the following elements:~~

~~a. The definition of the CRE and the CRE boundary.~~

~~b. Requirements for maintaining the CRE boundary in its design condition including configuration control and preventive maintenance.~~

(continued)

5.5 Procedures, Programs, and Manuals (continued)

~~5.5.2.16 Control Room Envelope Habitability Program (Continued)~~

~~c. Requirements for (i) determining the unfiltered air inleakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.~~

~~The following is an exception to Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.~~

~~Appropriate application of ASTM E-741 shall include the ability to take minor exceptions to the test methodology. These exceptions shall be documented in the test report.~~

~~d. Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by one train of the CREACUS, operating at the flow rate required by the VFTP, at a Frequency of 24 months on a STAGGERED TEST BASIS. The results shall be trended and used as part of the 24 month assessment of the CRE boundary.~~

~~e. The quantitative limits on unfiltered air inleakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by the testing described in paragraph c. The unfiltered air inleakage limit for radiological challenges is the inleakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air inleakage limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis.~~

~~f. The provisions of SR 3.0.2 are applicable to the Frequencies for assessing CRE habitability, determining CRE unfiltered inleakage, and measuring CRE pressure and assessing the CRE boundary as required by paragraphs c and d, respectively.~~

(continued)

5.5 Procedures, Programs, and Manuals (continued)

~~5.5.2.17~~ ~~Battery Monitoring and Maintenance Program~~

~~This program provides for battery restoration and maintenance, which includes the following:~~

~~a.~~ ~~Actions to restore battery cells with float voltage < 2.13 V, and~~

~~b.~~ ~~Actions to verify that the remaining cells are above 2.07 V when a battery cell or cells have been found less than 2.13 V, and~~

~~c.~~ ~~Actions to equalize and test battery cells that had been discovered with electrolyte level below the top of the plates.~~

5.0 ADMINISTRATIVE CONTROLS

5.6 ~~Safety Function Determination Program (SFDP)~~ Deleted

~~5.6.1~~ ~~This program ensures loss of safety function is detected and appropriate actions taken. Upon failure to meet two or more LCOs at the same time, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate limitations and remedial or compensatory actions may be identified to be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6.~~

~~5.6.2~~ ~~The SFDP shall contain the following:~~

~~a.~~ ~~Provisions for cross-train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected.~~

~~b.~~ ~~Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists.~~

~~c.~~ ~~Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities.~~

~~d.~~ ~~Other appropriate limitations and remedial or compensatory actions.~~

~~5.6.3~~ ~~A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:~~

~~a.~~ ~~A required system redundant to system(s) supported by the inoperable support system is also inoperable (Case A); or~~

~~b.~~ ~~A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable (Case B); or~~

~~c.~~ ~~A required system redundant to support system(s) for the supported systems (a) and (b) above is also inoperable (Case C).~~

(continued)

~~5.6 Safety Function Determination Program (SFDP)~~

~~5.6.3~~
~~(continued)~~

~~Generic Example:~~

~~Train A~~		~~Train B~~	
~~System i~~		~~System i~~	~~- Case C~~
~~+~~		~~+~~	
~~System ii~~	~~- (Support System) Inoperable~~	~~System ii~~	
~~+~~		~~+~~	
~~System iii~~		~~System iii~~	~~- Case A~~
~~+~~		~~+~~	
~~System iv~~		~~System iv~~	~~- Case B~~

~~5.6.4~~ ~~The Safety Function Determination Program identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.~~

(continued)

5.0 ADMINISTRATIVE CONTROLS

5.7 Reporting Requirements

5.7.1 Routine Reports

In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted in accordance with 10 CFR 50.4. The reports shall be addressed to the U.S. Nuclear Regulatory Commission, Attention: Document Control Desk, Washington, D.C., with a copy to the Regional Administrator of the Regional Office of the NRC, unless otherwise noted.

5.7.1.1 ~~Annual Reports~~ ← Deleted

~~-----NOTE-----~~
~~A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station.~~
~~----------~~

~~Annual Reports covering the activities of the unit as described below for the previous calendar year shall be submitted by March 31 of each year.~~

~~Reports required on an annual basis include:~~

~~a. (Deleted)~~

(continued)

5.7 Reporting Requirements (continued)

~~5.7.1.1 Annual Reports (continued)~~

~~b. Reactor Coolant System Specific Activity Report~~

~~Reports required on an annual basis shall include the results of specific activity analysis in which the primary coolant exceeded the limits of Specification 3.4.16. The following information shall be included in these reports:~~

~~1. Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded; and~~

~~2. Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while the limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than the limit. Each result should include date and time of sampling and the radioiodine concentrations; and~~

~~3. Cleanup system flow history starting 48 hours prior to the first sample in which the limit was exceeded; and~~

~~4. Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady-state level; and~~

~~5. The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.~~

5.7.1.2 Annual Radiological Environmental Operating Report

~~NOTE~~

~~A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station.~~

facility

The Annual Radiological Environmental Operating Report covering the operation of the ~~unit~~ during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the

(continued)

5.7 Reporting Requirements (continued)

5.7.1.2 Annual Radiological Environmental Operating Report (continued)

objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. The report shall identify the thermoluminescent dosimeter (TLD) results that represent collocated dosimeters in relation to the NRC TLD program and the exposure period associated with each result. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.7.1.3 Radioactive Effluent Release Report

~~---NOTE---~~
~~A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.~~
~~------~~

The Radioactive Effluent Release Report covering the operation of the facility ~~unit~~ during the previous calendar year shall be submitted before May 1 of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents released from the facility ~~unit~~. The report shall also include a summary of the quantities of solid radioactive waste shipped from the facility ~~unit~~ directly to the disposal site and quantities of solid radioactive waste shipped from the facility ~~unit~~'s intermediary processor to the disposal site. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program (PCP) and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

(continued)

5.7 Reporting Requirements (continued)

~~5.7.1.4~~ ~~(Deleted)~~

~~5.7.1.5~~ ~~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.~~ ~~Specification 3.1.1, "SHUTDOWN MARGIN (SDM) - T_{avg} >200°F;"~~

~~2.~~ ~~Specification 3.1.2, "SHUTDOWN MARGIN (SDM) - T_{avg} ≤200°F;"~~

~~3.~~ ~~Specification 3.1.4, "Moderator Temperature Coefficient;"~~

~~4.~~ ~~Specification 3.1.5, "Control Element Assembly (CEA) Alignment;"~~

~~5.~~ ~~Specification 3.1.7, "Regulating CEA Insertion Limits;"~~

~~6.~~ ~~Specification 3.1.8, "Part Length Control Element Assembly Insertion Limits;"~~

~~7.~~ ~~Specification 3.2.1, "Linear Heat Rate;"~~

~~8.~~ ~~Specification 3.2.4, "Departure From Nucleate Boiling Ratio;"~~

~~9.~~ ~~Specification 3.2.5, "Axial Shape Index;"~~

~~10.~~ ~~Specification 3.4.1, "RCS DNB (Pressure, Temperature, and Flow) Limits;~~

~~11.~~ ~~Specification 3.9.1, "Boron Concentration."~~

~~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:~~

(continued)

5.7 Reporting Requirements (continued)

~~5.7.1.5~~ ~~CORE OPERATING LIMITS REPORT (COLR) (continued)~~

~~1.~~ ~~CENPD-132P, "Calculative Methods for the C-E Large Break LOCA Evaluation Model"~~

~~2.~~ ~~CENPD-137P, "Calculative Methods for the C-E Small Break LOCA Evaluation Model"~~

~~3.~~ ~~CEN-356(V)-P-A, "Modified Statistical Combination of Uncertainties"~~

~~4.~~ ~~SCE-9801-P-A, "Reload Analysis Methodology for the San Onofre Nuclear Generating Station Units 2 and 3"~~

~~5.~~ ~~CEN-635(S), "Identification of NRC Safety Evaluation Report Limitations and/or Constraints on Reload Analysis Methodology"~~

~~6.~~ ~~Letter, dated May 16, 1986, G. W. Knighton (NRC) to K. P. Baskin (SCE), "Issuance of Amendment No. 47 to Facility Operating License NPF-10 and Amendment No. 36 to Facility Operating License NPF-15," San Onofre Nuclear Generating Station Units 2 and 3 (Cycle 3 SER)~~

~~7.~~ ~~Letter, dated January 9, 1985, G. W. Knighton (NRC) to K. P. Baskin, "Issuance of Amendment No. 30 to Facility Operating License NPF-10 and Amendment No. 19 to Facility Operating License NPF-15," San Onofre Nuclear Generating Station Units 2 and 3 (Cycle 2 SER)~~

~~8.~~ ~~"Implementation of ZIRLO™ Cladding Material in CE Nuclear Power Fuel Assembly Designs," CENPD-404-P-A~~

~~9.~~ ~~SCE-0901, "PWR Reactor Physics Methodology Using Studsvik Design Codes"~~

~~c.~~ ~~The core operating limits shall be determined assuming operation at RATED THERMAL POWER such that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal hydraulic limits, Emergency Core Cooling System (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.~~

~~5.7.1.6~~ ~~REACTOR COOLANT SYSTEM (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)~~

~~a.~~ ~~RCS pressure and temperature limits for heatup, cooldown, low temperature operation, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:~~

(continued)

5.7 Reporting Requirements (continued)

~~5.7.1.6~~ ~~REACTOR COOLANT SYSTEM (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) (continued)~~

~~Technical Specification 3.4.3 RCS Pressure and Temperature (P/T) Limits,~~

~~Technical Specification 3.4.6 RCS Loops - MODE 4,~~

~~Technical Specification 3.4.7 RCS Loops - MODE 5, Loops Filled,~~

~~Technical Specification 3.4.12.1 Low Temperature Overpressure Protection (LTOP) System RCS Temperature ≤ PTLR Limit, and~~

~~Technical Specification 3.4.12.2 Low Temperature Overpressure Protection (LTOP) System RCS Temperature > PTLR Limit.~~

~~b.~~ ~~The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following document:~~

~~CE NPSD-683-A, The Development of a RCS Pressure and Temperature Limits Report for the Removal of P-T Limits and LTOP Setpoints from the Technical Specifications.~~

~~c.~~ ~~The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.~~

~~5.7.1.7~~ ~~Hazardous Cargo Traffic Report~~

~~Hazardous cargo traffic on Interstate 5 (I-5) and the AT&SF railway shall be monitored and the results submitted to the NRC Regional Administrator once every three years.~~

(continued)

5.7 Reporting Requirements (continued)

~~5.7.2~~ ~~Special Reports~~

~~Special Reports may be required covering inspection, test, and maintenance activities. These special reports are determined on an individual basis for each unit and their preparation and submittal are designated in the Technical Specifications.~~

~~Special Reports shall be submitted to the U. S. Nuclear Regulatory Commission, Attention: Document Control Desk, Washington, D. C. 20555, with a copy to the Regional Administrator of the Regional Office of the NRC, in accordance with 10 CFR 50.4 within the time period specified for each report.~~

~~The following Special Reports shall be submitted:~~

~~a.~~ ~~When a pre-planned alternate method of monitoring post-accident instrumentation functions is required by Condition B or Condition G of LCO 3.3.11, a report shall be submitted within 30 days from the time the action is required. The report shall outline the action taken, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the function to OPERABLE status.~~

~~b.~~ ~~Any abnormal degradation of the containment structure detected during the tests required by the Pre-Stressed Concrete Containment Tendon Surveillance Program shall be reported to the NRC within 30 days. The report shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, and the corrective action taken.~~

~~c.~~ ~~A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.2.11, Steam Generator (SG) Program. The report shall include:~~

(continued)

5.7 Reporting Requirements (continued)

~~5.7.2~~ ~~Special Reports~~ ~~(continued)~~

~~1.~~ ~~The scope of inspections performed on each SG,~~

~~2.~~ ~~Active degradation mechanisms found,~~

~~3.~~ ~~Nondestructive examination techniques utilized for each degradation mechanism,~~

~~4.~~ ~~Location, orientation (if linear), and measured sizes (if available) of service induced indications,~~

~~5.~~ ~~Number of tubes plugged during the inspection outage for each active degradation mechanism,~~

~~6.~~ ~~Total number and percentage of tubes plugged to date,~~

~~7.~~ ~~The results of condition monitoring, including the results of tube pulls and in-situ testing.~~

5.0 ADMINISTRATIVE CONTROLS

5.8 High Radiation Area

5.8.1 Each high radiation area as defined in 10 CFR 20 shall be barricaded and conspicuously posted as a high radiation area, and entrance thereto shall be controlled by requiring issuance of a Radiation Exposure Permit (REP).

Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device that continuously indicates the radiation dose rate in the area,

b. A radiation monitoring device that continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rates in the area have been determined and personnel have been made knowledgeable of them,

c. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device. This individual is responsible for providing positive radiation protection control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified in the radiation protection procedures or the applicable REP.

(continued)

5.8. High Radiation Area (continued)

5.8.2 In addition, areas that are accessible to personnel and that have radiation levels greater than 1.0 rem (but less than 500 rads at 1 meter) in 1 hour at 30 cm from the radiation source, or from any surface penetrated by the radiation, shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the shift manager on duty or radiation protection supervisor. Doors shall remain locked except during periods of access by personnel under an approved REP that specifies the dose rates in the immediate work areas and the maximum allowable stay time for individuals in that area. In lieu of a stay time specification on the REP, direct or remote continuous surveillance (such as closed circuit TV cameras) may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area.

5.8.3 Individual high radiation areas that are accessible to personnel, that could result in radiation doses greater than 1.0 rem in 1 hour, and that are within large areas where no enclosure exists to enable locking and where no enclosure can be reasonably constructed around the individual area shall be barricaded and conspicuously posted. A flashing light shall be activated as a warning device whenever the dose rate in such an area exceeds or is expected to exceed 1.0 rem in 1 hour at 30 cm from the radiation source or from any surface penetrated by the radiation.

Attachment 2

Proposed Technical Specifications - Markup - Unit 3

1.0 USE AND APPLICATION

1.1 Definitions

-----------------------------------NOTE-----------------------------------
The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications and Bases.

Term	Definition
ACTIONS	ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.
INSERT 1 →	
~~AXIAL SHAPE INDEX (ASI)~~	~~ASI shall be the power generated in the lower half of the core less the power generated in the upper half of the core, divided by the sum of the power generated in the lower and upper halves of the core.~~ $\cancel{ASI = \frac{lower - upper}{lower + upper}}$
~~AZIMUTHAL POWER TILT (T_q)~~	~~AZIMUTHAL POWER TILT shall be the power asymmetry between azimuthally symmetric fuel assemblies.~~
~~CHANNEL CALIBRATION~~	~~A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds within the necessary range and accuracy to known values of the parameter that the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel, including the required sensor, alarm, display, and trip functions, and shall include the CHANNEL FUNCTIONAL TEST. Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an inplace cross calibration of the sensing elements and normal calibration of the remaining adjustable devices in the channel. Whenever a sensing element is replaced, the next required inplace cross calibration consists of comparing the other sensing elements with the recently installed sensing element.~~

(continued)

INSERT 1

CERTIFIED FUEL HANDLER	A CERTIFIED FUEL HANDLER is an individual who complies with provisions of the CERTIFIED FUEL HANDLER training program required by TS 5.3.2.

1.1 Definitions

~~CHANNEL CALIBRATION (continued)~~	~~The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping, or total channel steps so that the entire channel is calibrated.~~
~~CHANNEL CHECK~~	~~A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter.~~
~~CHANNEL FUNCTIONAL TEST~~	~~A CHANNEL FUNCTIONAL TEST shall be:~~ ~~a. Analog channels—the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY, including required alarms, interlocks, display and trip functions;~~ ~~b. Bistable channels (e.g., pressure switches and switch contacts)—the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY, including required alarm and trip functions; or~~ ~~c. Digital computer channels—the use of diagnostic programs to test digital computer hardware and the injection of simulated process data into the channel to verify OPERABILITY, including alarm and trip functions.~~ ~~The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps so that the entire channel is tested.~~
~~CORE ALTERATION~~	~~CORE ALTERATION shall be the movement or manipulation of any fuel, sources, reactivity control components, or other components, excluding control element assemblies (CEAs) withdrawn into the upper guide structure, affecting reactivity,~~

(continued)

1.1 Definitions

~~CORE ALTERATION (continued)~~	~~within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.~~
~~CORE OPERATING LIMITS REPORT (COLR)~~	~~The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 5.7.1.5. Plant operation within these limits is addressed in individual Specifications.~~
~~DOSE EQUIVALENT I-131~~	~~DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in ICRP-30, Supplement to Part 1, pages 192-212, Tables titled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity."~~
~~$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY~~	~~$\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives > 15 minutes, making up at least 95% of the total noniodine activity in the coolant.~~
~~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~~

(continued)

1.1 Definitions

~~ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME (Continued)~~	~~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.~~
~~LEAKAGE~~	~~LEAKAGE shall be:~~ ~~a. Identified LEAKAGE~~ ~~1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) 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 to the Secondary System (primary to secondary LEAKAGE).~~ ~~b. Unidentified LEAKAGE~~ ~~All LEAKAGE that is not identified LEAKAGE.~~ ~~c. Pressure Boundary LEAKAGE~~ ~~LEAKAGE (except primary to secondary LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.~~
~~MODE~~	~~A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.~~

(continued)

1.1 Definitions

OPERABLE — OPERABILITY	A system, subsystem, train, component, or device shall be OPERABLE when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
~~PHYSICS TESTS~~	~~PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are:~~ ~~a. Described in Chapter 14, Initial Test Program of the SONGS Units 2 and 3 UFSAR;~~ ~~b. Authorized under the provisions of 10 CFR 50.59; or~~ ~~c. Otherwise approved by the Nuclear Regulatory Commission.~~
~~RCS PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)~~	~~The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.7.1.6.~~
~~RATED THERMAL POWER (RTP)~~	~~RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3438 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~~

(continued)

1.1 Definitions

~~REACTOR PROTECTIVE SYSTEM (RPS) RESPONSE TIME (continued)~~	~~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 CEAs (shutdown and regulating) are fully inserted except for the single CEA of highest reactivity worth, which is assumed to be fully withdrawn. However, with all CEAs verified fully inserted by two independent means, it is not necessary to account for a stuck CEA in the SDM calculation. With any CEAs not capable of being fully inserted, the 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.~~ ~~With any CEAs not capable of being fully inserted, the reactivity worth of these CEAs must be accounted for in the determination of SDM.~~
~~STAGGERED TEST BASIS~~	~~A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function.~~
~~THERMAL POWER~~	~~THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.~~

~~Table 1.1-1 (page 1 of 1)~~
~~MODES~~

~~MODE~~	~~TITLE~~	~~REACTIVITY CONDITION (k_{eff})~~	~~% RATED THERMAL POWER[a]~~	~~AVERAGE REACTOR COOLANT TEMPERATURE (°F)~~
~~1~~	~~Power Operation~~	~~$\geq$ 0.99~~	~~> 5~~	~~NA~~
~~2~~	~~Startup~~	~~$\geq$ 0.99~~	~~$\leq$ 5~~	~~NA~~
~~3~~	~~Hot Standby~~	~~< 0.99~~	~~NA~~	~~$\geq$ 350~~
~~4~~	~~Hot Shutdown~~	~~< 0.99~~	~~NA~~	~~350 > T_{avg} > 200~~
~~5~~	~~Cold Shutdown[b]~~	~~< 0.99~~	~~NA~~	~~$\leq$ 200~~
~~6~~	~~Refueling[c]~~	~~NA~~	~~NA~~	~~NA~~

~~(a) Excluding decay heat.~~

~~(b) All reactor vessel head closure bolts fully tensioned.~~

~~(C) One or more reactor vessel head closure bolts less than fully tensioned.~~

1.0 USE AND APPLICATION

1.2 Logical Connectors

PURPOSE

The purpose of this section is to explain the meaning of logical connectors.

Logical connectors are used in Technical Specifications (TS) to discriminate between, and yet connect, discrete Conditions, Required Actions, Completion Times, Surveillances, and Frequencies. The only logical connectors that appear in TS are <u>AND</u> and <u>OR</u>. The physical arrangement of these connectors constitutes logical conventions with specific meanings.

BACKGROUND

Several levels of logic may be used to state Required Actions. These levels are identified by the placement (or nesting) of the logical connectors and by the number assigned to each Required Action. The first level of logic is identified by the first digit of the number assigned to a Required Action and the placement of the logical connector in the first level of nesting (i.e., left justified with the number of the Required Action). The successive levels of logic are identified by additional digits of the Required Action number and by successive indentions of the logical connectors.

When logical connectors are used to state a Condition, Completion Time, Surveillance, or Frequency, only the first level of logic is used, and the logical connector is left justified with the statement of the Condition, Completion Time, Surveillance, or Frequency.

EXAMPLE~~S~~

The following example~~s~~ illustrate the use of logical connectors.

(continued)

1.2 Logical Connectors

EXAMPLES (continued)

EXAMPLE 1.2-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Verify . . . AND A.2 Restore . . .	

In this example the logical connector AND is used to indicate that when in Condition A, both Required Actions A.1 and A.2 must be completed.

(continued)

1.2 Logical Connectors

~~EXAMPLES (continued)~~

~~EXAMPLE 1.2-2~~

~~ACTIONS~~

~~CONDITION~~	~~REQUIRED ACTION~~	~~COMPLETION TIME~~
~~A. LCO not met.~~	~~A.1 Trip . . .~~ ~~OR~~ ~~A.2.1 Verify . . .~~ ~~AND~~ ~~A.2.2.1 Reduce . . .~~ ~~OR~~ ~~A.2.2.2 Perform . . .~~ ~~OR~~ ~~A.3 Align . . .~~	

~~This example represents a more complicated use of logical connectors. Required Actions A.1, A.2, and A.3 are alternative choices, only one of which must be performed as indicated by the use of the logical connector OR and the left justified placement. Any one of these three Actions may be chosen. If A.2 is chosen, then both A.2.1 and A.2.2 must be performed as indicated by the logical connector AND. Required Action A.2.2 is met by performing A.2.2.1 or A.2.2.2. The indented position of the logical connector OR indicates that A.2.2.1 and A.2.2.2 are alternative choices, only one of which must be performed.~~

1.0 USE AND APPLICATION

1.3 Completion Times

PURPOSE

The purpose of this section is to establish the Completion Time convention and to provide guidance for its use.

storage of irradiated fuel

BACKGROUND

Limiting Condition for Operation (LCOs) specify minimum requirements for ensuring safe ~~operation of the unit~~. The ACTIONS associated with an LCO state Conditions that typically describe the ways in which the requirements of the LCO can fail to be met. Specified with each stated Condition are Required Action(s) and Completion Time(s).

DESCRIPTION

The Completion Time is the amount of time allowed for completing a Required Action. It is referenced to the time of discovery of a situation (e.g., inoperable equipment or variable not within limits) that requires entering an ACTIONS Condition unless otherwise specified, providing the facility ~~unit~~ is in a ~~MODE or~~ specified condition stated in the Applicability of the LCO. Required Actions must be completed prior to the expiration of the specified Completion Time. An ACTIONS Condition remains in effect and the Required Actions apply until the Condition no longer exists or the facility ~~unit~~ is not within the LCO Applicability.

~~If situations are discovered that require entry into more than one Condition at a time within a single LCO (multiple Conditions), the Required Actions for each Condition must be performed within the associated Completion Time. When in multiple Conditions, separate Completion Times are tracked for each Condition starting from the time of discovery of the situation that required entry into the Condition.~~

~~Once a Condition has been entered, subsequent trains, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition, unless specifically stated. The Required Actions of the Condition continue to apply to each additional failure, with Completion Times based on initial entry into the Condition.~~

(continued)

1.3 Completion Times (continued)

~~DESCRIPTION (continued)~~

~~However, when a subsequent train, subsystem, component, or variable expressed in the Condition is discovered to be inoperable or not within limits, the Completion Time(s) may be extended. To apply this Completion Time extension, two criteria must first be met. The subsequent inoperability:~~

~~a. Must exist concurrent with the first inoperability; and~~

~~b. Must remain inoperable or not within limits after the first inoperability is resolved.~~

~~The total Completion Time allowed for completing a Required Action to address the subsequent inoperability shall be limited to the more restrictive of either:~~

~~a. The stated Completion Time, as measured from the initial entry into the Condition, plus an additional 24 hours; or~~

~~b. The stated Completion Time as measured from discovery of the subsequent inoperability.~~

~~The above Completion Time extensions do not apply to those Specifications that have exceptions that allow completely separate re-entry into the Condition (for each train, subsystem, component, or variable expressed in the Condition) and separate tracking of Completion Times based on this re-entry. These exceptions are stated in individual Specifications.~~

~~The above Completion Time extension does not apply to a Completion Time with a modified "time zero." This modified "time zero" may be expressed as a repetitive time (i.e., "once per 8 hours," where the Completion Time is referenced from a previous completion of the Required Action versus the time of Condition entry) or as a time modified by the phrase "from discovery . . ." Example 1.3-3 illustrates one use of this type of Completion Time. The 10 day Completion Time specified for Conditions A and B in Example 1.3-3 may not be extended.~~

(continued)

1.3 Completion Times (continued)

EXAMPLE~~S~~

The following example~~s~~ illustrate s the use of Completion Times ~~with different types of Conditions and changing Conditions~~.

EXAMPLE 1.3-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1 Verify . . . ~~Be in MODE 3~~. AND B.2 Restore . . . ~~Be in MODE 5~~.	6 hours 36 hours

Condition B has two Required Actions. Each Required Action has its own separate Completion Time. Each Completion Time is referenced to the time that Condition B is entered.

The Required Actions of Condition B are to perform the verification ~~Be in MODE 3~~ within 6 hours AND perform the restoration ~~in MODE 5~~ within 36 hours. A total of 6 hours is allowed for performing the verification ~~reaching MODE 3~~ and a total of 36 hours (not 42 hours) is allowed for performing the restoration ~~reaching MODE 5~~ from the time that Condition B was entered. If verification is performed ~~MODE 3 is reached~~ within 3 hours, the time allowed for performing the restoration ~~reaching MODE 5~~ is the next 33 hours because the total time allowed for performing the restoration ~~reaching MODE 5~~ is 36 hours.

