Duke Energy®

JAMES R. MORRIS
Vice President

Duke Energy Corporation
Catawba Nuclear Station
4800 Concord Road
York, SC 29745

803-701-4251
803-701-3221 fax

March 31, 2010

U. S. Nuclear Regulatory Commission
ATTENTION: Document Control Desk
Washington, D.C. 20555

10CFR50.90

Subject: Duke Energy Carolinas, LLC (Duke Energy)
Catawba Nuclear Station, Units 1 and 2
Docket Nos. 50-413 and 50-414

Application for Technical Specification Change Regarding Risk-Informed Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program

In accordance with the provisions of 10 CFR 50.90, Duke Energy is submitting a request for an amendment to the Technical Specifications (TS) for Catawba Nuclear Station (Catawba) Units 1 and 2.

The proposed amendment would modify Catawba's Technical Specifications by relocating specific surveillance frequencies to a licensee controlled program with the implementation of Nuclear Energy Institute (NEI) 04-10, "Risk-Informed Technical Specification Initiative 5b, Risk-Informed Method for Control of Surveillance Frequencies."

The changes are consistent with the NRC approved Industry/TSTF Standard Technical Specification (STS) change TSTF-425, Revision 3 (ADAMS Accession No. ML080280275). Availability of this TSTF was published in the Federal Register notice on July 6, 2009.

Attachment 1 provides a description of the proposed changes, the requested confirmation of applicability, and plant specific verifications. Attachment 2 provides the Probabilistic Risk Assessment (PRA) technical adequacy documentation. Attachment 3 provides the existing TS pages marked up to illustrate the proposed changes. Attachment 4 provides the proposed TS Bases pages. Attachment 5 provides a cross reference table comparing the TSTF surveillance numbers to the Catawba surveillance numbers. Attachment 6 provides the Proposed No Significant Hazards Consideration.

Duke Energy requests the NRC's review and approval of the proposed license amendment within one year of this submittal. Duke Energy is requesting a 90-day implementation grace period due to the extensive document changes necessary to implement this license amendment. Also, Duke Energy will update applicable sections of the Catawba UFSAR, as necessary, and submit these changes per 10CFR 50.71(e).

In accordance with Duke Energy administrative procedures and the Quality Assurance Program Topical Report, this proposed amendment has been reviewed and approved by the Catawba Plant Operations Review Committee.

In accordance with 10 CFR 50.91, a copy of this proposed amendment is being forwarded to the appropriate South Carolina State officials.

A001
NRR

There are no new commitments being made as a result of this proposed change.

Inquiries regarding this submittal should be directed to Tony Jackson at 803-701-3742.

Sincerely,

James R. Morris

Attachments:

1. Description and Assessment
2. Documentation of PRA Technical Adequacy
3. Proposed Technical Specification Changes
4. Proposed Technical Specification Bases Changes
5. Surveillance Frequency Cross Reference Table
6. Proposed No Significant Hazards Consideration

cc: w/Attachments

L. A. Reyes
NRC Region II Administrator
U.S. Nuclear Regulatory Commission
Atlanta Federal Center
61 Forsyth St., SW, Suite 23T85
Atlanta, GA 30303

J. H. Thompson
Project Manager (Catawba)
U.S. Nuclear Regulatory Commission
One White Flint North, Mail Stop 8-G9A
11555 Rockville Pike
Rockville, MD 20852-2738

G. A. Hutto
NRC Senior Resident Inspector
Catawba Nuclear Station

Susan E. Jenkins, Manager
Radioactive & Infectious Waste Management
Division of Waste Management
South Carolina Department of Health and Environmental Control
2600 Bull St.
Columbia, SC 29201

OATH AND AFFIRMATION

James R. Morris affirms that he is the person who subscribed his name to the foregoing statement, and that all the matters and facts set forth herein are true and correct to the best of his knowledge.

