Response to

Request for Additional Information No. 122 (1332, 1334), Revision 0

11/05/2008

U. S. EPR Standard Design Certification
AREVA NP Inc.
Docket No. 52-020
SRP Section: 16 - Technical Specifications
Application Section: FSAR Ch. 16

QUESTIONS for Technical Specification Branch (CTSB)

Question 16-243:

Provide additional explanation for each deviation from the reference STS definition for terms defined differently, for terms deleted, and for terms added into the proposed US EPR STS.

Provide justification regarding the changed content for the following terms:

AXIAL OFFSET (AO)
AZIMUTHAL POWER
CALIBRATION (vs STS defined CHANNEL CALIBRATION),
DIVISION OPERATIONAL TEST (vs STS defined CHANNEL OPERATIONAL TEST),
SENSOR OPERATIONAL TEST (vs STS defined CHANNEL OPERATIONAL TEST),
STAGGERED TEST BASIS (vs STS defined STAGGERED TEST BASIS),
ACTUATING DEVICE OPERATIONAL TEST (vs STS defined "TRIP ACTUATING DEVICE OPERATIONAL TEST" or new term without justification),
EXTENDED SELF TESTS (new term without justification),
ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME, and
REACTOR TRIP SYSTEM (RTS) RESPONSE TIME (STS terms deleted without justification)

The definition and use of these terms in the proposed Technical Specifications did not always appear consistent with terms defined and used in FSAR Chapter 7, as well as definitions used in IEEE Std 603-1998 and IEEE Std 338-1987 / RG 1.118, which were identified in FSAR 7.1, as part of the proposed licensing and design basis. Identify and justify any difference(s) between the definitions as used in the proposed Technical Specifications and the definitions established by IEEE Std 603-1998 and IEEE Std 338-1987 / RG 1.118.

See examples below:

Example #1 - It was not clear whether CALIBRATION would include the analog/digital (A/D) converter as well as the process analog sensor; both could be considered elements of an analog channel, but the TS typically refers to sensor output rather than channel. By contrast, FSAR 7.2.2.3.5, Compliance to Requirements on System Testing and Inoperable Surveillance Requirements (Clause 5.7 of IEEE 603-1998), describes testing of input channels.

Example #2 - It was not clear why the term "division" was almost always used in place of "channel." It appears to the staff that the proposed protection system architecture could be described as a sense-command-execute structure, whereby the channels could be defined in accordance with IEEE Std 603. The term "channel" would typically apply to the sense portion, and would be associated with a specific protective action. The term "division" would typically be used to establish boundaries for achieving physical, electrical, or functional independence from other divisions. Unless clearly described, the scope of a division can be substantially different than the scope of a single channel.

Example #3 - The scope and application of response time testing was not clear in the definitions. The proposed SENSOR OPERATIONAL TEST (SOT) definition includes, in part, the verification of the accuracy and time constants of the analog input modules. It was not clear if the SOT would include the sensor as well as the A/D converter. The methods to verify the response times associated with data processing and actuation devices was also not clear .

Example #4 - Provide a technical justification for the new EPR definition of AXIAL OFFSET (AO). This definition appears to be similar or equivalent to a previous AXIAL FLUX DIFFERENCE definition.

Example #5 - Provide a technical justification for the new EPR definition of AZIMUTHAL POWER. (Also, it appears this definition is really AZIMUTHAL POWER IMBALANCE (API)). This definition appears to be similar or equivalent to a previous QUADRANT POWER TILT RATIO definition.

This additional information is needed to ensure accuracy of terms used in GTS, STS, and PTS.

Response to Question 16-243:

A response to this Question will be provided by March 19, 2009.

Question 16-244:

Incorporate the following editorials to the EPR GTS, Section 1.0

Justify not following the Standard Technical Specifications (STS) format style shown in NUREG-1431, Westinghouse STS for the EPR GTS, Section 1.0, as found in the TSTF-GG-05-01, Writer's Guide for Plant-Specific Improved Technical Specifications. Numerous inconsistencies are found throughout the EPR GTS in separating/dividing lines between sections and in the presentation of headers.

Revise the EPR GTS, Section 1.0 regarding the definition of CHANNEL CHECK, to remove the fifth line of that definition that begins with "status to other indication or." This sentence appears to be redundant or superfluous text.

These editorials are needed to ensure consistency amongst GTS, STS, and PTS.

Response to Question 16-244:

The separating and dividing lines between sections, Chapter titles, and Section titles in U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications, Chapter 1, "Use and Application," will be revised in accordance with TSTF-CG-05-01, "Writer's Guide for Plant-Specific Improved Technical Specifications," June 2007. No issues were identified with the format of the headers.

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 1.1 "Definitions," will be revised so that the definition of Channel Check is consistent with the Standard Technical Specifications for Westinghouse Plants (NUREG-1431).

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Chapter 1, "Use and Application," will be revised as described in the response and indicated on the enclosed markup.

Question 16-245:

Provide a technical justification for changing terms in EPR GTS.

In the EPR GTS, Sections 1.3 and 1.4, the applicant proposed a new term, "Division," to replace "Train" and "Channel," the terms currently used in Westinghouse STS. The staff determined that these changes do not add any value. The new term did not enhance the understanding of the examples used to illustrate completion times in the GTS action statements or frequencies in the GTS SRs. Moreover, the terms "Train" and "Channel" are still being used throughout the EPR GTS. Provide a technical justification for these changes or revise the EPR GTS to adopt the original text of the STS.

This technical justification or change is needed to ensure the consistency of terms amongst GTS, STS, and PTS.

Response to Question 16-245:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section1.3 "Completion Times," will be revised to replace the term "Division" with the terms "Train" and "Channel" consistent with the Standard Technical Specifications for Westinghouse Plants (NUREG-1431). The term "Division" was not used in U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 1.4 "Frequency".

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 1.3 "Completion Times," will be revised as described in the response and indicated on the enclosed markup.

Question 16-246:

Provide a technical justification for changing terms in EPR GTS. Section 1.0.

In Example 1.4-1, the description of the Surveillance has been changed from "Perform CHANNEL CHECK." to "Perform CALIBRATION." The first sentence of the accompanying text has not been changed from the supposed daily frequently reoccurring SR. The rest of the text was changed. The example serves only as a standardized illustrative definition for Frequency and extension times. The example is not to justify any plant unique design difference. Justify or restore to the STS terminology of "CHANNEL CHECK." in this example.

This technical justification or change is needed to ensure the consistency of terms amongst GTS, STS, and PTS.

Response to Question 16-246:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 1.4 "Frequency," Example 1.4-1 will be revised to be consistent with the Standard Technical Specifications for Westinghouse Plants (NUREG-1431).

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 1.4 "Frequency," will be revised as described in the response and indicated on the enclosed markup..

Question 16-247:

Provide a correction to the text for the EPR GTS, Section 3.0.

The last sentence of LCO 3.0.7 is a continuation of the explanation from above and needs to be part of the original single paragraph to complete the explanation rather than "dangling" as a new paragraph topic. In LCO 3.0.7 of NUREG-1431, Westinghouse STS, this is a one-paragraph description. In the EPR STS, the last sentence of this paragraph has been separated as a second paragraph. This correction maintains the original intent of LCO 3.0.7 and avoids any potential misunderstanding in the importance of this LCO Applicability requirement.

In the EPR Bases, SR 3.0.1, the last sentence of the fourth paragraph is not consistent with the information in the Westinghouse STS. It is missing the following sentence: "This allowance includes those SRs where performance is normally precluded in a given MODE or other specified condition." This missing sentence from the Westinghouse STS should be retained to maintain the original basis for SR 3.0.1 intact.

These editorials are needed to ensure consistency amongst GTS, STS, and PTS.

Response to Question 16-247:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications LCO 3.0.7 and Bases SR 3.0.1 will be revised to be consistent with NUREG-1431.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications LCO 3.0.7 and Bases SR 3.0.1 will be revised as described in the response and indicated on the enclosed markup.

Question 16-248:

Provide the additional information needed to clarify the EPR Bases, Section B 3.6 for consistency with the EPR FSAR.

Revise Bases 3.6.1 and Bases 3.6.6 to expand and integrate the relevant explanation of the EPR dual containment feature from FSAR Section 6.2.3, Secondary Containment Functional Design. The revision information should include a description of the secondary containment protective function of the Shield Building (inclusive of its doors, openings, and penetrations in its boundary) which complements the function of the primary (inner) Containment Building. Provide a description of the collective tests, methods, inspections, and verifications that together ensure the integrity of the Shield Building (secondary containment) can be maintained.

The inner Containment Building and its penetrations establish the leak limiting boundary of the primary containment.

This information is needed to ensure the accuracy and completeness of the EPR Bases.

Response to Question 16-248:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Bases Sections 3.6.1 and 3.6.6 will be revised to be consistent with NUREG-1431 for Dual Containment Bases Sections 3.6.1 and 3.6.6 in accordance with the U.S. EPR design.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Bases 3.6.1 and 3.6.6 will be revised as described in the response and indicated on the enclosed mark-up.

Question 16-249:

Include the following editorials in the EPR GTS:

In the EPR GTS, LCO 3.6.4, add a (+) sign in front of 1.2 psig for consistency with the Westinghouse STS specific limits where the operational range crosses zero.

This editorial is needed to ensure the accuracy and consistency of the EPR GTS.

Response to Question 16-249:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications LCO 3.6.4 will be revised to be consistent with NUREG-1431.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications LCO 3.6.4 will be revised as described in the response and indicated on the enclosed mark-up.

Question 16-250:

The following editorials need to be considered for inclusion into the EPR GTS Section 3.7 and Bases:

1. In EPR Bases, Section B 3.7.7, Actions C.1 and C.2, the last sentence has been incorrectly separated by misplaced carriage returns.
2. In EPR Bases, Section B 3.7.8, Actions, first sentence should read "The ACTIONS have two Notes added."
3. In EPR GTS 3.7.10, Condition E statement, there is a space needed between "inoperable" and "in."
4. In EPR GTS, SR 3.7.10.2, the word "filter" is missing between "train" and "testing."
5. In the EPR Bases, Section B 3.6.6, LCO Section (pg B 3.6.6-1) second paragraph: Replace "the control room envelop (CRE) boundary" with "the secondary containment boundary".

This additional information is needed to ensure the accuracy and completeness of the EPR GTS and Bases.

Response to Question 16-250:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.7.10 and Bases Sections 3.7.7, 3.7.8, and 3.6.6 will be revised as defined by the question.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.7.10 and Bases Sections 3.7.7, 3.7.8 and 3.6.6 will be revised as described in the response and indicated on the enclosed mark-up.

Question 16-251:

This question has been intentionally deleted by the NRC.

Response to Question 16-251:

Not applicable.

FSAR Impact:

Not applicable.

Question 16-252:

Provide additional information to justify differences between the EPR GTS, Sections 5.2.2.d and 5.2.2.f and the applicable STS.

Provide additional information to explain the change to EPR GTS, Section 5.2.2.d that uses the statement "Administrative controls shall be developed," rather than the wording used in the Westinghouse STS, Section 5.2.2.d , "Administrative procedures shall be developed." This change in wording does not appear to add any value and unless there is an explanation, the STS wording should be used.

Provide the technical justification for the added wording in the EPR GTS, Section 5.2.2.f, that states "when the reactor is operating in MODE 1, 2, 3, or 4, ".

This additional information is needed to ensure the accuracy and completeness of the EPR GTS.

Response to Question 16-252:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 5.2.2.d will be changed from "Administrative controls" to "Administrative procedures"

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 5.2.2.f will be revised to delete "when the reactor is operating in MODE 1, 2, 3, or 4".

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 5.2.2 will be revised as described in the response and indicated on the enclosed mark-up.

Question .16-253:

The following are editorial and typographical errors discovered in the text of the EPR GTS, Section 5.0 should be corrected:

1. The EPR GTS, Section 5.5.3.c, the phrase " in accordance " should be inserted between "effluent" and "with".
2. The EPR GTS, Section 5.5.17.d, the third line begins with "MODE" when it should be lower case; as in "the pressurization mode of operation".
3. The EPR GTS, Section 5.6.1, the title of this section should be underlined.
4. The EPR GTS, Section 5.6.7 g, the word "insitu" should be spelled as two words; as "in situ."
5. The EPR GTS, Sections 5.7.1.b and 5.7.2.b, the word "Specification" is capitalized. The EPR GTS has adopted the practice of using this to indicate reference to a specific numbered Specification within the GTS. In this case, the word usage does not apply to another numbered GTS Section; therefore no capitalization is required.
6. The EPR GTS, Section 5.5.11, " Gaseous Waste Processing System Radioactivity Monitoring Program," last paragraph should refers to the "Gaseous Waste Processing System Radioactivity Monitoring Program" rather than the "Explosive Gas and Storage Tank Radioactivity Monitoring Program."

These editorials are needed to ensure the accuracy and completeness of the EPR GTS.

Response to Question 16-253:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Sections 5.5.3.c, 5.5.17.d, 5.6.1, 5.6.7.g, 5.7.1.b, 5.7.2.b, and 5.5.11 will be revised as defined by the question.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Sections 5.5.3.c, 5.5.17.d, 5.6.1, 5.6.7.g, 5.7.1.b, 5.7.2.b and 5.5.11 will be revised as described in the response and indicated on the enclosed mark-up.

Question 16-254:

Provide additional information for the following statement included in the EPR GTS, Section 5.5.10, Ventilation Filter Testing Program:

The design versus operational flowrate of AVS and SBVS appear inconsistent with the flowrate used in the EPR GTS. Also, the test tolerance appears to exceed the +10% tolerance. For each group of filter systems tested, identify the FSAR Table which lists the nominal flowrate upon which this test should be based. The tolerances listed for the heater capacities in the EPR GTS, Section 5.5.10.e appear to exceed the +/- 10%.

In the EPR GTS, Section 5.5.10.e lists two filter banks for CREF, outside air and emergency filter banks, but does not identify the two filter banks in EPR GTS, Section 5.5.10.a thru d. The names used in the EPR GTS, Section 5.5.10.e do not match the names provided in the EPR GTS, Section 3.7.10 which are identified as iodine filtration train and fresh air intake train. Provide a technical justification for these differences and incorporate changes to make them consistent.

This additional information is needed to ensure the accuracy and completeness of the EPR GTS.

Response to Question 16-254:

A response to this Question will be provided by March 19, 2009.

Question 16-255:

Provide technical justification for differences between the EPR Administrative Programs and the applicable STS Administrative Programs.

The EPR GTS, Section 5.5.2, identifies Low Head Safety Injection, Medium Head Safety Injection, and Nuclear Sampling as the systems used for Primary Coolant Sources Outside of Containment. The Westinghouse STS identify other system including CVCS, Recirc Spray, Safety Injection, gas stripper and Hydrogen recombiner included in their program. The EPR GTS and Westinghouse STS both include safety injection systems, provide a technical justification for not including comparable systems such as CVCS or gaseous removing systems in the mix of EPR systems that serve this function.

This technical justification is need to ensure the effectiveness and completeness of Primary Coolant Sources Outside of Containment.

Response to Question 16-255:

The leakage pathways listed in U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 5.5.2 are consistent with the dose analysis assumptions in U.S. EPR FSAR Tier 2, Section 15.0.3.11. The systems list reflects plant specific differences. The chemical and volume control system (CVCS) and containment spray systems for the U.S. EPR are not relied upon to mitigate any Design Basis Accident. The U.S. EPR does utilize passive autocatalytic recombiners inside containment for accident mitigation, but does not employ an external gas stripper or hydrogen recombiner.

FSAR Impact:

U.S. EPR FSAR TIER 2 will not be changed as a result of this response.

Question 16-256:

Provide a revision to the proposed GTS which correctly adopts the TSTF-490-A. Alternatively, if the FSAR Applicant wishes to remove this COL Applicant option, then also remove the following final words of the definition: " or similar source."

The proposed EPR GTS has apparently adopted the ITS change TSTF-490-A, Rev O which permits deletion of definition for E-Bar - AVERAGE DISINTREGATION ENERGY, modifies DOSE EQUIVALENT I-131, and adds a new definition, DOSE EQUIVALENT XE-133. This is an acceptable approach; however, the TSTF-490-A has not been incorporated correctly.

In the modification of DOSE EQUIVALENT I-131, the "Reviewer's Note" for adopting this proposed GTS definition to a future Applicant, depending upon whether the unit is eventually licensed to 10 CFR 100.11 or the unit is licensed to 10 CFR 50.67, has been removed. This will be important information for COL Applicants with multiple unit sites and with current NRC licenses. Provide a revision to the EPR GTS which replicates the reviewers notes, and incorporates the method for determining dose using the alternative thyroid dose conversions factor references.

This definition also has an typographical error in the identified Federal Guidance Report No. 11, which is missing the "L" in the first word of the title, "Limiting".

Response to Question 16-256:

The Reviewer's Note in TSTF-490-A allows this change to be applied to current operating plants that are licensed to 10 CFR 100.11 or to plants licensed after January 10,1997 that are to be licensed to 10 CFR 100.21. Since the U.S. EPR GTS are only applicable to the latter category of plants, the Reviewer's Note and the 10 CFR 100.11 references are not needed and were deleted as unnecessary information.

The definition of Dose Equivalent I-131 will be revised to correctly spell "Limiting". The phrase "or similar source" will be deleted from the definition of Dose Equivalent XE-133.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications will be revised as described in the response and indicated on the enclosed mark-up.

Question 16-257:

Provide the additional applicability requirement for Mode 2 with Keff < 1.0 to the Applicability for LCO 3.1.1 or provide a technical justification for not needing this additional applicability requirement to the Bases for 3.1.1.

The Bases for 3.1.1, Background Section, 4th paragraph, currently states that "when the unit is in Mode 2 with the reactor subcritical, Shut Down Margin (SDM) control is ensured by operating with the shutdown banks fully withdrawn and the control (banks) within the estimated critical condition" indicating the need for a Mode 2 subcritical SDM. Any justification for not including a Mode 2 subcritical applicability requirement would also need to consider clarification of the Bases for 3.1.1, Background Section, 4th paragraph.

The additional justification is needed to ensure the accuracy and completeness of the LCO.

Response to Question 16-257:

The applicability for U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.1.1 will be revised to add MODE 2 with k_{eff} <1.0,

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.1.1 will be revised as described in the response and indicated on the enclosed markup.

Question 16-258:

Incorporate the following editorials into the EPR GTS, Section 3.1 and Bases, Section B 3.1:

Provide consistent editorial use of the term Keff with appropriate subscripting in TS Section 3.1 and where used elsewhere throughout the Technical Specifications. Example: In LCO 3.1.3 Applicability, Moderator Temperature Coefficient (MTC), Keff is subscripted; however in Required Action B.1, Keff is not written with a subscript.

Revise the last paragraph of the EPR Bases, Section B 3.1.3, LCO Section, to ensure the terms "BOC positive (upper) limit" and the "EOC negative (lower) limit" are used consistently throughout the Bases and the Technical Specifications. These terms are amplified by stating the positive or negative "sign" of the value. The LCO establishes a maximum positive value that cannot be exceeded. The beginning of cycle (BOC) positive (upper) limit and the end of cycle (EOC) negative (lower) limit are established in the COLR to allow specifying limits for each particular cycle. It is suggested that the "positive (upper) limit" and "negative (lower) limit" are the more definitive and descriptive terms and should be used throughout the Technical Specification as appropriate. In addition, deleted the following sentence from this paragraph that states that "this permits the unit to take advantage of improved fuel management and changes in unit operating schedule."