~~If Condition B is entered while in MODE 3, the time allowed for reaching MODE 5 is the next 36 hours.~~

(continued)

1.3 Completion Times (continued)

~~EXAMPLES (continued)~~

~~EXAMPLE 1.3-2~~

~~ACTIONS~~

~~CONDITION~~	~~REQUIRED ACTION~~	~~COMPLETION TIME~~
~~A. One pump inoperable.~~	~~A.1 Restore pump to OPERABLE status.~~	~~7 days~~
~~B. Required Action and associated Completion Time not met.~~	~~B.1 Be in MODE 3.~~ ~~AND~~ ~~B.2 Be in MODE 5.~~	~~6 hours~~ ~~36 hours~~

~~When a pump is declared inoperable, Condition A is entered. If the pump is not restored to OPERABLE status within 7 days, Condition B is also entered and the Completion Time clocks for Required Actions B.1 and B.2 start. If the inoperable pump is restored to OPERABLE status after Condition B is entered, Condition A and B are exited, and therefore the Required Actions of Condition B may be terminated.~~

~~When a second pump is declared inoperable while the first pump is still inoperable, Condition A is not re-entered for the second pump. LCO 3.0.3 is entered, since the ACTIONS do not include a Condition for more than one inoperable pump. The Completion Time clock for Condition A does not stop after LCO 3.0.3 is entered, but continues to be tracked from the time Condition A was initially entered.~~

~~While in LCO 3.0.3, if one of the inoperable pumps is restored to OPERABLE status and the Completion Time for Condition A has not expired, LCO 3.0.3 may be exited and operation continued in accordance with Condition A.~~

(continued)

1.3 Completion Times (continued)

~~EXAMPLES~~

~~EXAMPLE 1.3-2 (continued)~~

~~While in LCO 3.0.3, if one of the inoperable pumps is restored to OPERABLE status and the Completion Time for Condition A has expired, LCO 3.0.3 may be exited and operation continued in accordance with Condition B. The Completion Time for Condition B is tracked from the time the Condition A Completion Time expired.~~

~~On restoring one of the pumps to OPERABLE status, the Condition A Completion Time is not reset, but continues from the time the first pump was declared inoperable. This Completion Time may be extended if the pump restored to OPERABLE status was the first inoperable pump. A 24 hour extension to the stated 7 days is allowed, provided this does not result in the second pump being inoperable for > 7 days.~~

(continued)

1.3 Completion Times (continued)

~~EXAMPLES (continued)~~

~~EXAMPLE 1.3-3~~

~~ACTIONS~~

~~CONDITION~~	~~REQUIRED ACTION~~	~~COMPLETION TIME~~
~~A. One Function X train inoperable.~~	~~A.1 Restore Function X train to OPERABLE status.~~	~~7 days~~ ~~AND~~ ~~10 days from discovery of failure to meet the LCO~~
~~B. One Function Y train inoperable.~~	~~B.1 Restore Function Y train to OPERABLE status.~~	~~72 hours~~ ~~AND~~ ~~10 days from discovery of failure to meet the LCO~~
~~C. One Function X train inoperable.~~ ~~AND~~ ~~One Function Y train inoperable.~~	~~C.1 Restore Function X train to OPERABLE status.~~ ~~OR~~ ~~C.2 Restore Function Y train to OPERABLE status.~~	~~72 hours~~ ~~72 hours~~

(continued)

1.3 Completion Times (continued)

~~EXAMPLES~~

~~Example 1.3-3 (continued)~~

~~When one Function X train and one Function Y train are inoperable, Condition A and Condition B are concurrently applicable. The Completion Times for Condition A and Condition B are tracked separately for each train starting from the time each train was declared inoperable and the Condition was entered. A separate Completion Time is established for condition C and tracked from the time the second train was declared inoperable (i.e., the time the situation described in Condition C was discovered).~~

~~If Required Action C.2 is completed within the specified Completion Time, Conditions B and C are exited. If the Completion Time for Required Action A.1 has not expired, operation may continue in accordance with Condition A. The remaining Completion Time in Condition A is measured from the time the affected train was declared inoperable (i.e., initial entry into Condition A).~~

~~The Completion Times of Conditions A and B are modified by a logical connector, with a separate 10 day Completion Time measured from the time it was discovered the LCO was not met. In this example, without the separate Completion Time, it would be possible to alternate between Conditions A, B, and C in such a manner that operation could continue indefinitely without ever restoring systems to meet the LCO. The separate Completion Time modified by the phrase "from discovery of failure to meet the LCO" is designed to prevent indefinite continued operation while not meeting the LCO. This Completion Time allows for an exception to the normal "time zero" for beginning the Completion Time "clock." In this instance, the Completion Time "time zero" is specified as commencing at the time the LCO was initially not met, instead of at the time the associated Condition was entered.~~

(continued)

1.3 Completion Times (continued)

~~EXAMPLES (continued)~~

~~EXAMPLE 1.3-4~~

~~ACTIONS~~

~~CONDITION~~	~~REQUIRED ACTION~~	~~COMPLETION TIME~~
~~A. One or more valves inoperable.~~	~~A.1 Restore valve(s) to OPERABLE status.~~	~~4 hours~~
~~B. Required Action and associated Completion Time not met.~~	~~B.1 Be in MODE 3.~~ ~~AND~~ ~~B.2 Be in MODE 4.~~	~~6 hours~~ ~~12 hours~~

~~A single Completion Time is used for any number of valves inoperable at the same time. The Completion Time associated with Condition A is based on the initial entry into Condition A and is not tracked on a per valve basis. Declaring subsequent valves inoperable, while Condition A is still in effect, does not trigger the tracking of separate Completion Times.~~

~~Once one of the valves has been restored to OPERABLE status, the Condition A Completion Time is not reset, but continues from the time the first valve was declared inoperable. The Completion Time may be extended if the valve restored to OPERABLE status was the first inoperable valve. The Condition A Completion Time may be extended for up to 4 hours provided this does not result in any subsequent valve being inoperable for > 4 hours.~~

~~If the Completion Time of 4 hours (including any extensions) expires while one or more valves are still inoperable, Condition B is entered.~~

(continued)

1.3 Completion Times (continued)

~~EXAMPLES (continued)~~

~~EXAMPLE 1.3-5~~

~~ACTIONS~~

~~-----NOTE-----~~
~~Separate Condition entry is allowed for each inoperable valve.~~
~~----------~~

~~CONDITION~~	~~REQUIRED ACTION~~	~~COMPLETION TIME~~
~~A. One or more valves inoperable.~~	~~A.1 Restore valve to OPERABLE status.~~	~~4 hours~~
~~B. Required Action and associated Completion Time not met.~~	~~B.1 Be in MODE 3.~~ ~~AND~~ ~~B.2 Be in MODE 4.~~	~~6 hours~~ ~~12 hours~~

~~The Note above the ACTIONS table is a method of modifying how the Completion Time is tracked. If this method of modifying how the Completion Time is tracked was applicable only to a specific Condition, the Note would appear in that Condition rather than at the top of the ACTIONS Table.~~

~~The Note allows Condition A to be entered separately for each inoperable valve, and Completion Times tracked on a per valve basis. When a valve is declared inoperable, Condition A is entered and its Completion Time starts. If subsequent valves are declared inoperable, Condition A is entered for each valve and separate Completion Times start and are tracked for each valve.~~

(continued)

1.3 Completion Times (continued)

~~EXAMPLES~~

~~EXAMPLE 1.3-5 (continued)~~

~~If the Completion Time associated with a valve in Condition A expires, Condition B is entered for that valve. If the Completion Times associated with subsequent valves in Condition A expire, Condition B is entered separately for each valve and separate Completion Times start and are tracked for each valve. If a valve that caused entry into Condition B is restored to OPERABLE status, Condition B is exited for that valve.~~

~~Since the Note in this example allows multiple Condition entry and tracking of separate Completion Times, Completion Time extensions do not apply.~~

~~EXAMPLE 1.3-6~~

~~ACTIONS~~

~~CONDITION~~	~~REQUIRED ACTION~~	~~COMPLETION TIME~~
~~A. One channel inoperable.~~	~~A.1 Perform SR 3.x.x.x.~~ ~~OR~~ ~~A.2 Reduce THERMAL POWER to ≤ 50% RTP.~~	~~Once per 8 hours~~ ~~8 hours~~
~~B. Required Action and associated Completion Time not met.~~	~~B.1 Be in MODE 3.~~	~~6 hours~~

(continued)

1.3 Completion Times (continued)

~~EXAMPLES~~

~~EXAMPLE 1.3-6 (continued)~~

~~Entry into Condition A offers a choice between Required Action A.1 or A.2. Required Action A.1 has a "Once per" Completion Time, which qualifies for the 25% extension, per SR 3.0.2, to each performance after the initial performance. If Required Action A.1 is followed and the Required Action is not met within the Completion Time (including the 25% extension allowed by SR 3.0.2), Condition B is entered. If Required Action A.2 is followed and the Completion Time of 8 hours is not met, Condition B is entered.~~

~~If after entry into Condition B, Required Action A.1 or A.2 is met, Condition B is exited and operation may then continue in Condition A.~~

(continued)

1.3 Completion Times (continued)

~~EXAMPLES (continued)~~

~~EXAMPLE 1.3-7~~

~~ACTIONS~~

~~CONDITION~~	~~REQUIRED ACTION~~	~~COMPLETION TIME~~
~~A. One subsystem inoperable.~~	~~A.1 Verify affected subsystem isolated.~~	~~1 hour~~ ~~AND~~ ~~Once per 8 hours thereafter~~
	~~AND~~	
	~~A.2 Restore subsystem to OPERABLE status.~~	~~72 hours~~
~~B. Required Action and associated Completion Time not met.~~	~~B.1 Be in MODE 3.~~	~~6 hours~~
	~~AND~~	
	~~B.2 Be in MODE 5.~~	~~36 hours~~

~~Required Action A.1 has two Completion Times. The 1 hour Completion Time begins at the time the Condition is entered and each "Once per 8 hours thereafter" interval begins upon performance of Required Action A.1.~~

~~If after Condition A is entered, Required Action A.1 is not met within either the initial 1 hour or any subsequent 8 hour interval from the previous performance (including the 25% extension allowed by SR 3.0.2), Condition B is entered.~~

(continued)

1.3 Completion Times (continued)

~~EXAMPLES~~	~~EXAMPLE 1.3-7 (continued)~~ ~~The Completion Time clock for Condition A does not stop after Condition B is entered, but continues from the time Condition A was initially entered. If Required Action A.1 is met after Condition B is entered, Condition B is exited and operation may continue in accordance with Condition A, provided the Completion Time for Required Action A.2 has not expired.~~
IMMEDIATE COMPLETION TIME	When "Immediately" is used as a Completion Time, the Required Action should be pursued without delay and in a controlled manner.

1.0 USE AND APPLICATION

1.4 Frequency

PURPOSE

The purpose of this section is to define the proper use and application of Frequency requirements.

DESCRIPTION

Each Surveillance Requirement (SR) has a specified Frequency in which the Surveillance must be met in order to meet the associated LCO. An understanding of the correct application of the specified Frequency is necessary for compliance with the SR.

The "specified Frequency" is referred to throughout this section and each of the Specifications of Section 3.0, Surveillance Requirement (SR) Applicability. The "specified Frequency" consists of the requirements of the Frequency column of each SR, as well as certain Notes in the Surveillance column that modify performance requirements.

~~Situations where a Surveillance could be required (i.e., its Frequency could expire), but where it is not possible or not desired that it be performed until sometime after the associated LCO is within its Applicability, represent potential SR 3.0.4 conflicts. To avoid these conflicts, the SR (i.e., the Surveillance or the Frequency) is stated such that it is only "required" when it can be and should be performed. With an SR satisfied, SR 3.0.4 imposes no restriction.~~

EXAMPLES

The following examples illustrate the various ways that Frequencies are specified. In these examples, the Applicability of the LCO (LCO not shown) ~~is MODES 1, 2, and 3~~.

occurs whenever any fuel assembly is stored in the fuel storage pool

(continued)

EXAMPLES (continued)

EXAMPLE 1.4-1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify . . . ~~Perform CHANNEL CHECK.~~	7 days ~~12 hours~~

Example 1.4-1 contains the type of SR most often encountered in the Technical Specifications (TS). The Frequency specifies an interval (7 days ~~12 hours~~) during which the associated Surveillance must be performed at least one time. Performance of the Surveillance initiates the subsequent interval. Although the Frequency is stated as 7 days ~~12 hours~~, an extension of the time interval to 1.25 times the stated Frequency is allowed by SR 3.0.2 for operational flexibility. The measurement of this interval continues at all times, even when the SR is not required to be met per SR 3.0.1 (such as when the equipment is inoperable, a variable is outside specified limits, or the facility ~~unit~~ is outside the Applicability of the LCO). If the interval specified by SR 3.0.2 is exceeded while the facility ~~unit~~ is in a ~~MODE or other~~ specified condition in the Applicability of the LCO~~, and the performance of the Surveillance is not otherwise modified (refer to Example 1.4-3)~~, then SR 3.0.3 becomes applicable.

If the interval as specified by SR 3.0.2 is exceeded while the facility ~~unit~~ is not in a ~~MODE or other~~ specified condition in the Applicability of the LCO for which performance of the SR is required, the Surveillance must be performed within the Frequency requirements of SR 3.0.2 prior to entry into the ~~MODE or other~~ specified condition. Failure to do so would result in a violation of SR 3.0.4.

(continued)

1.4 Frequency (continued)

EXAMPLES (continued)

EXAMPLE 1.4-2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
~~Verify flow is within limits.~~ Verify . . .	~~Once within 12 hours after ≥ 25% RTP~~ Prior to moving a fuel assembly . . . ~~AND~~ ~~24 hours thereafter~~

Example 1.4-2 ~~has two Frequencies. The first is~~ illustrates a one time performance Frequency~~, and the second is of the type shown in Example 1.4-1~~. ~~The logical connector "AND" indicates that both Frequency requirements must be met. Each time reactor power is increased from a power level < 25% RTP to ≥ 25% RTP, the Surveillance must be performed within 12 hours.~~

~~The use of "once" indicates a single performance will satisfy the specified Frequency (assuming no other Frequencies are connected by "AND").~~ This type of Frequency does not qualify for the 25% extension allowed by SR 3.0.2. ~~"Thereafter" indicates future performances must be established per SR 3.0.2, but only after a specified condition is first met (i.e., the "once" performance in this example). If reactor power decreases to < 25% RTP, the measurement of both intervals stops. New intervals start upon reactor power reaching 25% RTP.~~

(continued)

1.4 Frequency (continued)

~~EXAMPLES (continued)~~

~~EXAMPLE 1.4-3~~

~~SURVEILLANCE REQUIREMENTS~~

~~SURVEILLANCE~~	~~FREQUENCY~~
~~-------------------NOTE-------------------~~ ~~Not required to be performed until 12 hours after ≥ 25% RTP.~~ ~~--~~	
~~Perform channel adjustment.~~	~~7 days~~

~~The interval continues, whether or not the unit operation is < 25% RTP between performances.~~

~~As the Note modifies the required performance of the Surveillance, it is construed to be part of the "specified Frequency." Should the 7 day interval be exceeded while operation is < 25% RTP, this Note allows 12 hours after power reaches ≥ 25% RTP to perform the Surveillance. The Surveillance is still considered to be performed within the "specified Frequency." Therefore, if the Surveillance were not performed within the 7 day (plus 25% per SR 3.0.2) interval, but operation was < 25% RTP, it would not constitute a failure of the SR or failure to meet the LCO. Also, no violation of SR 3.0.4 occurs when changing MODES, even with the 7 day Frequency not met, provided operation does not exceed 12 hours with power ≥ 25% RTP.~~

~~Once the unit reaches 25% RTP, 12 hours would be allowed for completing the Surveillance. If the Surveillance were not performed within this 12 hour interval, there would then be a failure to perform a Surveillance within the specified Frequency; MODE changes then would be restricted in accordance with SR 3.0.4 and the provisions of SR 3.0.3 would apply.~~

3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

LCO 3.0.1 — LCOs shall be met during the ~~MODES or other~~ specified conditions in the Applicability, except as provided in LCO 3.0.2 ~~and LCO 3.0.7~~.

LCO 3.0.2 — Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met~~, except as provided in LCO 3.0.5 and LCO 3.0.6~~.

If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required, unless otherwise stated.

~~LCO 3.0.3~~ — ~~When an LCO is not met and the associated ACTIONS are not met or an associated ACTION is not provided, the unit shall be placed in a MODE or other specified condition in which the LCO is not applicable. Action shall be initiated within 1 hour to place the unit, as applicable, in:~~

~~a. MODE 3 within 7 hours;~~

~~b. MODE 4 within 13 hours; and~~

~~c. MODE 5 within 37 hours.~~

~~Exceptions to this Specification are stated in the individual Specifications.~~

~~Where corrective measures are completed that permit operation in accordance with the LCO or ACTIONS, completion of the actions required by LCO 3.0.3 is not required.~~

~~LCO 3.0.3 is applicable in MODES 1, 2, 3, and 4.~~

~~LCO 3.0.4~~ — ~~When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall not be made except when the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time. This~~

(continued)

3.0 LCO APPLICABILITY (continued)

~~LCO 3.0.4 (continued)~~

~~Specification shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS.~~

~~Exceptions to this Specification are stated in the individual Specifications. These exceptions allow entry into MODES or other specified conditions in the Applicability when the associated ACTIONS to be entered allow unit operation in the MODE or other specified condition in the Applicability only for a limited period of time.~~

~~LCO 3.0.5~~

~~Equipment removed from service or declared inoperable to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to LCO 3.0.2 for the system returned to service under administrative control to perform the testing required to demonstrate OPERABILITY.~~

~~LCO 3.0.6~~

~~When a supported system LCO is not met solely due to a support system LCO not being met, the Conditions and Required Actions associated with this supported system are not required to be entered. Only the support system LCO ACTIONS are required to be entered. This is an exception to LCO 3.0.2 for the supported system. In this event, additional evaluations and limitations may be required in accordance with Specification 5.6, "Safety Function Determination Program (SFDP)." If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.~~

~~When a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.~~

(continued)

3.0 LCO APPLICABILITY (continued)

~~LCO 3.0.7~~

~~Special test exception (STE) LCOs in each applicable LCO section allow specified Technical Specifications (TS) LCO 3.0.7 requirements to be changed to permit performance of special tests and operations. Unless otherwise specified, all other TS requirements remain unchanged. Compliance with STE LCOs is optional. When an STE LCO is desired to be met but is not met, the ACTIONS of the STE LCO shall be met. When an STE LCO is not desired to be met, entry into a MODE or other specified condition in the Applicability shall only be made in accordance with the other applicable Specifications.~~

3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 3.0.1

SRs shall be met during the ~~MODES or other~~ specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.

SR 3.0.2

The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met.

~~For Frequencies specified as "once," the above interval extension does not apply.~~

~~If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance.~~

~~Exceptions to this Specification are stated in the individual Specifications.~~

SR 3.0.3

If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is greater. This delay period is permitted to allow performance of the Surveillance. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.

If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered. The Completion Times of the Required Actions begin immediately upon expiration of the delay period.

(continued)

3.0 SR APPLICABILITY

SR 3.0.3 (continued)

When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered. The Completion Times of the Required Actions begin immediately upon failure to meet the Surveillance.

SR 3.0.4

Entry into a ~~MODE or other~~ specified condition in the Applicability of an LCO shall not be made unless the LCO's Surveillances have been met within their specified Frequency. This provision shall not prevent entry into ~~MODES or other~~ specified conditions in the Applicability that are required to comply with Actions.

~~3.7~~ 3.1 PLANT SYSTEMS

~~3.7.16~~ 3.1.1 Fuel Storage Pool Water Level

LCO ~~3.7.16~~ 3.1.1 The fuel storage pool water level shall be ≥ 23 ft over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: During movement of fuel assemblies in the fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Fuel storage pool water level not within limit.	A.1 ~~--------NOTE---------~~ ~~LCO 3.0.3 is not applicable.~~ Suspend movement of fuel assemblies in fuel storage pool.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR ~~3.7.16.1~~ 3.1.1.1 Verify the fuel storage pool water level is ≥ 23 ft above the top of irradiated fuel assemblies seated in the storage racks.	7 days

~~3.7~~ 3.1 PLANT SYSTEMS

~~3.7.17~~ 3.1.2 Fuel Storage Pool Boron Concentration

LCO ~~3.7.17~~ 3.1.2 The fuel storage pool boron concentration shall be ≥ 2000 ppm.

APPLICABILITY: Whenever any fuel assembly is stored in the fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Fuel storage pool boron concentration not within limit.	~~------------NOTE------------~~ ~~LCO 3.0.3 is not applicable.~~ ~~----------------------------~~	
	A.1 Suspend movement of fuel assemblies in the fuel storage pool.	Immediately
	AND	
	A.2 Initiate action to restore fuel storage pool boron concentration to within limit.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR ~~3.7.17.1~~ 3.1.2.1 Verify the fuel storage pool boron concentration is within limit.	7 days

3.1 ~~3.7~~ PLANT SYSTEMS

3.1.3 ~~3.7.18~~ Spent Fuel Assembly Storage

LCO 3.1.3 ~~3.7.18~~ The combination of initial enrichment and burnup of each SONGS 2 and 3 spent fuel assembly stored in Region I shall be within the acceptable burnup domain of Figure ~~3.7.18-1~~ 3.1.3-1 or Figure ~~3.7.18-2~~ 3.1.3-2, or the fuel assembly shall be stored in accordance with Technical Specification 4.3.1.1.

The combination of initial enrichment and burnup of each SONGS 2 and 3 spent fuel assembly stored in Region II shall be within the acceptable burnup domain of Figure ~~3.7.18-3~~ 3.1.3-3 or Figure ~~3.7.18-4~~ 3.1.3-4, or the fuel assembly shall be stored in accordance with Technical Specification 4.3.1.1.

Each SONGS 1 uranium dioxide spent fuel assembly stored in Region II shall be stored in accordance with Technical Specification 4.3.1.1.

APPLICABILITY: Whenever any fuel assembly is stored in the fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 ~~----------NOTE----------~~ ~~LCO 3.0.3 is not applicable.~~ ~~---------------------~~ Initiate action to bring the noncomplying fuel assembly into compliance.	 Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR ~~3.7.18.1~~ 3.1.3.1	Verify by administrative means the initial enrichment, burnup, and cooling time of the fuel assembly are in accordance with LCO ~~3.7.18~~ 3.1.3, or Design Features 4.3.1.1, or LCS 4.3.100. Rev 2, dated 09/27/07.	Prior to moving a fuel assembly to any spent fuel pool storage location.

FIGURE ~~3.7.18-1~~ 3.1.3-1

MINIMUM BURNUP AND COOLING TIME VS. INITIAL ENRICHMENT
FOR
UNRESTRICTED PLACEMENT OF SONGS 2 AND 3 FUEL
IN
REGION I RACKS

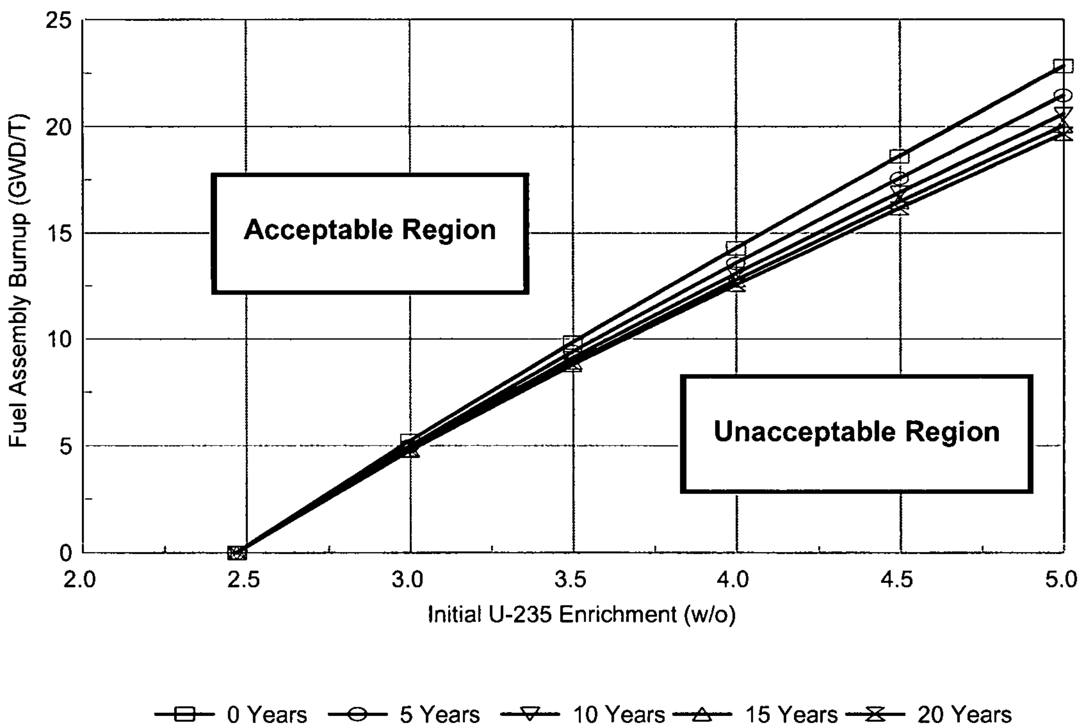


FIGURE ~~3.7.18-2~~
3.1.3-2

MINIMUM BURNUP AND COOLING TIME VS. INITIAL ENRICHMENT
FOR
PLACEMENT OF SONGS 2 AND 3 FUEL IN PERIPHERAL POOL LOCATIONS
IN
REGION I RACKS

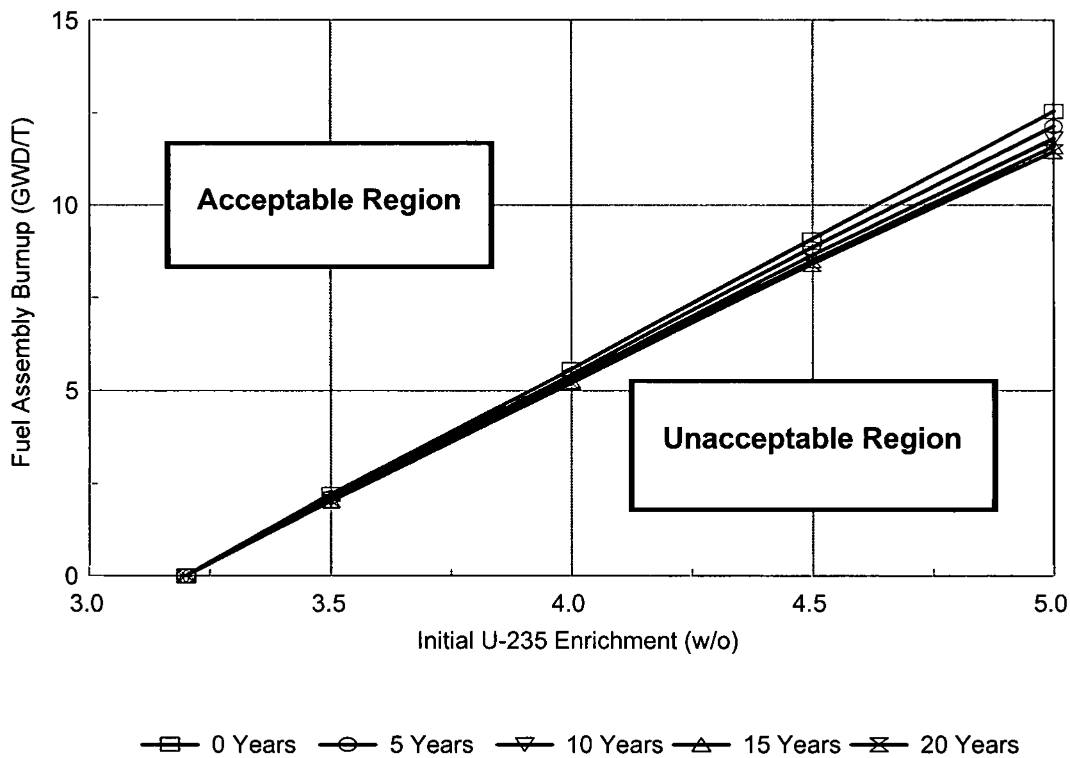


FIGURE ~~3.7.18-3~~ 3.1.3-3

MINIMUM BURNUP AND COOLING TIME VS. INITIAL ENRICHMENT
FOR
UNRESTRICTED PLACEMENT OF SONGS 2 AND 3 FUEL
IN
REGION II RACKS

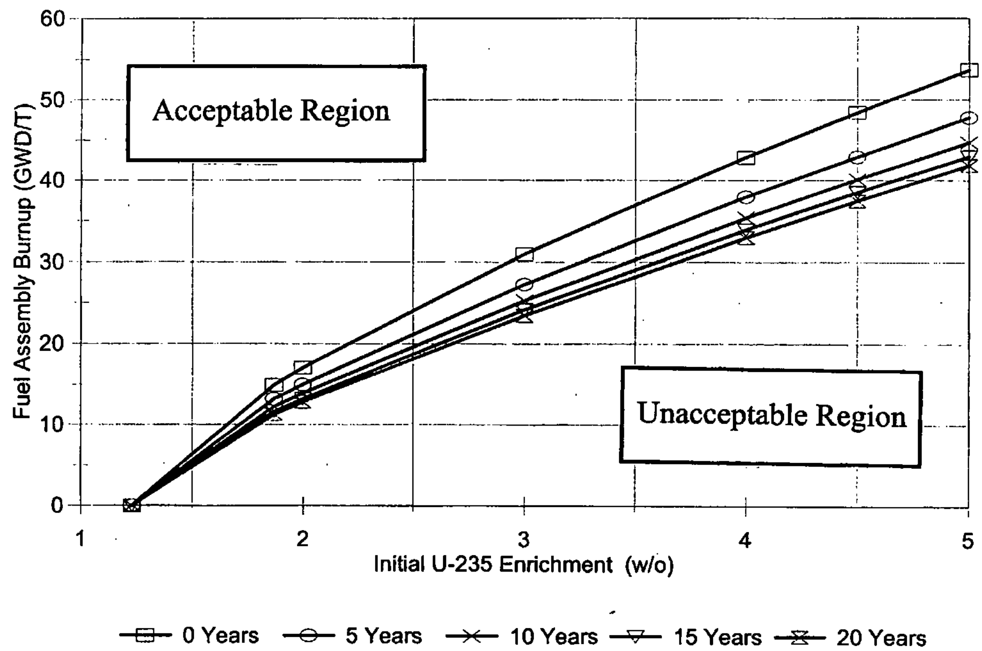
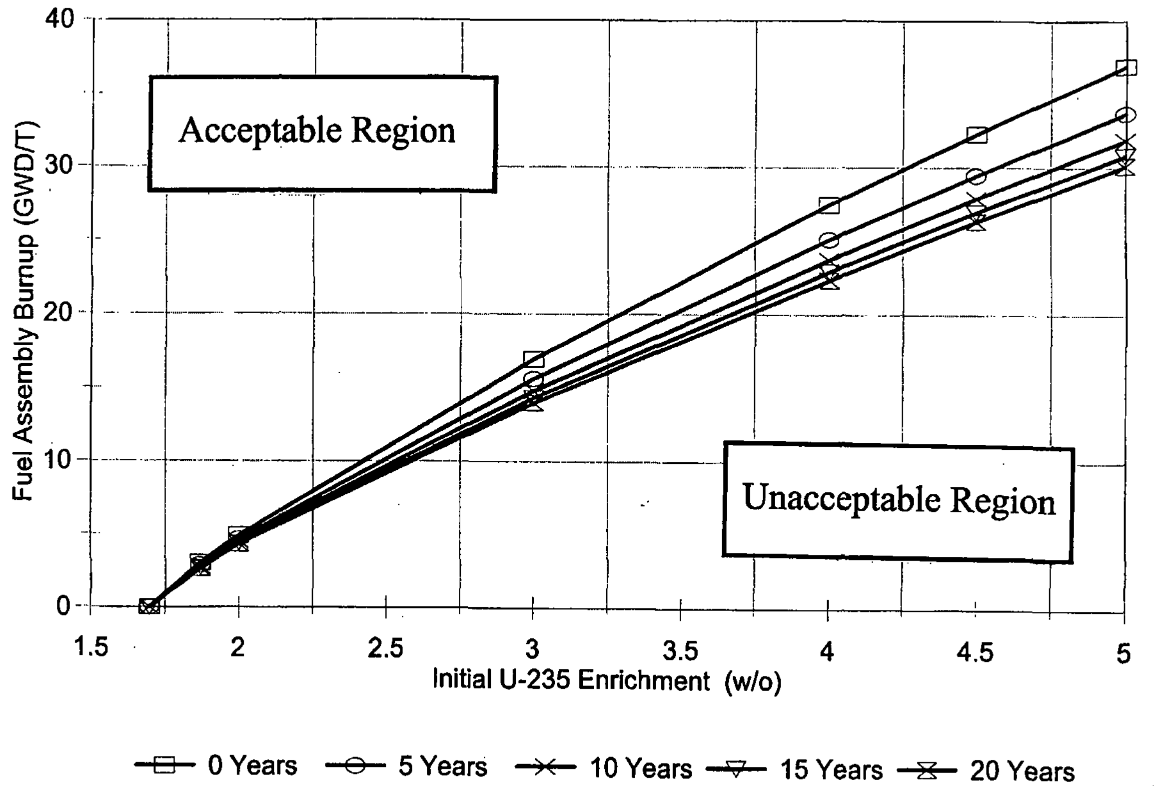


FIGURE ~~3.7.18-4~~
3.1.3-4

MINIMUM BURNUP AND COOLING TIME VS. INITIAL ENRICHMENT
FOR
PLACEMENT OF SONGS 2 AND 3 FUEL IN PERIPHERAL POOL LOCATIONS
IN
REGION II RACKS



4.0 DESIGN FEATURES

4.1 Site

4.1.1 Exclusion Area Boundary

The exclusion area boundary shall be as shown in Figure 4.1-1.