James R. Morris, Site Vice President

Subscribed and sworn to me: 3/31/10
Date

Notary Public

ANTHONY P. JACKSON
NOTARY PUBLIC
MY COMMISSION EXPIRES
JULY 2, 2014
STATE OF SOUTH CAROLINA

My commission expires: 7/2/2014
Date

ATTACHMENT 1

DESCRIPTION AND ASSESSMENT

1.0 Description

The proposed amendment would modify the Catawba Nuclear Station (Catawba) Technical Specifications (TS) by relocating specific surveillance frequencies to a licensee controlled program with the adoption of Technical Specification Task Force (TSTF)-425, Revision 3, "Relocate Surveillance Frequencies to Licensee Control – Risk Informed Technical Specification Task Force (RITSTF) Initiative 5b." Additionally, the change would add a new program, the Surveillance Frequency Control Program, to TS Section 5.0, "Administrative Controls."

The changes are consistent with the NRC approved Industry/TSTF Standard Technical Specification (STS) change TSTF-425, Revision 3 (ADAMS Accession No. ML080280275). Availability of this TSTF was published in the Federal Register notice on July 6, 2009.

2.0 Assessment

2.1 Applicability of Published Safety Evaluation

Duke has reviewed the Safety Evaluation Report (SER) dated July 6, 2009. This included a review of the NRC staff's evaluation of TSTF-425, Revision 3, and the requirements specified in NEI 04-10, Rev. 1, (ADAMS Accession No. ML071360456).

Attachment 2 includes Duke Energy's documentation with regards to the PRA technical adequacy, consistent with the requirements of Regulatory Guide 1.200, Rev. 1 (ADAMS Accession No. ML070240001), Section 4.2, and describes any PRA models without NRC endorsed standards, including documentation of the quality characteristics of the models.

Duke Energy has concluded that the justifications presented in the TSTF-425 proposal and the safety evaluation prepared by the NRC staff is applicable to Catawba Units 1 and 2, and justify this amendment to incorporate the changes to the Catawba TS.

2.2 Optional Changes and Variations

The proposed amendment is consistent with the STS changes described in TSTF-425, Revision 3, however, Duke Energy proposes variations or deviations from TSTF-425, as identified below.

1. The revised (re-typed) TS pages are not included in this proposed amendment due to the number of TS pages affected, the nature of the proposed changes, and the outstanding amendment requests that Catawba currently has under NRC review. Providing only the mark-ups of the proposed TS changes satisfies the requirements of 10 CFR 50.90. This is an administrative deviation from TSTF-425 with no exceptions to the NRC staff's model safety evaluation dated July 6, 2009. This administrative deviation is consistent with Exelon's Peach Bottom Atomic Power Station License Amendment application dated August 31, 2009 (NRC Accession No. ML092470153).

2. A note in Technical Specification 3.3.1 concerning the one-time extension for SR 3.3.1.5 will be deleted since it has expired and the same page is being revised for this amendment.

3. Attachment 5 provides a cross reference table between the NUREG-1431 surveillances included in TSTF-425 versus the Catawba surveillances included in this amendment request. This cross reference table highlights the following:

    a. TSTF-425 surveillances with identical corresponding Catawba Surveillance numbers,
    b. TSTF-425 surveillances and corresponding Catawba Surveillances but with differing Surveillance numbers,
    c. TSTF-425 surveillances that are not contained in the Catawba TS and therefore not applicable, and
    d. Catawba plant specific surveillances that are not contained in TSTF-425 surveillance mark-ups, but are applicable to these amendment requests.

    Concerning the above, Catawba surveillances with identical corresponding TSTF-425 surveillance numbers (item "a" above) are not deviations from TSTF-425.

    Catawba surveillance numbers that differ from the corresponding TSTF-425 surveillance numbers (item "b" above) are administrative deviations only from TSTF-425 with no impact on the NRC Staff's model SER.