Revise the EPR Bases, Section 3.1.4, SR 3.1.4.3 the last sentence of the first paragraph to include the word "than" after "testing at less."

Provide consistent editorial use of the term TAVG by subscripting and capitalization (or not) within the subscript, in TS Section 3.1 and where used elsewhere throughout the Technical Specifications. Example: In SR 3.1.4.3.a., the term Tavg is not subscripted and "avg" is lowercase; however, in LCO 3.1.7, Required Actions B.3, the term the term Tavg uses subscripts and lowercase; and in third paragraph of the Bases for Actions, B.1, B.2, and B.3, TAVG uses subscript with uppercase.

Revise the EPR Bases, Section B 3.1.7, Background Section second-to-last paragraph by removing the space between "RCCA" and "s."

These revisions are needed to ensure the accuracy and completeness of the EPR GTS and Bases.

Response to Question 16-258:

The U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications (TS) will be revised as follows:

TS Sections 3.1.1, 3.1.3, and 3.1.6 will be revised to include the term K_{eff}.

TS Sections 3.1.3 and Bases will be revised to include the positive (upper) limit and the negative (lower) limit.

The sentence "This permits the unit to take advantage of improved fuel management and changes in unit operating schedule" will be deleted from TS Bases Section 3.1.3.

The TS Bases for Surveillance Requirement (SR) 3.1.4.3 will be revised to add the word "than" after "testing at less."

TS Sections 3.1.4, 3.4.2, and Bases Sections 3.1.7 and 3.7.7 will be revised to include the term T_{avg}.

TS Bases Section 3.1.7 will be revised to remove the space between "RCCA" and "s".

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications appropriate Sections and Bases will be revised as described in the response and indicated on the enclosed markup.

Question 16-259:

Resolve the discrepancy between LCO 3.1.3 b that specifies a maximum MTC equal to 0 pcm/oF when THERMAL POWER ≥ 50% and the third paragraph of the Background section of Bases 3.1.3 which states that "beginning of cycle (BOC) MTC is less than zero when THERMAL POWER ≥ 50% RTP." Determine if any changes are need to address related inconsistencies between EPR GTS and Bases and the applicable information in the EPR FSAR.

This additional information and any changes are needed to ensure the accuracy, completeness and consistency amongst the EPR GTS, Bases, and FSAR.

Response to Question 16-259:

The U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications (TS) will be revised as follows:

TS Bases Section 3.1.3 Background will be revised to note that:

“Both initial and reload cores are designed so that the beginning of cycle (BOC) MTC is less than or equal to zero when THERMAL POWER is ≥ 50% RTP."

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Bases Section 3.1.3 will be revised as described in the response and indicated on the enclosed mark-up.

Question 16-260:

Provide an additional SR 3.1.3.2 surveillance requirement or a technical justification for not including a Beginning-of-Cycle (BOC) negative Moderator Temperature Coefficient (MTC) value verification as described in NUREG 1432, "Standard Technical Specifications Combustion Engineering Plants," that states that "within 7 days after reaching 40 effective full power days and a 2/3 core burnup, satisfies the confirmatory check of the most negative MTC value. . . ."

The additional surveillance requirement or technical justification is needed to ensure the completeness of SR 3.1.3.2.

Response to Question 16-260:

The Combustion Engineering (CE) Technical Specification Surveillance requirement states that "within 7 days after reaching 40 effective full power days and a 2/3 core burnup, satisfies the confirmatory check of the most negative MTC value…." This action is not appropriate for the 18-month EPR cycle 1 core or potential 24 month cycles where heavy loading of gadolinia fuel is involved. In these types of cores the gadolinia fuel depletes and the soluble boron increases from the beginning of cycle (BOC) conditions to the Most Reactive Exposure (MRE) point which is typically around 113 effective full power days (EFPD) (refer to U.S. EPR FSAR Tier 2, Figure 4.3-59). At this time in core life, the most positive moderator temperature coefficient (MTC) at hot full power (HFP) conditions is achieved. After this point the MTC becomes (less positive) more negative with core burnup. It may not be until about 200 EFPD before MTC behavior becomes linear with exposure allowing extrapolation. The CE TS requirement would not be useful for the U.S. EPR and was removed. The Westinghouse TS requirement also associates the MTC check to a soluble boron concentration of 300 ppm boron. The U.S. EPR core uses enriched boron B-10 that has been increased to 37 percent by isotopic weight. The isotopic B-10 abundance is contained in the Core Operating Limits Report (COLR) and could be changed on a cycle specific basis. For the U.S. EPR, the CE TS requirement is used with a burnup trigger for the end of life (EOL) MTC test instead of the Westinghouse TS model that would be impacted by boron enrichment changes.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 16-261:

Correct the SR 3.1.4.2 surveillance specification for "moving each RCCA not fully inserted in the core ≥ 16 steps in either direction." This surveillance states that each Rod Control Cluster Assembly (RCCA) not fully inserted should be moved greater than or equal to 16 steps in either directions. The Bases for 3.1.4, Surveillance Requirements, 2nd paragraph, states that "moving each control RCCA by 16 steps will not cause radial or axial tilts, or oscillations to occur." This Bases statement implies a maximum movement of 16 steps.

This change in information is needed to ensure the accuracy of the EPR GTS.

Response to Question 16-261:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications, Surveillance Requirement 3.1.4.2 and Bases will be modified to reflect a range of 16 to 20 steps.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications will be revised as described in the response and indicated on the enclosed mark-up.

Question 16-262:

Provide a definition for the term "bite position" as used only in the EPR Bases, Section B 3.1.4, RCCA Group Alignment Limits, Applicable Safety Analyses, second sentence in the fourth paragraph.

This additional information is needed to ensure the accuracy and completeness of the EPR Bases.

Response to Question 16-262:

The definition of “bite” will be added to U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Bases Section 3.1.4. The term “bite” corresponds to the rod position that minimizes the time for negative reactivity addition when the rods are inserted.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications will be revised as described in the response and indicated on the enclosed mark-up.

Question 16-263:

Provide the additional information necessary to technically justify the interpretation of the "Note" in the Applicability Section of LCO 3.1.6 that "allows the LCO to be not applicable during a partial trip, during loss of load events" as stated in the Bases for 3.1.6, Applicability Section, 2nd paragraph. An appropriate note would need to be developed to clarify the intent to remove the requirement of this LCO during a partial trip and would be contrary to the existing basis that states that "this condition is outside of the control bank insertion limits and requires prompt action to restore operation within insertion limits." If a clarifying note is developed, a technical justification will need to be added to the Bases for 3.1.6 adequately justifying the exception.

This additional information is needed to ensure the accuracy and completeness of EPR GTS and Bases.

Response to Question 16-263:

The note in U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Bases Section 3.1.6 pertaining to partial trip will be deleted because partial trips have not been evaluated and are not in the U.S. EPR requirements.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications will be revised as described in the response and indicated on the enclosed mark-up.

Question 16-264:

Provide the additional information needed for clarification or the necessary changes to correct the discrepancy between LCO 3.1.6, Required Action C.1, that requires "Be in Mode 3" and the Bases for 3.1.6, Action Section, last paragraph, that states that "the plant must be brought to Mode 2 with Keff < 1.0."

This additional information or changes are needed to ensure the accuracy and completeness of EPR GTS and Bases.

Response to Question 16-264:

This GTS is the same as the NUREG-1431. No change will be made other than the editorial changes and the corrections from Question 16-258.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications will be revised as described in the response to Question 16-258 and indicated on the enclosed mark-up.

Question 16-265:

Provide the additional information needed for clarification or the necessary changes to correct the discrepancy between LCO 3.1.7, Required Action A.1.2, that requires the use of the Self-Powered Neutron Detectors (SPNDs) to verify RCCA position, and the Bases for 3.1.7, Action Section, 3rd paragraph, that refers to the use of the Aeroball Measurement System (AMS) for such purpose. If SPNDs is to be used to verify RCCA position, a discussion should be added to the Bases describing how readings from the SPNDs will accurately measure RCCA position.

This additional information or changes are needed to ensure the accuracy and completeness of EPR GTS and Bases.

Response to Question 16-265:

The U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications, Required Action A.1.2 of Section 3.1.7, will be revised to read “Verify position of RCCAs with inoperable position detectors indirectly using the Aeroball Measurement System.”

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications will be revised as described in the response and indicated on the enclosed mark-up.

Question 16-266:

Provide a justification for not requiring verification of RCCA position under LCO 3.1.7, Condition B, with "one or more banks with two or more analog RCCA position indicators inoperable" using incore instrumentation to verify RCCA position.

This additional information is needed to ensure accuracy and completeness of the EPR GTS.

Response to Question 16-266:

When “one or more banks with two or more analog RCCA position indicators inoperable” exist, it is the intent of the Bases that the actions of U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.1.7, Required Actions for Condition A, be implemented in addition to the Required Actions for Condition B and within their intended time-frames.

A Required Action will be added for U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.1.7 Condition B to implement the Required Actions for Condition A.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications will be revised as described in the response and indicated on the enclosed mark-up.

Question 16-267:

Provide a technical justification for not having a Required Action associated with LCO 3.1.7 to determine rod position when one or more RCCAs with inoperable position indicators have been moved and current RCCA's position is unknown.

This additional information is needed to ensure accuracy and completeness of the EPR GTS.

Response to Question 16-267:

The conditions associated with U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.1.7, Condition A, cover those where "one or more RCCAs with inoperable position indicators have been moved and the current position indication is unknown." Since there are at least one or more inoperable position indicators, the Action response of U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.1.7, Condition A, applies regardless of whether or not rods have been moved.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 16-268:

Provide a justification for not including an additional Required Action for LCO 3.1.8 to restore the Operability of the Volume Control Tank (VCT) and letdown isolation valves.

This additional information is needed to ensure accuracy and completeness of the EPR GTS.

Response to Question 16-268:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.1.8 and Bases will be revised to add an additional Required Action for to restore the operability of the volume control tank (VCT) and letdown isolation valves.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications will be revised as described in the response and indicated on the enclosed mark-up.

Question 16-269:

Provide a justification or change to the "Frequency" of 24 months for SR 3.1.8.2 to be consistent with the Bases for 3.1.8, Surveillance Requirement Section, that identifies "the need to perform this Surveillance under conditions that apply during a plant outage," rather than a set period basis.

This additional information is needed to ensure accuracy and completeness of the EPR GTS and Bases

Response to Question 16-269:

Performing U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications SR 3.1.8.2 during an operating cycle would realign the charging pump suction valves from the volume control tank (VCT) to the in-containment refueling water storage tank (IRWST) and potentially cause a significant reactivity change. The justification for performing the surveillance during a plant outage is equivalent to the following NUREG-1431 Bases for SR 3.5.2.5 and SR 3.5.2.6: the same valves (VCT outlet) must realign for the Westinghouse plants that use a charging / safety injection pump for charging during normal plant operation.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 16-270:

Provide a justification for no surveillance requirement specifically linked to physics testing under SR 3.1.9 (or Table 3.3.1-1) requiring the performance of an OPERABILITY verification of the intermediate range and power range channels.

This additional information is needed to ensure accuracy and completeness of the EPR GTS and Bases.

Response to Question 16-270:

The U.S. EPR uses a SENSOR OPERABILITY TEST for the power and intermediate range divisions that are primarily composed of digital components. U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Surveillance Requirement 3.1.9.3 “Perform a SENSOR OPERATIONAL TEST on power range and intermediate range divisions” will be added.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications will be revised as described in the response and indicated on the enclosed mark-up.

Question 16-271:

Provide a justification for not specifying in LCO 3.1.9 the limit on RCS loop temperature discussed in the Bases for 3.1.9, Applicable Safety Analyses Section, last sentence in the 2nd paragraph, which states "the fuel design criteria are preserved as long as, reactor coolant temperature is maintained ≥ 568°F ."

This additional information is needed to ensure accuracy and completeness of the EPR GTS and Bases.

Response to Question 16-271:

The Westinghouse Technical Specifications (NUREG-1431) has a limited temperature band between the pressurizer safety valves lift setpoints and the minimum temperature for criticality to perform an adequate isothermal temperature coefficient (ITC) test. A common practice is to invoke the Special Test Exception for minimum temperature during the ITC test and allow the plant to go below the normal minimum temperature for criticality. The U.S. EPR has 10°F margin between the normal hot zero power temperature and the minimum temperature for criticality as opposed to 6°F for a typical Westinghouse plant, thus there is no need to invoke a Special Test Exception to allow a lower RCS temperature.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 16-272:

Provide a justification for LCO 3.1.9 not allowing the suspension of LCO 3.4.2 "RCS Minimum Temperature for Criticality" during performance of physics tests. The Bases for 3.1.9, Background Section, 2nd-to-last paragraph, states that the "performance of this test could challenge LCO 3.4.2, RCS Minimum Temperature for Criticality."

This additional information is needed to ensure accuracy and completeness of the EPR GTS and Bases.

Response to Question 16-272:

The U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Bases Section 3.1.9, Background section will be revised to indicate that performance of this test should not challenge LCO 3.4.2.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Bases Section 3.1.9 will be revised as described in the response and indicated on the enclosed mark-up.

Question 16-273:

Provide addition clarification that will distinguish between the Self-Powered Neutron Detectors (SPNDs) and the "fixed incore instrumentation" described in the Bases for 3.2.1, Background Section, 8th paragraph, or confirm that they are the same instrumentation. Revise the Bases document to include this additional information.

This additional information is needed to ensure accuracy and completeness of the EPR GTS and Bases.

Response to Question 16-273:

The fixed incore instrumentation in this paragraph refers to the self-powered neutron detectors (SPND). The phrase “and fixed incore instrumentation” will be removed for clarification.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Bases Section 3.2.1 will be revised as described in the response and indicated on the enclosed markup.

Question 16-274:

Determine if it would be more accurate to revise the wording in the second-to-last paragraph of the Background section of Bases for 3.2.1 from "first LPD LCO 1 threshold" to "First High LPD LCO 1 level."

This additional information is needed to ensure accuracy and completeness of the EPR GTS and Bases.

Response to Question 16-274:

The wording will be revised in the Background in U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Bases Section 3.2.1 from "first LPD LCO 1 threshold" to "First High LPD LCO 1 level."

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Bases Section 3.2.1 will be revised as described in the response and indicated on the enclosed markup.

Question 16-275:

The EPR GTS, LCO 3.2.1, Action B.1 states ≤ 10% RTP. The EPR Bases, Section 3.2.1, Action B.1 states from "< 10% RTP." Determine the correct RTP and revise the EPT GTS and EPR Bases accordingly.

This additional information is needed to ensure accuracy and consistency of the EPR GTS and Bases.

Response to Question 16-275:

The U.S. EPR Tier 2, Chapter 16 Technical Specifications Bases Section 3.2.1, Action B.1 will be changed from “<” to “≤” for accuracy and consistency of the U.S. EPR GTS and Bases.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Bases Section 3.2.1 Action B.1 will be revised as described in the response and indicated on the enclosed markup.

Question 16-276:

Provide additional justification for the 12 hour surveillance frequency for SR 3.2.1.1 with the Reactor Control Surveillance and Limitation System (RCSL) and its associated alarm not in service.

As stated in the Bases for 3.2.1, Surveillance Requirements, the 12 hour frequency is based on the ability to identify trends that could result in an approach to the Axial Offset (AO) limits. This statement, in itself, is not a sufficient justification for the 12 hour frequency. Operating history from current plant designs show that power distribution limits can be exceed in a shorter time frame. Note: NUREG 1432, "Standard Technical Specifications Combustion Engineering Plants" requires a two hour frequency for the similar application, under SR 3.2.1.1.

This additional information is needed to ensure accuracy and consistency of the EPR GTS and Bases.

Response to Question 16-276:

The U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Bases Section 3.2.1 refers to the Linear Power Density (LPD) limits, and not Axial Offset limits. The 12 hour surveillance frequency is intended to be conducted by the operators with the Reactor Control Surveillance and Limitation (RCSL) System in service to provide a redundant check of the LPD limits.

When the RCSL System is not in service, the Surveillance Requirement will be modified to be met within an hour of the RCSL being declared inoperable and occur once per hour thereafter until the RCSL System is back in service. This allows the operator to monitor trends in the LPD limits when the RCSL System is not in service. If the LPD is not within limits and the RCSL System is out of service, the operator must take immediate action to reduce power until the most limiting LPD is within the LPD limits.

The U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.2.1 and Bases will be revised by adding an action statement to address this issue.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.2.1 and Bases will be revised as described in the response and indicated on the enclosed markup.

Question 16-277:

Incorporate the following editorials into the EPR GTS and Bases as applicable:

The EPR Bases, Section B 3.2.2, Background Section, in the first paragraph change "an addition," to "in addition."

The EPR Bases, Section B 3.2.4, Background Section, in the first sentence of the third paragraph change "nuclear" to "neutron."

This revision is needed to ensure accuracy and consistency of the EPR GTS and Bases.

Response to Question 16-277:

The following changes to the U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications (TS) will be incorporated:

TS Bases Section 3.2.2, Background Section, in the first paragraph change "an addition," to "in addition."

TS Bases Section 3.2.4, Background Section, in the first sentence of the third paragraph change "nuclear" to "neutron."

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Bases Sections 3.2.2 and 3.2.4 will be revised as described in the response and indicated on the enclosed markup.

Question 16-278:

Clarify the purpose for LCO 3.2.2 regarding FΔHN. Various statements in the Bases for 3.2.2 appear inconsistent relative to the types of design basis events that are covered by the FΔHN LCO.

For example, the Bases for 3.2.2, Background Section, 6th paragraph states that "operation outside the LCO limits may produce unacceptable consequences if an anticipated operational occurrence (AOO) or other postulated accident occurs;" and the Applicable Safety Analyses section states that the FΔHN LCO "limits the scope of power distributions from which an accident may be initiated for all FSAR Chapter 15 events." The 3.2.2 Bases, Applicability Section, states that "this LCO applies only to LOCA analyses."

This additional information is needed to ensure accuracy of the EPR Bases.

Response to Question 16-278:

The U.S. EPR Tier 2, Chapter 16, Technical Specifications Bases Section 3.2.2, Applicable Safety Analyses Note b states the FΔHN LCO is used to "Limit the scope of power distributions from which an accident may be initiated for all FSAR Chapter 15 events." This includes all of Mode 1 operation. To be consistent with U.S. EPR FSAR Tier 2, Chapter 15 analyses, the Applicability section of U.S. EPR Tier 2, Technical Specifications Sections 3.2.2 and B.3.2.2 will be revised to include Mode 1 operation.

The Surveillance Requirements (SR) will be updated from 15 effective full power days (EFPD) to 31 EFPD to be consistent with the SR 3.2.4.2. A frequency of 31 EFPD is acceptable because the power distribution changes only slightly with the amount of fuel burnup. The FΔHN is first determined after each fuel loading when the THERMAL POWER is greater than 40 percent of rated thermal power (RTP), but prior to its exceeding 70 percent RTP, verifying that the core is properly loaded.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Sections 3.2.2 and Bases will be revised as described in the response and indicated on the enclosed markup.