4.1.2 Low Population Zone (LPZ)

The LPZ shall be as shown in Figure 4.1-2.

4.2 ~~Reactor Core~~ Deleted.

~~4.2.1 Fuel Assemblies~~

~~The reactor shall contain 217 fuel assemblies. Each assembly shall consist of a matrix of Zircaloy or ZIRLO™ clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Integral or Discrete Burnable Absorber Rods may be used. They may include: borosilicate glass - Na_2O-B_2O_3-SiO_2 components, boron carbide - B_4C, zirconium boride - ZrB_2, gadolinium oxide - Gd_2O_3, erbium oxide - Er_2O_3. Limited substitutions of zirconium alloy (such as ZIRLO™ or Zircaloy) 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.~~

~~4.2.2 Control Element Assemblies~~

~~The reactor core shall contain 83 full length and eight part length control element assemblies (CEAs). The control material shall be silver indium cadmium, boron carbide, and inconel as approved by the NRC.~~

(continued)

4.0 DESIGN FEATURES (continued)

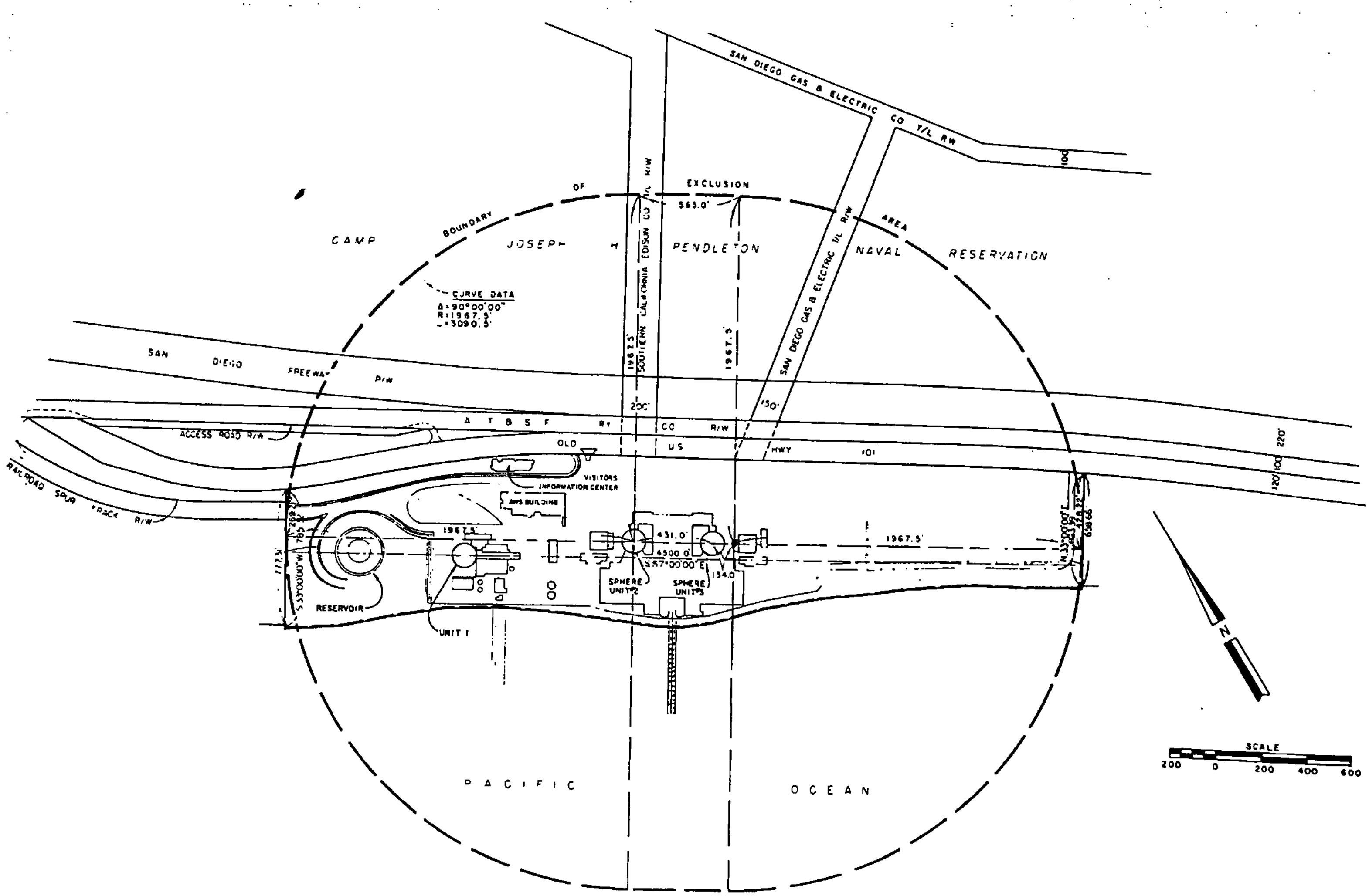


Figure 4.1-1 (page 1 of 1)
Exclusion Area Boundary

(continued)

4.0 DESIGN FEATURES (continued)

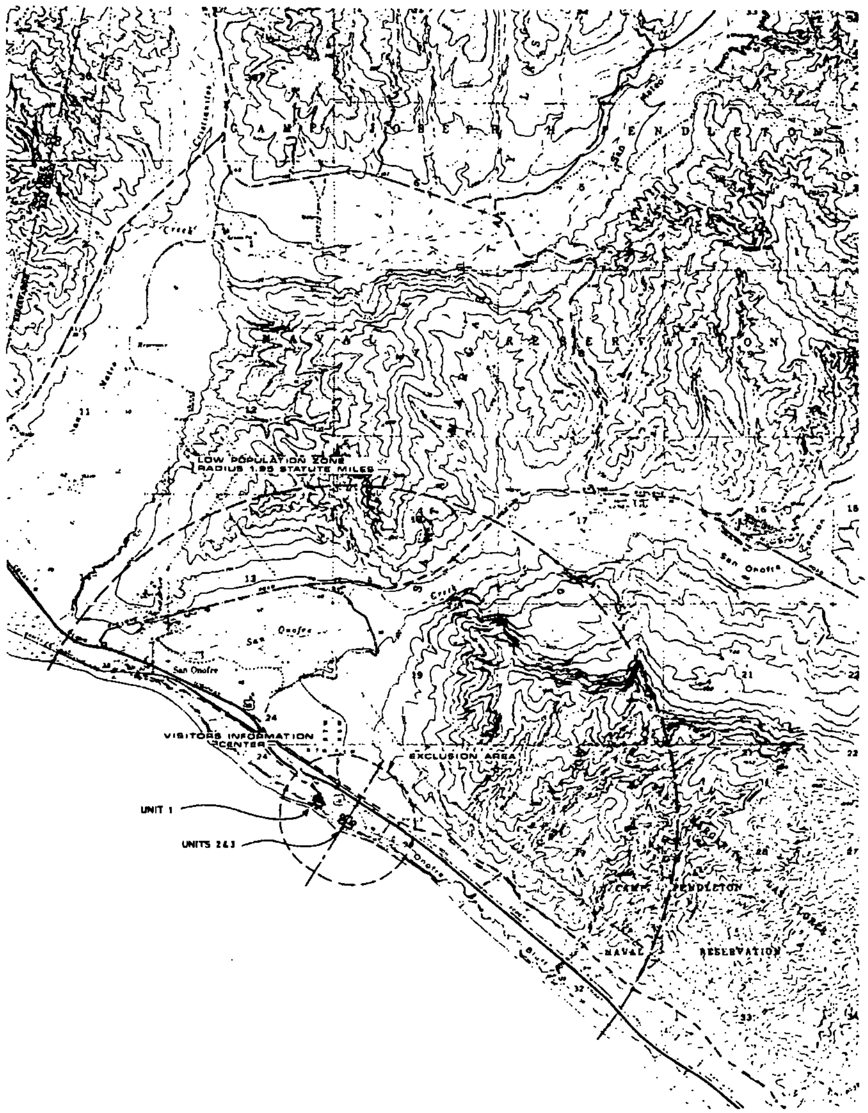


Figure 4.1-2 (page 1 of 1)
Low Population Zone

(continued)

4.0 DESIGN FEATURES (continued)

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 4.8 weight percent;

b. $K_{eff} < 1.0$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR;

c. $K_{eff} \leq 0.95$ if fully flooded with water borated to 1700 ppm, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR;

d. Three or five Borated stainless steel guide tube inserts (GT-Insert) may be used. When three Borated stainless steel guide tube inserts are used, they will be installed in an assembly's center guide tube, the guide tube associated with the serial number, and the diagonally opposite guide tube. Fuel containing GT-Inserts may be placed in either Region I or Region II. However, credit for GT-Inserts is only taken for Region II storage.

   A five-finger CEA may be installed in an assembly. Fuel containing a five-finger CEA may be placed in either Region I or Region II. Credit for inserted 5-finger CEAs is taken for both Region I and Region II.

e. A nominal 8.85 inch center to center distance between fuel assemblies placed in Region II;

f. A nominal 10.40 inch center to center distance between fuel assemblies placed in Region I;

(continued)

4.0 DESIGN FEATURES (continued)

4.3.1 Criticality (continued)

3.1.3

g. Prior to using the storage criteria of LCO ~~3.7.18~~ and LCS 4.0.100, the following uncertainties will be applied:

(1) The calculated discharge burnup of San Onofre Units 2 and 3 assemblies will be reduced by 6.6%.

(2) The calculated discharge burnup of San Onofre Unit 1 fuel assemblies will be reduced by 10.0%.

3.1.3-1 h. Units 2 and 3 fuel assemblies with a burnup in the "acceptable range" of Figure ~~3.7.18-1~~ are allowed unrestricted storage in Region I;

3.1.3-2 i. Units 2 and 3 fuel assemblies with a burnup in the "acceptable range" of Figure ~~3.7.18-2~~ are allowed unrestricted storage in the peripheral pool locations with 1 or 2 faces toward the spent fuel pool walls of Region I;

3.1.3-3 j. Units 2 and 3 fuel assemblies with a burnup in the "acceptable range" of Figure ~~3.7.18-3~~ are allowed unrestricted storage in Region II;

3.1.3-4 k. Units 2 and 3 fuel assemblies with a burnup in the "acceptable range" of Figure ~~3.7.18-4~~ are allowed unrestricted storage in the peripheral pool locations with 1 or 2 faces toward the spent fuel pool walls of Region II;

Figure 3.1.3-1, Figure 3.1.3-2, Figure 3.1.3-3, and Figure 3.1.3-4

l. Units 2 and 3 fuel assemblies with a burnup in the "unacceptable range" of ~~Figure 3.7.18-1, Figure 3.7.18-2, Figure 3.7.18-3, and Figure 3.7.18-4~~ will be stored in compliance with Licensee Controlled Specification 4.0.100 Rev. 2, dated 09/27/07; and

m. Each SONGS 1 uranium dioxide spent fuel assembly stored in Region II shall be stored in accordance with Licensee Controlled Specification 4.0.100 Rev. 2, dated 09/27/07.

(continued)

4.0 DESIGN FEATURES (continued)

4.3 Fuel Storage (continued)

~~4.3.1.2 The new fuel storage racks are designed and shall be maintained with:~~

~~a. Fuel assemblies having a maximum U-235 enrichment of 4.8 weight percent;~~

~~b. $K_{eff} \leq$ 0.95 if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR;~~

~~c. $K_{eff} \leq$ 0.98 if moderated by aqueous foam, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR; and~~

~~d. A minimum 29 inch center to center distance between fuel assemblies placed in the storage racks.~~

4.3.2 Drainage

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below Technical Specification 3.1.1 ~~3.7.16~~ value (23 feet above the top of irradiated fuel assemblies seated in the storage racks).

4.3.3 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1542 fuel assemblies.

5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

5.1.1 The corporate officer with direct responsibility for the plant shall be responsible for overall ~~unit operation and maintenance of~~ management of the ~~Units 2 and 3 at~~ San Onofre Nuclear Generating Station, and all site support functions. He shall delegate in writing the succession to this responsibility during his absence.

5.1.2 The Shift Manager shall be responsible for the ultimate command decision authority for all unit activities and operations which affect the safety of the plant, site personnel, and/or the general public. ~~A management directive to this effect, signed by the corporate officer with direct responsibility for the plant shall be reissued to all site/station personnel on an annual basis.~~

~~5.1.3~~ ~~The Control Room Supervisor (CRS) shall be responsible for the Control Room command function. A management directive to this effect, signed by the corporate officer with direct responsibility for the plant, shall be issued annually to all site/station personnel. The confines of the Control Room Area shall be defined as depicted in the Licensee Controlled Specification (LCS). During any absence of the CRS from the Control Room Area while the Unit is in MODE 1, 2, 3, or 4, an individual with an active Senior Reactor Operator's (SRO) license shall be designated to assume the Control Room command function. During any absence of the CRS from the Control Room Area while the Unit is in MODE 5 or 6, an individual with an active SRO license or Reactor Operator's license shall be designated to assume the Control Room command function.~~

5.0 ADMINISTRATIVE CONTROLS

5.2 Organization

5.2.1 Onsite and Offsite Organizations

Onsite and offsite organizations shall be established for ~~unit~~ plant operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting the safety of the nuclear ~~power plant~~ fuel.

a. Lines of authority, responsibility, and communication shall be established and defined for the highest management levels through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These relationships, including the plant-specific titles of those personnel fulfilling the responsibilities for the positions delineated in these Technical Specifications, are documented in the UFSAR.

b. The corporate officer with direct responsibility for the plant shall be responsible for overall ~~unit~~ safe handling and storage ~~operation~~ of nuclear fuel and shall have control over those onsite activities necessary for safe ~~operation and maintenance~~ handling and storage of the nuclear fuel ~~plant~~ .

c. A specified corporate officer (or officers) shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure safe management of nuclear fuel ~~safety~~.

CERTIFIED FUEL HANDLERS

d. The individuals who train ~~the operating staff~~ ~~Certified Fuel Handlers~~, and those who carry out ~~health physics~~ radiation protection and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their ~~independence from operating pressures~~ ability to perform their assigned functions.

(continued)

5.2 Organization (continued)

5.2.2 ~~UNIT~~ FACILITY STAFF

The ~~unit~~ facility staff organization shall include the following:

a. ~~A non-Licensed Operator shall be assigned to each reactor containing fuel and an additional non-Licensed Operator shall be assigned for each unit when a reactor is operating in MODES 1, 2, 3, or 4.~~ Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 5.2.2-1.

~~With both units shutdown or defueled, a total of three non-licensed operators are required for the two units.~~

~~b.~~ ~~At least one licensed Reactor Operator (RO) shall be in the Control Room when fuel is in the reactor. In addition, while the unit is in MODE 1, 2, 3 or 4, at least one licensed Senior Reactor Operator (SRO) shall be in the Control Room Area.~~

~~c.~~b. Shift crew composition may be less than the minimum requirement of ~~10 CFR 50.54(m)(2)(i) and 5.2.2.a~~ Table 5.2.2-1 for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements. INSERT 2

INSERT 3

~~d.~~c. ~~A radiation protection technician shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.~~ Oversight of fuel handling operations shall be provided by a ~~Certified Fuel Handler~~ CERTIFIED FUEL HANDLER.

~~e.~~ ~~Deleted~~

~~f.~~d. The Shift Manager~~, Plant Operations (at time of appointment), Shift Managers, and Control Room Supervisors~~ shall ~~hold~~ be a ~~Senior Reactor Operator's license.~~ ~~Certified Fuel Handler~~ CERTIFIED FUEL HANDLER.

~~g.~~ ~~The Shift Technical Advisor (STA) shall provide advisory technical support to the Shift Manager in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. The STA shall have a Bachelor's Degree or equivalent in a scientific or engineering discipline with specific training in plant design and in the response and analysis of the plant for transients and accidents.~~

(continued)

INSERT 2

During such absences, no fuel movement or movement of heavy loads over storage racks containing fuel is permitted.

INSERT 3

A radiation protection technician shall be on site during fuel handling operations and during movement of heavy loads over storage racks containing fuel.

5.2 Organization (continued)

CERTIFIED FUEL HANDLER

Table 5.2.2-1
Minimum Shift Crew Composition

POSITION	MINIMUM STAFFING
~~Certified Fuel Handler~~	1*
Certified Operator	1

Note: The Certified Operator position may be filled by a ~~Certified Fuel Handler~~.
* May be shared between Units 2 and 3.

CERTIFIED FUEL HANDLER

(continued)

5.0 ADMINISTRATIVE CONTROLS

5.3 ~~Unit~~ Facility Staff Qualifications

5.3.1 Each member of the ~~unit~~ facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except ~~a)~~ the radiation protection manager who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975~~, and b) multi-discipline supervisors who shall meet or exceed the qualifications listed below~~.

~~In addition, the Shift Technical Advisor shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift.~~

~~Multi-discipline supervisors shall meet or exceed the following requirements:~~

~~a.~~ ~~Education:~~ ~~Minimum of a high school diploma or equivalent~~.

~~b.~~ ~~Experience:~~ ~~Minimum of four years of related technical experience which shall include three years power plant experience of which one year is at a nuclear plant~~.

~~c.~~ ~~Training:~~ ~~Complete the multi-discipline supervisor training program~~.

5.3.2 An NRC approved training and retraining program for the ~~Certified Fuel Handlers~~ CERTIFIED FUEL HANDLERS shall be maintained.

5.0 ADMINISTRATIVE CONTROLS

5.4 Technical Specifications (TS) Bases Control

5.4.1 Changes to the Bases of the TS shall be made under appropriate administrative controls.

5.4.2 Changes to the Bases may be made without prior NRC approval provided the changes do not require either of the following:

a. A change in the TS incorporated in the license; or

b. A change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

5.4.3 The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the UFSAR.

5.4.4 Proposed changes that meet the criteria of (a) or (b) above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC every ~~within 6 months following every Unit 3 refueling, not to exceed~~ 24 months. ~~This schedule is consistent with SCE's submittal of UFSAR updates as allowed by the NRC approved exemption for 10 CFR 50.71(e) dated April 27, 1999.~~

5.0 ADMINISTRATIVE CONTROLS

5.5 Procedures, Programs, and Manuals

5.5.1 Procedures

5.5.1.1 Scope

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

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

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

Deleted

c. Quality assurance for effluent and environmental monitoring using the guidance in Regulatory Guide 4.15, Revision 1, 1979;

d. Fire Protection Program implementation; and

e. Programs, as specified in Specification 5.5.2.

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

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

(continued)

5.5.2 Programs and Manuals

The following programs and manuals shall be established, implemented, and maintained.

5.5.2.1 Offsite Dose Calculation Manual (ODCM)

a. The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the Radiological Environmental Monitoring Program;

b. The ODCM shall also contain the Radioactive Effluent Controls required by Specification 5.5.2.3 and the Radiological Environmental Monitoring programs required by the LCS, and descriptions of the information that should be included in the Annual Radiological Environmental Operating Report and the Radioactive Effluent Release Report required by Specification 5.7.1.2 and Specification 5.7.1.3.

5.5.2.1.1 Licensee-initiated changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

   1. Sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s);

   2. A determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.106, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

   3. Documentation of the fact that the change has been reviewed and found acceptable.

b. Shall become effective upon review and approval by the corporate officer with direct responsibility for the plant or designee.

(continued)

5.5.2.1.1 Licensee-initiated changes to the ODCM: (continued)

c. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

5.5.2.2 Deleted

5.5.2.3 Radioactive Effluent Controls Program

This program conforming to 10 CFR 50.36a provides for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by operating procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to 10 CFR 20, Appendix B, Table II, Column 2;

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.106 and with the methodology and parameters in the ODCM;

(continued)

5.5.2.3 Radioactive Effluent Controls Program (continued)

d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each unit to unrestricted areas, conforming to 10 CFR 50, Appendix I;

e. Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days;

f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2 percent of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I;

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the site boundary conforming to the dose associated with 10 CFR 20, Appendix B, Table II, Column 1;

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and

j. Limitations on the annual dose or dose commitment to any member of the public due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

5.5.2.4 ~~Component Cyclic or Transient Limit Program~~

Deleted

~~This program provides controls to track the UFSAR Table 3.9-1 cyclic and transient occurrences to ensure that components are maintained within the design limits.~~

(continued)

5.5 Procedures, Programs, and Manuals (continued)

5.5.2.5 ~~Reactor Coolant Pump Flywheel Inspection Program~~

Deleted

~~Surveillance of the primary coolant pump flywheels shall consist of a 100% volumetric inspection of the flywheels each 10 years.~~

5.5.2.6 ~~Secondary Water Chemistry Program~~

Deleted

~~This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation and low pressure turbine disc stress corrosion cracking.~~ ~~The program shall include:~~

~~a.~~ ~~Identification of a sampling schedule for the critical parameters and control points for these parameters;~~

~~b.~~ ~~Identification of the procedures used to measure the values of the critical parameters;~~

~~c.~~ ~~Identification of process sampling points;~~

~~d.~~ ~~Procedures for the recording and management of data;~~

~~e.~~ ~~Procedures defining corrective actions for all off-control point chemistry conditions; and~~

~~f.~~ ~~A procedure identifying (a) the authority responsible for interpretation of data and (b) the sequence and timing of administrative events, required to initiate corrective action.~~

5.5.2.7 ~~Explosive Gas and~~ Storage Tank Radioactivity Monitoring Program

This program provides controls for ~~potentially explosive gas mixtures contained in the Gaseous Radwaste System, the quantity of radioactivity contained in gas storage tanks, and~~ the quantity of radioactivity contained in unprotected outdoor liquid storage tanks. ~~The gaseous radioactivity quantities shall be determined following methodology comparable with Branch Technical Position (BTP) ETSB 11-5, "Postulated Radioactive Release due to Waste Gas System Leak or Failure".~~ The liquid radwaste quantities shall be determined in accordance with Standard Review Plan, Section 15.7.3, "Postulated Radioactive Release due to Tank Failures".

(continued)

5.5.2.7 ~~Explosive Gas and~~ Storage Tank Radioactivity Monitoring Program (continued)

The program shall include:

~~a.~~ ~~The limits for the concentrations of hydrogen and oxygen in the Gaseous Radwaste System and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion); and~~

~~b.~~ ~~A surveillance program to ensure that the quantity of radioactivity contained in each waste gas decay tank and fed into the gaseous radwaste vent system is less than the amount that would result in a whole body exposure of greater than or equal to 0.5 rem to any individual in the unrestricted area, in the event of an uncontrolled release of the tanks contents; and~~

~~c.~~ a ~~A~~ surveillance program to ensure that the quantity of radioactivity contained in all outdoor liquid radwaste tanks that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the Liquid Waste Management System is less than the amount that would result in concentrations less than the limits of 10 CFR Part 20, Appendix B, Table II, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tanks' contents.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the ~~Explosive Gas and~~ Storage Tank Radioactivity Monitoring Program surveillance frequencies.