    TSTF-425 surveillances that are not contained in the Catawba TS (item "c" above) are not applicable to these amendment requests. This also includes Catawba's corresponding surveillances that are event driven or performed in accordance with an existing program (safety evaluation scope exceptions). Not including these TSTF-425 surveillances is an administrative deviation from TSTF-425 with no impact on the NRC Staff's model SER.

    For Catawba plant specific surveillances that are not contained in TSTF-425 Surveillance mark-ups, but are applicable to this amendment request (item "d" above), Duke Energy has determined that the relocation of these surveillance frequencies is consistent with TSTF-425, Revision 3, and the NRC Staff's model SER. This includes the scope exceptions documented in Section 1.0, "Introduction," of the model SER since the Catawba plant specific surveillances involve fixed periodic frequencies and therefore do not meet any of the four exceptions.

    A similar cross reference table comparing the TSTF and plant specific surveillances was also provided by Exelon's Peach Bottom Atomic Power Station. The License Amendment applications to relocate specific surveillances in accordance with TSTF-425 are dated August 31, 2009 (NRC Accession No. ML092470153) and October 30, 2009 (NRC Accession No. ML093060126).

Duke Energy currently has seven license amendment requests that are pending approval from the NRC that affect surveillances modified in this amendment request. A listing of those amendment letters is provided in the table below, along with the surveillances (SRs) that are affected. Since the approval process of these amendments is in progress, Duke Energy will not know the final disposition of each request until later in 2010. Duke Energy will provide updated pages and mark-ups for affected SRs before final approval of this amendment.

Date of Amendment Letter	Affected SRs and SR Bases
09/02/08	This LAR modifies the SR 3.6.6.4 to be "not applicable". The SR Description for SR 3.5.4.2 is modified. The Bases only for SRs 3.3.2.7, 3.3.2.9 and 3.6.6.3 are revised, also.
10/02/08	This LAR modifies the SR Description for SRs 3.6.13.1, 3.6.13.4, and 3.6.13.5 and SR 3.6.13.6.
05/28/09	This LAR modifies the SR Description of SRs 3.8.1.2, 3.8.1.7, 3.8.1.9, 3.8.1.11, 3.8.1.12, 3.8.1.15, 3.8.1.19, and 3.8.1.20.
07/01/09	The SR Bases for SRs 3.3.1.7 and 3.3.1.8 are modified. Also, the SR Description for SR 3.3.1.11 is modified.
09/30/09	This LAR modifies the Frequency of SR 3.6.6.7 to be event-driven.
12/14/09	SRs 3.8.4.3 and 3.8.4.6 are modified to add a reference to a new table.
12/15/09	SR 3.4.16.1 description is modified by this LAR. Also, SR 3.4.16.3 is deleted.

3.0 Regulatory Analysis

3.1 No Significant Hazards Consideration

Duke Energy has reviewed the proposed no significant hazards consideration determination (NSHC) published in the Federal Register on July 6, 2009, 74 FR 31996-32006. Duke Energy has concluded that the proposed NSHC presented in the Federal Register notice is applicable to Catawba Units 1 and 2, and is provided in Attachment 6 to this amendment request. This satisfies the requirements of 10 CFR 50.91(a).

Attachment 2

Documentation of PRA Technical Adequacy

Catawba Nuclear Station

Application for Technical Specification Change Regarding Risk-Informed Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program (Adoption of TSTF-425, Revision 3)

Documentation of PRA Technical Adequacy

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2.1 Overview

The technical adequacy of the probabilistic risk assessment (PRA) must be compatible with the safety implications of the proposed Technical Specification (TS) changes and the role the PRA plays in justifying the changes. The Nuclear Regulatory Commission (NRC) has developed Regulatory Guide (RG) 1.200 (Ref. 1), which references the American Society of Mechanical Engineers (ASME) PRA standard RA-Sb-2005, Addenda to ASME RA-S-2002 (Ref. 2) for internal events at power. External events and shutdown risk impacts may be considered quantitatively or qualitatively. RG 1.200 also references the NEI peer review process NEI 00-02 (Ref. 3).