Question 16-279:

Provide a technical justification for the 12 hour surveillance frequency for SR 3.2.3.1 with the Reactor Control Surveillance and Limitation System (RCSL) and its associated alarm not in service.

As stated in the Bases for 3.2.3, Surveillance Requirements, the 12 hour frequency is based on the ability to identify trends that could result in an approach to the DNBR limits. This statement, in itself, is not a sufficient justification for the 12 hour frequency. Operating history from current plant designs show that power distribution limits can be exceed in a shorter time frame.

This additional information is needed to ensure accuracy and consistency of the EPR GTS and Bases.

Response to Question 16-279:

The 12 hour surveillance frequency is intended to be conducted by the operators with the Reactor Control Surveillance and Limitation (RCSL) System in service to provide a redundant check of the DNBR limits.

When the RCSL System is not in service, the Surveillance Requirement will be modified to be met within an hour of the RCSL being declared inoperable and occur once per hour thereafter until the RCSL System is back in service. This allows the operator to monitor trends of the DNBR limits when the RCSL System is not in service. If the departure from nucleate boiling ratio (DNBR) is not within limits and the RCSL System is out of service, the operator is required to take immediate action to reduce power until the most limiting DNBR is within limits.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Sections 3.2.3 and Bases will be revised as described in the response and indicated on the enclosed markup.

Question 16-280:

Identify any additional design basis Anticipated Operational Occurences (AOO), other than Loss of Coolant Flow event that, although not typical, is the most limiting event relative to maximum Δ DNBR.

The statement in the Bases for 3.2.3, Applicable Safety Analyses, 2nd paragraph, states that Loss of Coolant Flow is typically the limiting Δ DNBR event and therefore provides conservative limits for all other AOOs. For an event to provide conservative limits for all other AOOs, it has to be the most limiting. Revise the Bases document to include this clarification or additional information.

This clarification or additional information is needed to ensure the accuracy and completeness of the EPR Bases.

Response to Question 16-280:

The maximum change in the Departure from Nucleate Boiling Ratio (ΔDNBR) is evaluated for accidents in U.S. EPR FSAR Tier 2, Chapter 15 Safety Analyses that utilize the methodology describe in ANP-10287P “Incore Trip Setpoint and Transient Methodology for U.S. EPR Topical Report.” The change in ΔDNBR is addressed in detail by this topical report. The maximum ΔDNBR is evaluated for the rapid hot full power Anticipated Operational Occurrence (AOO) transient events including (but not limited to) Loss of Coolant Flow, Increase in Steam Flow, and Uncontrolled Control Rod Assembly Withdrawal at Power. The U.S. EPR FSAR Tier 2, Chapter 15 Safety Analyses show that the maximum ΔDNBR for an AOO is the Complete Loss of Coolant Flow event. The U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Bases Section 3.2.3 will be revised by adding a clarifying statement to address this issue.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Bases Section 3.2.3 will be revised as described in the response and indicated on the enclosed markup.

Question 16-281:

Provide the additional information or clarification with regard to the use of "accident," "accident analyses," "AOOs," and "postulated accidents." The Bases for 3.2.4, Background Section, 1st and 2nd paragraphs, refer only to "accident" and "accident analyses," whereas the same Bases, Applicable Safety Analyses Section refers to "AOOs" and "postulated accidents." Include AOOs in the Background Section.

This information is needed to ensure accuracy and completeness of the EPR Bases.

Response to Question 16-281:

The use of Anticipated Operational Occurrences (AOOs) and “postulated accidents” will be included in the Background Section of the U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Bases Section 3.2.4.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Bases Section 3.2.4 will be revised as described in the response and indicated on the enclosed markup.

Question 16-282:

Provide a technical justification for the 12 hour surveillance frequency for SR 3.2.4.1 with the Reactor Control Surveillance and Limitation System (RCSL) and its associated alarm not in service.

As stated in the Bases for 3.2.4, Surveillance Requirements, the 12 hour frequency is based on the ability to identify trends that could result in an approach to the AO limits. This statement, in itself, is not a sufficient justification for the 12 hour frequency. Operating history from current plant designs show that power distribution limits can be exceed in a shorter time frame.

This additional information is needed to ensure accuracy and consistency of the EPR GTS and Bases.

Response to Question 16-282:

The 12 hour surveillance frequency is intended to be conducted by the operators with the Reactor Control Surveillance and Limitation (RCSL) System in service to provide a redundant check of the Axial Offset (AO) limits.

When the RCSL System is not in service, the Surveillance Requirement will be modified to be met within an hour of the RCSL being declared inoperable and occur once per hour thereafter until the RCSL System is back in service. This allows the operator to monitor trends in the AO limits when the RCSL System is not in service. If the AO is not within limits and the RCSL System is out of service, the operator must take immediate action to reduce power until the most limiting AO is within the AO limits. The U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.2.4 and Bases will be revised by adding an action statement and a surveillance requirement to address this issue.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Sections 3.2.4 and Bases will be revised as described in the response and indicated on the enclosed markup.

Question 16-283:

This question has been intentionally deleted by the NRC.

Response to Question 16-283:

Not applicable.

FSAR Impact:

Not applicable.

Question 16-284:

Provide additional information to verify the accuracy of the information in the EPR Bases, Section B 3.2.1.

The EPR Bases, Section B 3.2.1, Applicability Section states that "power distribution is a concern any time the reactor is critical. The power distribution LCOs, however, are only applicable in MODE 1 above 10% RTP. This LCO is not a concern below 10% RTP because the core is operating well below its thermal limits." Confirm that it would be more accurate if the last sentence of this quote was worded to say "at or below 10%."

This additional information is needed to ensure the accuracy completeness of the EPR Bases for 3.2.1

Response to Question 16-284:

The U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Bases for Section 3.2.1, Applicability Section will be revised to clarify that the LPD LCO is applicable for power above 10 percent of rated thermal power (RTP).

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Bases Section 3.2.1 will be revised as described in the response and indicated on the enclosed markup.

Question 16-285:

Provide additional information to adequately describe the bases or reasons for information in the EPR Bases, Section B 3.2.1.

The EPR Bases, Section B 3.2.1, Background Section, 12th paragraph states that "the SPND signal gradually increases (conservative) and the gain constants must be periodically recalibrated to prevent unnecessary LPD penalties." Provide the additional information necessary to explain the reason for the Self-Powered Neutron Detector (SPND) signal (has to [can it decrease or go unchanged]) gradually increases between calibrations. Revise the Bases document to include this additional information.

This additional information is needed to ensure the accuracy completeness of the EPR Bases for 3.2.1.

Response to Question 16-285:

The self powered neutron detector (SPND) signal to flux ratio gradually increases due to buildup of Co-60 and Co-61. This effect is greater than the decrease in sensitivity due to Co-59 burnup. As a result, the SPND signal will become more conservative with burnup.

From the time of calibration, a decreasing (non-conservative) SPND signal can result from changes in core power distributions due to core burnup. The signal variation due to power distribution changes can cause either a decreased or increased SPND signal. This variation in the SPND signal requires periodic recalibration to minimize uncertainties. The setpoint analyses account for the uncertainty inherent for a given calibration frequency.

The U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Bases Sections 3.2.1 and 3.2.3 will be revised by adding clarifying paragraphs to address this issue. The U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Bases Section 3.2.4 will be revised by modifying existing statements to clarify this issue.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Bases Sections 3.2.1, 3.2.3, and 3.2.4 will be revised as described in the response and indicated on the enclosed markup.

Question 16-286:

Provide additional information to adequately describe the bases or reasons for information in the EPR Bases, Section B 3.2.1.

The EPR Bases, Section B 3.2.1, Background Section, 10th paragraph states that "after calibration, all twelve SPNDs within the same axial slice therefore provide the same value, which corresponds to the maximum linear power density value for that axial slice." Identify and justify the conditions for all twelve SPNDs within the same axial slice providing the same value after calibration.

This additional information is needed to ensure the accuracy and completeness of the EPR Bases for 3.2.1.

Response to Question 16-286:

At the time of calibration, each of the self powered neutron detectors (SPND) within the same axial elevation are adjusted by a unique calibration factor to indicate the maximum power density within the axial slice as determined by the 3-D power distribution.

A visual example of the SPNDs for Linear Power Density (LPD) calibration principle is shown in Figure 16-286-1—SPND Calibration for LPD. The calibration principle for SPND is found in Appendix B of ANP-10287P, “Incore Trip Setpoint and Transient Methodology for U.S. EPR Topical Report.”

The U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Bases Section 3.2.1 will be revised by modifying an existing statement to clarify this issue.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Bases Section 3.2.1 will be revised as described in the response and indicated on the enclosed markup.

[ ]

Figure 16-286-1—SPND Calibration for LPD

Question 16-287:

Provide additional information to adequately describe the bases or reasons for information in the EPR Bases, Section B 3.2.1.

Provide the additional information necessary to adequately explain the following statement from the EPR Bases, Section B 3.2.1, Background Section, 7th paragraph: "such a condition signifies a reduction in the capability of the plant to withstand an AOO or postulated accident, but does not necessarily imply a violation of fuel design limits." Revise the Bases document to include this additional information.

This additional information is needed to ensure the completeness of the EPR Bases, Section B 3.2.1.

Response to Question 16-287:

The U.S. EPR FSAR Bases for Section 3.2.1, Background Section specifies a condition where the Limiting Condition for Operation (LCO) limit is exceeded. A violation of the LCO signifies a reduction in the capability of the plant to withstand an anticipated operational occurrence (AOO) or postulated accident. This is due to the fact that an AOO initiated from this condition has less margin to the safety limit than was assumed in the safety analysis. This does not necessarily imply a violation of fuel design limits because the Protection System (PS) initiates a reactor trip when the reactor trip setpoint is reached. The U.S. EPR FSAR Bases for Section 3.2.1, Background Section will be revised by modifying a Bases statement to address this issue.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Bases Section 3.2.1 will be revised as described in the response and indicated on the enclosed markup.

Question 16-288:

Provide additional information to verify the accuracy of the information in the EPR Bases, Section B 3.2.2.

The EPR Bases, Section B 3.2.2, Applicability Section, 2nd paragraph currently states the following: "The LOCA safety analysis indirectly models $F^{N}{}_{\Delta H}$ as an input parameter. The Nuclear Heat Flux Hot Channel Factor (FQ(Z)) and the axial peaking factors are inserted directly into the LOCA safety analyses that verify the acceptability of the resulting peak cladding temperature (Ref. 1). Include additional information that explains the relationship between FΔHN and FQ (z).

This additional information is needed for clear understand of $F^{N}{}_{\Delta H}$ in the EPR Bases.

Response to Question 16-288:

The nuclear enthalpy rise hot channel factor ($F_{\Delta H}{}^{N}$) is used directly in the loss of coolant accident (LOCA) analysis to verify the acceptability of the resulting peak cladding temperature.

The relationship between the Nuclear Heat Flux Hot Channel Factor ($F_Q(Z)$) and the axial peaking factors ($F_Z(Z)$) in relationship to the radial peaking factor ($F_{\Delta H}{}^{N}$ {or FΔHN as stated in the question}) is defined as: $F_Q(Z) = F_{\Delta H}{}^{N} \cdot F_Z(Z)$. However, the mention of the Nuclear Heat Flux Hot Channel Factor and axial peaking factors will be removed for clarity.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Bases Section 3.2.2 will be revised as described in the response and indicated on the enclosed markup.

Question 16-289:

Provide additional information to adequately describe the bases or reasons for information in the EPR Bases, Section B 3.2.3.

The EPR Bases, Section B 3.2.3, Background Section, 11th paragraph states that "after calibration, all twelve SPND fingers therefore provide the same axial power shape representative of the power shape of the actual hot channel." Provide an explanation to identify and justify the conditions for all twelve SPNDs within the same axial slice will provide the same value after calibration.

This additional information is needed to ensure the accuracy and completeness of the EPR Bases for 3.2.3.

Response to Question 16-289:

The U.S. EPR Bases Section 3.2.3 in the background section specifies a calibration is performed for each of the self powered neutron detectors (SPND). This statement is modified to state that the SPND are adjusted within the same "finger" by a unique calibration factor to indicate the power density of the hot rod as determined by the 3-D power distribution.

A visual example of the SPNDs for departure from nucleate boiling (DNBR) are shown in Figure 16-289-1—SPND Calibration for DNBR. The calibration principle for SPND is found in Appendix B of ANP-10287P, "Incore Trip Setpoint and Transient Methodology for U.S. EPR Topical Report."

The U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Bases Section 3.2.3 will be revised by modifying an existing statement to clarify this issue.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Bases Section 3.2.3 will be revised as described in the response and indicated on the enclosed markup.

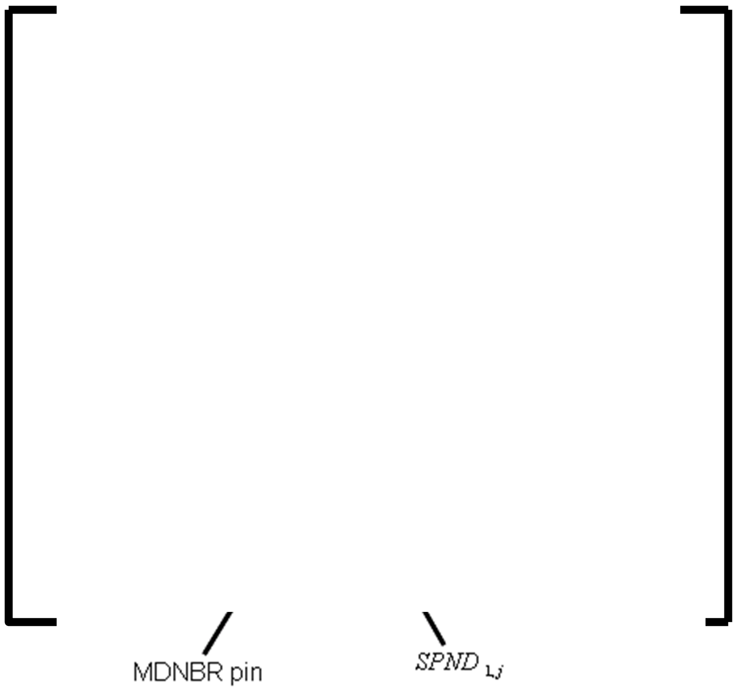

Figure 16-289-1—SPND Calibration for DNBR

Question 16-290:

Provide additional information to adequately describe the bases or reasons for information in the EPR Bases, Section B 3.2.3.

The EPR Bases, Section B3.2.3, Background Section, 8th paragraph states that "such a condition signifies a reduction in the capability of the plant to withstand an AOO or postulated accident, but does not necessarily imply a violation of fuel design limits." Provide addition information necessary to more clearly explain this statement. Revise the Bases document to include this additional information.

This additional information is needed to ensure the accuracy and completeness of the EPR Bases Section B 3.2.3.

Response to Question 16-290:

The U.S. EPR FSAR Bases for Section 3.2.3, Background Section specifies a condition where the Limiting Condition for Operation (LCO) limit is exceeded. A violation of the LCO signifies a reduction in the capability of the plant to withstand an anticipated operational occurrence (AOO) or postulated accident. This is due to the fact that an AOO initiated from this condition has less margin to the safety limit than was assumed in the safety analysis. This does not necessarily imply a violation of fuel design limits because the Protection System (PS) initiates a reactor trip when the reactor trip setpoint is reached. The U.S. EPR FSAR Bases for Section 3.2.3, Background Section will be revised by modifying a Bases statement to address this issue.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Bases Section 3.2.3 will be revised as described in the response and indicat`ed on the enclosed markup.

U.S. EPR Final Safety Analysis Report Markups

16-244

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.
ACTUATING DEVICE OPERATIONAL TEST (ADOT)	An ADOT shall consist of operating the ~~trip~~ actuating device and verifying the OPERABILITY of all devices in the division required for ~~trip~~ actuating device OPERABILITY. The ADOT may be performed by means of any series of sequential, overlapping, or total division steps.
AXIAL OFFSET (AO)	AO 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. $$AO = ((Upper - Lower) / (Lower + Upper)) * 100$$
AZIMUTHAL POWER IMBALANCE (API)	AZIMUTHAL POWER IMBALANCE shall be the maximum of the difference between the maximum power generated in any core quadrant (QN_{max}) and the minimum power generated in any core quadrant (QN_{min}), as measured by the power range excore detectors. $$API = QN_{\max} - QN_{\min}$$
CALIBRATION	A CALIBRATION shall be the adjustment, as necessary, of the sensor output such that it responds within the necessary range and accuracy to known values of the parameter that the sensor monitors. The CALIBRATION shall encompass all devices in the division required for sensor OPERABILITY. CALIBRATION of instrument divisions with resistance temperature detector (RTD) or thermocouple sensors may consist of an inplace qualitative assessment of sensor behavior and normal CALIBRATION of the remaining adjustable devices in the division. The CALIBRATION may be performed by means of any series of sequential, overlapping, or total steps.

16-244

1.1 Definitions

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 to other indications or status derived from independent instrumentation channels measuring the same parameter.
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 5.6.3. Plant operation within these limits is addressed in individual Specifications.
~~DIVISION OPERATIONAL TEST (DOT)~~	~~A DOT shall be the injection of a simulated or actual signal into the division as close to the sensor as practicable to verify OPERABILITY of all devices in the division required for OPERABILITY. The DOT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints required for division OPERABILITY such that the setpoints are within the necessary range and accuracy. The DOT may be performed by means of any series of sequential, overlapping, or total steps.~~
DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries per gram) that alone would produce the same dose when inhaled as the combined activities of iodine isotopes I-131, I-132, I-133, I-134, and I-135 actually present. The determination of DOSE EQUIVALENT I-131 shall be performed using thyroid dose conversion factors from EPA Federal Guidance Report No. 11, 1988, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion."

16-256

16-244

1.1 Definitions

DOSE EQUIVALENT XE-133	DOSE EQUIVALENT XE-133 shall be that concentration of Xe-133 (microcuries per gram) that alone would produce the same acute dose to the whole body as the combined activities of noble gas nuclides Kr-85m, Kr-85, Kr-87, Kr-88, Xe-131m, Xe-133m, Xe-133, Xe-135m Xe-135, and Xe-138 actually present. If a specific noble gas nuclide is not detected, it should be assumed to be present at the minimum detectable activity. The determination of DOSE EQUIVALENT XE-133 shall be performed using effective dose conversion factors for air submersion listed in Table III.1 of EPA Federal Guidance Report No. 12, 1993, "External Exposure to Radionuclides in Air, Water, and Soil" or the average gamma disintegration energies as provided in ICRP Publication 38, "Radionuclide Transformations" ~~or similar source~~.
EXTENDED SELF TESTS	Testing of the Protection System signal processors that cannot be performed during power operation are performed during the start-up of a computer. These tests can also be initiated by pushing a reset button on the computer. These tests include a basic hardware test using the internal diagnostics monitor, a self-test of the operating system, and basic hardware tests.