~~5.5.2.8~~ ~~Primary Coolant Sources Outside Containment Program~~

~~This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The systems include high pressure safety injection recirculation, the shutdown cooling system, the reactor coolant sampling system (post-accident sampling piping only until such time as a modification eliminates the post-accident piping as a potential leakage path), the containment spray system, the radioactive waste gas system (post-accident sampling return piping only until such time as a modification eliminates the post-accident piping as a potential leakage path), and the liquid radwaste~~

(continued)

~~5.5.2.8~~ ~~Primary Coolant Sources Outside Containment Program (continued)~~

~~system (post-accident sampling return piping only until such time as a modification eliminates the post-accident piping as a potential leakage path). The program shall include the following:~~

~~a.~~ ~~Preventive maintenance and periodic visual inspection requirements; and~~

~~b.~~ ~~Integrated leak test requirements for each system at refueling cycle intervals or less.~~

~~5.5.2.9~~ ~~Pre-Stressed Concrete Containment Tendon Surveillance Program~~

~~This program provides controls for monitoring any tendon degradation in pre-stressed concrete containment, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. Program itself is relocated to the LCS.~~

~~5.5.2.10~~ ~~Inservice Inspection and Testing Program~~

~~This program provides controls for inservice inspection of ASME Code Class 1, 2, and 3 components and Code Class CC and MC components including applicable supports. The program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components. The program itself is located in the LCS.~~

~~5.5.2.11~~ ~~Steam Generator (SG) Program~~

~~A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following provisions:~~

~~a.~~ ~~Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged, to confirm that the performance criteria are being met.~~

(continued)

5.5 Procedures, Programs, and Manuals (continued)

~~5.5.2.11 Steam Generator (SG) Program (continued)~~

~~b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.~~

~~1. Structural integrity performance criterion: All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.~~

~~2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 0.5 gpm per SG and 1 gpm through both SGs.~~

~~3. The operational LEAKAGE performance criterion is specified in LCO 3.4.13, "RCS Operational LEAKAGE."~~

(continued)

~~5.5.2.11 Steam Generator (SG) Program (continued)~~

~~c. Provisions for SG tube repair criteria.~~

~~1. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 35% of the nominal tube wall thickness shall be plugged.~~

~~d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. The tube-to-tubesheet weld is not part of the tube.~~

~~In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.~~

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

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

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

~~e. Provisions for monitoring operational primary to secondary LEAKAGE.~~

5.5 Procedures, Programs, and Manuals (continued)

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5.5 Procedures, Programs, and Manuals (continued)

~~5.5.2.12~~ ~~Ventilation Filter Testing Program (VFTP)~~

~~This Program establishes the required testing of the Engineered Safety Feature filter ventilation system "Control Room Emergency Air Cleanup System." The frequency of testing shall be in accordance with Regulatory Guide 1.52, Revision 2. As a minimum the VFTP program shall include the following:~~

~~a. Inplace testing of the high efficiency particulate air (HEPA) filters to demonstrate acceptable penetration and system bypass when tested at the appropriate system flowrate in accordance with Regulatory Guide 1.52, Revision 2, and ANSI N510-1975 (see Note 1); and~~

~~b. Inplace testing of the charcoal adsorber to demonstrate acceptable penetration and system bypass when tested at the appropriate system flowrate in accordance with Regulatory Guide 1.52, Revision 2, and ANSI N510-1975 (see Note 1); and~~

~~c. Laboratory testing of charcoal adsorber samples obtained in accordance with Regulatory Guide 1.52, Revision 2 and tested per the methodology of ASTM D3803-1989 at 30°C and 70% relative humidity to show acceptable methyl iodide penetration; and~~

~~d. Testing to demonstrate the pressure drop across the combined HEPA filters, the prefilters, and the charcoal adsorbers, when tested at the appropriate system flowrate.~~

~~Note 1: Sample and injection points shall be qualified per ANSI N510-1975 unless manifolds have been qualified per ASME N510-1989. HEPA testing will be conducted with DOP aerosol or suitable alternate.~~

(continued)

5.5 Procedures, Programs, and Manuals (continued)

~~5.5.2.12~~ ~~Ventilation Filter Testing Program (VFTP) (continued)~~

~~The provisions of Technical Specification Surveillance Requirement 3.0.2 and Technical Specification Surveillance Requirement 3.0.3 are applicable to the VFTP test frequencies.~~

~~5.5.2.13~~ ~~Diesel Fuel Oil Testing Program~~

~~This program implements required testing of both new fuel oil and stored fuel oil. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM standards. The purpose of the program is to establish the following:~~

~~a. Acceptability of new fuel oil use prior to addition to storage tanks by determining that the fuel oil has:~~

~~1. an API gravity or an absolute specific gravity within limits,~~

~~2. a flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and~~

~~3. a water and sediment content within limits.~~

~~b. Other properties for ASTM 2D fuel oil are within limits within 31 days following sampling and addition to the storage tanks, with exceptions noted in the Bases for Surveillance Requirement 3.8.3.3; and,~~

~~c. Total particulate concentration of fuel oil is ≤ 10 mg/l when tested every 92 days in accordance with ASTM D-2276, Method A.~~

~~5.5.2.14~~ ~~Deleted~~

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5.5 Procedures, Programs, and Manuals (continued)

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5.5 Procedures, Programs, and Manuals (continued)

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5.5 Procedures, Programs, and Manuals (continued)

~~5.5.2.15 Containment Leakage Rate Testing Program~~

~~A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995 as modified by the following exception:~~

~~NEI 94-01 - 1995, Section 9.2.3: The first Type A Test performed after the September 10, 1995 Type A Test shall be performed prior to startup from Unit 3 Cycle 16 refueling outage, which is scheduled to commence in the fall of 2010 and to end in the first quarter of 2011. SONGS Unit 3 shall not operate past September 9, 2011 until the Type A Test is satisfactorily completed.~~

~~The calculated peak containment internal pressure related to the design basis loss-of-coolant accident,~~ $\cancel{P_a}$, ~~is 48.0 psig (~~$\cancel{P_a}$ ~~will conservatively be assumed to be equal to the calculated peak containment internal pressure for the design basis Main Steam Line Break (51.5 psig) for the purpose of containment testing in accordance with this Technical Specification).~~

~~The maximum allowable containment leakage rate,~~ $\cancel{L_a}$, ~~at~~ $\cancel{P_a}$, ~~shall be 0.10% of containment air weight per day.~~

~~Leakage rate acceptance criteria are:~~

~~a. The Containment overall leakage rate acceptance criterion is~~ $\cancel{\leq 1.0\ L_a}$. ~~During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are~~ $\cancel{\leq 0.60\ L_a}$ ~~for the Type B and Type C tests and~~ $\cancel{\leq 0.75\ L_a}$ ~~for the Type A tests;~~

~~b. Air lock testing acceptance criteria are:~~

~~1) Overall air lock leakage rate is~~ $\cancel{\leq 0.05\ L_a}$ ~~when tested at~~ $\cancel{\geq P_a}$.

~~2) For each door, the leakage rate is~~ $\cancel{\leq 0.01\ L_a}$ ~~when pressurized to~~ $\cancel{\geq 9.0}$ ~~psig.~~

(continued)

5.5 Procedures, Programs, and Manuals (continued)

~~5.5.2.15~~ ~~Containment Leakage Rate Testing Program (Continued)~~

~~The provisions of Surveillance Requirement 3.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program. However, test frequencies specified in this Program may be extended consistent with the guidance provided in NEI 94-01, "Industry Guideline For Implementing Performance-Based Option Of 10CFR 50, Appendix J," as endorsed by Regulatory Guide 1.163. Specifically, NEI 94-01 has these provisions for test frequencies extension:~~

~~1.~~ ~~Consistent with standard scheduling practices for Technical Specifications Required Surveillances, intervals for recommended Type A testing may be extended by up to 15 months. This option should be used only in cases where refueling schedules have been changed to accommodate other factors.~~

~~2.~~ ~~Consistent with standard scheduling practices for Technical Specifications Required Surveillances, intervals for the recommended surveillance frequency for Type B and Type C testing may be extended by up to 25 percent of the test interval, not to exceed 15 months.~~

~~The provisions of Surveillance Requirement 3.0.3 are applicable to the Containment Leakage Rate Testing Program.~~

~~5.5.2.16~~ ~~Control Room Envelope Habitability Program~~

~~A Control Room Envelope Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Emergency Air Cleanup System (CREACUS), CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem total effective dose equivalent (TEDE) for the duration of the accident. The program shall include the following elements:~~

~~a.~~ ~~The definition of the CRE and the CRE boundary.~~

~~b.~~ ~~Requirements for maintaining the CRE boundary in its design condition including configuration control and preventive maintenance.~~

(continued)

5.5 Procedures, Programs, and Manuals (continued)

~~5.5.2.16 Control Room Envelope Habitability Program (Continued)~~

~~c.~~ ~~Requirements for (i) determining the unfiltered air inleakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.~~

~~The following is an exception to Sections C.1 and C.2 of regulatory Guide 1.197, Revision 0.~~

~~Appropriate application of ASTM E-741 shall include the ability to take minor exceptions to the test methodology. These exceptions shall be documented in the test report.~~

~~d.~~ ~~Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by one train of the CREACUS, operating at the flow rate required by the VFTP, at a Frequency of 24 months on a STAGGERED TEST BASIS. The results shall be trended and used as part of the 24 month assessment of the CRE boundary.~~

~~e.~~ ~~The quantitative limits on unfiltered air inleakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by the testing described in paragraph c. The unfiltered air inleakage limit for radiological challenges is the inleakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air inleakage limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis.~~

~~f.~~ ~~The provisions of SR 3.0.2 are applicable to the Frequencies for assessing CRE habitability, determining CRE unfiltered inleakage, and measuring CRE pressure and assessing the CRE boundary as required by paragraphs c and d, respectively.~~

(continued)

5.5 Procedures, Programs, and Manuals (continued)

~~5.5.2.17 Battery Monitoring and Maintenance Program~~

~~This program provides for battery restoration and maintenance, which includes the following:~~

~~a. Actions to restore battery cells with float voltage < 2.13 V, and~~

~~b. Actions to verify that the remaining cells are above 2.07 V when a battery cell or cells have been found less than 2.13 V, and~~

~~c. Actions to equalize and test battery cells that had been discovered with electrolyte level below the top of the plates.~~

5.0 ADMINISTRATIVE CONTROLS

5.6 ~~Safety Function Determination Program (SFDP)~~ ← Deleted

~~5.6.1~~ ~~This program ensures loss of safety function is detected and appropriate actions taken. Upon failure to meet two or more LCOs at the same time, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate limitations and remedial or compensatory actions may be identified to be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6.~~

~~5.6.2~~ ~~The SFDP shall contain the following:~~

~~a.~~ ~~Provisions for cross-train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected.~~

~~b.~~ ~~Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists.~~

~~c.~~ ~~Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities.~~

~~d.~~ ~~Other appropriate limitations and remedial or compensatory actions.~~

~~5.6.3~~ ~~A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:~~

~~a.~~ ~~A required system redundant to system(s) supported by the inoperable support system is also inoperable (Case A); or~~

~~b.~~ ~~A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable (Case B); or~~

~~c.~~ ~~A required system redundant to support system(s) for the supported systems (a) and (b) above is also inoperable (Case C).~~

(continued)

~~5.6 Safety Function Determination Program (SFDP)~~

~~5.6.3~~
~~(continued)~~

~~Generic Example:~~

~~Train A~~		~~Train B~~	
~~System i~~		~~System i~~	~~- Case C~~
~~+~~		~~+~~	
~~System ii~~	~~- (Support System) Inoperable~~	~~System ii~~	
~~+~~		~~+~~	
~~System iii~~		~~System iii~~	~~- Case A~~
~~+~~		~~+~~	
~~System iv~~		~~System iv~~	~~- Case B~~

~~5.6.4~~ ~~The Safety Function Determination Program identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.~~

(continued)

5.0 ADMINISTRATIVE CONTROLS

5.7 Reporting Requirements

5.7.1 Routine Reports

In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted in accordance with 10 CFR 50.4. The reports shall be addressed to the U.S. Nuclear Regulatory Commission, Attention: Document Control Desk, Washington, D.C., with a copy to the Regional Administrator of the Regional Office of the NRC, unless otherwise noted.

5.7.1.1 ~~Annual Reports~~ Deleted

~~NOTE~~

~~A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station.~~

~~Annual Reports covering the activities of the unit as described below for the previous calendar year shall be submitted by March 31 of each year.~~

~~Reports required on an annual basis include:~~

~~a. (Deleted)~~

(continued)

5.7 Reporting Requirements (continued)

~~5.7.1.1 Annual Reports (continued)~~

~~b. Reactor Coolant System Specific Activity Report~~

~~Reports required on an annual basis shall include the results of specific activity analysis in which the primary coolant exceeded the limits of Specification 3.4.16. The following information shall be included in these reports:~~

~~1. Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded; and~~

~~2. Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while the limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than the limit. Each result should include date and time of sampling and the radioiodine concentrations; and~~

~~3. Cleanup system flow history starting 48 hours prior to the first sample in which the limit was exceeded; and~~

~~4. Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady-state level; and~~

~~5. The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.~~

5.7.1.2 Annual Radiological Environmental Operating Report

~~-----NOTE-----~~
~~A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station.~~
~~----------~~

facility

The Annual Radiological Environmental Operating Report covering the operation of the ~~unit~~ during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the

(continued)

5.7 Reporting Requirements (continued)

5.7.1.2 Annual Radiological Environmental Operating Report (continued)

objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. The report shall identify the thermoluminescent dosimeter (TLD) results that represent collocated dosimeters in relation to the NRC TLD program and the exposure period associated with each result. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.7.1.3 Radioactive Effluent Release Report

~~-----NOTE-----~~

~~A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.~~

~~----------~~

The Radioactive Effluent Release Report covering the operation of the ~~unit~~ facility during the previous calendar year shall be submitted before May 1 of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents released from the ~~unit~~ facility. The report shall also include a summary of the quantities of solid radioactive waste shipped from the ~~unit~~ facility directly to the disposal site and quantities of solid radioactive waste shipped from the ~~unit~~ facility's intermediary processor to the disposal site. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program (PCP) and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

(continued)

5.7 Reporting Requirements (continued)

~~5.7.1.4~~ ~~(Deleted)~~

~~5.7.1.5~~ ~~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.~~ ~~Specification 3.1.1, "SHUTDOWN MARGIN (SDM) - T_{avg} >200°F;"~~

~~2.~~ ~~Specification 3.1.2, "SHUTDOWN MARGIN (SDM) - T_{avg} ≤200°F;"~~

~~3.~~ ~~Specification 3.1.4, "Moderator Temperature Coefficient;"~~

~~4.~~ ~~Specification 3.1.5, "Control Element Assembly (CEA) Alignment;"~~

~~5.~~ ~~Specification 3.1.7, "Regulating CEA Insertion Limits;"~~

~~6.~~ ~~Specification 3.1.8, "Part Length Control Element Assembly Insertion Limits;"~~

~~7.~~ ~~Specification 3.2.1, "Linear Heat Rate;"~~

~~8.~~ ~~Specification 3.2.4, "Departure From Nucleate Boiling Ratio;"~~

~~9.~~ ~~Specification 3.2.5, "Axial Shape Index;"~~

~~10.~~ ~~Specification 3.4.1, "RCS DNB (Pressure, Temperature, and Flow) Limits;~~

~~11.~~ ~~Specification 3.9.1, "Boron Concentration."~~

~~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:~~

(continued)

5.7 Reporting Requirements (continued)

~~5.7.1.5 CORE OPERATING LIMITS REPORT (COLR) (continued)~~

~~1. CENPD-132P, "Calculative Methods for the C-E Large Break LOCA Evaluation Model"~~

~~2. CENPD-137P, "Calculative Methods for the C-E Small Break LOCA Evaluation Model"~~

~~3. CEN-356(V)-P-A, "Modified Statistical Combination of Uncertainties"~~

~~4. SCE-9801-P-A, "Reload Analysis Methodology for the San Onofre Nuclear Generating Station Units 2 and 3"~~

~~5. CEN-635(S), "Identification of NRC Safety Evaluation Report Limitations and/or Constraints on Reload Analysis Methodology"~~

~~6. Letter, dated May 16, 1986, G. W. Knighton (NRC) to K. P. Baskin (SCE), "Issuance of Amendment No. 47 to Facility Operating License NPF-10 and Amendment No. 36 to Facility Operating License NPF-15," San Onofre Nuclear Generating Station Units 2 and 3 (Cycle 3 SER)~~

~~7. Letter, dated January 9, 1985, G. W. Knighton (NRC) to K. P. Baskin, "Issuance of Amendment No. 30 to Facility Operating License NPF-10 and Amendment No. 19 to Facility Operating License NPF-15," San Onofre Nuclear Generating Station Units 2 and 3 (Cycle 2 SER)~~

~~8. "Implementation of ZIRLO™ Cladding Material in CE Nuclear Power Fuel Assembly Designs," CENPD-404-P-A~~

~~9. SCE-0901, "PWR Reactor Physics Methodology Using Studsvik Design Codes"~~

~~c. The core operating limits shall be determined assuming operation at RATED THERMAL POWER such that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal hydraulic limits, Emergency Core Cooling System (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.~~

~~5.7.1.6 REACTOR COOLANT SYSTEM (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)~~

~~a. RCS pressure and temperature limits for heatup, cooldown, low temperature operation, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:~~

(continued)

5.7 Reporting Requirements (continued)

~~5.7.1.6~~ ~~REACTOR COOLANT SYSTEM (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) (continued)~~

~~Technical Specification 3.4.3 RCS Pressure and Temperature (P/T) Limits,~~

~~Technical Specification 3.4.6 RCS Loops - MODE 4,~~

~~Technical Specification 3.4.7 RCS Loops - MODE 5, Loops Filled,~~

~~Technical Specification 3.4.12.1 Low Temperature Overpressure Protection (LTOP) System RCS Temperature ≤ PTLR Limit, and~~

~~Technical Specification 3.4.12.2 Low Temperature Overpressure Protection (LTOP) System RCS Temperature > PTLR Limit.~~

~~b.~~ ~~The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following document:~~

~~CE NPSD-683-A, The Development of a RCS Pressure and Temperature Limits Report for the Removal of P-T Limits and LTOP Setpoints from the Technical Specifications.~~

~~c.~~ ~~The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.~~

~~5.7.1.7~~ ~~Hazardous Cargo Traffic Report~~

~~Hazardous cargo traffic on Interstate 5 (I-5) and the AT&SF railway shall be monitored and the results submitted to the NRC Regional Administrator once every three years.~~

(continued)

5.7 Reporting Requirements (continued)

~~5.7.2~~ ~~Special Reports~~

~~Special Reports may be required covering inspection, test, and maintenance activities. These special reports are determined on an individual basis for each unit and their preparation and submittal are designated in the Technical Specifications.~~

~~Special Reports shall be submitted to the U. S. Nuclear Regulatory Commission, Attention: Document Control Desk, Washington, D. C. 20555, with a copy to the Regional Administrator of the Regional Office of the NRC, in accordance with 10 CFR 50.4 within the time period specified for each report.~~

~~The following Special Reports shall be submitted:~~

~~a.~~ ~~When a pre-planned alternate method of monitoring post-accident instrumentation functions is required by Condition B or Condition G of LCO 3.3.11, a report shall be submitted within 30 days from the time the action is required. The report shall outline the action taken, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the function to OPERABLE status.~~

~~b.~~ ~~Any abnormal degradation of the containment structure detected during the tests required by the Pre-Stressed Concrete Containment Tendon Surveillance Program shall be reported to the NRC within 30 days. The report shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, and the corrective action taken.~~

~~c.~~ ~~A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.2.11, Steam Generator (SG) Program. The report shall include:~~

(continued)

5.7 Reporting Requirements (continued)

~~5.7.2~~ ~~Special Reports (continued)~~

~~1. The scope of inspections performed on each SG,~~

~~2. Active degradation mechanisms found,~~

~~3. Nondestructive examination techniques utilized for each degradation mechanism,~~

~~4. Location, orientation (if linear), and measured sizes (if available) of service induced indications,~~

~~5. Number of tubes plugged during the inspection outage for each active degradation mechanism,~~

~~6. Total number and percentage of tubes plugged to date,~~

~~7. The results of condition monitoring, including the results of tube pulls and in-situ testing.~~

5.0 ADMINISTRATIVE CONTROLS

5.8 High Radiation Area

5.8.1 Each high radiation area as defined in 10 CFR 20 shall be barricaded and conspicuously posted as a high radiation area, and entrance thereto shall be controlled by requiring issuance of a Radiation Exposure Permit (REP).

Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device that continuously indicates the radiation dose rate in the area,

b. A radiation monitoring device that continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rates in the area have been determined and personnel have been made knowledgeable of them,

c. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device. This individual is responsible for providing positive radiation protection control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified in the radiation protection procedures or the applicable REP.

(continued)

5.8. High Radiation Area (continued)

5.8.2 In addition, areas that are accessible to personnel and that have radiation levels greater than 1.0 rem (but less than 500 rads at 1 meter) in 1 hour at 30 cm from the radiation source, or from any surface penetrated by the radiation, shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the shift manager on duty or radiation protection supervisor. Doors shall remain locked except during periods of access by personnel under an approved REP that specifies the dose rates in the immediate work areas and the maximum allowable stay time for individuals in that area. In lieu of a stay time specification on the REP, direct or remote continuous surveillance (such as closed circuit TV cameras) may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area.

5.8.3 Individual high radiation areas that are accessible to personnel, that could result in radiation doses greater than 1.0 rem in 1 hour, and that are within large areas where no enclosure exists to enable locking and where no enclosure can be reasonably constructed around the individual area shall be barricaded and conspicuously posted. A flashing light shall be activated as a warning device whenever the dose rate in such an area exceeds or is expected to exceed 1.0 rem in 1 hour at 30 cm from the radiation source or from any surface penetrated by the radiation.

Attachment 3

Proposed Technical Specifications Bases Markup Pages, Unit 2 (For Information Only)

B 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

BASES

LCOs

and LCO 3.0.2

LCO 3.0.1 ~~through LCO 3.0.7~~ establish the general requirements applicable to all Specifications and apply at all times unless otherwise stated.

LCO 3.0.1

LCO 3.0.1 establishes the Applicability statement within each individual Specification as the requirement for when the LCO is required to be met (i.e., when the ~~unit~~ facility is in the ~~MODES or other~~ specified conditions of the Applicability statement of each Specification).

LCO 3.0.2

LCO 3.0.2 establishes that upon discovery of a failure to meet an LCO, the associated ACTIONS shall be met. The Completion Time of each Required Action for an ACTIONS Condition is applicable from the point in time that an ACTIONS Condition is entered. The Required Actions establish those remedial measures that must be taken within specified Completion Times when the requirements of an LCO are not met. This Specification establishes that:

a. Completion of the Required Actions within the specified Completion Times constitutes compliance with a Specification; and

b. Completion of the Required Actions is not required when an LCO is met within the specified Completion Time, unless otherwise specified.

~~There are two basic types of Required Actions. The first type of Required Action specifies a time limit in which the LCO must be met. This time limit is the Completion Time to restore an inoperable system or component to OPERABLE status or to restore variables to within specified limits. If this type of Required Action is not completed within the specified Completion Time, a shutdown may be required to place the unit in a MODE or condition in which the Specification is not applicable. (Whether stated as a Required Action or not, correction of the entered Condition is an action that may always be considered upon entering~~

(continued)

BASES (continued)

LCO 3.0.2 (continued)

~~ACTIONS.) The second type of Required Action specifies the remedial measures that permit continued operation of the unit that is not further restricted by the Completion Time. In this case, compliance with the Required Actions provides an acceptable level of safety for continued operation.~~

Completing the Required Actions is not required when an LCO is met or is no longer applicable, unless otherwise stated in the individual Specifications.

~~The nature of some Required Actions of some Conditions necessitates that, once the Condition is entered, the Required Actions must be completed even though the associated Conditions no longer exist. The individual LCO's ACTIONS specify the Required Actions where this is the case. An example of this is in LCO 3.8.1, "AC Sources—Operating."~~

The Completion Times of the Required Actions are also applicable when ~~a system or component is removed from service~~ a specified Condition in the Applicability is entered intentionally. The reasons for intentionally relying on the ACTIONS include, but are not limited to, performance of Surveillances, preventive maintenance, corrective maintenance, or investigation of ~~operational~~ problems. Entering ACTIONS for these reasons must be done in a manner that does not compromise ~~safety~~ the safe storage of irradiated fuel. Intentional entry into ACTIONS should not be made for ~~operational~~ convenience. ~~Alternatives that would not result in redundant equipment being inoperable should be used instead. Doing so limits the time both subsystems/trains of a safety function are inoperable and limits the time other conditions exist which result in LCO 3.0.3 being entered. Individual Specifications may specify a time limit for performing an SR when equipment is removed from service or bypassed for testing. In this case, the Completion Times of the Required Actions are applicable when this time limit expires, if the equipment remains removed from service or bypassed.~~

~~When a change in MODE or other specified condition is required to comply with Required Actions, the unit may enter a MODE or other specified condition in which another Specification becomes applicable. In this case, the Completion Times of the associated Required Actions would apply from the point in time that the new Specification becomes applicable and the ACTIONS Condition(s) are entered.~~

(continued)

BASES (continued)

~~LCO 3.0.3~~

~~LCO 3.0.3 establishes the actions that must be implemented when an LCO is not met and:~~

~~a.~~ ~~An associated Required Action and Completion Time is not met and no other Condition applies; or~~

~~b.~~ ~~The condition of the unit is not specifically addressed by the associated ACTIONS. This means that no combination of Conditions stated in the ACTIONS can be made that exactly corresponds to the actual condition of the unit. Sometimes, possible combinations of Conditions are such that entering LCO 3.0.3 is warranted; in such cases, the ACTIONS specifically state a Condition corresponding to such combinations and also that LCO 3.0.3 be entered immediately.~~

~~This Specification delineates the time limits for placing the unit in a safe MODE or other specified condition when operation cannot be maintained within the limits for safe operation as defined by the LCO and its ACTIONS. It is not intended to be used as an operational convenience that permits routine voluntary removal of redundant systems or components from service in lieu of other alternatives that would not result in redundant systems or components being inoperable.~~

~~Upon entering LCO 3.0.3, 1 hour is allowed to prepare for an orderly shutdown before initiating a change in unit operation. This includes time to permit the operator to coordinate the reduction in electrical generation with the load dispatcher to ensure the stability and availability of the electrical grid. The time limits specified to reach lower MODES of operation permit the shutdown to proceed in a controlled and orderly manner that is well within the specified maximum cooldown rate and within the capabilities of the unit, assuming that only the minimum required equipment is OPERABLE. This reduces thermal stresses on components of the Reactor Coolant System and the potential for a plant upset that could challenge safety systems under conditions to which this Specification applies. The use and interpretation of specified times to complete the actions of LCO 3.0.3 are consistent with the discussion of Section 1.3, Completion Times.~~

(continued)

BASES (continued)

~~LCO 3.0.3~~
~~(continued)~~

~~Voluntary entry into LCO 3.0.3 is permissible but requires prior approval (approval may be verbal) from either the Operations Manager, Station Manager or corporate officer with direct responsibility for the plant. The approval must subsequently be documented in written retrievable manner. Inadvertent entry still allows for the one hour preparation period before Actions to change MODES must begin.~~

~~A unit shutdown required in accordance with LCO 3.0.3 may be terminated and LCO 3.0.3 exited if any of the following occurs:~~

~~a. The LCO is now met.~~

~~b. A Condition exists for which the Required Actions have now been performed.~~

~~c. ACTIONS exist that do not have expired Completion Times. These Completion Times are applicable from the point in time that the Condition is initially entered and not from the time LCO 3.0.3 is exited.~~

~~The time limits of Specification 3.0.3 allow 37 hours for the unit to be in MODE 5 when a shutdown is required during MODE 1 operation. If the unit is in a lower MODE of operation when a shutdown is required, the time limit for reaching the next lower MODE applies. If a lower MODE is reached in less time than allowed, however, the total allowable time to reach MODE 5, or other applicable MODE, is not reduced. For example, if MODE 3 is reached in 2 hours, then the time allowed for reaching MODE 4 is the next 11 hours, because the total time for reaching MODE 4 is not reduced from the allowable limit of 13 hours. Therefore, if remedial measures are completed that would permit a return to MODE 1, a penalty is not incurred by having to reach a lower MODE of operation in less than the total time allowed.~~

~~In MODES 1, 2, 3, and 4, LCO 3.0.3 provides actions for Conditions not covered in other Specifications. The requirements of LCO 3.0.3 do not apply in MODES 5 and 6 because the unit is already in the most restrictive Condition required by LCO 3.0.3.~~

~~The requirements of LCO 3.0.3 do not apply in other specified conditions of the Applicability (unless in MODE 1, 2, 3, or 4) because the ACTIONS of individual Specifications~~

(continued)

BASES (continued)

~~LCO 3.0.3 (continued)~~

~~sufficiently define the remedial measures to be taken. Exceptions to LCO 3.0.3 are provided in instances where requiring a unit shutdown, in accordance with LCO 3.0.3, would not provide appropriate remedial measures for the associated condition of the unit. An example of this is in LCO 3.7.16, "Fuel Storage Pool Water Level." LCO 3.7.16 has an Applicability of "During movement of irradiated fuel assemblies in the fuel storage pool." Therefore, this LCO can be applicable in any or all MODES. If the LCO and the Required Actions of LCO 3.7.16 are not met while in MODE 1, 2, or 3, there is no safety benefit to be gained by placing the unit in a shutdown condition. The Required Action of LCO 3.7.16 of "Suspend movement of irradiated fuel assemblies in fuel storage pool" is the appropriate Required Action to complete in lieu of the actions of LCO 3.0.3. These exceptions are addressed in the individual Specifications.~~

~~LCO 3.0.4~~

~~LCO 3.0.4 establishes limitations on changes in MODES or other specified conditions in the Applicability when an LCO is not met. It precludes placing the unit in a different MODE or other specified condition when the following exist:~~

~~a.~~ ~~The requirements of an LCO, in the MODE or other specified condition to be entered, are not met; and~~

~~b.~~ ~~Continued noncompliance with these LCO requirements would result in the unit being required to be placed in a MODE or other specified condition in which the LCO does not apply to comply with the Required Actions.~~

~~Compliance with Required Actions that permit continued operation of the unit for an unlimited period of time in a MODE or other specified condition provides an acceptable level of safety for continued operation. This is without regard to the status of the unit before or after the MODE change. Therefore, in such cases, entry into a MODE or other specified condition in the Applicability may be made in accordance with the provisions of the Required Actions. The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before unit startup.~~

(continued)

BASES (continued)

~~LCO 3.0.4 (continued)~~

~~The provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from a normal shutdown.~~

~~Exceptions to LCO 3.0.4 are stated in the individual Specifications. Exceptions may apply to all the ACTIONS or to a specific Required Action of a Specification.~~