16-256

16-244

1.1 ~~—~~ Definitions

LEAKAGE

LEAKAGE shall be:

a. Identified LEAKAGE

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

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

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

b. Unidentified LEAKAGE

All LEAKAGE (except RCP seal water injection or leakoff) that is not identified LEAKAGE; and

c. Pressure Boundary LEAKAGE

LEAKAGE (except ~~SG~~ 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.

OPERABLE – OPERABILITY

A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY 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).

16-244

1.1 ~~—~~_Definitions

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 FSAR Chapter 14, "Verification Programs"; 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 and the low temperature overpressure protection setpoints, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.4, "Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)."
PROTECTION SYSTEM (PS) RESPONSE TIME	The PS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its PS trip setpoint at the division sensor until the PS equipment is capable of performing its safety function (i.e., loss of stationary gripper coil voltage, 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.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 4590 MWt.
SENSOR OPERATIONAL TEST (SOT)	A SOT shall be the injection of a simulated or actual signal into the division as close to the sensor as practicable to verify OPERABILITY of all devices in the input circuit required for OPERABILITY. The SOT shall include the verification of the accuracy and time constants of the analog input modules. The SOT may be performed by means of any series of sequential, overlapping, or total steps.

16-244

1.1 Definitions

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 rod cluster control assemblies (RCCAs) are fully inserted except for the single RCCA of highest reactivity worth, which is assumed to be fully withdrawn. However, with all RCCAs verified fully inserted by two independent means, it is not necessary to account for a stuck RCCA in the SDM calculation. With any RCCA not capable of being fully inserted, the reactivity worth of the RCCA must be accounted for in the determination of SDM; and b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the nominal zero power design level.
STAGGERED TEST BASIS	A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, trains, divisions, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, trains, divisions, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, divisions, 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.

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16-244

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 indentations 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.

EXAMPLES

The following examples illustrate 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.

16-244

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.

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16-244 single line

16-244
2 spaces
no indent

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 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 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 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.

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

16-244

1.3 ~~—~~_Completion Times

DESCRIPTION (continued)

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 . . ."

EXAMPLES

The following examples illustrate 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 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

16-244

1.3 Completion Times

EXAMPLES (continued)

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 be in MODE 3 within 6 hours AND in MODE 5 within 36 hours. A total of 6 hours is allowed for reaching MODE 3 and a total of 36 hours (not 42 hours) is allowed for reaching MODE 5 from the time that Condition B was entered. If MODE 3 is reached within 3 hours, the time allowed for reaching MODE 5 is the next 33 hours because the total time allowed for 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.

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, Conditions A and B are exited, and therefore, the Required Actions of Condition B may be terminated.

16-244

1.3 Completion Times

EXAMPLES (continued)

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.

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.

16-244

1.3 ~~—~~ _Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-3

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One Function X ~~division~~train inoperable.	A.1 Restore Function X ~~division~~train to OPERABLE status.	7 days
B. One Function Y ~~division~~train inoperable.	B.1 Restore Function Y ~~division~~train to OPERABLE status.	72 hours
C. One Function X ~~division~~train inoperable. AND One Function Y ~~division~~train inoperable.	C.1 Restore Function X ~~division~~train to OPERABLE status. OR C.2 Restore Function Y ~~division~~train to OPERABLE status.	72 hours 72 hours

16-245

When one Function X ~~division~~train and one Function Y ~~division~~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 ~~division~~train starting from the time each ~~division~~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 ~~division~~train was declared inoperable (i.e., the time the situation described in Condition C was discovered).

16-245

16-244

1.3 ~~—~~ Completion Times

EXAMPLES (continued)

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.

It is 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. However, doing so would be inconsistent with the basis of the Completion Times. Therefore, there shall be administrative controls to limit the maximum time allowed for any combination of Conditions that result in a single contiguous occurrence of failing to meet the LCO. These administrative controls shall ensure that the Completion Times for those Conditions are not inappropriately extended.

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.

16-244

1.3 Completion Times

EXAMPLES (continued)

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 the extension) expires while one or more valves are still inoperable, Condition B is entered.

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.

16-244

1.3 ~~—~~_Completion Times

EXAMPLES (continued)

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.

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 ~~division~~ 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

16-245

16-244

1.3 Completion Times

EXAMPLES (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. The initial 8 hour interval of Required Action A.1 begins when Condition A is entered and the initial performance of Required Action A.1 must be complete within the first 8 hour interval. If Required Action A.1 is followed, and the Required Action is not met within the Completion Time (plus the 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.

<u>EXAMPLE 1.3-7</u>

ACTIONS

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

16-244

1.3 ~~—~~ Completion Times

EXAMPLES (continued)

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 (plus the extension allowed by SR 3.0.2), Condition B is entered. 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.

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IMMEDIATE COMPLETION TIME

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

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16-244

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.

Sometimes special situations dictate when the requirements of a Surveillance are to be met. They are "otherwise stated" conditions allowed by SR 3.0.1. They may be stated as clarifying Notes in the Surveillance, as part of the Surveillance, or both.

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.

The use of "met" or "performed" in these instances conveys specific meanings. A Surveillance is "met" only when the acceptance criteria are satisfied. Known failure of the requirements of a Surveillance, even without a Surveillance specifically being "performed," constitutes a Surveillance not "met." "Performance" refers only to the requirement to specifically determine the ability to meet the acceptance criteria.

Some Surveillances contain Notes that modify the Frequency of performance or the conditions during which the acceptance criteria must be satisfied. For these Surveillances, the MODE-entry restrictions of SR 3.0.4 may not apply. Such a Surveillance is not required to be performed prior to entering a MODE or other specified condition in the Applicability of the associated LCO if any of the following three conditions are satisfied:

16-244

1.4 ~~—~~ _Frequency

DESCRIPTION (continued)

a. The Surveillance is not required to be met in the MODE or other specified condition to be entered;

b. The Surveillance is required to be met in the MODE or other specified condition to be entered, but has been performed within the specified Frequency (i.e., it is current) and is known not to be failed; or

c. The Surveillance is required to be met, but not performed, in the MODE or other specified condition to be entered, and is known not to be failed.

Examples 1.4-3, 1.4-4, 1.4-5, and 1.4-6 discuss these special situations.

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.

EXAMPLE 1.4-1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Perform ~~CALIBRATION~~CHANNEL CHECK	~~24 months~~12 hours

16-246

Example 1.4-1 contains the type of SR most often encountered in the Technical Specifications (TS). The Frequency specifies an interval (~~24 months~~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 ~~24 months~~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 unit is outside the Applicability of the LCO). If the interval specified by SR 3.0.2 is exceeded while the 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.

16-244

1.4 Frequency

EXAMPLES (continued)

If the interval as specified by SR 3.0.2 is exceeded while the unit is not in a MODE or other specified condition in the Applicability of the LCO for which performance of the SR is required, then SR 3.0.4 becomes applicable. The Surveillance must be performed within the Frequency requirements of SR 3.0.2, as modified by SR 3.0.3, prior to entry into the MODE or other specified condition or the LCO is considered not met (in accordance with SR 3.0.1) and LCO 3.0.4 becomes applicable.

EXAMPLE 1.4-2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify flow is within limits.	Once within 12 hours after ≥ 25% RTP AND 24 hours thereafter

Example 1.4-2 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.

16-244

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 was not performed within the 7 day (plus the extension allowed by 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 was not performed within this 12 hour interval, there would then be a failure to perform a Surveillance within the specified Frequency, and the provisions of SR 3.0.3 would apply.

16-244

1.4 ~~—~~ Frequency

EXAMPLES (continued)

EXAMPLE 1.4-4

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
----------------------------NOTE-------------------------- Only required to be met in MODE 1. --- Verify leakage rates are within limits.	24 hours

Example 1.4-4 specifies that the requirements of this Surveillance do not have to be met until the unit is in MODE 1. The interval measurement for the Frequency of this Surveillance continues at all times, as described in Example 1.4-1. However, the Note constitutes an "otherwise stated" exception to the Applicability of this Surveillance. Therefore, if the Surveillance was not performed within the 24 hour interval (plus the extension allowed by SR 3.0.2), but the unit was not in MODE 1, there would be no failure of the SR nor failure to meet the LCO. Therefore, no violation of SR 3.0.4 occurs when changing MODES, even with the 24 hour Frequency exceeded, provided the MODE change was not made into MODE 1. Prior to entering MODE 1 (assuming again that the 24 hour Frequency was not met), SR 3.0.4 would require satisfying the SR.

EXAMPLE 1.4-5

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
----------------------------NOTE-------------------------- Only required to be performed in MODE 1. --- Perform complete cycle of the valve.	7 days

16-244

1.4 Frequency

EXAMPLES (continued)

The interval continues, whether or not the unit operation is in MODES 1, 2, or 3 (the assumed Applicability of the associated LCO) between performances.

As the Note modifies the required performance of the Surveillance, the Note is construed to be part of the "specified Frequency." Should the 7 day interval be exceeded while operation is not in MODE 1, this Note allows entry into and operation in MODES 2 and 3 to perform the Surveillance. The Surveillance is still considered to be performed within the "specified Frequency" if completed prior to entering MODE 1. Therefore, if the Surveillance was not performed within the 7 day (plus the extension allowed by SR 3.0.2) interval, but operation was not in MODE 1, 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 result in entry into MODE 1.

Once the unit reaches MODE 1, the requirement for the Surveillance to be performed within its specified Frequency applies and would require that the Surveillance had been performed. If the Surveillance was not performed prior to entering MODE 1, there would then be a failure to perform a Surveillance within the specified Frequency, and the provisions of SR 3.0.3 would apply.

EXAMPLE 1.4-6

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
----------------------------NOTE-------------------------- Not required to be met in MODE 3. --- Verify parameter is within limits.	24 hours

16-244

1.4 Frequency

EXAMPLES (continued)

Example 1.4-6 specifies that the requirements of this Surveillance do not have to be met while the unit is in MODE 3 (the assumed Applicability of the associated LCO is MODES 1, 2, and 3). The interval measurement for the Frequency of this Surveillance continues at all times, as described in Example 1.4-1. However, the Note constitutes an "otherwise stated" exception to the Applicability of this Surveillance. Therefore, if the Surveillance was not performed within the 24 hour interval (plus the extension allowed by SR 3.0.2), and the unit was in MODE 3, there would be no failure of the SR nor failure to meet the LCO. Therefore, no violation of SR 3.0.4 occurs when changing MODES to enter MODE 3, even with the 24 hour Frequency exceeded, provided the MODE change does not result in entry into MODE 2. Prior to entering MODE 2 (assuming again that the 24 hour Frequency was not met), SR 3.0.4 would require satisfying the SR.

16-244 blank line

3.0 LCO Applicability

LCO 3.0.4 (continued)

c. When an allowance is stated in the individual value, parameter, or other Specification.

This Specification shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

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, an evaluation shall be performed in accordance with Specification 5.5.14, "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.

LCO 3.0.7

Test Exception LCO 3.1.9, "PHYSICS TESTS Exceptions – MODE 2," and LCO 3.4.17, "RCS Loops – Test Exceptions," allow specified Technical Specification (TS) requirements to be changed to permit performance of special tests and operations. Unless otherwise specified, all other TS requirements remain unchanged. Compliance with Test Exception LCOs is optional. When a Test Exception LCO is desired to be met but is not met, the ACTIONS of the Test Exception LCO shall be met.

16-247 run in

When a Test Exception LCO is not desired to be met, entry into a MODE or other specified condition in the Applicability shall be made in accordance with the other applicable Specifications.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.1 SHUTDOWN MARGIN (SDM)

LCO 3.1.1 SDM shall be within the limits specified in the COLR.

16-257 and 16-258

APPLICABILITY: MODE 2 with K_{eff} < 1.0
MODES 3, 4, and 5.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SDM not within limits.	A.1 Initiate boration to restore SDM to within limits.	15 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.1.1 Verify SDM to be within the limits specified in the COLR.	24 hours

3.1 REACTIVITY CONTROL SYSTEMS

3.1.3 Moderator Temperature Coefficient (MTC)

LCO 3.1.3 The MTC shall be maintained within the limits specified in the COLR and a maximum positive (upper) limit as specified below:

a. 5 pcm/°F when THERMAL POWER < 50% RTP; and

b. 0 pcm/°F when THERMAL POWER is ≥ 50% RTP.

16-258

APPLICABILITY: MODE 1 and MODE 2 with $k_{eff} \geq 1.0$ for the positive (upper) MTC limit, MODES 1, 2, and 3 for the negative (lower) MTC limit.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. MTC not within positive (upper) limit.	A.1	Establish administrative withdrawal limits for control banks to maintain MTC within limit.	24 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1	Be in MODE 2 with k_{eff} < 1.0.	6 hours
C. MTC not within negative (lower) limit.	C.1	Be in MODE 4.	12 hours

16-258

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.3.1	Verify MTC is within positive (upper) limit.	Once prior to entering MODE 1 after each refueling
SR 3.1.3.2	-------------------------------NOTE------------------------------ If the MTC is more negative than the COLR limit when extrapolated to the end of cycle, SR 3.1.3.2 must be repeated prior to exceeding the minimum allowable boron concentration at which MTC is projected to exceed the negative (lower) limit. -- Verify the MTC is within the negative (lower) limit specified in the COLR.	16-258 Once each fuel cycle within 7 ~~EFPD~~effective full power days of reaching 2/3 of expected core burnup

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Two or more RCCAs not within alignment limits.	D.1.1 Verify SDM to be within the limits specified in the COLR. OR D.1.2 Initiate boration to restore SDM to within limit. AND D.2 Be in MODE 3.	1 hour 1 hour 6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.4.1	Verify individual RCCA positions within alignment limit.	12 hours
SR 3.1.4.2	Verify RCCA freedom of movement (trippability) by moving each RCCA not fully inserted in the core ~~≥ 16~~ 16 to 20 steps in either direction.	92 days
SR 3.1.4.3	Verify drop time of each RCCA, from the fully withdrawn position, is ≤ 3.5 seconds from opening of the reactor trip breaker to the centerline of lowest RCCA position indication coil, with: a. T_{avg} ≥ 500°F; and b. All reactor coolant pumps operating.	Prior to criticality after each removal of the reactor head

16-261

16-258

3.1 REACTIVITY CONTROL SYSTEMS

3.1.6 Control Bank Insertion Limits

LCO 3.1.6 Each rod control cluster assembly (RCCA) control bank shall be within insertion, sequence, and overlap limits specified in the COLR.

16-258

APPLICABILITY: MODE~~S~~ 1, ~~and 2.~~
MODE 2 with K_{eff} ≥1.0.

--NOTE--
This LCO is not applicable while performing SR 3.1.4.2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more RCCA control banks with insertion limits not met.	A.1.1 Verify SDM is within the limits specified in the COLR.	1 hour
	OR	
	A.1.2 Initiate boration to restore SDM to within limit.	1 hour
	AND	
	A.2 Restore RCCA control bank(s) to within insertion limits.	2 hours
B. One or more RCCA control banks with sequence or overlap limits not met.	B.1.1 Verify SDM is within the limits specified in the COLR.	1 hour
	OR	
	B.1.2 Initiate boration to restore SDM to within limit.	1 hour
	AND	

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
	B.2	Restore RCCA control bank(s) to within sequence and overlap limits.	2 hours
C. Required Action and associated Completion Time not met.	C.1	Be in MODE ~~3~~2 with K_{eff} <1.	6 hours

16-258

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.6.1	Verify estimated critical RCCA control bank position is within the limits specified in the COLR.	Once within 4 hours prior to achieving criticality
SR 3.1.6.2	Verify each RCCA control bank position is within the insertion, sequence, and overlap limits specified in the COLR.	12 hours

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Rod Control Cluster Assembly (RCCA) Position Indication.

LCO 3.1.7 The Analog and Digital RCCA Position Indication shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

---NOTE---

Separate Condition entry is allowed for each inoperable analog RCCA position indicator and each inoperable digital RCCA position indicator.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more banks with one individual analog RCCA position indicator inoperable.	A.1.1 Implement the imbalance/dropped RCCA penalty on the Low DNBR reactor trip setpoint.	8 hours
	<u>AND</u>	
	A.1.2 Verify position of RCCAs with inoperable position detectors indirectly using the ~~SPNDs~~Aeroball Measurement System.	Once per 8 hours <u>AND</u> Once within 4 hours after an RCCA with an inoperable analog RCCA position indicator has been moved in excess of 20 steps in one direction since the last determination of the RCCA's position
	16-265 and 16-266	
	<u>OR</u>	

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
	A.2	Reduce THERMAL POWER to ≤ 50% RTP.	8 hours
B. One or more banks with two or more analog RCCA position indicators inoperable.	B.1	Implement Required Actions for Condition A.	Immediately
	AND		
16-266	B.~~1~~2	Verify three rod control cluster assembly units (RCCAUs) are OPERABLE.	Immediately
	AND		
	B.~~2~~3	Place the RCCAs under manual control.	Immediately
	AND		
	B.~~3~~4	Determine the Reactor Coolant System T_{avg}.	Once per 1 hour
	AND		
	B.~~4~~5	Restore inoperable analog RCCA position indicator(s) to OPERABLE status such that a maximum of one analog RCCA position indicator in the associated bank is inoperable.	24 hours

3.1 REACTIVITY CONTROL SYSTEMS

3.1.8 Boron Dilution Protection (BDP)

LCO 3.1.8 The volume control tank (VCT) and letdown isolation valves shall be OPERABLE.

APPLICABILITY: MODES 3, 4, 5, and 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more VCT or letdown isolation valves inoperable.	A.1 Restore VCT or letdown isolation valves to OPERABLE status.	1 hour
~~A~~B. ~~One or more VCT or letdown isolation valves inoperable.~~ Required Action and associated Completion Time of Condition A not met.	~~A~~B.1 Isolate the affected boron dilution flow path by use of at least one closed and de-activated automatic valve or closed manual valve.	8 hours
	AND	
	~~A~~B.2 -------------NOTE--------------- Isolation devices that are locked, sealed, or otherwise secured may be verified by use of administrative means. ------------------------------------- Verify the affected boron dilution flow path is isolated.	31 days

16-268

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.9.1	Verify THERMAL POWER is ≤ 5% RTP.	30 minutes
SR 3.1.9.2	Verify SDM is within the limits specified in the COLR.	24 hours
SR 3.1.9.3	Perform a SENSOR OPERATIONAL TEST on power range and intermediate range divisions.	10 days prior to initiation of PHYSICS TESTS

16-270

3.2 POWER DISTRIBUTION LIMITS

3.2.1 Linear Power Density (LPD)

LCO 3.2.1 The LPD shall not exceed the limits specified in the COLR.

APPLICABILITY: MODE 1 with THERMAL POWER > 10% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LPD not within limits.	A.1 Restore LPD to within limits.	1 hour
B. LPD not within region of acceptable operation when the RCSL System is out of service.	B.1 Invite action to reduce power until LPD is within limits.	Immediately
~~B~~C. Required Action and associated Completion Time not met.	~~B~~C.1 Reduce THERMAL POWER to ≤ 10% RTP.	6 hours

16-276

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.1.1	Verify LPD is within limits specified in the COLR.	12 hours
SR 3.2.1.2	---------------------------NOTE--------------------------- Only required to be met when the RCSL System monitoring of the LPD channel is out of service. With RCSL in service, this parameter is continuously monitored. -- Verify LPD, as indicated on the most limiting reading protection system LPD channel, is within its limit.	Within 1 hour AND Once per hour thereafter

16-276

3.2 POWER DISTRIBUTION LIMITS

3.2.2 Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)

LCO 3.2.2 $F^N_{\Delta H}$ shall be within the limits specified in the COLR.

APPLICABILITY: MODE 1 ~~with THERMAL POWER > 90% RTP~~.

16-278

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. $F^N_{\Delta H}$ not within limits.	A.1 Reduce THERMAL POWER by 1% for each 1% that $F^N_{\Delta H}$ exceeds the limits. AND A.2 Restore $F^N_{\Delta H}$ to within limits.	1 hour 4 hours
B. Required Action and associated Completion Time not met.	B.1 ~~Reduce THERMAL POWER ≤ 90% RTP~~Be in MODE 2.	~~2~~ 6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.2.1	-------------------------------NOTE------------------------------ Not required to be performed until 24 hours after exceeding ~~90~~40% power. --- Verify $F^N_{\Delta H}$ is within limits specified in the COLR.	Once after each refueling outage prior to exceeding ~~98~~70% RTP AND ~~15~~ 31 effective full power days thereafter

16-278

3.2 POWER DISTRIBUTION LIMITS

3.2.3 Departure From Nucleate Boiling Ratio (DNBR)

LCO 3.2.3 The DNBR shall not exceed the limits specified in the COLR.

APPLICABILITY: MODE 1 with THERMAL POWER > 10% RTP.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. DNBR not within limits.	A.1	Restore DNBR to within limits.	1 hour
B. DNBR not within region of acceptable operation when the RCSL System is out of service.	B.1	Invite action to reduce power until DNBR is within limits.	Immediately
~~B~~C. Required Action and associated Completion Time not met.	~~B~~C.1	Reduce THERMAL POWER to $\leq$ 10% RTP.	6 hours

16-279

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.3.1	Verify DNBR is within limits specified in the COLR.	12 hours
SR 3.2.3.2	---------------------------NOTE--------------------------- Only required to be met when the RCSL System monitoring of the DNBR channel is out of service. With RCSL in service, this parameter is continuously monitored. --- Verify DNBR, as indicated on the most limiting reading protection system DNBR channel, is within its limit.	Within 1 hour AND Once per hour thereafter

16-279

3.2 POWER DISTRIBUTION LIMITS

3.2.4 AXIAL OFFSET (AO)

LCO 3.2.4 The AO shall not exceed the limits specified in the COLR.

APPLICABILITY: MODE 1 with THERMAL POWER ≥ 50% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. AO not within limits.	A.1 Restore AO to within limits.	1 hour
B. AO not within region of acceptable operation when the RCSL System is out of service.	B.1 Inititate action to reduce power until AO is within limits.	Immediately
~~B~~C. Required Action and associated Completion Time not met.	~~B~~C.1 Reduce THERMAL POWER to < 50% RTP.	4 hours

16-282

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.4.1	Verify AO is within the limits specified in the COLR.	12 hours
SR 3.2.4.2	Determine target AO in conjunction with a full core flux map.	31 effective full power days
SR 3.2.4.3	---------------------------NOTE--------------------------------- Only required to be met when the RCSL System monitoring of the AO is out of service. With RCSL in service, this parameter is continuously monitored. --- Verify AO is within its limit.	Within 1 hour AND Once per hour thereafter

16-282

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.2 RCS Minimum Temperature for Criticality

LCO 3.4.2 Each RCS loop average coolant temperature (T_{avg}) shall be ≥ 568°F.

APPLICABILITY: MODE 1,
MODE 2 with k_{eff} ≥ 1.0.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. $\boxed{T_{avg}}$ in one or more RCS loops not within limit.	A.1 Be in MODE 2 with k_{eff} < 1.0.	30 minutes

16-258

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.2.1	Verify RCS $\boxed{T_{avg}}$ in each loop ≥ 568°F.	12 hours

3.6 CONTAINMENT SYSTEMS

3.6.4 Containment Pressure

LCO 3.6.4 Containment pressure shall be ≥ -0.2 psig and ≤ $\boxed{+1.2}$ psig.

16-249

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Containment pressure not within limits.	A.1 Restore containment pressure to within limits.	1 hour
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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.1	Verify containment pressure is within limits.	12 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, 3, or 4.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 36 hours
D. Required Action and associated Completion Time of Condition A not met in MODE 5 or 6, or during movement of irradiated fuel assemblies.	~~--------REVIEWER'S NOTE---------~~ ~~The need for the toxic gas isolation state will be determined by the COL Applicant.~~ ~~--~~ D.1 --------------NOTE-------------- Place CREF train in toxic gas isolation state if automatic transfer to toxic gas isolation state is inoperable. ------------------------------------- Place OPERABLE CREF train in emergency mode. OR D.2 Suspend movement of irradiated fuel assemblies.	 Immediately Immediately
E. Two CREF trains inoperable in MODE 5 or 6, or during movement of irradiated fuel assemblies.	E.1 Suspend movement of irradiated fuel assemblies.	Immediately
F. Two CREF trains inoperable in MODE 1, 2, 3, or 4 for reasons other than Condition B.	F.1 Enter LCO 3.0.3.	Immediately

16-250

16-250

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.10.1	Operate each CREF train for ≥ 15 minutes with the heaters operating.	31 days
SR 3.7.10.2	Perform required CREF filter train testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.7.10.3	Verify each CREF train actuates on an actual or simulated actuation signal.	24 months
SR 3.7.10.4	Perform required CRE unfiltered air inleakage testing in accordance with the Control Room Envelope Habitability Program.	In accordance with the Control Room Envelope Habitability Program

5.2 Organization

5.2.2 Unit Staff (continued)

b. Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i) and Specifications 5.2.2.a and 5.2.2.f 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;

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;

d. Administrative ~~controls~~ procedures shall be developed and implemented to limit the working hours of personnel who perform safety related functions (e.g., licensed Senior Operators, licensed Operators, health physicists, auxiliary operators, and key maintenance personnel).

16-252

The controls shall include guidelines on working hours that ensure adequate shift coverage shall be maintained without routine heavy use of overtime.

Any deviation from the above guidelines shall be authorized in advance by the plant manager or designee, in accordance with approved administrative procedures, and with documentation of the basis for granting the deviation. Routine deviation from the working hour guidelines shall not be authorized.

Controls shall be included in the procedures to require a periodic independent review be conducted to ensure that excessive hours have not been assigned;

e. The operations manager or assistant operations manager shall hold a Senior Operator license; and

f. ~~When the reactor is operating in MODE 1, 2, 3, or 4, a~~An individual shall provide advisory technical support to the unit operations shift crew in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. This individual shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift.

5.5 Programs and Manuals

5.5.11 Gaseous Waste Processing System Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the Gaseous Waste Processing System and the quantity of radioactivity contained in gas delay beds. The gaseous radioactivity quantities shall be determined following the methodology in Branch Technical Position (BTP) 11-5, "Postulated Radioactive Release due to Waste Gas System Leak or Failure".

The program shall include:

a. The limits for concentrations of hydrogen and oxygen in the Gaseous Waste Processing 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

16-253

b. A surveillance program to ensure that the quantity of radioactivity contained in the gas delay beds is less than the amount that would result in a whole body exposure of ≥ 0.5 rem to any individual in an unrestricted area, in the event of an uncontrolled release of the beds' contents.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the ~~Explosive Gas and Storage Tank~~ Gaseous Waste Processing System Radioactivity Monitoring Program Surveillance Frequencies.

---REVIEWER'S NOTE---
The U.S. EPR does not have outdoor liquid radwaste tanks. If a COL Applicant adds outdoor liquid radwaste tanks, this program will be modified accordingly.

5.5.12 Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. 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 for 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

5.5 Programs and Manuals

5.5.17 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;

d. Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization ~~MODE~~ mode of operation by one train of the CREFS, 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;

16-253

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 Specification 5.5.17.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 or~~]]~~ smoke must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis; and

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 Specifications 5.5.17.c and 5.5.17.d, respectively.

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

The following reports shall be submitted in accordance with 10 CFR 50.4.

5.6.1 Annual Radiological Environmental Operating Report

16-253 underline

--------------------------------------REVIEWER'S NOTE--------------------------------------
[A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station.]

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 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. 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.

5.6.2 Radioactive Effluent Release Report

--------------------------------------REVIEWER'S 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 in the previous year shall be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR Part 50, Appendix I, Section IV.B.1.

5.6 Reporting Requirements

5.6.6 Containment Post Tensioning Surveillance Report

Any abnormal degradation of the containment structure detected during the tests required by the Containment Post Tensioning 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.

5.6.7 Steam Generator Tube Inspection Report

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 5.5.8, "Steam Generator (SG) Program". The report shall include:

a. The scope of inspections performed on each SG;

b. Active degradation mechanisms found;

c. Nondestructive examination techniques utilized for each degradation mechanism;

d. Location, orientation (if linear), and measured sizes (if available) of service induced indications;

e. Number of tubes plugged during the inspection outage for each active degradation mechanism;

f. Total number and percentage of tubes plugged to date;

16-253

g. The results of condition monitoring, including the results of tube pulls and in situ testing; and

~~h. The plugging percentage for all plugging in each SG.~~

5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

As provided in paragraph 20.1601(c) of 10 CFR Part 20, the following controls shall be applied to high radiation areas in place of the controls required by paragraph 20.1601(a) and (b) of 10 CFR Part 20:

5.7.1 High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation

a. Each entryway to such an area shall be barricaded and conspicuously posted as a high radiation area. Such barricades may be opened as necessary to permit entry or exit of personnel or equipment.

b. Access to, and activities in, each such area shall be controlled by means of Radiation Work Permit (RWP) or equivalent that includes ~~Specification~~ specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

16-253

c. Individuals qualified in radiation protection procedures and personnel continuously escorted by such individuals may be exempted from the requirement for an RWP or equivalent while performing their assigned duties provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual (whether alone or in a group) entering such an area shall possess one of the following:

1. A radiation monitoring device that continuously displays radiation dose rates in the area;

2. A radiation monitoring device that continuously integrates the radiation dose rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint;

3. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area; or

4. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

5.5 Programs and Manuals

5.5.2 Primary Coolant Sources Outside Containment

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 Low Head Safety Injection, Medium Head Safety Injection, and Nuclear Sampling. The program shall include the following:

a. Preventive maintenance and periodic visual inspection requirements; and

b. Integrated leak test requirements for each system once per 24 months.

The provisions of SR 3.0.2 are applicable.

5.5.3 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 ten times the concentration values in Appendix B, Table 2, Column 2 to 10 CFR 20.1001-20.2402;

16-253

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM;

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 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. Determination of projected dose contributions from radioactive effluents in accordance with the methodology in the ODCM at least every 31 days;

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 in Sections 3.1 through 3.9 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 to verify the OPERABILITY of systems and components, and that variables 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 may be performed by means of any series of sequential, overlapping, or total steps provided the entire Surveillance is performed within the specified Frequency. Additionally, the definitions related to instrument testing (e.g., CALIBRATION) specify that these tests are performed by means of any series of sequential, overlapping, or total steps.

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 not to be met between required Surveillance performances.

Surveillances do not have to be performed when the unit 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 test exception are only applicable when the test exception is used as an allowable exception to the requirements of a Specification.

16-247

Unplanned events may satisfy the requirements (including applicable acceptance criteria) for a given SR. In this case, the unplanned event may be credited as fulfilling the performance of the SR. This allowance includes those SRs whose performance is normally precluded in a given MODE or other specified condition.

Surveillances, including Surveillances invoked by Required Actions, do not have to be performed on inoperable equipment because the

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.3 Moderator Temperature Coefficient (MTC)

BASES

BACKGROUND

According to GDC 11 (Ref. 1), the reactor core and its interaction with the Reactor Coolant System (RCS) must be designed for inherently stable power operation, even in the possible event of an accident. In particular, the net reactivity feedback in the system must compensate for any unintended reactivity increases.

The MTC relates a change in core reactivity to a change in reactor coolant temperature (a positive MTC means that reactivity increases with increasing moderator temperature; conversely, a negative MTC means that reactivity decreases with increasing moderator temperature). The reactor is designed to operate with a negative MTC over the largest possible range of fuel cycle operation. Therefore, a coolant temperature increase will cause a reactivity decrease, so that the coolant temperature tends to return toward its initial value. Reactivity increases that cause a coolant temperature increase will thus be self limiting, and stable power operation will result.

16-259

MTC values are predicted at selected burnups during the safety evaluation analysis and are confirmed to be acceptable by measurements. Both initial and reload cores are designed so that the beginning of cycle (BOC) MTC is less than or equal to zero when THERMAL POWER is ≥ 50% RTP. The actual value of the MTC is dependent on core characteristics, such as fuel loading and reactor coolant soluble boron concentration. The core design may require additional fixed distributed poisons to yield an MTC at BOC within the range analyzed in the plant accident analysis. The end of cycle (EOC) MTC is also limited by the requirements of the accident analysis. Fuel cycles are evaluated to ensure that the MTC does not exceed the EOC limit.

The limitations on MTC are provided to ensure that the value of this coefficient remains within the limiting conditions assumed in the accident and transient analyses.

If the LCO limits are not met, the unit response during transients may not be as predicted. The core could violate criteria that prohibit a return to criticality, or the departure from nucleate boiling ratio criteria of the approved correlation may be violated, which could lead to a loss of the fuel cladding integrity.

BASES

LCO

LCO 3.1.3 requires the MTC to be within specified limits of the COLR to ensure that the core operates within the assumptions of the accident analysis. During the reload core safety evaluation, the MTC is analyzed to determine that its values remain within the bounds of the original accident analysis during operation.

Assumptions made in safety analyses require that the MTC be less positive than a given upper bound and more positive than a given lower bound. The MTC is most positive at BOC; this upper bound must not be exceeded. This maximum positive (upper) limit occurs at BOC, all rods out (ARO), hot zero power conditions. At EOC the MTC takes on its most negative (lower) value, when the lower bound becomes important. This LCO exists to ensure that both the upper and lower bounds are not exceeded.

16-258

During operation, therefore, the conditions of the LCO can only be ensured through measurement. The Surveillance checks at BOC and EOC on MTC provide confirmation that the MTC is behaving as anticipated so that the acceptance criteria are met.

The LCO establishes a maximum positive value that cannot be exceeded. The BOC positive (upper) limit and the EOC negative (lower) limit are established in the COLR to allow specifying limits for each particular cycle. ~~This permits the unit to take advantage of improved fuel management and changes in unit operating schedule.~~

APPLICABILITY

Technical Specifications place both LCO and SR values on MTC, based on the safety analysis assumptions described above.

In MODE 1, the limits on MTC must be maintained to ensure that any accident initiated from THERMAL POWER operation will not violate the design assumptions of the accident analysis. In MODE 2 with the reactor critical, the upper limit must also be maintained to ensure that startup and subcritical accidents (such as the uncontrolled control rod assembly or group withdrawal) will not violate the assumptions of the accident analysis. The lower MTC limit must be maintained in MODES 2 and 3, in addition to MODE 1, to ensure that cooldown accidents will not violate the assumptions of the accident analysis. In MODES 4, 5, and 6, this LCO is not applicable, since no postulated accidents using the MTC as an analysis assumption are initiated from these MODES.

ACTIONS

A.1

If the upper MTC limit is violated, administrative withdrawal limits for control banks must be established to maintain the MTC within its limits. The MTC becomes more negative with control bank insertion and decreased boron concentration. A Completion Time of 24 hours provides

BASES

SURVEILLANCE REQUIREMENTS (continued)

Meeting the requirements of ANS/ANSI-19.6.1-2005 prior to entering MODE 1 ensures that this LCO for the positive MTC limit are met at other reactivity or power conditions by validating the accuracy of the physics predictions over the entire cycle.

The BOC (upper) MTC value for ARO will be inferred from isothermal temperature coefficient measurements obtained during the PHYSICS TESTS after refueling. The ARO value can be directly compared to the BOC MTC limit of the LCO. If required, measurement results and predicted design values can be used to establish administrative withdrawal limits for control banks.

SR 3.1.3.2

In similar fashion, the LCO demands that the MTC be less negative than the specified value at the EOC full power conditions. The EOC MTC measurement may be performed at any THERMAL POWER, but the results must be extrapolated to the conditions at RTP with all banks withdrawn with the expected EOC boron concentration. This is required to have a valid comparison with the LCO value. Because the RTP MTC value gradually becomes more negative with higher core exposure and a decrease in boron concentration the projected EOC MTC may be evaluated before the reactor actually reaches the EOC condition. Performing the Surveillance upon reaching 2/3 of expected projected burnup minimizes the extrapolation errors while providing sufficient warning prior to reaching the EOL (lower) MTC limit. Allowing 7 effective full power days to complete the surveillance allows time to schedule the surveillance with other plant activities and still maintain sufficient margin to the end of life MTC value that the surveillance is designed to protect. MTC values must be extrapolated and compensated to permit direct comparison to the specified MTC limits.

16-258

SR 3.1.3.2 is modified by a Note, which indicates that if the extrapolated MTC is more negative than the EOC COLR limit, the Surveillance must be repeated, and that shutdown must occur prior to exceeding the minimum allowable boron concentration at which MTC is projected to exceed the negative (lower) limit.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 11~~, August, 2007~~.
2. FSAR Chapter 15.
3. ANSI/ANS-19.6.1-2005, Reload Startup Physics Tests For Pressurized Reactors.

BASES

APPLICABLE SAFETY ANALYSES (continued)

a. There will be no violations of:

1. Specified acceptable fuel design limits; or

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

b. The core remains subcritical after AOOs and postulated accidents.

Two types of misalignment are distinguished. During movement of a control RCCA group, one RCCA may stop moving, while the other RCCAs in the group continue. This condition may cause excessive power peaking. The second type of misalignment occurs if one RCCA fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition requires an evaluation to determine that sufficient reactivity worth is held in the control RCCAs to meet the SDM requirement, with the maximum worth RCCA stuck fully withdrawn.

16-262

Analysis is performed as follows with regard to static RCCA misalignment (Ref. 3). With the control banks at their insertion limit and again at the bite position, which corresponds to the nearly withdrawn position that minimizes the time for negative reactivity insertion, a power distribution is created at every combination of one and two RCCAs inserted and withdrawn 20 steps from the bank position. Satisfying limits on departure from nucleate boiling ratio in these cases bounds the situation when RCCAs are misaligned from their group by 8 steps. It is assumed that no physical mechanism can cause more than two RCCAs to be misaligned at one time within the time frame between self powered neutron detector (SPND) CALIBRATIONS.

Another type of misalignment occurs if one RCCA fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition is assumed in the evaluation to determine that the required SDM is met with the maximum worth RCCA also fully withdrawn (Ref. 3).

The Required Actions in this LCO ensure that either deviations from the alignment limits will be corrected or that THERMAL POWER will be adjusted so that excessive local linear power densities will not occur, and that the requirements on SDM and ejected rod worth are preserved.