~~Surveillances do not have to be performed on the associated inoperable equipment (or on variables outside the specified limits), as permitted by SR 3.0.1. Therefore, changing MODES or other specified conditions while in an ACTIONS Condition, in compliance with LCO 3.0.4 or where an exception to LCO 3.0.4 is stated, is not a violation of SR 3.0.1 or SR 3.0.4 for those Surveillances that do not have to be performed due to the associated inoperable equipment. However, SRs must be met to ensure OPERABILITY prior to declaring the associated equipment OPERABLE (or variable within limits) and restoring compliance with the affected LCO.~~

~~LCO 3.0.5~~

~~LCO 3.0.5 establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service or declared inoperable to comply with ACTIONS. The sole purpose of this Specification is to provide an exception to LCO 3.0.2 (e.g., to not comply with the applicable Required Action(s)) to allow the performance of SRs to demonstrate:~~

~~a.~~ ~~The OPERABILITY of the equipment being returned to service; or~~

~~b.~~ ~~The OPERABILITY of other equipment.~~

~~The administrative controls ensure the time the equipment is returned to service in conflict with the requirements of the ACTIONS is limited to the time absolutely necessary to perform the allowed SRs. This Specification does not provide time to perform any other preventive or corrective maintenance.~~

(continued)

BASES (continued)

~~LCO 3.0.5 (continued)~~

~~An example of demonstrating the OPERABILITY of the equipment being returned to service is reopening a containment isolation valve that has been closed to comply with Required Actions and must be reopened to perform the SRs.~~

~~An example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to prevent the trip function from occurring during the performance of an SR on another channel in the other trip system. A similar example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to permit the logic to function and indicate the appropriate response during the performance of an SR on another channel in the same trip system.~~

~~LCO 3.0.6~~

~~LCO 3.0.6 establishes an exception to LCO 3.0.2 for support systems that have an LCO specified in the Technical Specifications (TS). This exception is provided because LCO 3.0.2 would require that the Conditions and Required Actions of the associated inoperable supported system LCO be entered solely due to the inoperability of the support system. This exception is justified because the actions that are required to ensure the unit is maintained in a safe condition are specified in the support system LCO's Required Actions. These Required Actions may include entering the supported system's Conditions and Required Actions or may specify other Required Actions.~~

~~When a support system is inoperable and there is an LCO specified for it in the TS, the supported system(s) are required to be declared inoperable if determined to be inoperable as a result of the support system inoperability. However, it is not necessary to enter into the supported systems' Conditions and Required Actions unless directed to do so by the support system's Required Actions. The potential confusion and inconsistency of requirements related to the entry into multiple support and supported systems' LCOs' Conditions and Required Actions are eliminated by providing all the actions that are necessary to ensure the unit is maintained in a safe condition in the support system's Required Actions.~~

(continued)

BASES (continued)

~~LCO 3.0.6 (continued)~~

~~However, there are instances where a support system's Required Action may either direct a supported system to be declared inoperable or direct entry into Conditions and Required Actions for the supported system. This may occur immediately or after some specified delay to perform some other Required Action. Regardless of whether it is immediate or after some delay, when a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.~~

~~Specification 5.6, "Safety Function Determination Program (SFDP)," ensures loss of safety function is detected and appropriate actions are taken. Upon failure to meet two or more LCOs concurrently, an evaluation shall be made to determine if loss of safety function exists. Additionally, other limitations, remedial actions, or compensatory actions may be identified as a result of the support system inoperability and corresponding exception to entering supported system Conditions and Required Actions. The SFDP implements the requirements of LCO 3.0.6.~~

~~Cross train checks to identify a loss of safety function for those support systems that support multiple and redundant safety systems are required. The cross train check verifies that the supported systems of the redundant OPERABLE support system are OPERABLE, thereby ensuring safety function is retained. If this evaluation determines that a loss of safety function exists, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.~~

~~LCO 3.0.7~~

~~Special tests and operations are required at various times over the unit's life to demonstrate performance characteristics, to perform maintenance activities, and to perform special evaluations. Because TS normally preclude these tests and operations, special test exceptions (STEs) allow specified requirements to be changed or suspended under controlled conditions. STEs are included in applicable sections of the Specifications. Unless otherwise specified, all other TS requirements remain unchanged and in~~

(continued)

BASES (continued)

~~LCO 3.0.7 (continued)~~

~~effect as applicable. This will ensure that all appropriate requirements of the MODE or other specified condition not directly associated with or required to be changed or suspended to perform the special test or operation will remain in effect.~~

~~The Applicability of an STE LCO represents a condition not necessarily in compliance with the normal requirements of the TS. Compliance with STE LCOs is optional.~~

~~A special test may be performed under either the provisions of the appropriate STE LCO or the other applicable TS requirements. If it is desired to perform the special test under the provisions of the STE LCO, the requirements of the STE LCO shall be followed. This includes the SRs specified in the STE LCO.~~

~~Some of the STE LCOs require that one or more of the LCOs for normal operation be met (i.e., meeting the STE LCO requires meeting the specified normal LCOs). The Applicability, ACTIONS, and SRs of the specified normal LCOs, however, are not required to be met in order to meet the STE LCO when it is in effect. This means that, upon failure to meet a specified normal LCO, the associated ACTIONS of the STE LCO apply, in lieu of the ACTIONS of the normal LCO. Exceptions to the above do exist. There are instances when the Applicability of the specified normal LCO must be met, where its ACTIONS must be taken, where certain of its Surveillances must be performed, or where all of these requirements must be met concurrently with the requirements of the STE LCO.~~

~~Unless the SRs of the specified normal LCOs are suspended or changed by the special test, those SRs that are necessary to meet the specified normal LCOs must be met prior to performing the special test. During the conduct of the special test, those Surveillances need not be performed unless specified by the ACTIONS or SRs of the STE LCO.~~

~~ACTIONS for STE LCOs provide appropriate remedial measures upon failure to meet the STE LCO. Upon failure to meet these ACTIONS, suspend the performance of the special test and enter the ACTIONS for all LCOs that are then not met. Entry into LCO 3.0.3 may possibly be required, but this determination should not be made by considering only the failure to meet the ACTIONS of the STE LCO.~~

B 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

BASES

SRs

SR 3.0.1 through SR 3.0.4 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.

SR 3.0.1

SR 3.0.1 establishes the requirement that SRs must be met during the ~~MODES or other~~ specified conditions in the Applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This Specification is to ensure that Surveillances are performed in order to verify the ~~OPERABILITY of systems and components, and that variables~~ facility conditions are within specified limits. Failure to meet a Surveillance within the specified Frequency, in accordance with SR 3.0.2, constitutes a failure to meet an LCO.

~~Systems and components are assumed to be OPERABLE when the associated SRs have been met. Nothing in this Specification, however, is to be construed as implying that systems or components are OPERABLE when:~~

~~a. The systems or components are known to be inoperable, although still meeting the SRs; or~~

~~b. The requirements of the Surveillance(s) are known to be not met between required Surveillance performances.~~

Surveillances do not have to be performed when the ~~unit~~ facility is in a ~~MODE or other~~ specified condition for which the requirements of the associated LCO are not applicable, unless otherwise specified. ~~The SRs associated with a special test exception (STE) are only applicable when the STE is used as an allowable exception to the requirements of a Specification.~~

~~Surveillances, including Surveillances invoked by Required Actions, do not have to be performed on inoperable equipment because the ACTIONS define the remedial measures that apply. Surveillances have to be met and performed in accordance with SR 3.0.2, prior to returning equipment to OPERABLE status.~~

(continued)

BASES (continued)

~~SR 3.0.1~~
~~(continued)~~

~~Upon completion of maintenance, appropriate post maintenance testing is required to declare equipment OPERABLE. This includes ensuring applicable Surveillances are not failed and their most recent performance is in accordance with SR 3.0.2. Post maintenance testing may not be possible in the current MODE or other specified conditions in the Applicability due to the necessary unit parameters not having been established. In these situations, the equipment may be considered OPERABLE provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function. This will allow operation to proceed to a MODE or other specified condition where other necessary post maintenance tests can be completed.~~

~~Some examples of this process are:~~

~~a.~~ ~~Auxiliary feedwater (AFW) pump turbine maintenance during refueling that requires testing at steam pressures > 800 psi. However, if other appropriate testing is satisfactorily completed, the AFW System can be considered OPERABLE. This allows startup and other necessary testing to proceed until the plant reaches the steam pressure required to perform the testing.~~

~~b.~~ ~~High pressure safety injection (HPSI) maintenance during shutdown that requires system functional tests at a specified pressure. Provided other appropriate testing is satisfactorily completed, startup can proceed with HPSI considered OPERABLE. This allows operation to reach the specified pressure to complete the necessary post maintenance testing.~~

SR 3.0.2

~~SR 3.0.2 establishes the requirements for meeting the specified Frequency for Surveillances and any Required Action with a Completion Time that requires the periodic performance of the Required Action on a "once per..." interval.~~

SR 3.0.2 permits a 25% extension of the interval specified in the Frequency. This extension facilitates Surveillance scheduling and considers ~~plant operating~~ facility conditions that may

(continued)

BASES (continued)

SR 3.0.2 (continued)

not be suitable for conducting the Surveillance (e.g., ~~transient conditions or~~ other ongoing Surveillance or maintenance activities).

The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs. ~~The exceptions to SR 3.0.2 are those Surveillances for which the 25% extension of the interval specified in the Frequency does not apply. These exceptions are stated in the individual Specifications. An example of where SR 3.0.2 does not apply is the Containment Leakage Rate Testing Program. Test frequencies specified in the Containment Leakage Rate Testing Program may be extended consistent with the guidance provided in NEI 94-01," Industry Guideline For Implementing Performance-Based Option Of 10CFR 50, Appendix J," as endorsed by Regulatory Guide 1.163.~~

~~As stated in SR 3.0.2, the 25% extension also does not apply to the initial portion of a periodic Completion Time that requires performance on a "once per..." basis. The 25% extension applies to each performance after the initial performance. The initial performance of the Required Action, whether it is a particular Surveillance or some other remedial action, is considered a single action with a single Completion Time. One reason for not allowing the 25% extension to this Completion Time is that such an action usually verifies that no loss of function has occurred by checking the status of redundant or diverse components or accomplishes the function of the inoperable equipment in an alternative manner.~~

The provisions of SR 3.0.2 are not intended to be used repeatedly merely as a~~n~~ ~~operational~~ convenience to extend Surveillance intervals or periodic Completion Time intervals beyond those specified.

SR 3.0.3

SR 3.0.3 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a Surveillance has not

(continued)

BASES (continued)

SR 3.0.3 (continued)

been completed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified frequency , whichever is greater, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met.

This delay period provides an adequate time to complete Surveillances that have been missed. This delay period permits the completion of a Surveillance before complying with Required Actions or other remedial measures that might preclude completion of the Surveillance.

The basis for this delay period includes consideration of facility ~~unit~~ conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements.

When a Surveillance with a Frequency based not on time intervals, but upon specified facility ~~unit~~ conditions~~, operating situations, or requirements of regulations (e.g., prior to entering MODE 1 after each fuel loading, or in accordance with 10CFR50, Appendix J, as modified by approved exemptions, etc.)~~ is discovered to not have been performed when specified, SR 3.0.3 allows for the full delay period of up to the specified Frequency to perform the Surveillance. However, since there is not a time interval specified, the missed Surveillance should be performed at the first reasonable opportunity.

~~SR 3.0.3 provides a time limit for, and allowances for the performance of, Surveillances that become applicable as a consequence of MODE changes imposed by Required Actions.~~

(continued)

BASES (continued)

SR 3.0.3 (continued)

Failure to comply with specified Frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 3.0.3 is a flexibility which is not intended to be used as a~~n~~ ~~operational~~ convenience to extend Surveillance intervals. While up to 24 hours or the limit of the specified Frequency is provided to perform the missed Surveillance, it is expected that the missed Surveillance will be performed at the first reasonable opportunity. The determination of the first reasonable opportunity should include consideration of the impact on ~~plant~~ risk (from delaying the Surveillances ~~as well as any plant configuration changes required or shutting the plant down to perform the Surveillance~~) and impact on any analysis assumptions, in addition to facility ~~unit~~ conditions, planning, availability of personnel, and the time required to perform the Surveillance. This risk impact should be managed through the program in place to implement 10CFR50.65(a)(4) and its implementation guidance, NRC Regulatory Guide 1.182, 'Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants.' This Regulatory Guide addresses consideration of temporary and aggregate risk impacts, determination of risk management action thresholds, and risk management action ~~up to and including plant shutdown~~. The missed Surveillance should be treated as an emergent condition as discussed in the Regulatory Guide. The risk evaluation may use quantitative, qualitative, or blended methods. The degree of depth and rigor of the evaluation should be commensurate with the importance of the component. Missed Surveillances for important components should be analyzed quantitatively. If the results of the risk evaluation determine the risk increase is significant, this evaluation should be used to determine the safest course of action. All missed Surveillances will be placed in the licensee's Corrective Action Program.

If a Surveillance is not completed within the allowed delay period, then the equipment is considered inoperable or the variable is considered outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then the equipment is inoperable, or the variable is outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon the failure of the Surveillance.

(continued)

BASES (continued)

SR 3.0.3 (continued)

Completion of the Surveillance within the delay period allowed by this Specification, or within the Completion Time of the ACTIONS, restores compliance with SR 3.0.1.

SR 3.0.4

SR 3.0.4 establishes the requirement that all applicable SRs must be met before entry into a ~~MODE or other~~ specified Condition in the Applicability.

This Specification ensures that system and component OPERABILITY requirements and variable limits are met before entry into ~~MODES or other~~ specified conditions in the Applicability for which these systems and components ensure safe operation of the facility ~~unit~~. ~~This Specification applies to changes in MODES or other specified conditions in the Applicability associated with unit shutdown as well as startup.~~

The provisions of SR 3.0.4 shall not prevent changes in ~~MODES or other~~ specified conditions in the Applicability that are required to comply with ACTIONS.

The precise requirements for performance of SRs are specified such that exceptions to SR 3.0.4 are not necessary. ~~The specific time frames and conditions necessary for meeting the SRs are specified in the Frequency, in the Surveillance, or both. This allows performance of Surveillances when the prerequisite condition(s) specified in a Surveillance procedure require entry into the MODE or other specified condition in the Applicability of the associated LCO prior to the performance or completion of a Surveillance. A Surveillance that could not be performed until after entering the LCO Applicability, would have its Frequency specified such that it is not "due" until the specific conditions needed are met. Alternately, the Surveillance may be stated in the form of a Note as not required (to be met or performed) until a particular event, condition, or time has been reached. Further discussion of the specific formats of SRs' annotation is found in Section 1.4, Frequency.~~

3.1

B ~~3.7~~ PLANT SYSTEMS

3.1.1

B ~~3.7.16~~ Fuel Storage Pool Water Level

BASES

BACKGROUND

The minimum water level in the fuel storage pool meets the assumptions of iodine decontamination factors following a fuel handling accident. The specified water level shields and minimizes the general area dose when the storage racks are filled to their maximum capacity. The water also provides shielding during the movement of spent fuel.

A general description of the fuel storage pool design is given in the UFSAR, Section 9.1.2, Reference 1, and the Spent Fuel Pool Cooling and Cleanup System is given in the UFSAR, Section 9.1.3 (Ref. 2). The assumptions of the fuel handling accident are given in the UFSAR, Section 15.7.3.4 ~~and 15.7.3.6~~ (Ref. 3 ~~and Ref. 6~~).

APPLICABLE SAFETY ANALYSES

The minimum water level in the fuel storage pool meets the assumptions of the fuel handling accident described in Regulatory Guide ~~1.25~~ 1.183 (Ref. 4). The resultant ~~2 hour thyroid~~ dose to a person at the exclusion area boundary or low population zone is a small fraction of the 10 CFR ~~100~~ 50.67 (Ref. 5) limits.

According to Reference 4, there is 23 ft of water between the top of the damaged fuel bundle and the fuel pool surface for a fuel handling accident. With this 23 ft of water, the assumptions of Reference 4 can be used directly. In practice, this LCO preserves this assumption for the bulk of the fuel in the storage racks. In the case of a single bundle, dropped and lying horizontally on top of the spent fuel racks, there would be < 23 ft of water above the top of the bundle.

~~However, when the potential of a dropped fuel assembly exists (which is when fuel is being moved) a water level is maintained that would ensure that there would be >23 feet above the fuel assembly laying on top of the racks. This increased water level is required by LCO 3.9.6 when the fuel storage pool is connected to the refueling cavity and by station procedures whenever fuel is being moved.~~

(continued)

APPLICABLE SAFETY ANALYSES (continued)

The fuel storage pool water level satisfies Criterion 3 of the NRC Policy Statement.

LCO

The specified water level preserves the assumptions of the fuel handling accident analysis (Ref. 3). As such, it is the minimum required for fuel storage and movement within the fuel storage pool.

APPLICABILITY

This LCO applies during movement of fuel assemblies (i.e., irradiated fuel, non-irradiated fuel, and the dummy fuel assembly) in the fuel storage pool since the potential for a release of fission products exists.

ACTIONS

A.1

~~Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply.~~

When the initial conditions for an accident cannot be met, steps should be taken to preclude the accident from occurring. When the fuel storage pool water level is lower than the required level, the movement of fuel assemblies in the fuel storage pool is immediately suspended. This effectively precludes a spent fuel handling accident from occurring. This does not preclude moving a fuel assembly to a safe position.

~~If moving fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving fuel assemblies while in MODES 1, 2, 3, and 4, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown.~~

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR ~~3.7.16.1~~ 3.1.1.1

This SR verifies sufficient fuel storage pool water is available in the event of a fuel handling accident. The water level in the fuel storage pool must be checked periodically. The 7 day Frequency is appropriate because the volume in the pool is normally stable. Water level changes are controlled by unit procedures and are acceptable, based on operating experience.

~~During refueling operations, the level in the fuel storage pool is at equilibrium with that of the refueling canal, and the level in the refueling canal is checked daily in accordance with LCO 3.9.6, "Refueling Water Level."~~

REFERENCES

1. UFSAR, Section 9.1.2.

2. UFSAR, Section 9.1.3.

3. UFSAR, Section 15.7.3.4.

4. Regulatory Guide ~~1.25~~ 1.183

5. 10 CFR ~~100.11~~ 50.67.

~~6. UFSAR, Section 15.7.3.6~~

B ~~3.7~~ 3.1 PLANT SYSTEMS

B ~~3.7.17~~ 3.1.2 Fuel Storage Pool Boron Concentration

BASES

BACKGROUND	As described in LCO ~~3.7.18~~ 3.1.3, "Spent Fuel Assembly Storage," fuel assemblies are stored in the spent fuel racks in accordance with criteria based on initial enrichment, discharge burnup, and cooling time (plutonium decay). Although the water in the spent fuel pool is normally borated to ≥ 2000 ppm, the criteria that limit the storage of a fuel assembly to specific rack locations is conservatively developed without taking credit for boron while maintaining K_{eff} < 1.0. Credit for boron is taken to maintain K_{eff} ≤ 0.95.
APPLICABLE SAFETY ANALYSES	Soluble boron in the spent fuel pool is credited in criticality analyses for normal and accident conditions. The relevant accidents are 1) Fuel Assembly Dropped Horizontally On Top of the Racks, 2) Fuel Assembly Dropped Vertically Into a Storage Location Already Containing a Fuel Assembly, 3) Fuel Assembly Dropped to the SFP Floor, and 4) Fuel Misloading in either Region I or Region II. The limiting accident is Fuel Misloading in either Region I or Region II. A fuel assembly could be inadvertently loaded into a spent fuel rack location not allowed by LCO ~~3.7.18~~ 3.1.3 (e.g., an un-irradiated fuel assembly or an insufficiently depleted fuel assembly). This accident is analyzed assuming the misloading of one fresh assembly with the maximum permissible enrichment. However, the negative reactivity effect of the soluble boron compensates for the increased reactivity caused by the postulated accident scenario. Under normal, non-accident conditions, the soluble boron needed to maintain K_{eff} less than or equal to 0.95, including uncertainties, is 970 ppm. Under accident conditions, the soluble boron needed to maintain K_{eff} less than or equal to 0.95, including uncertainties, is 1700 ppm. A SFP boron dilution analysis shows that dilution from 2000 ppm to below 1700 is not credible. Therefore, the minimum required soluble boron concentration is 2000 ppm. The concentration of dissolved boron in the fuel pool satisfies Criterion 2 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO

The specified concentration of 2000 ppm dissolved boron in the fuel pool preserves the assumptions used in the analyses described above. This concentration of dissolved boron is the minimum required concentration for fuel assembly storage and movement within the spent fuel pool.

APPLICABILITY

This LCO applies whenever fuel assemblies are stored in the spent fuel pool.

ACTIONS

A.1 and A.2

~~The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply.~~

When the concentration of boron in the spent fuel pool is less than required 2000 ppm, immediate action must be taken to preclude an accident from happening or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies. This does not preclude the movement of fuel assemblies to a safe position. In addition, action must be immediately initiated to restore boron concentration to the required 2000 ppm.

~~If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown.~~

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR ~~3.7.17.1~~ 3.1.2.1

This SR verifies that the concentration of boron in the spent fuel pool is within the required limit. As long as this SR is met, the analyzed incidents are fully addressed. The 7 day Frequency is appropriate because no major replenishment of pool water is expected to take place over a short period of time.

REFERENCES

1. UFSAR, Section 9.1.

B ~~3.7~~ 3.1 PLANT SYSTEMS

B ~~3.7.18~~ 3.1.3 Spent Fuel Assembly Storage

BASES

BACKGROUND

The spent fuel storage facility is designed to store either new (nonirradiated) nuclear fuel assemblies, or burned (irradiated) fuel assemblies in a vertical configuration underwater. The storage pool is sized to store 1542 fuel assemblies. Two types/sizes of spent fuel storage racks are used (Region I and Region II). The two Region I racks each contain 156 storage locations each spaced 10.40 inches on center in a 12x13 array. Four Region II storage racks each contain 210 storage locations in a 14x15 array. The remaining two Region II racks each contain 195 locations in a 13x15 array. All Region II locations are spaced 8.85 inches on center.

To maintain $K_{eff} \leq 0.95$ for spent fuel of maximum enrichment up to 4.8 w/o, (1) soluble boron is credited, and (2) the following storage patterns and borated stainless steel guide tube inserts are used as needed:

(1) unrestricted storage, minimum discharge burnup and cooling time requirements vs. initial enrichment,

(2) SFP Peripheral storage, minimum discharge burnup and cooling time requirements vs. initial enrichment,

(3) 2x2 storage patterns, minimum discharge burnup and cooling time requirements vs. initial enrichment,

(4) 3x3 storage patterns, minimum discharge burnup and cooling time requirements vs. initial enrichment,

(5) credit for inserted Control Element Assemblies (CEAs),

(6) credit for erbia in fresh assemblies,

(7) credit for cooling time (Pu-241 decay), and,

(8) credit for borated stainless steel guide tube inserts.

(continued)

BACKGROUND (continued)

When soluble boron is credited, the following acceptance criteria apply:

(1) Under normal conditions, the 95/95 neutron multiplication factor (K_{eff}), including all uncertainties, shall be less than 1.0 when flooded with unborated water, and,

(2) Under normal and accident conditions, the 95/95 neutron multiplication factor (K_{eff}), including all uncertainties, shall be less than or equal to 0.95 when flooded with borated water.

APPLICABLE SAFETY ANALYSES

The spent fuel storage facility is designed for noncriticality by use of adequate spacing, neutron absorbing stainless steel cans, borated water with a minimum soluble boron concentration of 970 ppm, and storage of fuel assemblies in accordance with the administrative controls in LCO ~~3.7.18~~ 3.1.3 and LCS 4.0.100, "Fuel Storage Patterns".

The spent fuel assembly storage satisfies Criterion 2 of the NRC Policy Statement.

LCO

The restrictions on the placement of fuel assemblies within the spent fuel pool, in the accompanying LCO, ensure that the K_{eff} of the spent fuel pool will always remain < 1.00 under normal, non-accident conditions assuming the pool to be flooded with unborated water. The K_{eff} of the spent fuel pool will always remain ≤ 0.95 under normal, non-accident conditions assuming the pool to be flooded with borated water with a minimum soluble boron concentration of 970 ppm. The K_{eff} of the spent fuel pool will always remain ≤ 0.95 under accident conditions assuming the pool to be flooded with borated water with a minimum soluble boron concentration of 1700 ppm. The restrictions are consistent with the criticality safety analysis performed for the spent fuel pool.

(continued)

APPLICABILITY

This LCO applies whenever any fuel assembly is stored in Regions I and II of the spent fuel pool.

ACTIONS

A.1

~~Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply.~~

When the configuration of fuel assemblies stored in Regions I and II of the spent fuel pool is not in accordance with LCO ~~3.7.18~~ 3.1.3, immediate action must be taken to make the necessary fuel assembly movement(s) to bring the configuration into compliance.

~~If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, in either case, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.~~

SURVEILLANCE REQUIREMENTS

SR ~~3.7.18.1~~ 3.1.3.1

This SR verifies by administrative means that the fuel assembly is stored in accordance with LCO ~~3.7.18~~ 3.1.3, or Design Features 4.3.1.1, or LCS 4.0.100. For fuel assemblies not stored in accordance with LCO ~~3.7.18~~ 3.1.3, performance of this SR will ensure compliance with Specification 4.3.1.1.

This surveillance is performed prior to the initial storage of a fuel assembly in a spent fuel pool location and prior to each subsequent movement to a new location.

REFERENCES

UFSAR, Section 9.1.2.2.

Attachment 4

Proposed Technical Specifications Bases Markup Pages, Unit 3 (For Information Only)

B 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

BASES

and LCO 3.0.2

LCOs	LCO 3.0.1 ~~through LCO 3.0.7~~ establish the general requirements applicable to all Specifications and apply at all times unless otherwise stated.
LCO 3.0.1	LCO 3.0.1 establishes the Applicability statement within each individual Specification as the requirement for when the LCO is required to be met (i.e., when the ~~unit~~ facility is in the ~~MODES or other~~ specified conditions of the Applicability statement of each Specification).

LCO 3.0.2

LCO 3.0.2 establishes that upon discovery of a failure to meet an LCO, the associated ACTIONS shall be met. The Completion Time of each Required Action for an ACTIONS Condition is applicable from the point in time that an ACTIONS Condition is entered. The Required Actions establish those remedial measures that must be taken within specified Completion Times when the requirements of an LCO are not met. This Specification establishes that:

a. Completion of the Required Actions within the specified Completion Times constitutes compliance with a Specification; and

b. Completion of the Required Actions is not required when an LCO is met within the specified Completion Time, unless otherwise specified.

~~There are two basic types of Required Actions. The first type of Required Action specifies a time limit in which the LCO must be met. This time limit is the Completion Time to restore an inoperable system or component to OPERABLE status or to restore variables to within specified limits. If this type of Required Action is not completed within the specified Completion Time, a shutdown may be required to place the unit in a MODE or condition in which the Specification is not applicable. (Whether stated as a Required Action or not, correction of the entered Condition is an action that may always be considered upon entering~~

(continued)

BASES

LCO 3.0.2 (continued)

~~ACTIONS.) The second type of Required Action specifies the remedial measures that permit continued operation of the unit that is not further restricted by the Completion Time. In this case, compliance with the Required Actions provides an acceptable level of safety for continued operation.~~

Completing the Required Actions is not required when an LCO is met or is no longer applicable, unless otherwise stated in the individual Specifications.

~~The nature of some Required Actions of some Conditions necessitates that, once the Condition is entered, the Required Actions must be completed even though the associated Conditions no longer exist. The individual LCO's ACTIONS specify the Required Actions where this is the case. An example of this is in LCO 3.8.1, "AC Sources—Operating."~~

a specified Condition in the Applicability is entered

the safe storage of irradiated fuel

The Completion Times of the Required Actions are also applicable when ~~a system or component is removed from service~~ intentionally. The reasons for intentionally relying on the ACTIONS include, but are not limited to, performance of Surveillances, preventive maintenance, corrective maintenance, or investigation of ~~operational~~ problems. Entering ACTIONS for these reasons must be done in a manner that does not compromise ~~safety~~. Intentional entry into ACTIONS should not be made for ~~operational~~ convenience. ~~Alternatives that would not result in redundant equipment being inoperable should be used instead. Doing so limits the time both subsystems/trains of a safety function are inoperable and limits the time other conditions exist which result in LCO 3.0.3 being entered. Individual Specifications may specify a time limit for performing an SR when equipment is removed from service or bypassed for testing. In this case, the Completion Times of the Required Actions are applicable when this time limit expires, if the equipment remains removed from service or bypassed.~~

~~When a change in MODE or other specified condition is required to comply with Required Actions, the unit may enter a MODE or other specified condition in which another Specification becomes applicable. In this case, the Completion Times of the associated Required Actions would apply from the point in time that the new Specification becomes applicable and the ACTIONS Condition(s) are entered.~~

(continued)

BASES (continued)

~~LCO 3.0.3~~

~~LCO 3.0.3 establishes the actions that must be implemented when an LCO is not met and:~~

~~a.~~ ~~An associated Required Action and Completion Time is not met and no other Condition applies; or~~

~~b.~~ ~~The condition of the unit is not specifically addressed by the associated ACTIONS. This means that no combination of Conditions stated in the ACTIONS can be made that exactly corresponds to the actual condition of the unit. Sometimes, possible combinations of Conditions are such that entering LCO 3.0.3 is warranted; in such cases, the ACTIONS specifically state a Condition corresponding to such combinations and also that LCO 3.0.3 be entered immediately.~~

~~This Specification delineates the time limits for placing the unit in a safe MODE or other specified condition when operation cannot be maintained within the limits for safe operation as defined by the LCO and its ACTIONS. It is not intended to be used as an operational convenience that permits routine voluntary removal of redundant systems or components from service in lieu of other alternatives that would not result in redundant systems or components being inoperable.~~

~~Upon entering LCO 3.0.3, 1 hour is allowed to prepare for an orderly shutdown before initiating a change in unit operation. This includes time to permit the operator to coordinate the reduction in electrical generation with the load dispatcher to ensure the stability and availability of the electrical grid. The time limits specified to reach lower MODES of operation permit the shutdown to proceed in a controlled and orderly manner that is well within the specified maximum cooldown rate and within the capabilities of the unit, assuming that only the minimum required equipment is OPERABLE. This reduces thermal stresses on components of the Reactor Coolant System and the potential for a plant upset that could challenge safety systems under conditions to which this Specification applies. The use and interpretation of specified times to complete the actions of LCO 3.0.3 are consistent with the discussion of Section 1.3, Completion Times.~~

(continued)

BASES (continued)

~~LCO 3.0.3 (continued)~~

~~Voluntary entry into LCO 3.0.3 is permissible but requires prior approval (approval may be verbal) from either the Operations Manager, Station Manager or corporate officer with direct responsibility for the plant. The approval must subsequently be documented in written retrievable manner. Inadvertent entry still allows for the one hour preparation period before Actions to change MODES must begin.~~

~~A unit shutdown required in accordance with LCO 3.0.3 may be terminated and LCO 3.0.3 exited if any of the following occurs:~~

~~a. The LCO is now met.~~

~~b. A Condition exists for which the Required Actions have now been performed.~~

~~c. ACTIONS exist that do not have expired Completion Times. These Completion Times are applicable from the point in time that the Condition is initially entered and not from the time LCO 3.0.3 is exited.~~

~~The time limits of Specification 3.0.3 allow 37 hours for the unit to be in MODE 5 when a shutdown is required during MODE 1 operation. If the unit is in a lower MODE of operation when a shutdown is required, the time limit for reaching the next lower MODE applies. If a lower MODE is reached in less time than allowed, however, the total allowable time to reach MODE 5, or other applicable MODE, is not reduced. For example, if MODE 3 is reached in 2 hours, then the time allowed for reaching MODE 4 is the next 11 hours, because the total time for reaching MODE 4 is not reduced from the allowable limit of 13 hours. Therefore, if remedial measures are completed that would permit a return to MODE 1, a penalty is not incurred by having to reach a lower MODE of operation in less than the total time allowed.~~

~~In MODES 1, 2, 3, and 4, LCO 3.0.3 provides actions for Conditions not covered in other Specifications. The requirements of LCO 3.0.3 do not apply in MODES 5 and 6 because the unit is already in the most restrictive Condition required by LCO 3.0.3.~~

~~The requirements of LCO 3.0.3 do not apply in other specified conditions of the Applicability (unless in MODE 1, 2, 3, or 4) because the ACTIONS of individual Specifications~~

(continued)

BASES

~~LCO~~ ~~3.0.3~~ ~~(continued)~~

~~sufficiently define the remedial measures to be taken. Exceptions to LCO 3.0.3 are provided in instances where requiring a unit shutdown, in accordance with LCO 3.0.3, would not provide appropriate remedial measures for the associated condition of the unit. An example of this is in LCO 3.7.16, "Fuel Storage Pool Water Level." LCO 3.7.16 has an Applicability of "During movement of irradiated fuel assemblies in the fuel storage pool." Therefore, this LCO can be applicable in any or all MODES. If the LCO and the Required Actions of LCO 3.7.16 are not met while in MODE 1, 2, or 3, there is no safety benefit to be gained by placing the unit in a shutdown condition. The Required Action of LCO 3.7.16 of "Suspend movement of irradiated fuel assemblies in fuel storage pool" is the appropriate Required Action to complete in lieu of the actions of LCO 3.0.3. These exceptions are addressed in the individual Specifications.~~

~~LCO 3.0.4~~

~~LCO 3.0.4 establishes limitations on changes in MODES or other specified conditions in the Applicability when an LCO is not met. It precludes placing the unit in a different MODE or other specified condition when the following exist:~~

~~a.~~ ~~The requirements of an LCO, in the MODE or other specified condition to be entered, are not met; and~~

~~b.~~ ~~Continued noncompliance with these LCO requirements would result in the unit being required to be placed in a MODE or other specified condition in which the LCO does not apply to comply with the Required Actions.~~

~~Compliance with Required Actions that permit continued operation of the unit for an unlimited period of time in a MODE or other specified condition provides an acceptable level of safety for continued operation. This is without regard to the status of the unit before or after the MODE change. Therefore, in such cases, entry into a MODE or other specified condition in the Applicability may be made in accordance with the provisions of the Required Actions. The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before unit startup.~~

(continued)

BASES

~~LCO 3.0.4~~
~~(continued)~~

~~The provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from a normal shutdown.~~

~~Exceptions to LCO 3.0.4 are stated in the individual Specifications. Exceptions may apply to all the ACTIONS or to a specific Required Action of a Specification.~~

~~Surveillances do not have to be performed on the associated inoperable equipment (or on variables outside the specified limits), as permitted by SR 3.0.1. Therefore, changing MODES or other specified conditions while in an ACTIONS Condition, in compliance with LCO 3.0.4 or where an exception to LCO 3.0.4 is stated, is not a violation of SR 3.0.1 or SR 3.0.4 for those Surveillances that do not have to be performed due to the associated inoperable equipment. However, SRs must be met to ensure OPERABILITY prior to declaring the associated equipment OPERABLE (or variable within limits) and restoring compliance with the affected LCO.~~

~~LCO 3.0.5~~

~~LCO 3.0.5 establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service or declared inoperable to comply with ACTIONS. The sole purpose of this Specification is to provide an exception to LCO 3.0.2 (e.g., to not comply with the applicable Required Action(s)) to allow the performance of SRs to demonstrate:~~

~~a.~~ ~~The OPERABILITY of the equipment being returned to service; or~~

~~b.~~ ~~The OPERABILITY of other equipment.~~

~~The administrative controls ensure the time the equipment is returned to service in conflict with the requirements of the ACTIONS is limited to the time absolutely necessary to perform the allowed SRs. This Specification does not provide time to perform any other preventive or corrective maintenance.~~

(continued)

BASES

~~LCO 3.0.5~~ ~~(continued)~~

~~An example of demonstrating the OPERABILITY of the equipment being returned to service is reopening a containment isolation valve that has been closed to comply with Required Actions and must be reopened to perform the SRs.~~

~~An example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to prevent the trip function from occurring during the performance of an SR on another channel in the other trip system. A similar example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to permit the logic to function and indicate the appropriate response during the performance of an SR on another channel in the same trip system.~~

~~LCO 3.0.6~~

~~LCO 3.0.6 establishes an exception to LCO 3.0.2 for support systems that have an LCO specified in the Technical Specifications (TS). This exception is provided because LCO 3.0.2 would require that the Conditions and Required Actions of the associated inoperable supported system LCO be entered solely due to the inoperability of the support system. This exception is justified because the actions that are required to ensure the unit is maintained in a safe condition are specified in the support system LCO's Required Actions. These Required Actions may include entering the supported system's Conditions and Required Actions or may specify other Required Actions.~~

~~When a support system is inoperable and there is an LCO specified for it in the TS, the supported system(s) are required to be declared inoperable if determined to be inoperable as a result of the support system inoperability. However, it is not necessary to enter into the supported systems' Conditions and Required Actions unless directed to do so by the support system's Required Actions. The potential confusion and inconsistency of requirements related to the entry into multiple support and supported systems' LCOs' Conditions and Required Actions are eliminated by providing all the actions that are necessary to ensure the unit is maintained in a safe condition in the support system's Required Actions.~~

(continued)

BASES

~~LCO 3.0.6~~ ~~(continued)~~

~~However, there are instances where a support system's Required Action may either direct a supported system to be declared inoperable or direct entry into Conditions and Required Actions for the supported system. This may occur immediately or after some specified delay to perform some other Required Action. Regardless of whether it is immediate or after some delay, when a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.~~

~~Specification 5.6, "Safety Function Determination Program (SFDP)," ensures loss of safety function is detected and appropriate actions are taken. Upon failure to meet two or more LCOs concurrently, an evaluation shall be made to determine if loss of safety function exists. Additionally, other limitations, remedial actions, or compensatory actions may be identified as a result of the support system inoperability and corresponding exception to entering supported system Conditions and Required Actions. The SFDP implements the requirements of LCO 3.0.6.~~

~~Cross train checks to identify a loss of safety function for those support systems that support multiple and redundant safety systems are required. The cross train check verifies that the supported systems of the redundant OPERABLE support system are OPERABLE, thereby ensuring safety function is retained. If this evaluation determines that a loss of safety function exists, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.~~

~~LCO 3.0.7~~

~~Special tests and operations are required at various times over the unit's life to demonstrate performance characteristics, to perform maintenance activities, and to perform special evaluations. Because TS normally preclude these tests and operations, special test exceptions (STEs) allow specified requirements to be changed or suspended under controlled conditions. STEs are included in applicable sections of the Specifications. Unless otherwise specified, all other TS requirements remain unchanged and in~~

(continued)

BASES

~~LCO 3.0.7~~
~~(continued)~~

~~effect as applicable. This will ensure that all appropriate requirements of the MODE or other specified condition not directly associated with or required to be changed or suspended to perform the special test or operation will remain in effect.~~

~~The Applicability of an STE LCO represents a condition not necessarily in compliance with the normal requirements of the TS. Compliance with STE LCOs is optional.~~

~~A special test may be performed under either the provisions of the appropriate STE LCO or the other applicable TS requirements. If it is desired to perform the special test under the provisions of the STE LCO, the requirements of the STE LCO shall be followed. This includes the SRs specified in the STE LCO.~~

~~Some of the STE LCOs require that one or more of the LCOs for normal operation be met (i.e., meeting the STE LCO requires meeting the specified normal LCOs). The Applicability, ACTIONS, and SRs of the specified normal LCOs, however, are not required to be met in order to meet the STE LCO when it is in effect. This means that, upon failure to meet a specified normal LCO, the associated ACTIONS of the STE LCO apply, in lieu of the ACTIONS of the normal LCO. Exceptions to the above do exist. There are instances when the Applicability of the specified normal LCO must be met, where its ACTIONS must be taken, where certain of its Surveillances must be performed, or where all of these requirements must be met concurrently with the requirements of the STE LCO.~~

~~Unless the SRs of the specified normal LCOs are suspended or changed by the special test, those SRs that are necessary to meet the specified normal LCOs must be met prior to performing the special test. During the conduct of the special test, those Surveillances need not be performed unless specified by the ACTIONS or SRs of the STE LCO.~~

~~ACTIONS for STE LCOs provide appropriate remedial measures upon failure to meet the STE LCO. Upon failure to meet these ACTIONS, suspend the performance of the special test and enter the ACTIONS for all LCOs that are then not met. Entry into LCO 3.0.3 may possibly be required, but this determination should not be made by considering only the failure to meet the ACTIONS of the STE LCO.~~

B 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

BASES

SRs	SR 3.0.1 through SR 3.0.4 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.

SR 3.0.1

SR 3.0.1 establishes the requirement that SRs must be met during the ~~MODES or other~~ specified conditions in the Applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This Specification is to ensure that Surveillances are performed in order to verify the ~~OPERABILITY of systems and components, and that variables~~ facility conditions are within specified limits. Failure to meet a Surveillance within the specified Frequency, in accordance with SR 3.0.2, constitutes a failure to meet an LCO.

~~Systems and components are assumed to be OPERABLE when the associated SRs have been met. Nothing in this Specification, however, is to be construed as implying that systems or components are OPERABLE when:~~

~~a. The systems or components are known to be inoperable, although still meeting the SRs; or~~

~~b. The requirements of the Surveillance(s) are known to be not met between required Surveillance performances.~~

Surveillances do not have to be performed when the ~~unit~~ facility is in a ~~MODE or other~~ specified condition for which the requirements of the associated LCO are not applicable, unless otherwise specified. ~~The SRs associated with a special test exception (STE) are only applicable when the STE is used as an allowable exception to the requirements of a Specification.~~

~~Surveillances, including Surveillances invoked by Required Actions, do not have to be performed on inoperable equipment because the ACTIONS define the remedial measures that apply. Surveillances have to be met and performed in accordance with SR 3.0.2, prior to returning equipment to OPERABLE status.~~

(continued)

BASES

~~SR 3.0.1 (continued)~~

~~Upon completion of maintenance, appropriate post maintenance testing is required to declare equipment OPERABLE. This includes ensuring applicable Surveillances are not failed and their most recent performance is in accordance with SR 3.0.2. Post maintenance testing may not be possible in the current MODE or other specified conditions in the Applicability due to the necessary unit parameters not having been established. In these situations, the equipment may be considered OPERABLE provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function. This will allow operation to proceed to a MODE or other specified condition where other necessary post maintenance tests can be completed.~~

~~Some examples of this process are:~~

~~a.~~ ~~Auxiliary feedwater (AFW) pump turbine maintenance during refueling that requires testing at steam pressures > 800 psi. However, if other appropriate testing is satisfactorily completed, the AFW System can be considered OPERABLE. This allows startup and other necessary testing to proceed until the plant reaches the steam pressure required to perform the testing.~~

~~b.~~ ~~High pressure safety injection (HPSI) maintenance during shutdown that requires system functional tests at a specified pressure. Provided other appropriate testing is satisfactorily completed, startup can proceed with HPSI considered OPERABLE. This allows operation to reach the specified pressure to complete the necessary post maintenance testing.~~

SR 3.0.2

~~SR 3.0.2 establishes the requirements for meeting the specified Frequency for Surveillances and any Required Action with a Completion Time that requires the periodic performance of the Required Action on a "once per..." interval.~~

SR 3.0.2 permits a 25% extension of the interval specified in the Frequency. This extension facilitates Surveillance scheduling and considers ~~plant operating~~ facility conditions that may

(continued)

BASES

SR 3.0.2 (continued)

not be suitable for conducting the Surveillance (e.g., ~~transient conditions or~~ other ongoing Surveillance or maintenance activities).

The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs. ~~The exceptions to SR 3.0.2 are those Surveillances for which the 25% extension of the interval specified in the Frequency does not apply. These exceptions are stated in the individual Specifications. An example of where SR 3.0.2 does not apply is the Containment Leakage Rate Testing Program. Test frequencies specified in the Containment Leakage Rate Testing Program may be extended consistent with the guidance provided in NEI 94-01," Industry Guideline For Implementing Performance-Based Option Of 10CFR 50, Appendix J," as endorsed by Regulatory Guide 1.163.~~

~~As stated in SR 3.0.2, the 25% extension also does not apply to the initial portion of a periodic Completion Time that requires performance on a "once per..." basis. The 25% extension applies to each performance after the initial performance. The initial performance of the Required Action, whether it is a particular Surveillance or some other remedial action, is considered a single action with a single Completion Time. One reason for not allowing the 25% extension to this Completion Time is that such an action usually verifies that no loss of function has occurred by checking the status of redundant or diverse components or accomplishes the function of the inoperable equipment in an alternative manner.~~

The provisions of SR 3.0.2 are not intended to be used repeatedly merely as a~~n~~ ~~operational~~ convenience to extend Surveillance intervals or periodic Completion Time intervals beyond those specified.

SR 3.0.3

SR 3.0.3 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a Surveillance has not

(continued)

BASES (continued)

SR 3.0.3 (continued)

been completed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified frequency , whichever is greater, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met.

This delay period provides an adequate time to complete Surveillances that have been missed. This delay period permits the completion of a Surveillance before complying with Required Actions or other remedial measures that might preclude completion of the Surveillance.

The basis for this delay period includes consideration of facility ~~unit~~ conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements.

When a Surveillance with a Frequency based not on time intervals, but upon specified facility ~~unit~~ conditions~~, operating situations, or requirements of regulations (e.g., prior to entering MODE 1 after each fuel loading, or in accordance with 10CFR50, Appendix J, as modified by approved exemptions, etc.)~~ is discovered to not have been performed when specified, SR 3.0.3 allows for the full delay period of up to the specified Frequency to perform the Surveillance. However, since there is not a time interval specified, the missed Surveillance should be performed at the first reasonable opportunity.

~~SR 3.0.3 provides a time limit for, and allowances for the performance of, Surveillances that become applicable as a consequence of MODE changes imposed by Required Actions.~~

(continued)

BASES (continued)

SR 3.0.3 (continued)

Failure to comply with specified Frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 3.0.3 is a flexibility which is not intended to be used as a ~~n operational~~ convenience to extend Surveillance intervals. While up to 24 hours or the limit of the specified Frequency is provided to perform the missed Surveillance, it is expected that the missed Surveillance will be performed at the first reasonable opportunity. The determination of the first reasonable opportunity should include consideration of the impact on ~~plant~~ risk (from delaying the Surveillances ~~as well as any plant configuration changes required or shutting the plant down to perform the Surveillance~~) and impact on any analysis assumptions, in addition to ~~unit~~ facility conditions, planning, availability of personnel, and the time required to perform the Surveillance. This risk impact should be managed through the program in place to implement 10CFR50.65(a)(4) and its implementation guidance, NRC Regulatory Guide 1.182, 'Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants.' This Regulatory Guide addresses consideration of temporary and aggregate risk impacts, determination of risk management action thresholds, and risk management action ~~up to and including plant shutdown~~. The missed Surveillance should be treated as an emergent condition as discussed in the Regulatory Guide. The risk evaluation may use quantitative, qualitative, or blended methods. The degree of depth and rigor of the evaluation should be commensurate with the importance of the component. Missed Surveillances for important components should be analyzed quantitatively. If the results of the risk evaluation determine the risk increase is significant, this evaluation should be used to determine the safest course of action. All missed Surveillances will be placed in the licensee's Corrective Action Program.

If a Surveillance is not completed within the allowed delay period, then the equipment is considered inoperable or the variable is considered outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then the equipment is inoperable, or the variable is outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon the failure of the Surveillance.

~~(continued)~~

BASES (continued)

SR 3.0.3 (continued)

Completion of the Surveillance within the delay period allowed by this Specification, or within the Completion Time of the ACTIONS, restores compliance with SR 3.0.1.

SR 3.0.4

SR 3.0.4 establishes the requirement that all applicable SRs must be met before entry into a ~~MODE or other~~ specified Condition in the Applicability.

This Specification ensures that system and component OPERABILITY requirements and variable limits are met before entry into ~~MODES or other~~ specified conditions in the Applicability for which these systems and components ensure safe operation of the facility ~~unit~~. ~~This Specification applies to changes in MODES or other specified conditions in the Applicability associated with unit shutdown as well as startup.~~

The provisions of SR 3.0.4 shall not prevent changes in ~~MODES or other~~ specified conditions in the Applicability that are required to comply with ACTIONS.

The precise requirements for performance of SRs are specified such that exceptions to SR 3.0.4 are not necessary. ~~The specific time frames and conditions necessary for meeting the SRs are specified in the Frequency, in the Surveillance, or both. This allows performance of Surveillances when the prerequisite condition(s) specified in a Surveillance procedure require entry into the MODE or other specified condition in the Applicability of the associated LCO prior to the performance or completion of a Surveillance. A Surveillance that could not be performed until after entering the LCO Applicability, would have its Frequency specified such that it is not "due" until the specific conditions needed are met. Alternately, the Surveillance may be stated in the form of a Note as not required (to be met or performed) until a particular event, condition, or time has been reached. Further discussion of the specific formats of SRs' annotation is found in Section 1.4, Frequency.~~

B ~~3.7~~ 3.1 PLANT SYSTEMS

B ~~3.7.16~~ 3.1.1 Fuel Storage Pool Water Level

BASES

BACKGROUND

The minimum water level in the fuel storage pool meets the assumptions of iodine decontamination factors following a fuel handling accident. The specified water level shields and minimizes the general area dose when the storage racks are filled to their maximum capacity. The water also provides shielding during the movement of spent fuel.

A general description of the fuel storage pool design is given in the UFSAR, Section 9.1.2, Reference 1, and the Spent Fuel Pool Cooling and Cleanup System is given in the UFSAR, Section 9.1.3 (Ref. 2). The assumptions of the fuel handling accident are given in the UFSAR, Section 15.7.3.4 ~~and 15.7.3.6~~ (Ref. 3 ~~and Ref. 6~~).

APPLICABLE SAFETY ANALYSES

The minimum water level in the fuel storage pool meets the assumptions of the fuel handling accident described in Regulatory Guide ~~1.25~~ 1.183 (Ref. 4). The resultant ~~2 hour thyroid~~ dose to a person at the exclusion area boundary or low population zone is a small fraction of the 10 CFR ~~100~~ 50.67 (Ref. 5) limits.

According to Reference 4, there is 23 ft of water between the top of the damaged fuel bundle and the fuel pool surface for a fuel handling accident. With this 23 ft of water, the assumptions of Reference 4 can be used directly. In practice, this LCO preserves this assumption for the bulk of the fuel in the storage racks. In the case of a single bundle, dropped and lying horizontally on top of the spent fuel racks, there would be < 23 ft of water above the top of the bundle.

~~However, when the potential of a dropped fuel assembly exists (which is when fuel is being moved) a water level is maintained that would ensure that there would be >23 feet above the fuel assembly laying on top of the racks. This increased water level is required by LCO 3.9.6 when the fuel storage pool is connected to the refueling cavity and by station procedures whenever fuel is being moved.~~

(continued)

APPLICABLE SAFETY ANALYSES (continued)	The fuel storage pool water level satisfies Criterion 3 of the NRC Policy Statement.
LCO	The specified water level preserves the assumptions of the fuel handling accident analysis (Ref. 3). As such, it is the minimum required for fuel storage and movement within the fuel storage pool.
APPLICABILITY	This LCO applies during movement of fuel assemblies (i.e., irradiated fuel, non-irradiated fuel, and the dummy fuel assembly) in the fuel storage pool since the potential for a release of fission products exists.

ACTIONS

A.1

~~Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply.~~

When the initial conditions for an accident cannot be met, steps should be taken to preclude the accident from occurring. When the fuel storage pool water level is lower than the required level, the movement of fuel assemblies in the fuel storage pool is immediately suspended. This effectively precludes a spent fuel handling accident from occurring. This does not preclude moving a fuel assembly to a safe position.

~~If moving fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving fuel assemblies while in MODES 1, 2, 3, and 4, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown.~~

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR ~~3.7.16.1~~ 3.1.1.1

This SR verifies sufficient fuel storage pool water is available in the event of a fuel handling accident. The water level in the fuel storage pool must be checked periodically. The 7 day Frequency is appropriate because the volume in the pool is normally stable. Water level changes are controlled by unit procedures and are acceptable, based on operating experience.

~~During refueling operations, the level in the fuel storage pool is at equilibrium with that of the refueling canal, and the level in the refueling canal is checked daily in accordance with LCO 3.9.6, "Refueling Water Level."~~

REFERENCES

1. UFSAR, Section 9.1.2.

2. UFSAR, Section 9.1.3.

3. UFSAR, Section 15.7.3.4.

4. Regulatory Guide ~~1.25~~ 1.183

5. 10 CFR ~~100.11~~ 50.67.

~~6. UFSAR, Section 15.7.3.6~~

B ~~3.7~~ 3.1 PLANT SYSTEMS

B ~~3.7.17~~ 3.1.2 Fuel Storage Pool Boron Concentration

BASES

BACKGROUND

As described in LCO ~~3.7.18~~ 3.1.3, "Spent Fuel Assembly Storage," fuel assemblies are stored in the spent fuel racks in accordance with criteria based on initial enrichment, discharge burnup, and cooling time (plutonium decay). Although the water in the spent fuel pool is normally borated to ≥ 2000 ppm, the criteria that limit the storage of a fuel assembly to specific rack locations is conservatively developed without taking credit for boron while maintaining K_{eff} < 1.0. Credit for boron is taken to maintain K_{eff} ≤ 0.95.

APPLICABLE SAFETY ANALYSES

Soluble boron in the spent fuel pool is credited in criticality analyses for normal and accident conditions. The relevant accidents are 1) Fuel Assembly Dropped Horizontally On Top of the Racks, 2) Fuel Assembly Dropped Vertically Into a Storage Location Already Containing a Fuel Assembly, 3) Fuel Assembly Dropped to the SFP Floor, and 4) Fuel Misloading in either Region I or Region II. The limiting accident is Fuel Misloading in either Region I or Region II.

A fuel assembly could be inadvertently loaded into a spent fuel rack location not allowed by LCO ~~3.7.18~~ 3.1.3 (e.g., an un-irradiated fuel assembly or an insufficiently depleted fuel assembly). This accident is analyzed assuming the misloading of one fresh assembly with the maximum permissible enrichment. However, the negative reactivity effect of the soluble boron compensates for the increased reactivity caused by the postulated accident scenario.

Under normal, non-accident conditions, the soluble boron needed to maintain K_{eff} less than or equal to 0.95, including uncertainties, is 970 ppm. Under accident conditions, the soluble boron needed to maintain K_{eff} less than or equal to 0.95, including uncertainties, is 1700 ppm. A SFP boron dilution analysis shows that dilution from 2000 ppm to below 1700 is not credible. Therefore, the minimum required soluble boron concentration is 2000 ppm.

The concentration of dissolved boron in the fuel pool satisfies Criterion 2 of the NRC Policy Statement.

(continued)

LCO

The specified concentration of 2000 ppm dissolved boron in the fuel pool preserves the assumptions used in the analyses described above. This concentration of dissolved boron is the minimum required concentration for fuel assembly storage and movement within the spent fuel pool.

APPLICABILITY

This LCO applies whenever fuel assemblies are stored in the spent fuel pool.

ACTIONS

A.1 and A.2

~~The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply.~~

When the concentration of boron in the spent fuel pool is less than required 2000 ppm, immediate action must be taken to preclude an accident from happening or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies. This does not preclude the movement of fuel assemblies to a safe position. In addition, action must be immediately initiated to restore boron concentration to the required 2000 ppm.

~~If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown.~~

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR ~~3.7.17.1~~ 3.1.2.1

This SR verifies that the concentration of boron in the spent fuel pool is within the required limit. As long as this SR is met, the analyzed incidents are fully addressed. The 7 day Frequency is appropriate because no major replenishment of pool water is expected to take place over a short period of time.

REFERENCES

1. UFSAR, Section 9.1.

B ~~3.7~~ 3.1 PLANT SYSTEMS

B ~~3.7.18~~ 3.1.3 Spent Fuel Assembly Storage

BASES

BACKGROUND

The spent fuel storage facility is designed to store either new (nonirradiated) nuclear fuel assemblies, or burned (irradiated) fuel assemblies in a vertical configuration underwater. The storage pool is sized to store 1542 fuel assemblies. Two types/sizes of spent fuel storage racks are used (Region I and Region II). The two Region I racks each contain 156 storage locations each spaced 10.40 inches on center in a 12x13 array. Four Region II storage racks each contain 210 storage locations in a 14x15 array. The remaining two Region II racks each contain 195 locations in a 13x15 array. All Region II locations are spaced 8.85 inches on center.

To maintain $K_{eff} \leq 0.95$ for spent fuel of maximum enrichment up to 4.8 w/o, (1) soluble boron is credited, and (2) the following storage patterns and borated stainless steel guide tube inserts are used as needed:

(1) unrestricted storage, minimum discharge burnup and cooling time requirements vs. initial enrichment,

(2) SFP Peripheral storage, minimum discharge burnup and cooling time requirements vs. initial enrichment,

(3) 2x2 storage patterns, minimum discharge burnup and cooling time requirements vs. initial enrichment,

(4) 3x3 storage patterns, minimum discharge burnup and cooling time requirements vs. initial enrichment,

(5) credit for inserted Control Element Assemblies (CEAs),

(6) credit for erbia in fresh assemblies,

(7) credit for cooling time (Pu-241 decay), and,

(8) credit for borated stainless steel guide tube inserts.