BASES

ACTIONS (continued)

D.2

If more than one RCCA is found to be misaligned or becomes misaligned because of bank movement, the unit conditions fall outside of the accident analysis assumptions. Since automatic bank sequencing would continue to cause misalignment, the unit must be brought to a MODE or Condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to MODE 3 within 6 hours.

The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.4.1

Verification that individual RCCA positions are within alignment limits at a Frequency of 12 hours provides a history that allows the operator to detect a RCCA that is beginning to deviate from its expected position. The specified Frequency takes into account other RCCA position information that is continuously available to the operator in the control room, so that during actual RCCA motion, deviations can immediately be detected.

16-261

SR 3.1.4.2

Verifying each control RCCA is OPERABLE would require that each RCCA be tripped. However, in MODES 1 and 2 with $k_{eff} \geq 1.0$, tripping each control RCCA would result in azimuthal or axial power tilts, or oscillations. Exercising each individual control RCCA every 92 days provides increased confidence that all RCCAs continue to be OPERABLE without exceeding the alignment limit, even if they are not regularly tripped. Moving each control RCCA by ~~16~~ 16 to 20 steps will not cause radial or axial power tilts, or oscillations, to occur. The 92 day Frequency takes into consideration other information available to the operator in the control room and SR 3.1.4.1, which is performed more frequently and adds to the determination of OPERABILITY of the RCCAs. Between required performances of SR 3.1.4.2 (determination of control RCCA OPERABILITY by movement), if a control RCCA(s) is discovered to be immovable, but remains trippable, the control RCCA(s) is considered to be OPERABLE. At any time, if a control RCCA(s) is immovable, a determination of the trippability (OPERABILITY) of the control RCCA(s) must be made and appropriate action taken.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.4.3

Verification of RCCA drop times allows the operator to determine that the maximum RCCA drop time permitted is consistent with the assumed RCCA drop time used in the safety analysis. Measuring RCCA drop times prior to reactor criticality, after each reactor vessel head removal, ensures that the reactor internals and rod drive mechanism will not interfere with RCCA motion or RCCA drop time, and that no degradation in these systems has occurred that would adversely affect control RCCA motion or drop time. This testing is performed with all RCPs operating and the average moderator temperature ≥ 500°F to simulate a reactor trip under actual conditions. Performing rod drop testing at less than the temperature specified for hot zero power is conservative due to increased reactor coolant density at lower temperature and the associated increase in rod drop resistance.

16-258

This Surveillance is performed prior to criticality after each removal of the reactor head, due to the plant conditions needed to perform the SR and the potential for an unplanned plant transient if the Surveillance were performed with the reactor at power.

REFERENCES

1. 10 CFR 50, Appendix A~~, August, 2007~~.
2. 10 CFR 50.46~~, August, 2007~~.
3. FSAR Chapter 15.

BASES

APPLICABILITY (continued)

16-263

The Applicability requirements have been modified by a Note indicating the LCO requirement is suspended during SR 3.1.4.2. SR 3.1.4.2 verifies the freedom of the RCCAs to move, and requires the control bank RCCAs to move below the LCO limits, which would normally violate the LCO. ~~The Note also allows the LCO to be not applicable during a partial trip, which inserts a selected RCCA group, specified in the COLR, during loss of load events. This condition is outside of the control bank insertion limits and requires prompt action to restore operation within insertion limits.~~

ACTIONS

A.1.1, A.1.2, and A.2

Operation beyond the insertion limits may result in a loss of SDM and excessive peaking factors. This restoration can occur in two ways:

a. Reducing power to be consistent with rod position; or

b. Moving rods to be consistent with power.

The insertion limits should not be violated during normal operation; this violation, however, may occur during transients when the operator is manually controlling the RCCAs in response to changing plant conditions. When the Control Bank groups are inserted beyond the insertion limits, actions must be taken to either withdraw the Control Bank groups beyond the insertion limits or to reduce THERMAL POWER to less than or equal to that allowed for the actual RCCA insertion limit. Verification of SDM or initiation of boration to regain SDM is required within 1 hour, since the SDM in MODES 1 and 2 normally ensured by adhering to the control and shutdown bank insertion limits (see LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") has been upset. If control banks are not within their insertion limits, then SDM will be verified by performing a reactivity balance calculation, considering the effects listed in the BASES for SR 3.1.1.1. The allowed Completion Time of 2 hours provides a reasonable time to restore control banks within insertion limits, allowing the operator to deal with current plant conditions while limiting peaking factors to acceptable levels.

BASES

BACKGROUND (continued)

16-258

The Digital RCCA Position Indication counts the pulses from the control rod drive control system that moves the RCCA~~-~~s. The Digital RCCA Position Indication tracks individual RCCA positions and can display the individual RCCA position or the position for each group of RCCAs. Individual RCCAs in a group all receive the same signal to move and should, therefore, all be at the same position indicated by the group position. The Digital RCCA Position Indication is considered highly precise (1 step = 10 mm ~ ⅜ inch). If a RCCA does not move one step for each demand pulse, the Digital RCCA Position Indication may still count the pulse and incorrectly reflect the position of the RCCA.

The Analog RCCA Position Indication provides a totally independent indication of actual RCCA position, but at a lower precision than the step counters. This indication is based on inductive analog signals from a series of coils spaced along a hollow tube. The Analog RCCA Position Indication is capable of measuring RCCA position within at least ± 8 steps.

APPLICABLE SAFETY ANALYSES

Control and shutdown RCCA position accuracy is essential during power operation. Power peaking, ejected rod or SDM limits may be violated in the event of an AOO or postulated accident (Ref. 2), with control or shutdown RCCAs operating outside their limits undetected. Therefore, the acceptance criteria for RCCA position indication is that RCCA positions must be known with sufficient accuracy in order to verify the core is operating within the group sequence, overlap, design peaking limits, ejected rod worth, and with minimum SDM (LCO 3.1.5, "Shutdown Bank Insertion Limits," and LCO 3.1.6, "Control Bank Insertion Limits"). The RCCA positions must also be known in order to verify the alignment limits are preserved (LCO 3.1.4, "RCCA Group Alignment Limits"). Control RCCA positions are continuously monitored to provide operators with information that ensures the plant is operating within the bounds of the accident analysis assumptions.

In addition, RCCA position indication is necessary to support automatic adjustment of Low DNBR protection system setpoints in the event of a sensed RCCA or multiple RCCA drops.

The control RCCA position indicator divisions satisfy Criterion 2 of 10 CFR 50.36(~~d~~c)(2)(ii). The control RCCA position indicators monitor control RCCA position, which is an initial condition of the accident.

BASES

ACTIONS (continued)

B.1, B.2, ~~and~~ B.3 and B.4

16-266

When more than one Analog RCCA Position Indication in one or more banks fail, additional actions are necessary to ensure that acceptable power distribution limits are maintained, minimum SDM is maintained, and the potential effects of RCCA misalignment on associated accident analyses are limited.

Verifying that at least three rod control cluster assembly units (RCCAUs) are OPERABLE ensures that the majority of the analog RCCA position indicators remain available to provide necessary plant information. Placing the Control Rod Drive Control System in manual assures unplanned RCCA motion will not occur. The immediate Completion Time for placing the Control Rod Drive Control System in manual reflects the urgency with which unplanned RCCA motion must be prevented while in this Condition as well as the need to maintain adequate monitoring of RCCA position.

With the RCCAs under manual control it is now possible to more effectively monitor for a misaligned or dropped RCCA based on changes in RCS ~~T_{AVG}~~ T_{avg} since only minor fluctuations are expected during steady-state operation. A Completion Time of Once per hour is sufficient to identify abnormal trends.

16-258

B.~~4~~5

16-266

The 24 hour Completion Time provides sufficient time to troubleshoot and restore the Analog RCCA Position Indication to operation while avoiding the plant challenges associated with the shutdown without full RCCA position indication.

C.1.1 and C.1.2

With one Digital RCCA Position Indicator per bank not capable of indicating RCCA position, the RCCA positions can be determined by the Analog RCCA Position Indication. Since normal power operation does not require excessive movement of RCCAs, verification by administrative means the RCCA positions of all affected banks that the most withdrawn RCCA and the least withdrawn RCCA are ≤ 8 steps apart within the allowed Completion Time of once every 8 hours is adequate.

BASES

APPLICABLE SAFETY ANALYSES

The BCMS responds to abnormal increases in neutron counts per second (flux rate) and actuates the VCT and letdown isolation valves to mitigate the consequences of an inadvertent boron dilution event as described in FSAR Chapter 15 (Ref. 1). The accident analyses rely on the VCT and letdown isolation valves to terminate a boron dilution event. The IRWST isolation valve is also sent a signal to open to protect the CVCS charging pump and provide uninterrupted flow to the RCP seals but it is not credited in the Chapter 15 accident analyses.

The single failure criterion required to be imposed in the conduct of plant safety analyses was considered in the original design of the VCT and letdown isolation valves. Two valves in series in the charging pump suction line provide assurance that the dilution path could be isolated even if a single failure occurred.

The BDP VCT and letdown isolation valves satisfies Criterion 3 of 10 CFR 50.36(~~d~~c)(3)(ii) (Ref. 2)

LCO

LCO 3.1.8 provides the requirements for OPERABILITY of the VCT and letdown isolation valves that mitigate the consequences of an inadvertent boron dilution event as described in FSAR Chapter 15 (Ref. 1). The VCT and letdown isolation valves are as follows:

- VCT Outlet (KBA21 AA001)
- Letdown to Charging Pump Suction Header (KBA25~~1~~ AA017)
- VCT and Letdown to Charging Pump Common Isolation (KBA21 AA009)

This LCO provides assurance that the VCT and letdown isolation valves will perform their designed safety functions to mitigate the consequences of inadvertent boron dilution events.

APPLICABILITY

The VCT and letdown isolation valves must be OPERABLE in MODES 3, 4, 5, and 6 because the safety analysis identifies the VCT and letdown isolation valves as the primary means to mitigate an inadvertent boron dilution of the RCS.

ACTIONS

A.1 ~~and A.2~~

16-268

If one or more of the VCT and letdown isolation valves are inoperable, the automatic capability for mitigation of dilution events is no longer available. In this case, the isolation valves are required to be restored to OPERABLE status within 1~~8~~ hour~~s~~. In this Condition, ability to maintain subcriticality may be reduced. Thus, 1 hour is allowed to return the VCT and letdown isolation valves to operable.

16-268

BASES

ACTIONS (continued)

B.1 and B.2

If the VCT and letdown isolation valves cannot be returned to OPERABLE status within the associated Completion Time, the plant must be brought to a configuration in which the LCO does not apply. As an alternative (Required Actions B.1 and B.2), the VCT and letdown return line line must be closed and secured within 8 hours to isolate the unborated water sources. The allowed Completion Time of 8 hours is reasonable, based on operating experience, to return the isolation valves to an OPERABLE condition in an orderly manner. The 31 day frequency is based on engineering judgment and is considered reasonable in view of other administrative controls that make valve opening an unlikely possibility.

SURVEILLANCE REQUIREMENTS

SR 3.1.8.1

Periodic surveillance testing of VCT and letdown isolation valves is required by the ASME Code. This verifies that the measured performance, receipt of isolation signal to full closure, is within an acceptable tolerance of the performance assumed in the plant safety analysis. SRs are specified in the Inservice Testing Program of the ASME Code.

The ASME Code provides the activities and Frequencies necessary to satisfy the requirements.

SR 3.1.8.2

This periodic surveillance is performed on the VCT and letdown isolation valves to verify that the actuation signal causes the appropriate valves to move to their correct position within the allowable design basis response time.

The 24 month frequency is based on the need to perform this Surveillance under conditions that apply during a plant outage and because of the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The Frequency is acceptable based on consideration of the design reliability of the equipment. The actuation logic is tested as part of Protection System testing.

REFERENCES

1. FSAR Chapter 15.
2. 10 CFR 50.36, Technical Specifications~~, August, 2007~~.

BASES

BACKGROUND (continued)

The boron reactivity coefficient is determined by dividing the measured bank worth by the measured boron concentration difference. Performance of this test could violate LCO 3.1.4, "RCCA Group Alignment Limits," LCO 3.1.5, "Shutdown Bank Insertion Limit," or LCO 3.1.6, "Control Bank Insertion Limits."

c. The Control Rod Worth Test is used to measure the reactivity worth of selected control banks. This test is performed at HZP and has three alternative methods of performance. The first method, the Boron Exchange Method, varies the reactor coolant boron concentration and moves the selected control bank in response to the changing boron concentration. The reactivity changes are measured with a reactivity computer. This sequence is repeated for the remaining control banks. The second method, the Rod Swap Method, measures the worth of a predetermined reference bank using the Boron Exchange Method above. The reference bank is then nearly fully inserted into the core. The selected bank is then inserted into the core as the reference bank is withdrawn. The HZP critical conditions are then determined with the selected bank fully inserted into the core. The worth of the selected bank is inferred, based on the position of the reference bank with respect to the selected bank. This sequence is repeated as necessary for the remaining control banks. The third method, the Boron Endpoint Method, moves the selected control bank over its entire length of travel and then varies the reactor coolant boron concentration to achieve HZP criticality again. The difference in boron concentration is the worth of the selected control bank. This sequence is repeated for the remaining control banks. Performance of this test could violate LCO 3.1.3, 3.1.4, LCO 3.1.5, or LCO 3.1.6.

d. The ITC Test measures the ITC of the reactor. This test is performed at HZP and has two methods of performance. The first method, the Slope Method, varies RCS temperature in a slow and continuous manner. The reactivity change is measured with a reactivity computer as a function of the temperature change. The ITC is the slope of the reactivity versus the temperature plot. The test is repeated by reversing the direction of the temperature change, and the final ITC is the average of the two calculated ITCs. The second method, the Endpoint Method, changes the RCS temperature and measures the reactivity at the beginning and end of the temperature change. The ITC is the total reactivity change divided by the total temperature change. The test is repeated by reversing the direction of the temperature change, and the final ITC is the average of the two calculated ITCs. Performance of this test ~~could~~ should not challenge LCO 3.4.2, "RCS Minimum Temperature for Criticality" with a cooldown of ≤ 8°F.~~"~~

16-272

BASES

SURVEILLANCE REQUIREMENTS (continued)

g. Isothermal temperature coefficient (ITC), when below the point of adding heat (POAH);

h. Moderate defect, when above the POAH; and

i. Doppler defect, when above the POAH.

Using the ITC accounts for Doppler reactivity in this calculation when the reactor is subcritical or critical but below the POAH and the fuel temperature will be changing at the same rate as the RCS.

The Frequency of 24 hours is based on the generally slow change in required boron concentration and on the low probability of an accident occurring without the required SDM.

16-270

SR 3.1.9.3

The power range and intermediate range neutron detectors must be verified to be OPERABLE in MODE 2 by LCO 3.3.1, “Protection System (PS).” A SENSOR OPERATIONAL TEST is performed on each power range and intermediate range channel prior to initiation of the PHYSICS TESTS. This will ensure that the PS is properly aligned to provide the required degree of core protection during the performance of the PHYSICS TESTS.

REFERENCES

1. 10 CFR 50, Appendix B, Section XI~~, March, 2006~~.

~~2. 10 CFR 50.52 "Licenses, Certifications, and Approvals for Nuclear Power Plants," August, 2007.~~

~~3~~2. 10 CFR 50.59 "Changes, Tests and Experiments.~~,~~" ~~August, 2007.~~

~~4~~3. Regulatory Guide 1.68, Revision 2, August, 1978.

~~5~~4. ANSI/ANS-19.6.1-2005, “Reload Startup Physics Tests for Pressurized Water Reactors,”~~November 29,~~ 2005.

~~6~~5. ANP-10263P Codes and Methods Applicability Report for the U.S. EPR, August, 2006.

BASES

BACKGROUND (continued)

insertion, and by calculating an actual value of DNBR and LPD, for comparison to:

- The respective trip setpoints in the PS and;
- The limits for critical operation in the RCSL system.

16-287

Thus the RCSL system continuously indicates to the operator how far the core is from the operating LPD and DNBR limits, and provides an alarm in the Main Control Room if any limit is exceeded. Such a condition signifies a reduction in the capability of the plant to withstand an AOO or postulated accident. This~~, but~~ does not necessarily imply a violation of fuel design limits because the PS initiates a reactor trip when the High LPD reactor trip setpoint is reached..

The calculation of the maximum linear power density (in kW/ft) in the PS (High Linear Power Density function) and in the RCSL system (High Linear Power Density Limitation function) is based on the readings of the SPNDs.~~ and fixed incore instrumentation.~~

16-273

Twelve fuel assemblies are instrumented with SPND fingers which are distributed radially over the core such that their signals are representative of the key core parameters for different perturbation modes and fuel management schemes. Each of the twelve SPND fingers contains six detectors. In each finger, three SPNDs are located in the top core half and the other three in the bottom core half to detect the peak power density occurring in either the top or bottom halves. Thus they can cover all possible power distributions normal or transient. Axial locations are always situated between two grids to rule out the effect of flux depression in the vicinity of the grids.

Flux mapping is performed periodically with the Aeroball Measurement System (AMS), including reference heat balance, to provide an accurate image of the absolute (i.e. in kW/ft) 3D-power distribution. Based on this flux map, each SPND signal is ~~calibrated~~ adjusted by a unique calibration factor to the peak power density within its axial slice. After calibration, all twelve SPNDs within the same axial slice therefore provide the same value, which corresponds to the maximum linear power density value for that axial slice.

16-287

The SPND signals are also calibrated to reproduce the power distribution of the hot channel, for minimum DNBR calculation (see LCO 3.2.~~2~~ 3 "Departure from Nucleate Boiling Ratio"), and to reproduce the average axial power of each core half, for AXIAL OFFSET calculation (see LCO 3.2.4 "AXIAL OFFSET (AO)").

BASES

BACKGROUND (continued)

~~The SPND signal gradually increases (conservative) and the gain constants must be periodically recalibrated to prevent unnecessary LPD penalties. Likewise, core burnup can produce power distribution changes that result in non-conservative SPND signals and also require periodic recalibration. The setpoint analyses account for the uncertainty inherent for a given AMS calibration frequency.~~The SPND signal to flux ratio gradually increases due to buildup of ^{60}Co and ^{61}Co. This effect is greater than the decrease in sensitivity due to ^{59}Co burnup. As a result, the SPND signal will always get more conservative with burnup.

16-285

From the time of calibration, a decreasing (non-conservative) SPND signal can result from changes in core power distributions due to core burnup. The signal variation due to power distribution changes can cause either a decreased or increased SPND signal. This variation in the SPND signal requires periodic recalibration to minimize uncertainties. The setpoint analyses account for the uncertainty inherent for a given AMS calibration frequency.

The maximum linear power density is monitored continuously by the "High Linear Power Density LCO" function of the RCSL System. Separate LCO setpoints exist for both the upper and lower half of the core. Violation of the linear power density operating limit initiates the following automatic and staggered countermeasures:

First High LPD LCO 1 level

- Audible alarm in the control room
- Prevent dilution signal (only for LPD LCO 1 signal in lower half of the core)
- RCCA bank withdrawal blocking signal
- Turbine generator power increase blocking signal
- RCCA bank insertion blocking signal (only for LPD LCO 1 signal in lower half of the core)

Second High LPD LCO 2 level

- Reduce turbine generator power signal
- Insert RCCA bank signal (only for LPD LCO 2 signal in upper half of the core)

16-274

The surveillance setpoint corresponds to the ~~first~~ First High LPD LCO 1 level ~~threshold~~. The objective of these staggered actions is to prevent operations leading to a further increase of linear power density so that the maximum LPD value can be quickly restored to below its limit.