(continued)

BACKGROUND (continued)

When soluble boron is credited, the following acceptance criteria apply:

(1) Under normal conditions, the 95/95 neutron multiplication factor (K_{eff}), including all uncertainties, shall be less than 1.0 when flooded with unborated water, and,

(2) Under normal and accident conditions, the 95/95 neutron multiplication factor (K_{eff}), including all uncertainties, shall be less than or equal to 0.95 when flooded with borated water.

APPLICABLE SAFETY ANALYSES

The spent fuel storage facility is designed for noncriticality by use of adequate spacing, neutron absorbing stainless steel cans, borated water with a minimum soluble boron concentration of 970 ppm, and storage of fuel assemblies in accordance with the administrative controls in LCO ~~3.7.18~~ 3.1.3 and LCS 4.0.100, "Fuel Storage Patterns".

The spent fuel assembly storage satisfies Criterion 2 of the NRC Policy Statement.

LCO

The restrictions on the placement of fuel assemblies within the spent fuel pool, in the accompanying LCO, ensure that the K_{eff} of the spent fuel pool will always remain < 1.00 under normal, non-accident conditions assuming the pool to be flooded with unborated water. The K_{eff} of the spent fuel pool will always remain ≤ 0.95 under normal, non-accident conditions assuming the pool to be flooded with borated water with a minimum soluble boron concentration of 970 ppm. The K_{eff} of the spent fuel pool will always remain ≤ 0.95 under accident conditions assuming the pool to be flooded with borated water with a minimum soluble boron concentration of 1700 ppm. The restrictions are consistent with the criticality safety analysis performed for the spent fuel pool.

(continued)

APPLICABILITY

This LCO applies whenever any fuel assembly is stored in Regions I and II of the spent fuel pool.

ACTIONS

A.1

~~Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply.~~

When the configuration of fuel assemblies stored in Regions I and II of the spent fuel pool is not in accordance with LCO ~~3.7.18~~ 3.1.3, immediate action must be taken to make the necessary fuel assembly movement(s) to bring the configuration into compliance.

~~If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, in either case, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.~~

SURVEILLANCE REQUIREMENTS

SR ~~3.7.18.1~~ 3.1.3.1

This SR verifies by administrative means that the fuel assembly is stored in accordance with LCO ~~3.7.18~~ 3.1.3, or Design Features 4.3.1.1, or LCS 4.0.100. For fuel assemblies not stored in accordance with LCO ~~3.7.18~~ 3.1.3, performance of this SR will ensure compliance with Specification 4.3.1.1.

This surveillance is performed prior to the initial storage of a fuel assembly in a spent fuel pool location and prior to each subsequent movement to a new location.

REFERENCES

UFSAR, Section 9.1.2.2.

Attachment 5

Proposed Technical Specifications - Clean - Units 2 and 3

1.0 USE AND APPLICATION

1.1 Definitions

---NOTE---
The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications and Bases.

Term	Definition
ACTIONS	ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.
CERTIFIED FUEL HANDLER	A CERTIFIED FUEL HANDLER is an individual who complies with provisions of the CERTIFIED FUEL HANDLER training program required by TS 5.3.2.
OPERABLE – OPERABILITY	A system, subsystem, train, component, or device shall be OPERABLE when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).

1.0 USE AND APPLICATION

1.2 Logical Connectors

PURPOSE

The purpose of this section is to explain the meaning of logical connectors.

Logical connectors are used in Technical Specifications (TS) to discriminate between, and yet connect, discrete Conditions, Required Actions, Completion Times, Surveillances, and Frequencies. The only logical connectors that appear in TS are AND and OR. The physical arrangement of these connectors constitutes logical conventions with specific meanings.

BACKGROUND

Several levels of logic may be used to state Required Actions. These levels are identified by the placement (or nesting) of the logical connectors and by the number assigned to each Required Action. The first level of logic is identified by the first digit of the number assigned to a Required Action and the placement of the logical connector in the first level of nesting (i.e., left justified with the number of the Required Action). The successive levels of logic are identified by additional digits of the Required Action number and by successive indentions of the logical connectors.

When logical connectors are used to state a Condition, Completion Time, Surveillance, or Frequency, only the first level of logic is used, and the logical connector is left justified with the statement of the Condition, Completion Time, Surveillance, or Frequency.

EXAMPLE

The following example illustrates the use of logical connectors.

EXAMPLE 1.2-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Verify . . . AND A.2 Restore . . .	

In this example the logical connector AND is used to indicate that when in Condition A, both Required Actions A.1 and A.2 must be completed.

1.0 USE AND APPLICATION

1.3 Completion Times

PURPOSE

The purpose of this section is to establish the Completion Time convention and to provide guidance for its use.

BACKGROUND

Limiting Conditions for Operation (LCOs) specify minimum requirements for ensuring safe storage of irradiated fuel. The ACTIONS associated with an LCO state Conditions that typically describe the ways in which the requirements of the LCO can fail to be met. Specified with each stated Condition are Required Action(s) and Completion Time(s).

DESCRIPTION

The Completion Time is the amount of time allowed for completing a Required Action. It is referenced to the time of discovery of a situation (e.g., inoperable equipment or variable not within limits) that requires entering an ACTIONS Condition unless otherwise specified, providing the facility is in a specified condition stated in the Applicability of the LCO. Required Actions must be completed prior to the expiration of the specified Completion Time. An ACTIONS Condition remains in effect and the Required Actions apply until the Condition no longer exists or the facility is not within the LCO Applicability.

EXAMPLE

The following example illustrates the use of Completion Times.

<u>EXAMPLE 1.3-1</u>

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1 Verify . . . <u>AND</u> B.2 Restore . . .	6 hours 36 hours

Condition B has two Required Actions. Each Required Action has its own separate Completion Time. Each Completion Time is referenced to the time that Condition B is entered.

The Required Actions of Condition B are to perform the verification within 6 hours <u>AND</u> perform the restoration within 36 hours. A total of 6 hours is allowed for performing the verification and a total of 36 hours (not 42 hours) is allowed performing the restoration from the time that Condition B was entered. If verification is performed within 3 hours, the

1.3 Completion Times

EXAMPLE (continued)

time allowed for performing the restoration is the next 33 hours because the total time allowed performing the restoration is 36 hours.

IMMEDIATE COMPLETION TIME	When "Immediately" is used as a Completion Time, the Required Action should be pursued without delay and in a controlled manner.

1.0 USE AND APPLICATION

1.4 Frequency

PURPOSE	The purpose of this section is to define the proper use and application of Frequency requirements.
DESCRIPTION	Each Surveillance Requirement (SR) has a specified Frequency in which the Surveillance must be met in order to meet the associated LCO. An understanding of the correct application of the specified Frequency is necessary for compliance with the SR. The "specified Frequency" is referred to throughout this section and each of the Specifications of Section 3.0, Surveillance Requirement (SR) Applicability. The "specified Frequency" consists of the requirements of the Frequency column of each SR, as well as certain Notes in the Surveillance column that modify performance requirements.

EXAMPLES

The following examples illustrate the various ways that Frequencies are specified. In these examples, the Applicability of the LCO (LCO not shown) occurs whenever any fuel assembly is stored in the fuel storage pool.

EXAMPLE 1.4-1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify . . .	7 days

Example 1.4-1 contains the type of SR most often encountered in the Technical Specifications (TS). The Frequency specifies an interval (7 days) during which the associated Surveillance must be performed at least one time. Performance of the Surveillance initiates the subsequent interval. Although the Frequency is stated as 7 days, an extension of the time interval to 1.25 times the stated Frequency is allowed by SR 3.0.2 for operational flexibility. The measurement of this interval continues at all times, even when the SR is not required to be met per SR 3.0.1 (such as when the equipment is inoperable, a variable is outside specified limits, or the facility is outside the Applicability of the LCO). If the interval specified by SR 3.0.2 is exceeded while the facility is in a specified condition in the Applicability of the LCO, then SR 3.0.3 becomes applicable.

1.4 Frequency

EXAMPLES (continued)

If the interval as specified by SR 3.0.2 is exceeded while the facility is not in a specified condition in the Applicability of the LCO for which performance of the SR is required, the Surveillance must be performed within the Frequency requirements of SR 3.0.2 prior to entry into the specified condition. Failure to do so would result in a violation of SR 3.0.4.

EXAMPLE 1.4-2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify . . .	Prior to moving a fuel assembly . . .

Example 1.4-2 illustrates a one time performance Frequency.

This type of Frequency does not qualify for the 25% extension allowed by SR 3.0.2.

3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

LCO 3.0.1	LCOs shall be met during the specified conditions in the Applicability, except as provided in LCO 3.0.2.
LCO 3.0.2	Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met. If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required, unless otherwise stated.

3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 3.0.1	SRs shall be met during the specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.
SR 3.0.2	The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met.
SR 3.0.3	If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is greater. This delay period is permitted to allow performance of the Surveillance. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed. If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered. The Completion Times of the Required Actions begin immediately upon expiration of the delay period. When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered. The Completion Times of the Required Actions begin immediately upon failure to meet the Surveillance.
SR 3.0.4	Entry into a specified condition in the Applicability of an LCO shall not be made unless the LCO's Surveillances have been met within their specified Frequency. This provision shall not prevent entry into specified conditions in the Applicability that are required to comply with ACTIONS.

3.1 PLANT SYSTEMS

3.1.1 Fuel Storage Pool Water Level

LCO 3.1.1 The fuel storage pool water level shall be ≥ 23 ft over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: During movement of fuel assemblies in the fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Fuel storage pool water level not within limit.	A.1 Suspend movement of fuel assemblies in fuel storage pool.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.1.1	Verify the fuel storage pool water level is ≥ 23 ft above the top of irradiated fuel assemblies seated in the storage racks.	7 days

3.1 PLANT SYSTEMS

3.1.2 Fuel Storage Pool Boron Concentration

LCO 3.1.2 The fuel storage pool boron concentration shall be ≥ 2000 ppm.

APPLICABILITY: When any fuel assembly is stored in the fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Fuel storage pool boron concentration not within limit.	A.1	Suspend movement of fuel assemblies in the fuel storage pool.	Immediately
	AND		
	A.2	Initiate action to restore fuel storage pool boron concentration to within limit.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.2.1	Verify the fuel storage pool boron concentration is within limit.	7 days

3.1 PLANT SYSTEMS

3.1.3 Spent Fuel Assembly Storage

LCO 3.1.3

The combination of initial enrichment and burnup of each SONGS 2 and 3 spent fuel assembly stored in Region 1 shall be within the acceptable burnup domain of Figure 3.1.3-1 or Figure 3.1.3-2 or in accordance with Technical Specification 4.3.1.1.

The combination of initial enrichment and burnup of each SONGS 2 and 3 spent fuel assembly stored in Region II shall be within the acceptable burnup domain of Figure 3.1.3-3 or Figure 3.1.3-4, or in accordance with Technical Specification 4.3.1.1.

Each SONGS 1 uranium dioxide spent fuel assembly stored in Region II shall be stored in accordance with Technical Specification 4.3.1.1.

APPLICABILITY: Whenever any fuel assembly is stored in of the fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Requirements of the LCO not met.	A.1	Initiate action to bring the noncomplying fuel assembly into compliance.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.3.1	Verify by administrative means the initial enrichment, burnup, and cooling time of the fuel assembly are in accordance with LCO 3.1.3, or Design Features 4.3.1.1, or LCS 4.0.100. Rev 2, dated 09/27/07.	Prior to moving a fuel assembly to any spent fuel pool storage location.

FIGURE 3.1.3-1

MINIMUM BURNUP AND COOLING TIME VS. INITIAL ENRICHMENT
FOR
UNRESTRICTED PLACEMENT OF SONGS 2 AND 3 FUEL
IN
REGION I RACKS

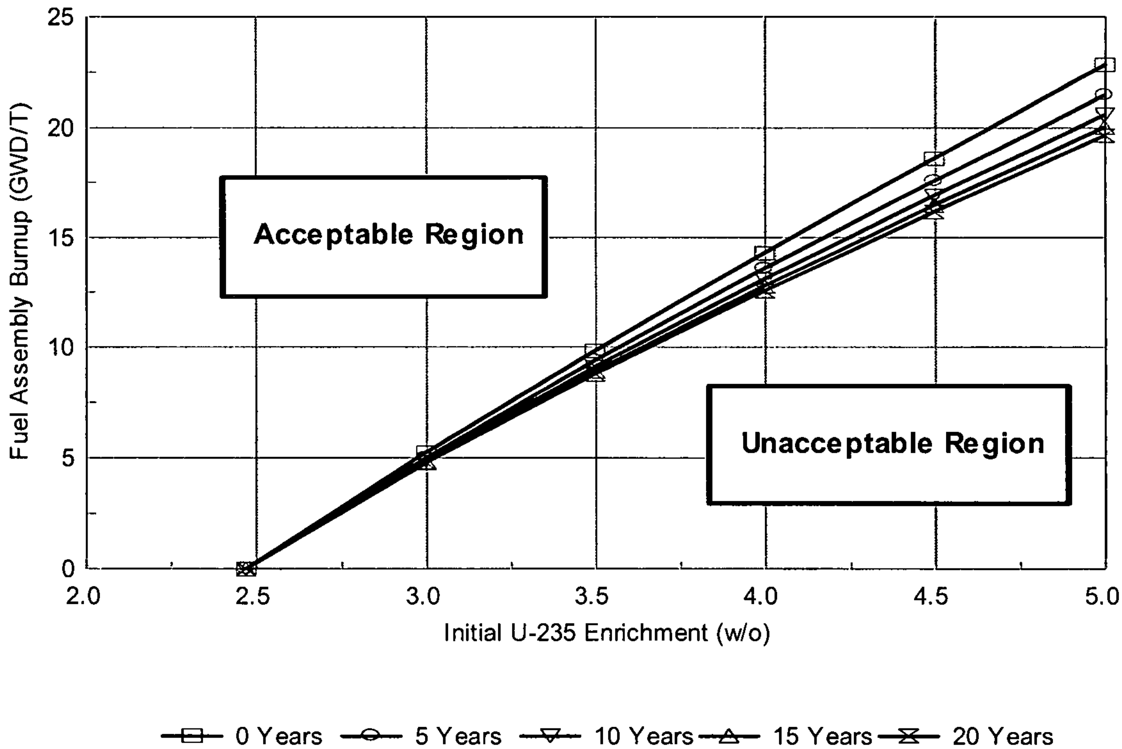


FIGURE 3.1.3-2

MINIMUM BURNUP AND COOLING TIME VS. INITIAL ENRICHMENT
FOR
PLACEMENT OF SONGS 2 AND 3 FUEL IN PERIPHERAL POOL LOCATIONS
IN
REGION I RACKS

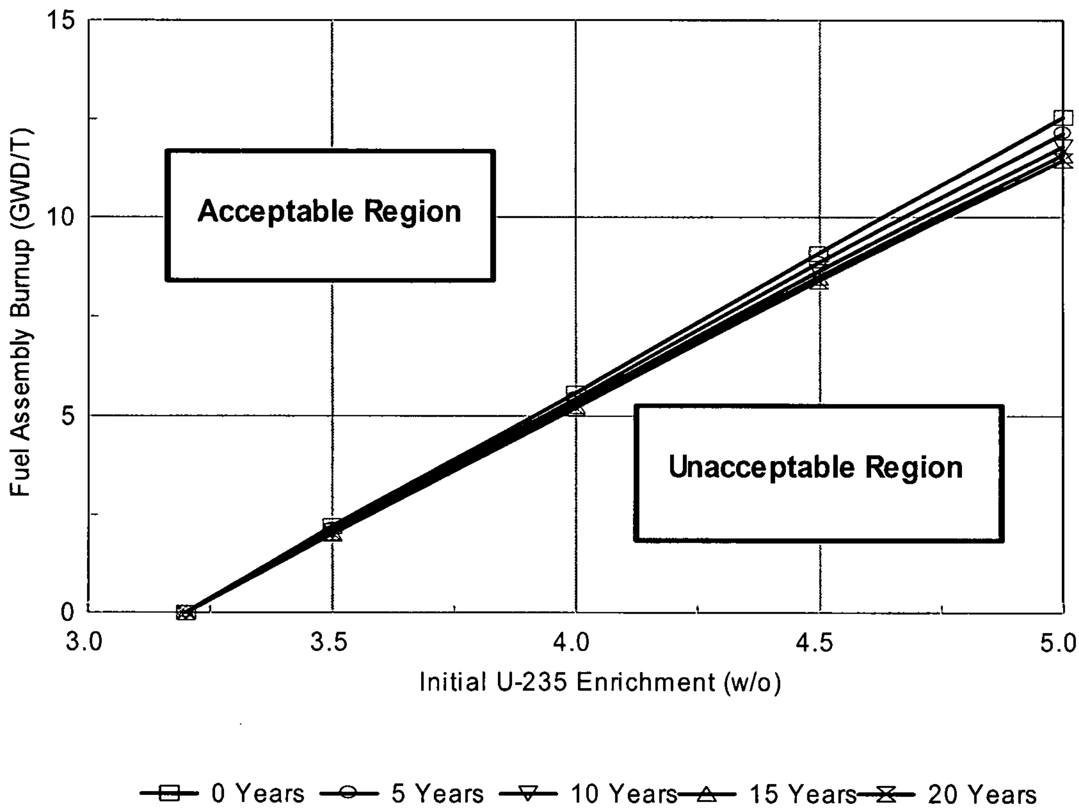


FIGURE 3.1.3-3

MINIMUM BURNUP AND COOLING TIME VS. INITIAL ENRICHMENT
FOR
UNRESTRICTED PLACEMENT OF SONGS 2 AND 3 FUEL
IN
REGION II RACKS

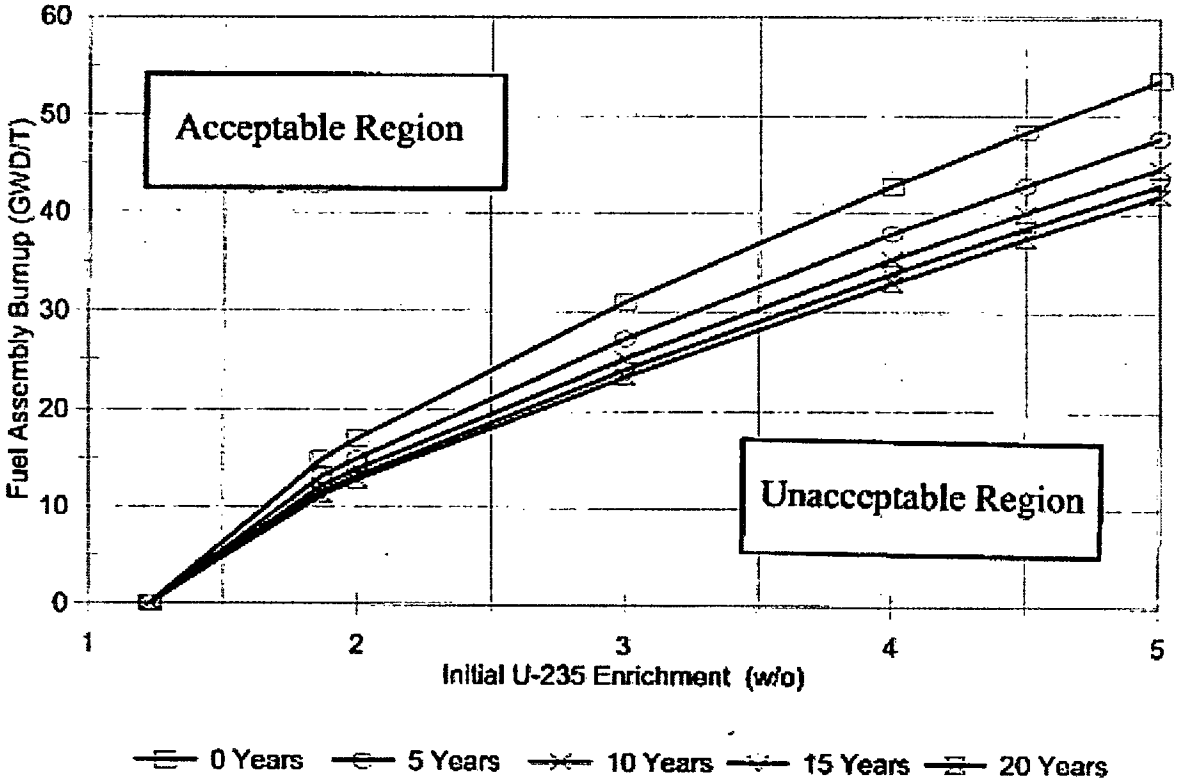
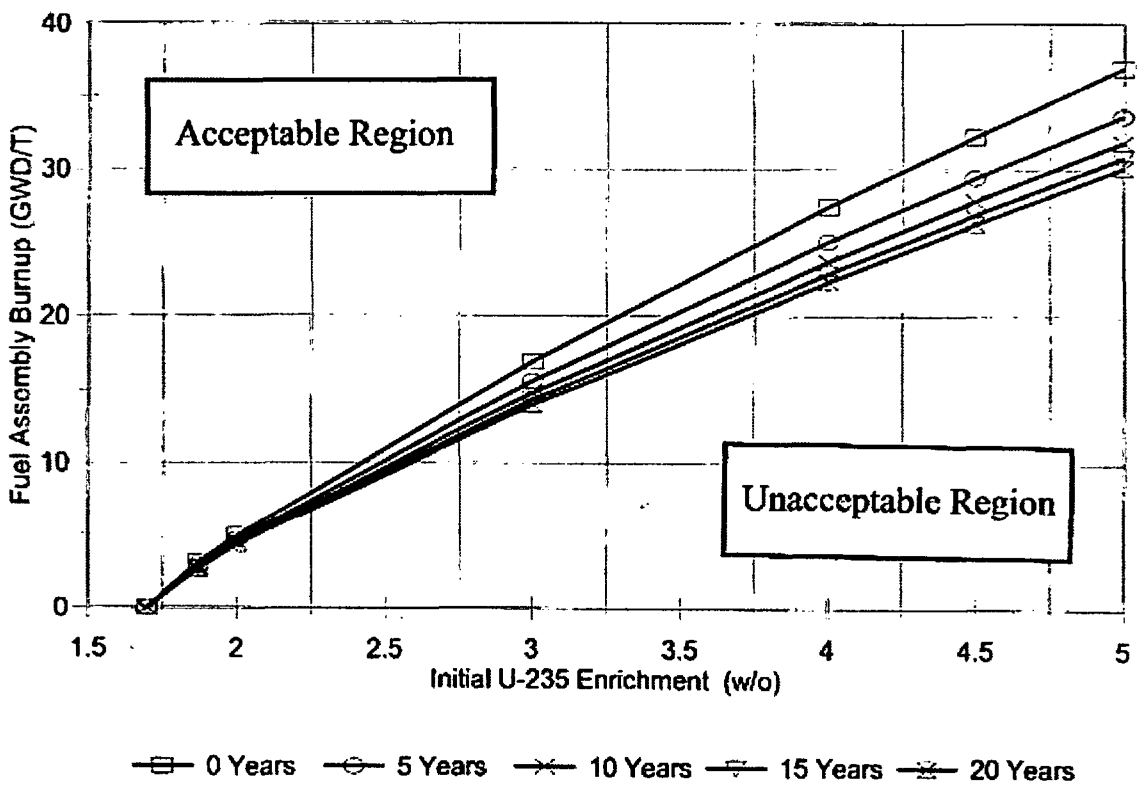


FIGURE 3.1.3-4

MINIMUM BURNUP AND COOLING TIME VS. INITIAL ENRICHMENT
FOR
PLACEMENT OF SONGS 2 AND 3 FUEL IN PERIPHERAL POOL LOCATIONS
IN
REGION II RACKS



4.0 DESIGN FEATURES

4.1 Site Location

4.1.1 Exclusion Area Boundary

The exclusion area boundary shall be as shown in Figure 4.1-1.

4.1.2 Low Population Zone (LPZ)

The LPZ shall be as shown in Figure 4.1-2.

4.2 Deleted.

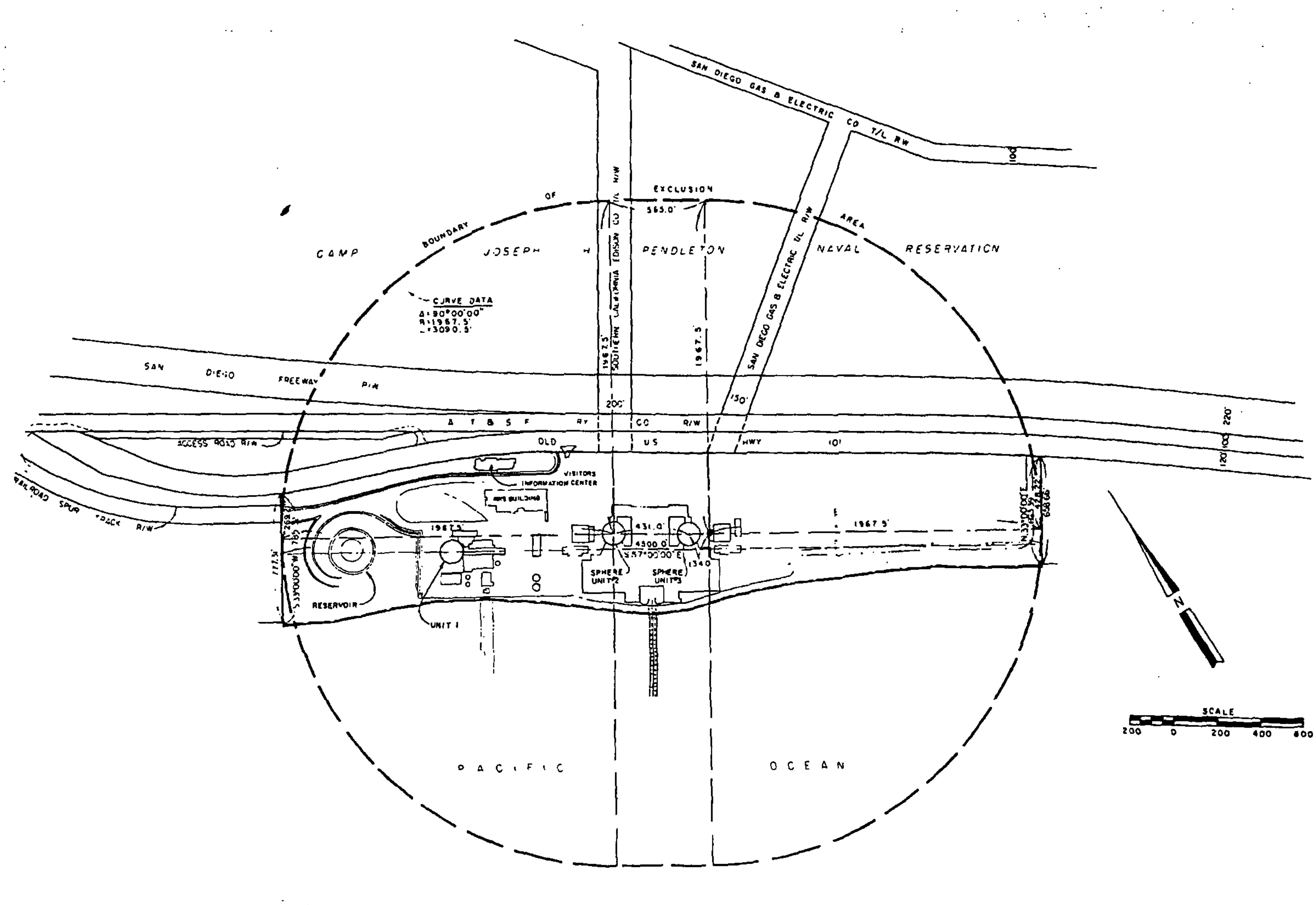


Figure 4.1-1 (page 1 of 1)
Exclusion Area Boundary

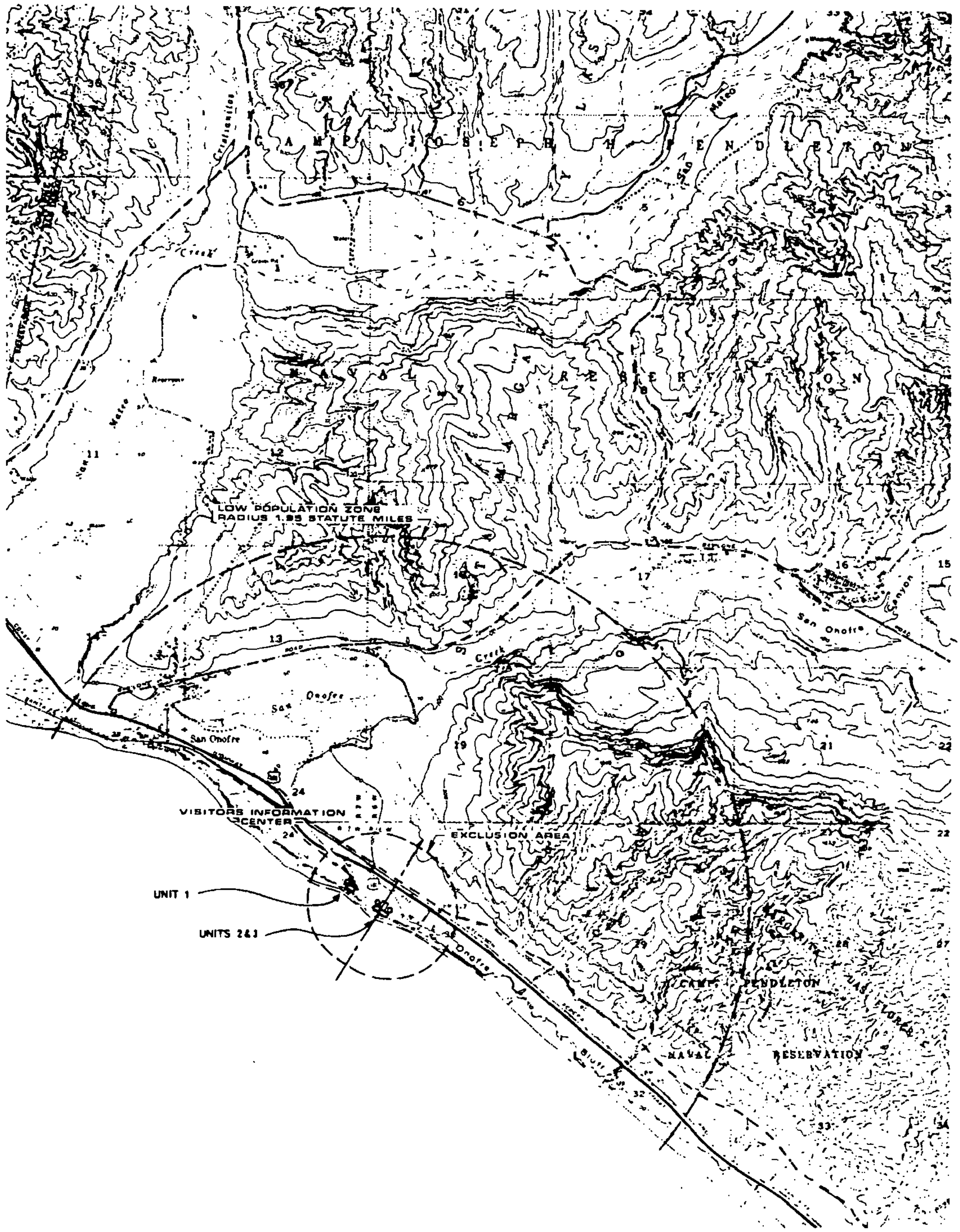


Figure 4.1-2 (page 1 of 1)
Low Population Zone

4.0 DESIGN FEATURES

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 4.8 weight percent;

b. $K_{eff} < 1.0$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR;

c. $K_{eff} \leq 0.95$ if fully flooded with water borated to 1700 ppm, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR;

d. Three or five borated stainless steel guide tube inserts (GT-Insert) may be used. When three borated stainless steel guide tube inserts are used, they will installed in an assembly's center guide tube, the guide tube associated with the serial number, and the diagonally opposite guide tube. Fuel containing GT-Inserts may be placed in Region I or Region II. However, credit for GT-Inserts is only taken for Region II storage.

   A five-finger CEA may be installed in an assembly. Fuel containing a five-finger CEA may be placed in either Region I or Region II. Credit for inserted 5-finger CEAs is taken for both Region I and Region II.

e. A nominal 8.85 inch center to center distance between fuel assemblies placed in Region II;

f. A nominal 10.40 inch center to center distance between fuel assemblies placed in Region I;

g. Prior to using the storage criteria of LCO 3.1.3 and LCS 4.0.100, the following uncertainties will be applied:

   (1) The calculated discharge burnup of San Onofre Units 2 and 3 assemblies will be reduced by 6.6%.

   (2) The calculated discharge burnup of San Onofre Unit 1 fuel assemblies will be reduced by 10.0%.

4.0 DESIGN FEATURES

4.3 Fuel Storage (continued)

h. Units 2 and 3 fuel assemblies with a burnup in the "acceptable range" of Figure 3.1.3-1 are allowed unrestricted storage in Region I;

i. Units 2 and 3 fuel assemblies with a burnup in the "acceptable range" of Figure 3.1.3-2 are allowed unrestricted storage in the peripheral pool locations with 1 or 2 faces toward the spent fuel pool walls of Region I;

j. Units 2 and 3 fuel assemblies with a burnup in the "acceptable range" of Figure 3.1.3-3 are allowed unrestricted storage in Region II;

k. Units 2 and 3 fuel assemblies with a burnup in the "acceptable range" of Figure 3.1.3-4 are allowed unrestricted storage in the peripheral pool locations with 1 or 2 faces toward the spent fuel pool walls of Region II;

l. Units 2 and 3 fuel assemblies with a burnup in the "unacceptable range" of Figure 3.1.3-1, Figure 3.1.3-2, Figure 3.1.3-3, and Figure 3.1.3-4 will be stored in compliance with Licensee Controlled Specification 4.0.100 Rev. 2, dated 9/27/07; and

m. Each SONGS 1 uranium dioxide spent fuel assembly stored in Region II shall be stored in accordance with Licensee Controlled Specification 4.0.100 Rev. 2, dated 9/27/07.

4.3.2 Drainage

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below Technical Specification 3.1.1 value (23 feet above the top of irradiated fuel assemblies seated in the storage racks).

4.3.3 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1542 fuel assemblies.

5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

5.1.1 The corporate officer with direct responsibility for the plant shall be responsible for overall management of the San Onofre Nuclear Generating Station, and all site support functions. He shall delegate in writing the succession to this responsibility during his absence.

5.1.2 The Shift Manager shall be responsible for the ultimate command decision authority for all unit activities and operations which affect the safety of the plant, site personnel, and/or the general public.

5.0 ADMINISTRATIVE CONTROLS

5.2 Organization

5.2.1 Onsite and Offsite Organizations

Onsite and offsite organizations shall be established for plant operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting the safety of the nuclear fuel.

a. Lines of authority, responsibility, and communication shall be established and defined throughout highest management levels through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These relationships, including the plant-specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications, as documented in the UFSAR.

b. The corporate officer with direct responsibility for the plant shall be responsible for overall safe handling and storage of nuclear fuel and shall have control over those onsite activities necessary for safe handling and storage of the nuclear fuel.

c. A specified corporate officer (or officers) shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure safe management of nuclear fuel.

d. The individuals who train CERTIFIED FUEL HANDLERS, and those who carry out radiation protection and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their ability to perform their assigned functions.

5.2.2 FACILITY STAFF

The facility staff organization shall include the following:

a. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 5.2.2-1;

b. Shift crew composition may be less than the minimum requirement of Table 5.2.2-1 for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate

5.2 Organization

5.2.2 FACILITY STAFF (continued)

action is taken to restore the shift crew composition to within the minimum requirements. During such absences, no fuel movement or movement of heavy loads over storage racks containing fuel is permitted.

c. A radiation protection technician shall be on site during fuel handling operations and during movement of heavy loads over storage racks containing fuel. Oversight of fuel handling operations shall be provided by a CERTIFIED FUEL HANDLER.

d. The Shift Manager shall be a CERTIFIED FUEL HANDLER.

Table 5.2.2-1
Minimum Shift Crew Composition

POSITION	MINIMUM STAFFING
CERTIFIED FUEL HANDLER	1*
Certified Operator	1

Note: The Certified Operator position may be filled by a CERTIFIED FUEL HANDLER.
* May be shared between Units 2 and 3.

5.0 ADMINISTRATIVE CONTROLS

5.3 Facility Staff Qualifications

5.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except the radiation protection manager who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975.

5.3.2 An NRC approved training and retraining program for the CERTIFIED FUEL HANDLERS shall be maintained.

5.0 ADMINISTRATIVE CONTROLS

5.4 Technical Specifications (TS) Bases Control

5.4.1 Changes to the Bases of the TS shall be made under appropriate administrative controls.

5.4.2 Changes to the Bases may be made without prior NRC approval provided the changes do not require either of the following:

a. A change in the TS incorporated in the license; or

b. A change to the updated UFSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

5.4.3 The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the UFSAR.

5.4.4 Proposed changes that meet the criteria of (a) or (b) above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC every 24 months.

5.0 ADMINISTRATIVE CONTROLS

5.5 Procedures, Programs, and Manuals

5.5.1 Procedures

5.5.1.1 Scope

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

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

b. Deleted.

c. Quality assurance for effluent and environmental monitoring using the guidance in Regulatory Guide 4.15, Revision 1, 1979;

d. Fire Protection Program implementation; and

e. Programs, as specified in Specification 5.5.2.

5.5.2. Programs and Manuals

The following programs and manuals shall be established, implemented, and maintained.

5.5.2.1 Offsite Dose Calculation Manual (ODCM)

a. The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the Radiological Environmental Monitoring Program;

b. The ODCM shall also contain the Radioactive Effluent Controls required by Specification 5.5.2.3 and Radiological Environmental Monitoring programs required by LCS, and descriptions of the information that should be included in the Annual Radiological Environmental Operating Report and the Radioactive Effluent Release Report required by Specification 5.7.1.2 and Specification 5.7.1.3.

5.5.2.1.1 Licensee-initiated changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

5.5 Procedures, Programs, and Manuals (continued)

5.5.2.1.1 Licensee-initiated changes to the ODCM (continued):

1. Sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s);

2. A determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.106, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

3. Documentation of the fact that the change has been reviewed and found acceptable.

b. Shall become effective upon review and approval by the corporate officer with direct responsibility for the plant or designee.

c. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

5.5.2.2 Deleted

5.5.2.3 Radioactive Effluent Controls Program

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to 10 CFR 20, Appendix B, Table II, Column 2;

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.106 and with the methodology and parameters in the ODCM;

5.5 Procedures, Programs, and Manuals (continued)

5.5.2.3 Radioactive Effluent Controls Program (continued)

d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each unit to unrestricted areas, conforming to 10 CFR 50, Appendix I;

e. Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days;

f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2 percent of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I;

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the site boundary conforming to the dose associated with 10 CFR 20, Appendix B, Table II, Column 1;

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives greater than 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and

j. Limitations on the annual dose or dose commitment to any member of the public due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

5.5.2.4 Deleted

5.5.2.5 Deleted

5.5.2.6 Deleted

5.5 Procedures, Programs, and Manuals (continued)

5.5.2.7 Storage Tank Radioactivity Monitoring Program

This program provides controls for the quantity of radioactivity contained in unprotected outdoor liquid storage tanks. The liquid radwaste quantities shall be determined in accordance with Standard Review Plan, Section 15.7.3, "Postulated Radioactive Release due to Tank Failures".

The program shall include a surveillance program to ensure that the quantity of radioactivity contained in all outdoor liquid radwaste tanks that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the Liquid Waste Management System is less than the amount that would result in concentrations less than the limits of 10 CFR Part 20, Appendix B, Table II, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tanks' contents.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Storage Tank Radioactivity Monitoring Program surveillance frequencies.

5.0 ADMINISTRATIVE CONTROLS

5.6 Deleted

5.0 ADMINISTRATIVE CONTROLS

5.7 Reporting Requirements

5.7.1 Routine Reports

In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted in accordance with 10 CFR 50.4. The reports shall be addressed to U.S. Nuclear Regulatory Commission, Attention: Document Control Desk, Washington, D.C., with a copy to the Regional Administrator of the Regional Office of the NRC, unless otherwise noted.

5.7.1.1 Deleted

5.7.1.2 Annual Radiological Environmental Operating Report

The Annual Radiological Environmental Operating Report covering the operation of the facility during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. The report shall identify the thermoluminescent dosimeter (TLD) results that represent collocated dosimeters in relation to the NRC TLD program and the exposure period associated with each result. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.7 Reporting Requirements (continued)

5.7.1.3 Radiological Effluent Release Report

The Radioactive Effluent Release Report covering the operation of the facility in the previous calendar year shall be submitted prior to May 1 of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the facility. The report shall also include a summary of the quantities of solid radioactive waste shipped from the facility directly to the disposal site and quantities of solid radioactive waste shipped from the facility's intermediary processor to the disposal site. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program (PCP) and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

5.0 ADMINISTRATIVE CONTROLS

5.8 High Radiation Area

5.8.1 Each high radiation area as defined 10 CFR 20 shall be barricaded and conspicuously posted as a high radiation area, and entrance thereto shall be controlled by requiring issuance of a Radiation Exposure Permit (REP).

Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device that continuously indicates the radiation dose rate in the area,

b. A radiation monitoring device that continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rates in the area have been determined and personnel have been made knowledgeable of them,

c. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device. This individual is responsible for providing positive radiation protection control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified in the radiation protection procedures or the applicable REP.

5.8.2 In addition, areas that are accessible to personnel and that have radiation levels greater than 1.0 rem (but less than 500 rads at 1 meter) in 1 hour at 30 cm from the radiation source, or from any surface penetrated by the radiation, shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the shift manager on duty or radiation protection supervisor. Doors shall remain locked except during periods of access by personnel under an approved REP that specifies the dose rates in the immediate work areas and the maximum allowable stay time for individuals in that area. In lieu of a stay time specification on the REP, direct or remote continuous surveillance (such as closed circuit TV cameras) may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area.

5.8.3 Individual high radiation areas that are accessible to personnel, that could result in radiation doses greater than 1.0 rem in 1 hour, and that are within large areas where no enclosure exists to enable locking and where no enclosure can be reasonably constructed around the individual area shall be barricaded and conspicuously posted. A flashing light shall be activated as a warning device whenever the dose rate in such an area exceeds or is expected to exceed 1.0 rem in 1 hour at 30 cm from the radiation source or from any surface penetrated by the radiation.

Attachment 6

Proposed Technical Specifications Bases Pages - Clean, Units 2 and 3 (For Information Only)

B 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

BASES

LCOs	LCO 3.0.1 and LCO 3.0.2 establish the general requirements applicable to all Specifications and apply at all times unless otherwise stated.
LCO 3.0.1	LCO 3.0.1 establishes the Applicability statement within each individual Specification as the requirement for when the LCO is required to be met (i.e., when the facility is in the specified conditions of the Applicability statement of each Specification).
LCO 3.0.2	LCO 3.0.2 establishes that upon discovery of a failure to meet an LCO, the associated ACTIONS shall be met. The Completion Time of each Required Action for an ACTIONS Condition is applicable from the point in time that an ACTIONS Condition is entered. The Required Actions establish those remedial measures that must be taken within specified Completion Times when the requirements of an LCO are not met. This Specification establishes that: a. Completion of the Required Actions within the specified Completion Times constitutes compliance with a Specification; and b. Completion of the Required Actions is not required when an LCO is met within the specified Completion Time, unless otherwise specified. Completing the Required Actions is not required when an LCO is met or is no longer applicable, unless otherwise stated in the individual Specifications. The Completion Times of the Required Actions are also applicable when a specified Condition in the Applicability is entered intentionally. The reasons for intentionally relying on the ACTIONS include, but are not limited to, performance of Surveillances, preventive maintenance, corrective maintenance, or investigation of problems. Entering ACTIONS for these reasons must be done in a manner that does not compromise the safe storage of irradiated fuel. Intentional entry into ACTIONS should not be made for convenience.

B 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

BASES

SRs	SR 3.0.1 through SR 3.0.4 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.
SR 3.0.1	SR 3.0.1 establishes the requirement that SRs must be met during the specified conditions in the Applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This Specification is to ensure that Surveillances are performed in order to verify the facility conditions are within specified limits. Failure to meet a Surveillance within the specified Frequency, in accordance with SR 3.0.2, constitutes a failure to meet an LCO. Surveillances do not have to be performed when the facility is in a specified condition for which the requirements of the associated LCO are not applicable, unless otherwise specified.
SR 3.0.2	SR 3.0.2 permits a 25% extension of the interval specified in the Frequency. This extension facilitates Surveillance scheduling and considers facility conditions that may not be suitable for conducting the Surveillance (e.g., other ongoing Surveillance or maintenance activities). The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs. Any exceptions to SR 3.0.2 are stated in the individual Specifications. The provisions of SR 3.0.2 are not intended to be used repeatedly merely as a convenience to extend Surveillance intervals (other than those consistent with refueling intervals) or periodic Completion Time intervals beyond those specified.
SR 3.0.3	SR 3.0.3 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a Surveillance has not been completed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified Frequency, whichever is greater, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met. This delay period provides adequate time to complete Surveillances that have been missed. This delay period permits the completion of a Surveillance before complying with Required Actions or other remedial measures that might preclude completion of the Surveillance.

BASES (continued)

SR 3.0.3 (continued)

The basis for this delay period includes consideration of facility conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements.

When a Surveillance with a Frequency based not on time intervals, but upon specified facility conditions is discovered to not have been performed when specified, SR 3.0.3 allows for the full delay period of up to the specified Frequency to perform the Surveillance. However, since there is not a time interval specified, the missed Surveillance should be performed at the first reasonable opportunity.

Failure to comply with specified Frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 3.0.3 is a flexibility which is not intended to be used as a convenience to extend Surveillance intervals.

SR 3.0.4

SR 3.0.4 establishes the requirement that all applicable SRs must be met before entry into a specified condition in the Applicability.

This Specification ensures that system and component OPERABILITY requirements and variable limits are met before entry into specified conditions in the Applicability for which these systems and components ensure safe operation of the facility.

The provisions of SR 3.0.4 shall not prevent changes in specified conditions in the Applicability that are required to comply with ACTIONS.

The precise requirements for performance of SRs are specified such that exceptions to SR 3.0.4 are not necessary. The specific time frames and conditions necessary for meeting the SRs are specified in the Frequency, in the Surveillance, or both. This allows performance of Surveillances when the prerequisite condition(s) specified in a Surveillance procedure require entry into specified condition in the Applicability of the associated LCO prior to the performance or completion of a Surveillance. A Surveillance that could not be performed until after entering the LCO's Applicability, would have its Frequency specified such that it is not "due" until the specific conditions needed are met. Alternately, the Surveillance may be stated in the form of a Note, as not required (to be met or performed) until a particular event, condition, or time has been reached. Further discussion of the specific formats of SRs' annotation is found in Section 1.4, Frequency.

B 3.1 PLANT SYSTEMS

B 3.1.1 Fuel Storage Pool Water Level

BASES

BACKGROUND	The minimum water level in the fuel storage pool meets the assumptions of iodine decontamination factors following a fuel handling accident. The specified water level shields and minimizes the general area dose when the storage racks are filled to their maximum capacity. The water also provides shielding during the movement of spent fuel. A general description of the fuel storage pool design is given in the UFSAR, Section 9.1.2, Reference 1, and the Spent Fuel Pool Cooling and Cleanup System is given in the UFSAR, Section 9.1.3 (Ref. 2). The assumptions of the fuel handling accident are given in the UFSAR, Section 15.7.3.4 (Ref. 3).
APPLICABLE SAFETY ANALYSES	The minimum water level in the fuel storage pool meets the assumptions of the fuel handling accident described in Regulatory Guide 1.183 (Ref. 4). The resultant dose to a person at the exclusion area boundary or low population zone is a small fraction of the 10 CFR 50.67 (Ref. 5) limits. According to Reference 4, there is 23 ft of water between the top of the damaged fuel bundle and the fuel pool surface for a fuel handling accident. With a 23 ft water level, the assumptions of Reference 4 can be used directly. In practice, this LCO preserves this assumption for the bulk of the fuel in the storage racks. In the case of a single bundle, dropped and lying horizontally on top of the spent fuel racks, however, there would be < 23 ft of water above the top of the bundle. The fuel storage pool water level satisfies Criterion 3 of the NRC Policy Statement.
LCO	The specified water level preserves the assumptions of the fuel handling accident analysis (Ref. 3). As such, it is the minimum required for fuel storage and movement within the fuel storage pool.
APPLICABILITY	This LCO applies during movement of fuel assemblies (i.e., irradiated fuel, non-irradiated fuel, and the dummy fuel assembly) in the fuel storage pool since the potential for a release of fission products exists.

BASES (continued)

ACTIONS	A.1 When the initial conditions for an accident cannot be met, steps should be taken to preclude the accident from occurring. When the fuel storage pool water level is lower than the required level, the movement of fuel assemblies in the fuel storage pool is immediately suspended. This effectively precludes a spent fuel handling accident from occurring. This does not preclude moving a fuel assembly to a safe position.
SURVEILLANCE REQUIREMENTS	SR 3.1.1.1 This SR verifies sufficient fuel storage pool water is available in the event of a fuel handling accident. The water level in the fuel storage pool must be checked periodically. The 7 day Frequency is appropriate because the volume in the pool is normally stable. Water level changes are controlled by unit procedures and are acceptable, based on operating experience.
REFERENCES	1. UFSAR, Section 9.1.2. 2. UFSAR, Section 9.1.3. 3. UFSAR, Section 15.7.3.4. 4. Regulatory Guide 1.183. 5. 10 CFR 50.67.

B 3.1 PLANT SYSTEMS

B 3.1.2 Fuel Storage Pool Boron Concentration

BASES

BACKGROUND

As described in LCO 3.1.3, "Spent Fuel Assembly Storage," fuel assemblies are stored in the spent fuel racks in accordance with criteria based on initial enrichment, discharge burnup, and cooling time (plutonium decay). Although the water in the spent fuel pool is normally borated to ≥ 2000 ppm, the criteria that limit the storage of a fuel assembly to specific rack locations is conservatively developed without taking credit for boron while maintaining K_{eff} < 1.0. Credit for boron is taken to maintain K_{eff} ≤ 0.95.

APPLICABLE SAFETY ANALYSES

Soluble boron in the spent fuel pool is credited in criticality analyses for normal and accident conditions. The relevant accidents are 1) Fuel Assembly Dropped Horizontally On Top of the Racks, 2) Fuel Assembly Dropped Vertically Into a Storage Location Already Containing a Fuel Assembly, 3) Fuel Assembly Dropped to the SFP Floor, and 4) Fuel Misloading in either Region I or Region II. The limiting accident is Fuel Misloading in either Region I or Region II.

A fuel assembly could be inadvertently loaded into a spent fuel rack location not allowed by LCO 3.1.3 (e.g., an unirradiated fuel assembly or an insufficiently depleted fuel assembly). This accident is analyzed assuming the misloading of one fresh assembly with the maximum permissible enrichment. However, the negative reactivity effect of the soluble boron compensates for the increased reactivity caused by the postulated accident scenario.

Under normal, non-accident conditions, the soluble boron needed to maintain K_{eff} less than or equal to 0.95, including uncertainties, is 970 ppm. Under accident conditions, the soluble boron needed to maintain K_{eff} less than or equal to 0.95, including uncertainties, is 1700 ppm. A SFP boron dilution analysis shows that dilution from 2000 ppm to below 1700 ppm is not credible. Therefore, the minimum required soluble boron concentration is 2000 ppm.

The concentration of dissolved boron in the fuel pool satisfies Criterion 2 of the NRC Policy Statement.

LCO

The specified concentration of 2000 ppm dissolved boron in the fuel pool preserves the assumptions used in the analyses described above. This concentration of dissolved boron is the minimum required concentration for fuel assembly storage and movement within the fuel pool.

APPLICABILITY

This LCO applies whenever fuel assemblies are stored in the spent fuel pool.

BASES (continued)

ACTIONS	<u>A.1, A.2.1, and A.2.2</u> When the concentration of boron in the spent fuel pool is less than required 2000 ppm, immediate action must be taken to preclude an accident from happening or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies. This does not preclude the movement of fuel assemblies to a safe position. In addition, action must be immediately initiated to restore boron concentration to the required 2000 ppm.
SURVEILLANCE REQUIREMENTS	<u>SR 3.1.2.1</u> This SR verifies that the concentration of boron in the spent fuel pool is within the required limit. As long as this SR is met, the analyzed incidents are fully addressed. The 7 day Frequency is appropriate because no major replenishment of pool water is expected to take place over a short period of time.
REFERENCES	1. UFSAR, Section 9.1.

B 3.1 PLANT SYSTEMS

B 3.1.3 Spent Fuel Assembly Storage

BASES

BACKGROUND

The spent fuel storage facility is designed to store either new (nonirradiated) nuclear fuel assemblies, or burned (irradiated) fuel assemblies in a vertical configuration underwater. The storage pool is sized to store 1542 fuel assemblies. Two types/sizes of spent fuel storage racks are used (Region I and Region II). The two Region I racks each contain 156 storage locations each spaced 10.40 inches on center in a 12x13 array. Four Region II storage racks each contain 210 storage locations in a 14x15 array. The remaining two Region II racks each contain 195 locations in a 13x15 array. All Region II locations are spaced 8.85 inches on center.

To maintain $K_{eff} < 0.95$ for spent fuel of maximum enrichment up to 4.8 w/o, (1) soluble boron is credited, and (2) the following storage patterns and borated stainless steel guide tube inserts are used as needed:

(1) unrestricted storage, minimum discharge burnup and cooling time requirements vs. initial enrichment,

(2) SFP Peripheral storage, minimum discharge burnup and cooling time requirements vs. initial enrichment,

(3) 2x2 storage patterns, minimum discharge burnup and cooling time requirements vs. initial enrichment,

(4) 3x3 storage patterns, minimum discharge burnup and cooling time requirements vs. initial enrichment,

(5) credit for inserted Control Element Assemblies (CEAs),

(6) credit for erbia in fresh assemblies,

(7) credit for cooling time (Pu-241 decay), and,

(8) credit for borated stainless steel guide tube inserts.

When soluble boron is credited, the following acceptance criteria apply:

(1) Under normal conditions, the 95/95 neutron multiplication factor (K_{eff}), including all uncertainties, shall be less than 1.0 when flooded with unborated water, and,

BASES (continued)

BACKGROUND (continued)

(2) Under normal and accident conditions, the 95/95 neutron multiplication factor (K_{eff}), including all uncertainties, shall be less than or equal to 0.95 when flooded with borated water.

APPLICABLE SAFETY ANALYSES

The spent fuel storage facility is designed for noncriticality by use of adequate spacing, neutron absorbing stainless steel cans, borated water with a minimum soluble boron concentration of 970 ppm, and storage of fuel assemblies in accordance with the administrative controls in LCO 3.1.3 and LCS 4.0.100, "Fuel Storage Patterns".

The spent fuel pool storage satisfies Criterion 2 of the NRC Policy Statement.

LCO

The restrictions on the placement of fuel assemblies within the spent fuel pool, in the accompanying LCO, ensure that the K_{eff} of the spent fuel pool will always remain < 1.00 under normal, non-accident conditions assuming the pool to be flooded with unborated water. The K_{eff} of the spent fuel pool will always remain ≤ 0.95 under normal, non-accident conditions assuming the pool to be flooded with borated water with a minimum soluble boron concentration of 970 ppm. The K_{eff} of the spent fuel pool will always remain ≤ 0.95 under accident conditions assuming the pool to be flooded with borated water with a minimum soluble boron concentration of 1700 ppm. The restrictions are consistent with the criticality safety analysis performed for the spent fuel pool.

APPLICABILITY

This LCO applies whenever any fuel assembly is stored in Regions I and II of the spent fuel pool.

ACTIONS

A.1

When the configuration of fuel assemblies stored in Regions I and II of the spent fuel pool is not in accordance with LCO 3.1.3, immediate action must be taken to make the necessary fuel assembly movement(s) to bring the configuration into compliance.

SURVEILLANCE REQUIREMENTS

SR 3.1.3.1

This SR verifies by administrative means that the fuel assembly is stored in accordance with LCO 3.1.3 or Design Features 4.3.1.1, or LCS 4.0.100. For fuel not stored in accordance with LCO 3.1.3, performance of this SR will ensure compliance with Specification 4.3.1.1.

This surveillance is performed prior to the initial storage of a fuel assembly in the spent fuel pool location and prior to each subsequent movement to a new location.

BASES (continued)

REFERENCES	UFSAR, Section 9.1.2.2.