BASES

APPLICABLE SAFETY ANALYSES

The power distribution and RCCA insertion and alignment LCOs prevent core power distributions from reaching levels that violate acceptance criteria regarding fuel design and coolability. The power density at any point in the core must be limited to maintain the fuel design criteria. This is accomplished by maintaining the power distribution and reactor coolant conditions such that the peak power density and minimum DNBR are within operating limits supported by accident analyses (Ref. 1).

Maximum LPD limit assumed in the LOCA analysis (Ref. 1) is typically limiting relative to the maximum LPD assumed in safety analyses for other AOO and postulated accidents. Therefore this LCO provides conservative limits for other AOOs such as uncontrolled RCCA bank withdrawal.

Fuel cladding damage does not typically occur while the unit is operating at conditions outside the limits of this LCO during normal operation. Fuel cladding damage could result, however, if an AOO event occurs from initial conditions outside the limits of this LCO. The potential for fuel cladding damage exists because changes in the power distribution can cause increased power peaking during the transient.

LPD satisfies Criterion 2 of 10 CFR 50.36(~~d~~c)(2)(ii) (Ref. 3).

LCO

The LOCA safety analysis generally determines the maximum permitted linear power density for the upper half of the core. The LCO limit ensures that the post-LOCA fuel cladding temperature does not exceed a specified maximum limit of 2200°F. As a consequence the LCO ensures that the maximum LPD in the core is not exceeded in the event of a LOCA. A separate limit is also provided for the lower half of the core. This limit provides margin for those events that result in a axial redistribution of power towards the bottom of the core. Both limits ensure margin to fuel centerline melt and maintain clad strain < 1% during all AOOs. The LPD limits are provided in the COLR.

APPLICABILITY

16-284

Power distribution is a concern any time the reactor ~~is critical~~ thermal power is greater than approximately 10% RTP. Therefore, the~~. The~~ power distribution LCOs~~, however,~~ are ~~only~~ applicable in MODE 1 above 10% RTP. This LCO is not a concern at or below 10% RTP because the core is operating well below its thermal limits.

BASES

ACTIONS

A.1

With the LPD exceeding its limit, excessive fuel damage could occur following an AOO or postulated accident. In this condition, prompt action must be taken to restore the LPD to within the specified limits.

The one hour limit to restore the LPD to within its specified limits is reasonable since the likelihood of an accident happening over this short period is negligible. The one hour Completion Time also allows the operator sufficient time for evaluating core conditions and confirming automatic actions have been effective or initiating proper corrective actions to restore the LPD to within its specified limits.

B.1

If the RCSL System is not available the OPERABLE LPD channels are monitored to ensure that the LPD limit is not exceeded. Operation within this limit ensures that no postulated accident results in consequences more severe than those described in the FSAR, Tier 2, Chapter 15.

When operating with the RCSL System out of service there is a possibility of a slow undetectable transient that degrades the LPD slowly over a 1 hour period and is then followed by an AOO or an accident. To remedy this, the PS calculated values of LPD are monitored every hour when the RCSL System is out of service. The 1 hour Frequency is adequate to allow the operator to identify an adverse trend in conditions that could result in an approach to the LPD LCO.

When the RCSL System is out of service and the LPD is not within a region of acceptable operation, immediate action is required to reduce power until the LPD is within the existing RCSL System out of service TS limits.

~~B~~C.1

16-276

If the value of LPD is not restored to within its limits within the required Completion Time; the unit must be brought in a MODE or condition where the LCO is no longer applicable. This is done by placing the plant in at least MODE 1 with THERMAL POWER ≤ ~~<~~10% RTP within 6 hours.

16-275

The allowed Completion Time of 6 hours is reasonable to reach MODE 1 with THERMAL POWER ≤ ~~<~~10% RTP from full power operation in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1

The Surveillance requires the operator to verify that the LPD is within limits. This verification is in addition to the automatic checking performed by the RCSL System. The Surveillance can be performed by obtaining the current LPD generated by the RCSL System (providing the RCSL System is in service and has been properly calibrated) and verifying the value is within limits specified in the COLR. Alternately, the verification may also be performed by manually monitoring each OPERABLE PS LPD division and verifying the value is within limits specified in the COLR. Since there are four different divisions based on individual loop conditions, it is necessary to monitor the most limiting LPD division. A 12 hour Frequency is sufficient to allow the operator to identify trends that would result in an approach to the LPD limits.

16-276

SR 3.2.1.2

With the RCSL System out of service, the operator must monitor the LPD with each OPERABLE LPD channel. A 1 hour Frequency is sufficient to allow the operator to identify trends that would result in an approach to the LPD limits.

This SR is modified by a Note that states that the SR is only required to be met when the LPD System is out of service. Continuous monitoring of the LPD is provided by the RCSL System, which calculates core power and core power operating limits based on the LPD and continuously displays these limits to the operator. A RCSL System margin alarm is annunciated in the event that the THERMAL POWER exceeds the core power operating limit based on the LPD.

REFERENCES

1. FSAR Chapter 15.
2. FSAR Chapter 6.
3. 10 CFR 50.36, Technical Specifications~~, August, 2007~~.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)

BASES

BACKGROUND

The purpose of this LCO is to establish limits on the power density at any point in the core so that the fuel design criteria are not exceeded and the accident analysis assumptions remain valid specifically for the loss of cooling accident (LOCA) analyses. ~~An~~ In addition this limit allows for further constraining the initial operating conditions assumed in other accident analyses. The design limits on local (pellet) and integrated fuel rod peak power density are expressed in terms of hot channel factors. Control of the core power distribution with respect to these factors ensures that local conditions in the fuel rods and coolant channels do not challenge core integrity at any location during either normal operation or a postulated accident analyzed in the safety analyses.

16-277

$F^N_{\Delta H}$ is defined as the ratio of the highest integrated linear power along any fuel rod to the average integrated fuel rod power. Therefore, $F^N_{\Delta H}$ is a measure of the maximum total power produced in a fuel rod.

$F^N_{\Delta H}$ is sensitive to fuel loading patterns, bank insertion, and fuel burnup. $F^N_{\Delta H}$ typically increases with control bank insertion and typically decreases with fuel burnup.

$F^N_{\Delta H}$ is not directly measurable but is inferred from a power distribution map obtained with the Aeroball Measurement System (AMS). Specifically, the results of a three dimensional power distribution map are analyzed by computer to determine $F^N_{\Delta H}$. This factor is calculated at least every ~~15~~ 31 effective full power days (EFPD). However, during power operation, the global power distribution is continuously monitored by LCO 3.2.4, "AXIAL OFFSET (AO)," and LCO 3.2.5, "AZIMUTHAL POWER IMBLANCE (API)," which address directly and continuously measured process variables.

16-278

Since DNBR and LPD are monitored independently and protected with separate LCOs which specifically account for the 3D power distribution in the core, $F^N_{\Delta H}$ limits are used to verify the acceptability of the resulting limiting peak cladding temperatures that are used in the LOCA safety analyses.

Operation outside the LCO limits may produce unacceptable consequences if an anticipated operation occurrence (AOO) or other postulated accident occurs.

BASES

APPLICABLE SAFETY ANALYSES

This LCO provides limits on $F^N_{\Delta H}$ for the following purposes:

a. Restrict initial LPD to a value which ensures that during a LOCA the peak clad temperatures do not exceed 2200°F; and

16-278

b. Limit the scope of power distributions from which an accident may be initiated for all FSAR Chapter 15 events.

~~The LOCA safety analysis indirectly models~~The nuclear enthalpy rise hot channel factor ($F^N_{\Delta H}$) is used ~~as an input parameter. The Nuclear Heat Flux Hot Channel Factor ($F_Q(Z)$) and the axial peaking factors are inserted~~ directly in~~to~~ the LOCA ~~safety~~ analys~~e~~is ~~that~~to verify the acceptability of the resulting peak cladding temperature (Ref. 1).

$F^N_{\Delta H}$ shall be maintained within the limits specified in the COLR. The $F^N_{\Delta H}$ limit identifies the coolant flow channel with the maximum enthalpy rise. This channel has the least heat removal capability and thus the highest probability for a departure from nucleate boiling (DNB). The limiting value of $F^N_{\Delta H}$, described by the equation contained in the COLR, is the design radial peaking factor used in the unit safety analyses.

$F^N_{\Delta H}$ satisfies Criterion 2 of 10 CFR 50.36(~~d~~c)(2)(ii) (Ref. 2).

LCO

$F^N_{\Delta H}$ shall be maintained within the limits specified in the COLR. The $F^N_{\Delta H}$ limit identifies the coolant flow channel with the maximum enthalpy rise. This channel has the least heat removal capability and thus the highest probability for a DNB. The limiting value of $F^N_{\Delta H}$, described by the equation contained in the COLR, is the design radial peaking factor used in the unit safety analyses.

APPLICABILITY

MODE 1 with THERMAL POWER > 90% RTP.

Applicability in MODE 1 ~~with THERMAL POWER ≤ 90% RTP is not required because this LCO applies only to LOCA analyses.~~is required to limit the scope of power distributions from which an accident may be initiated for all FSAR Chapter 15 events. LOCA events are limiting at ~~HFP~~ hot full power because at lower powers there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to challenge licensing criteria.

16-278

BASES

ACTIONS

A.1 and A.2

When $F^N_{\Delta H}$ exceeds its limit there is little concern regarding DNBR or LPD since these parameters are independently monitored and protected with other trips. However, the LOCA analyses assume this limit at the initiation of the transient therefore exceeding it could result in peak clad temperatures in excess of the acceptance criteria.

The 1 hour limit to reduce power by 1% for each 1% that $F^N_{\Delta H}$ limit is exceeded by allows for an orderly power reduction that reduces the hot fuel rod integrated power to near its 100% limit.

The 4 hour limit then provides adequate time to confirm that $F^N_{\Delta H}$ has been restored or make necessary adjustments through control rod movements or further power reductions. This completion time also provides a reasonable limit on the amount of time which the plant may outside the $F^N_{\Delta H}$ limit.

B.1

16-278

When the Required Action cannot be met or completed within the required Completion Time; the unit must be brought in a MODE or condition where the LCO is no longer applicable. This is done by placing the plant in at least MODE ~~1~~2 ~~with THERMAL POWER ≤ 90% RTP~~

The allowed Completion Time of ~~2~~ 6 hours is reasonable to reach MODE ~~1~~ 2 ~~with THERMAL POWER ≤ 90% RTP~~ from full power operation in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

The value of $F^N_{\Delta H}$ is determined by taking an AMS flux map. A data reduction computer program (POWERTRAX™) then calculates the maximum value of $F^N_{\Delta H}$ from the measured flux distribution. The measured value of $F^N_{\Delta H}$ must be multiplied by the appropriate measurement uncertainty before making comparisons to the $F^N_{\Delta H}$ limit.

BASES

SURVEILLANCE REQUIREMENTS (continued)

Confirming $F^N_{\Delta H}$ in MODE 1 after an outage and before exceeding ~~98~~70% power ensures that plant is operating within the limit given the major change in power distributions resulting from the core reload.

16-278

The ~~15~~ 31 EFPD frequency between $F^N_{\Delta H}$ confirmations is also acceptable since power distributions change relatively slowly over this amount of fuel burnup. Accordingly, this frequency is short enough that the $F^N_{\Delta H}$ limit cannot be exceeded for any significant period of operation.

This SR is modified by a Note that states the SR is not required to be performed until 24 hours after exceeding ~~90~~40% power. This time period allows sufficient time to perform the required surveillance.

REFERENCES

1. 10 CFR 50.46, Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors~~, August, 2007~~.
2. 10 CFR 50.36, Technical Specifications~~, August, 2007~~.

BASES

BACKGROUND (continued)

The PS and the RCSL system are capable of verifying that the LPD and the DNBR do not exceed their limits. The PS and the RCSL system perform this function by continuously monitoring incore Self-Powered Neutron Detectors (SPND) measurements, thermal-hydraulic data, and Rod Cluster Control Assemblies (RCCA) insertion, and by calculating an actual value of DNBR and LPD, for comparison to:

- The respective trip setpoints in the PS and;
- The limits of acceptable operation in the RCSL system.

16-290

Thus the RCSL system indicates continuously to the operator how far the core is from the operating LPD and DNBR limits, and provides an alarm in the Main Control Room if any limit is exceeded. Such a condition signifies a reduction in the capability of the plant to withstand an AOO or postulated accident. This~~, but~~ does not necessarily imply a violation of fuel design limits because the PS initiates a reactor trip when the Low DNBR reactor trip setpoint is reached.

The calculation of the minimum DNBR in the PS (Low DNBR trip function) and in the RCSL system (Low DNBR LCO function) is based for both systems on:

- The power density distribution in the hot channel, which is based on the readings of the SPND (reconstruction) fixed incore instrumentation;
- The inlet reactor coolant temperature;
- The pressurizer pressure and;
- The RCS flow rate.

Twelve fuel assemblies are instrumented with SPND fingers which are distributed radially over the core such that their signals are representative of the key core parameters for different perturbation modes and fuel management schemes. Each of the twelve SPND fingers contains six detectors. In each finger, three SPNDs are located in the top core half and the other three in the bottom core half to detect the peak power density occurring in either the top or bottom halves. Thus they can cover all possible power distributions normal or transient. Axial SPND locations are always situated between two grids to rule out the effect of flux depression in the vicinity of the grids.

16-289

Flux mapping is performed periodically with the Aeroball Measurement System (AMS), including reference heat balance, to provide an accurate image of the absolute (i.e. in kW/ft) 3D-power distribution. In each finger, the six SPND signals are then ~~calibrated~~adjusted by a unique calibration factor to the power density of the hot rod integrated on the length of the

BASES

BACKGROUND (continued)

SPND. After calibration, all twelve SPND fingers therefore provide the same axial power shape representative of the power shape of the actual hot channel.

16-285

The SPND signal to flux ratio gradually increases due to buildup of ^{60}Co and ^{61}Co. This effect is greater than the decrease in sensitivity due to ^{59}Co burnup. As a result, the SPND signal will always get more conservative with burnup.

From the time of calibration, a decreasing (non-conservative) SPND signal can result from changes in core power distributions due to core burnup. The signal variation due to power distribution changes can cause either a decreased or increased SPND signal. This variation in the SPND signal requires periodic recalibration to minimize uncertainties. The setpoint analyses account for the uncertainty inherent for a given AMS calibration frequency.
~~The SPND signal gradually increases (conservative) with core burnup and the gain constants must be periodically recalibrated to prevent unnecessary DNBR penalties. Likewise, core burnup can produce power distribution changes that result in non-conservative SPND signals and also require periodic recalibration. Setpoint analyses account for the uncertainty inherent for a given AMS calibration frequency.~~

The PS and the RCSL system use these measurements and a proprietary algorithm to reconstruct the local thermal-hydraulic conditions at the minimum DNBR point in the core and apply the chosen Critical Flux Predictor (Ref. 3) to calculate the DNBR.

The minimum DNBR is monitored continuously by the "Low DNBR LCO" function of the RCSL system and indicated to the operator. Violation of the DNBR operating limit initiates the following automatic and staggered countermeasures:

First Low DNBR LCO 1 level

- RCCA bank withdrawal blocking signal, and
- Turbine generator power increase blocking signal

Second Low DNBR LCO 2 level

- Reduce turbine generator power signal, and
- Insert RCCA bank signal

The surveillance setpoint corresponds to the first Low DNBR LCO threshold. The objective of these staggered actions is to prevent operations leading to a further decrease of the DNBR such that the minimum DNBR value can be quickly restored to above its limit.

BASES

APPLICABLE SAFETY ANALYSES

The power distribution and RCCA insertion and alignment LCOs prevent core power distributions from reaching levels that violate acceptance criteria regarding fuel design and coolability. The DNBR at any point in the core must be limited to maintain the fuel design criteria. This is accomplished by maintaining the power distribution and reactor coolant conditions such that the peak power density and minimum DNBR are within operating limits supported by accident analyses.

16-280

~~The minimum DNBR limit is typically established based on the Loss of Coolant Flow accident which is limiting relative to the maximum ΔDNBR assumed in safety analyses for all other AOOs. Therefore this LCO provides conservative limits for all other AOOs.~~The maximum change in the Departure from Nucleate Boiling Ratio (ΔDNBR) is evaluated for Chapter 15 Safety Analyses that utilize the methodology described in Reference 5. From the Chapter 15 Safety Analyses, the Complete Loss of Flow event typically provides the maximum ΔDNBR (based upon core loading characteristics). As described in Reference 5, the DNB LCO established with the maximum ΔDNBR in conjunction with the Low DNBR trip function assures protection from DNB for all AOOs.

Fuel cladding damage does not normally occur while the unit is operating at conditions outside the limits of this LCO during normal operation. Fuel cladding damage could result, however, if an AOO event occurs from initial conditions outside the limits of this LCO. The potential for fuel cladding damage exists because changes in the power distribution can cause a reduction in DNB margin at the initiation of a fast transient such that other plant trips can no longer respond in time to protect the fuel design limits.

DNBR satisfies Criterion 2 of 10 CFR 50.36(~~d~~c)(2)(ii) (Ref. 4).

LCO

The Loss of Flow accident generally establishes the DNB LCO limits as this transient is too fast to be protected by the Low DNBR trip in the PS. The DNB LCO therefore ensures that the plant operates far enough away from the DNBR design limit that in the event of a very fast transient sufficient time exists for other plants trips to intervene prior to exceeding the DNBR design limit. The DNBR limits are provided in the COLR.

APPLICABILITY

The DNB LCO is only applicable in MODE 1 above 10% RTP. This LCO is not a concern below 10% RTP and for lower operating MODES because the stored energy in the fuel and the energy being transferred to the reactor coolant are sufficiently low that DNBR is no longer a concern.

ACTIONS

A.1

With the DNBR exceeding its limit, excessive fuel damage could occur following an AOO or postulated accident. In this condition, prompt action must be taken to restore the DNBR to within the specified limits.

BASES

ACTIONS (continued)

The 1 hour limit to restore the DNBR to within its specified limits is reasonable since the likelihood of an accident happening over this short period is negligible. The 1 hour Completion Time also allows the operator sufficient time for evaluating core conditions and either confirm automatic actions have been effective or initiating proper corrective actions to restore the DNBR to within its specified limits.

16-279

B.1

If the RCSL System is not available the OPERABLE DNBR channels are monitored to ensure that the DNBR limit is not exceeded. Operation within this limit ensures that no postulated accident results in consequences more severe than those described in the FSAR, Tier 2, Chapter 15.

When operating with the RCSL System out of service there is a possibility of a slow undetectable transient that degrades the DNBR slowly over a 1 hour period and is then followed by an AOO or an accident. To remedy this, the PS calculated values of DNBR are monitored every hour when the RCSL System is out of service. The 1 hour Frequency is adequate to allow the operator to identify an adverse trend in conditions that could result in an approach to the DNBR LCO.

When the RCSL System is out of service and the DNBR is not within a region of acceptable operation, immediate action is required to reduce power until the DNBR is within the existing RCSL System out of service TS limits.

~~B~~C.1

If the value of DNBR is not restored within its specified limits within the required Completion Time; the unit must be brought in a MODE or condition where the LCO is no longer applicable. This is done by placing the plant in at least MODE 1 with THERMAL POWER ≤ 10% RTP within 6 hours.

The allowed Completion Time of 6 hours is reasonable to reach MODE 1 with THERMAL POWER ≤ 10% RTP from full power operation in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1

The Surveillance requires the operator to verify that the DNBR is within limits. This verification is in addition to the automatic checking performed

BASES

SURVEILLANCE REQUIREMENTS (continued)

by the RCSL System. The Surveillance can be performed by obtaining the current DNBR generated by the RCSL System (providing the RCSL System is in service and has been properly calibrated) and verifying the value is within limits specified in the COLR. Alternately, the verification may also be performed by manually monitoring each OPERABLE PS DNBR division and verifying the value is within limits specified in the COLR. Since there are four different divisions based on individual loop conditions, it is necessary to monitor the most limiting DNBR division. A 12 hour Frequency is sufficient to allow the operator to identify trends that would result in an approach to the DNBR limits.

16-279

SR 3.2.3.2

With the RCSL System out of service, the operator must monitor the DNBR with each OPERABLE DNBR channel. A 1 hour Frequency is sufficient to allow the operator to identify trends that would result in an approach to the DNBR limits.

This SR is modified by a Note that states that the SR is only required to be met when the RCSL System is out of service. Continuous monitoring of the DNBR is provided by the RCSL System, which calculates core power and core power operating limits based on the DNBR and continuously displays these limits to the operator. A RCSL System margin alarm is annunciated in the event that the THERMAL POWER exceeds the core power operating limit based on the DNBR.

BASES

REFERENCES

1. FSAR Chapter 15.

2. FSAR Chapter 6.

3. ANP-10269P-A, “The ACH-2 CHF Correlation for the U.S. EPR Topical Report,” AREVA NP Inc, 2008. ~~and Supplement 1, Rev. 0, August, 2007.~~

4. ~~4.~~ 10 CFR 50.36, Technical Specifications~~, August, 2007~~.

5. ANP-10287P, Revision 0, “Incore Trip Setpoint and Transient Methodology for U.S. EPR Topical Report,” AREVA NP Inc, 2007.

16-280

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 AXIAL OFFSET (AO)

BASES

BACKGROUND

The purpose of this LCO is to limit the core power distribution to the initial values assumed in the anticipated operational occurances (AOOs) and postulated accident analysis. Operation within the limits imposed by this LCO either limits or prevents potential fuel cladding failures that could breach the primary fission product barrier and release fission products to the reactor coolant in the event of a loss of coolant accident (LOCA), loss of flow accident, ejected rod cluster control assembly (RCCA) accident, or other postulated accident requiring termination by a Protection System (PS) trip function. This LCO limits the amount of damage to the fuel cladding during an accident by ensuring that the plant is operating within acceptable conditions at the onset of a transient (Ref. 2).

16-281

AXIAL OFFSET (AO) is a measure of the axial power distribution in the core. The purpose for a limit on AO is to limit the axial power distributions to initial values assumed in the AOOs and postulated accident analyses. Extreme shifts in power towards either the top or bottom of the core can have adverse impacts during an accident. In general, top-peaked power shapes have lower minimum departure from nucleate boiling ratio(s) (MDNBRs) to start with while bottom-peaked shapes tend to result in more significant DNBR degradation during a transient. Significant shifts in either direction can lead to increased linear power densities (LPDs). Minimizing power distribution skewing over time also minimizes xenon distribution skewing, which is a significant factor in controlling axial power distribution.

16-281

16-277

The reactor control, surveillance and limitations (RCSL) system continuously monitors the AXIAL OFFSET based on evaluations of the core power distribution using the incore self-powered ~~nuclear~~ neutron detectors (SPNDs). Twelve fuel assemblies are instrumented with SPND fingers which are distributed radially over the core such that their signals are representative of the key core parameters for different perturbation modes and fuel management schemes. Each of the twelve SPND fingers contains six detectors. In each finger, three SPNDs are located in the top core half and the other three in the bottom core half to detect power density occurring in top and bottom halves. Thus they can cover all possible power distributions, normal or accidental. Axial locations are always between two grids to rule out the effect of flux depression in the vicinity of the grids.

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BASES

BACKGROUND (continued)

16-285

The SPND signal to flux ratio gradually increases due to buildup of ^{60}Co and ^{61}Co. This effect is greater than the decrease in sensitivity due to ^{59}Co burnup. As a result, the SPND signal will always get more conservative with burnup.

From the time of calibration, a decreasing (non-conservative) SPND signal can result from changes in core power distributions due to core burnup. The signal variation due to power distribution changes can cause either a decreased or increased SPND signal. This variation in the SPND signal requires periodic recalibration to minimize uncertainties. The setpoint analyses account for the uncertainty inherent for a given AMS calibration frequency.
~~The SPND signal gradually increases (conservative) with core burnup and the gain constants must be periodically recalibrated to prevent unnecessary AO penalties. Likewise, core burnup can produce power distribution changes that result in non-conservative SPND signals and also require periodic recalibration. Setpoint analyses account for the uncertainty inherent for a given AMS calibration frequency.~~

The AXIAL OFFSET is monitored continuously by the "Axial Power Shape LCO" function of the RCSL system. The Axial Power Shape LCO function aims at informing the operator if the AO limit is violated. The AO setpoint is function of the core THERMAL POWER level. Violation of the AO operating limits initiates an alarm in the main control room and an automatic signal blocking any turbine generator power increase. Active countermeasures are not initiated so as not to interfere with those automatic actions initiated by the "AXIAL OFFSET Control" function in RCSL, which will tend to restore the AO to within limits.During power operation with the RCSL system out of service, SPND signals from the PS may be manually monitored to determine AO and verify the LCO limit is maintained.

APPLICABLE SAFETY ANALYSES

The maximum AO limit is established for the following purposes:

a. Restrict initial AO to a value which ensures that during a LOCA the peak clad temperatures do not exceed 2200°F (Ref. 1) and;

b. Restrict the scope of power distributions assumed as initial conditions in analyzing anticipated operational occurrences (AOOs) and postulated accidents.

Fuel cladding damage does not typically occur while the unit is operating at conditions outside the limits of this LCO during normal operation. Fuel cladding damage could result, however, if an AOO event occurs from initial conditions outside the limits of this LCO. The potential for fuel

BASES

LCO

The positive AO limit is generally established to minimize or eliminate the consequences of the rod ejection or uncontrolled RCCA withdrawal transient. The negative AO limit is generally established by The Loss of Flow transient. The limits are established around a target AO that is a function of the core management scheme. The AO limits are provided in the COLR.

Violation of this LCO could produce unacceptable consequences if an AOO or postulated accident occurs while the AO is outside its specified limits.

APPLICABILITY

The AO LCO is only applicable in MODE 1 above 50% RTP. This LCO is not a concern below 50% RTP and for lower operating MODES because xenon transients generated within the lower power level range are not severe. In addition, significant margin to thermal limits exists at lower power levels and therefore thermal limits are not significantly challenged.

ACTIONS

A.1

With the AO exceeding its limit, excessive fuel damage could occur following an AOO or postulated accident. In this condition, prompt action must be taken to restore the AO to within the specified limit.
The 1 hour time period to restore the AO to within its specified limit is reasonable and ensures that the core does not continue to operate in this Condition. The 1 hour Completion Time also allows the operator sufficient time for evaluating core conditions and confirming automatic actions have been effective or initiating proper corrective actions to restore the AO to within its specified limit.

16-282

B.1

If the RCSL System is not available the OPERABLE AO channels are monitored to ensure that the AO limit is not exceeded. Operation within this limit ensures that no postulated accident results in consequences more severe than those described in the FSAR, Tier 2, Chapter 15.

When operating with the RCSL System out of service there is a possibility of a slow undetectable transient that degrades the AO slowly over a 1 hour period and is then followed by an AOO or an accident. To remedy this, the PS calculated values of AO are monitored every hour when the RCSL System is out of service. The 1 hour Frequency is adequate to allow the operator to identify an adverse trend in conditions that could result in an approach to the AO LCO.

When the RCSL System is out of service and the AO is not within a region of acceptable operation, immediate action is required to reduce

BASES

ACTIONS (continued)

power until the AO is within the existing RCSL System out of service TS limits.

~~B~~C.1

16-282

If the value of AO is not restored to within its specified limit within the required Completion Time; the unit must be placed in a MODE or condition where the LCO is no longer applicable. This is done by placing the plant in a least MODE 1 with THERMAL POWER to < 50% RTP within 4 hours.

The allowed Completion Time of 4 hours is reasonable to reach MODE 1 with THERMAL POWER < 50% RTP from full power operation in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1

The Surveillance requires the operator to verify that the AO is within limits about the target AO. This verification is in addition to the automatic checking performed by the RCSL System. The Surveillance can be performed by obtaining the current AO generated by the RCSL System (providing the RCSL System is in service and has been properly calibrated) and verifying the value is within limits specified in the COLR. Alternately, the verification may also be performed by manually monitoring each OPERABLE PS AO division and verifying the value is within limits specified in the COLR. Since there are four different divisions based on individual loop conditions, it is necessary to monitor the most limiting AO division. A 12 hour Frequency is sufficient to allow the operator to identify trends that would result in an approach to the AO limits.

Another option is to monitor the AO through the generation of an AMS flux map. A data reduction computer program (POWERTRAX™) then calculates the core wide assembly nodal power distribution from the measured flux distribution.

SR 3.2.4.2

Target AO is determined at equilibrium xenon conditions in conjunction with a full core flux map.

BASES

SURVEILLANCE REQUIREMENTS (continued)

16-282

SR 3.2.4.3

With the RCSL System out of service, the operator must monitor the AO with each OPERABLE PS AO division. A 1 hour Frequency is sufficient to allow the operator to identify trends that would result in an approach to the AO limits.

This SR is modified by a Note that states that the SR is only required to be met when the RCSL System is out of service. Continuous monitoring of the AO is provided by the RCSL System, which calculates core power and core power operating limits based on the AO and continuously displays these limits to the operator. A RCSL System margin alarm is annunciated in the event that the THERMAL POWER exceeds the core power operating limit based on the AO.

REFERENCES

1. 10 CFR 50.46, Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors~~, August, 2007~~.

2. FSAR Chapter 15.

3. 10 CFR 50.36, Technical Specifications~~, August, 2007~~.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1 Containment

BASES

BACKGROUND

The containment consists of a cylindrical reinforced concrete outer Shield Building, a cylindrical post-tensioned concrete inner Containment Building with a 0.25-inch thick steel liner, and an annular space between the two buildings. A common basemat supports both structures. The containment, including all its penetrations, is a low leakage shell designed to contain radioactive material that may be released from the reactor core following a design basis loss of coolant accident (LOCA). Additionally, the containment provides shielding from the fission products that may be present in the containment atmosphere following accident conditions.

16-248

An annular space exists between the walls and domes of the Containment Building and the Shield Building to permit inservice inspection and collection of containment outleakage. The Shield Building provides protection from external hazards and the Containment Building provides for leak tightness.

Containment piping penetration assemblies provide for the passage of process, service, sampling and instrumentation pipelines into the reactor vessel while maintaining containment OPERABILITY. The Shield Building provides shielding and allows controlled release of the annulus atmosphere under accident conditions, as well as ~~protecting~~ environmental missle protection for the Containment Building ~~from external hazards~~and the Nuclear Steam Supply System.

The inner Containment Building 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. SR 3.6.1.1 leakage rate requirements comply with 10 CFR 50, Appendix J, Option B (Ref. 1), as modified by approved exemptions.

The isolation devices for the penetrations in the containment boundary are a part of the containment leak tight barrier. To maintain this leak tight barrier:

a. All penetrations required to be closed during accident conditions are either:

   1. Capable of being closed by an OPERABLE automatic containment isolation system; or

B 3.6 CONTAINMENT SYSTEMS

B 3.6.6 Shield Building

BASES

BACKGROUND

The shield building is a concrete structure that surrounds the Containment Building. Between the Containment Building and the shield building inner wall is an annular space that collects containment leakage that may occur following a loss of coolant accident (LOCA). This space also allows for periodic inspection of the outer surface of the Containment Building.

16-248

The Annulus Ventilation System (AVS) establishes a negative pressure in the annulus between the shield building and the containment building. Filters in the system then control the release of radioactive contaminants to the environment. The description of the AVS is provided in the Bases for Specification 3.6.7. The shield building is required to be OPERABLE to ensure retention of containment leakage and proper operation of the AVS.

To ensure the retention of containment leakage within the Containment Building:

a. The door in each access opening is closed except when the access opening is being used for normal transit entry and exit, and

b. The sealing mechanism associated with each penetration (e.g., welds, bellows, or O-rings) is OPERABLE.

APPLICABLE SAFETY ANALYSES

The design basis for shield building OPERABILITY is a LOCA. Maintaining shield building OPERABILITY ensures that the release of radioactive material from the containment atmosphere is restricted to those leakage paths and associated leakage rates assumed in the accident analyses.

The shield building satisfies Criterion 3 of 10 CFR 50.36(~~d~~c)(2)(ii).

LCO

Shield building OPERABILITY must be maintained to ensure proper operation of the AVS and to limit radioactive leakage from the containment to those paths and leakage rates assumed in the accident analyses.

The LCO is modified by a Note allowing the shield building boundary to be opened intermittently under administrative controls. This Note only applies to openings in the shield building boundary that can be rapidly restored to the design conditions, such as doors, hatches, floor plugs, and access panels. For entry and exit through doors, the administrative

BASES

LCO (continued)

16-250

control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls should be accomplished by procedures, and consist of stationing a dedicated individual at the opening who is in continuous communication with the operators in the ~~CRE~~control room. This individual will have a method to rapidly close the opening and to restore the ~~CRE~~ shield building boundary to a condition equivalent to the design condition when a need for ~~CRE~~ shield building isolation is indicated.

APPLICABILITY

Maintaining shield building OPERABILITY prevents leakage of radioactive material from the shield building. Radioactive material may enter the shield building from the containment following a LOCA. Therefore, shield building OPERABILITY is required in MODES 1, 2, 3, and 4 when a main steam line break, LOCA, or rod ejection accident could release radioactive material to the containment atmosphere.

In MODES 5 and 6, the probability and consequences of these events are low due to the Reactor Coolant System temperature and pressure limitations in these MODES. Therefore, shield building OPERABILITY is not required in MODE 5 or 6.

ACTIONS

A.1

In the event shield building OPERABILITY is not maintained, shield building OPERABILITY must be restored within 24 hours. Twenty-four hours is a reasonable Completion Time considering the limited leakage design of containment and the low probability of a Design Basis Accident occurring during this time period.

B.1 and B.2

If the shield building cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.6.1

Verifying that shield building annulus negative pressure is within limit ensures that operation remains within the limit assumed in the containment analysis. The 12 hour Frequency of this SR was developed considering operating experience related to shield building annulus

16-258

BASES

APPLICABLE SAFETY ANALYSES (continued)

The CCW System is designed to perform its function with a single failure of any active component, assuming a loss of offsite power.

The CCW System also functions to cool the unit from residual heat removal (RHR) entry conditions ($T_{\cancel{cold}avg}$ < 350°F), to MODE 5 ($\cancel{T_{cold}}$ T_{avg} ≤~~<~~ 200°F), during normal and post accident operations. The time required to cool from 350°F to 200°F is a function of the number of CCW and RHR loops operating. Two CCW trains are sufficient to remove decay heat during subsequent operations with $\cancel{T_{cold}}$ T_{avg} ≤~~<~~ 200°F. This assumes a maximum service water temperature of 95°F occurring simultaneously with the maximum heat loads on the system.

To meet single failure criteria for the RCP thermal barrier cooling function, the load is required to be cooled by a common header which is capable of being connected to two OPERABLE CCW trains. A single failure of a train initiates an automatic system response to transfer the common header to the remaining train.

The CCW System satisfies Criterion 3 of 10 CFR 50.36(~~d~~c)(2)(ii).

LCO

The CCW System consists of four trains. Four CCW trains are required to be OPERABLE to provide the required redundancy to ensure that the system functions to remove post accident heat loads.

A CCW train is considered OPERABLE when:

a. The pump and associated surge tank are OPERABLE; and

b. The associated piping, valves, heat exchanger, and instrumentation and controls required to perform the safety related function are OPERABLE.

With the exception of the RCP thermal barrier cooling common loop, the isolation of CCW from other components or systems not required for safety may render those components or systems inoperable but does not affect the OPERABILITY of the CCW System.

APPLICABILITY

In MODES 1, 2, 3, and 4, the CCW System is a normally operating system, which must be prepared to perform its post accident safety functions, primarily RCS heat removal, which is achieved by cooling the Low Head Safety Injection heat exchanger.

BASES

ACTIONS (continued)

C.1 and C.2

16-250 run in

If a CCW train cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.7.1

This SR is modified by a Note indicating that the isolation of the CCW flow to individual components, other than the RCP thermal barrier cooling common loop, may render those components inoperable but does not affect the OPERABILITY of the CCW System.

Verifying the correct alignment for manual, power operated, and automatic valves in the CCW flow path provides assurance that the proper flow paths exist for CCW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This Surveillance does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.7.7.2

This SR verifies proper automatic operation of the CCW valves on an actual or simulated actuation signal. The CCW System is a normally operating system that cannot be fully actuated as part of routine testing during normal operation. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 24 month Frequency is based on the need to

BASES

APPLICABILITY

In MODES 1, 2, 3, and 4, the ESW System is a normally operating system that is required to support the OPERABILITY of the equipment serviced by the ESW System and required to be OPERABLE in these MODES.

16-250

In MODES 5 and 6, the OPERABILITY requirements of the ESW System are determined by the systems it supports.

ACTIONS

The ~~actions~~ ACTIONS have two Notes added. The first Note indicates that the applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources - Operating," should be entered if an inoperable ESW train results in an inoperable EDG. The second Note indicates that the applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," should be entered if an inoperable ESW train results in an inoperable decay heat removal train. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.

A.1

If one ESW train is inoperable, action must be taken to restore OPERABLE status within 120 days. In this condition, the remaining OPERABLE ESW trains are adequate to perform the heat removal function.

The 120 day Completion Time to restore an ESW train to OPERABLE is reasonable since its operation is not assumed in the safety analysis to mitigate the consequences of postulated accidents or AOOs, it provides a reasonable time for repairs, and the low probability of a postulated accident or AOO occurring during this period.

B.1

If two ESW trains are inoperable, action must be taken to restore one to OPERABLE status within 72 hours. In this condition, the two remaining OPERABLE ESW train are adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in one of the OPERABLE ESW trains could result in loss of ESW System function.

The 72 hour Completion Time is based on the redundant capabilities afforded by the two OPERABLE trains, and the low probability of a postulated accident occurring during this time period.