The industry guidance document for the implementation of Initiative 5b is NEI 04-10, "Risk-Informed Method for Control of Surveillance Frequencies". The NRC issued a Final Safety Evaluation for NEI 04-10 Revision 0, on September 28, 2006 (Ref. 4). The Staff found that NEI 04-10, Revision 0, was acceptable for referencing by licensees proposing to amend their TSs to establish a Surveillance Frequency Control Program (SFCP), provided that the following conditions are satisfied:

1. The licensee submits documentation with regard to PRA technical adequacy consistent with the requirements of RG 1.200, Section 4.2.

2. When a licensee proposes to use PRA models for which NRC-endorsed standards do not exist, the licensee submits documentation, which identifies the quality characteristics of those models, consistent with RG 1.200, Sections 1.2 and 1.3. Otherwise, the licensee identifies and justifies the methods to be applied for assessing the risk contribution for sources of risk not addressed by PRA models.

Subsequently NEI 04-10 Revision 1 was approved (Ref. 5) and is the current document of record.

The implementation of the SFCP at the Catawba Nuclear Station will follow the guidance provided in NEI 04-10, Revision 1 in evaluating proposed surveillance frequency changes.

The Catawba PRA is a full scope PRA including both internal and external events (i.e., flood, seismic, fire, high winds (tornado)). Having previously completed a self-assessment against the supporting requirements of ASME PRA Standard through addenda RA-Sc-2007 (Ref. 6), Duke Energy is planning to perform a self-assessment against the supporting requirements of ASME/ANS PRA standard RA-Sa-2009, Addendum A to RA-S-2008 (Ref. 7) for the current Catawba PRA model of record (including fire, flood, seismic, and high winds (tornado) models) in 2010. Also there is currently significant work being performed at Duke Energy in the area of fire PRAs. This will be discussed further in the Fire PRA Model section.

The following information is submitted by Duke Energy to address the conditions of the NRC SER for Initiative 5b.

2.2 Historical Summary

The original Catawba PRA was initiated in July 1984 by Duke Power Company (Duke Power) staff, Delian Corporation, and Safety and Reliability Optimization Services (SAROS), Incorporated. Law Engineering Testing Company and Structural Mechanics Associates provided specific input to the seismic analysis. Science Applications International Corporation (SAIC) provided support for the human reliability analysis. External reviews were conducted by personnel from SAIC, Delian Corporation, and SAROS Incorporated. It was a full scope Level 3 PRA with internal and external events (i.e., seismic, flood, high winds (tornado), fire). A peer review of the draft PRA was conducted by Electric Power Research Institute (EPRI) (Ref. 8). The final study, which incorporated the comments of all the reviews, was completed in August 1987 and resulted in an internal Duke Power report (Ref. 9) as Revision 0 to the PRA.

On November 23, 1988, the NRC issued Generic Letter (GL) 88-20 (Ref. 10), which requested that licensees conduct an Individual Plant Examination (IPE) in order to identify potential severe accident vulnerabilities at their plants. In response, Duke Power initiated a review and update of the original Catawba study in April 1991. The Catawba response to GL 88-20 was provided by letter dated September 10, 1992 (Ref. 11). In this letter Duke Power noted that the enclosed Revision 1 of the PRA consisted of a complete Level 3 PRA with a detailed analysis of both internal and external events. By letter dated June 7, 1994 (Ref. 12), the NRC provided a SER of the internal events portion of the above Catawba IPE submittal.

In response to Generic Letter 88-20, Supplement 4 (Ref. 13), Duke Power completed an Individual Plant Examination of External Events (IPEEE) for severe accidents. This IPEEE was submitted to the NRC by letter dated June 21, 1994 (Ref. 14). The IPEEE report contained a detailed write-up of the Catawba seismic and fire PRA analysis methods, results and conclusions. It also addressed other events such as high winds, floods, and transportation accidents. The IPEEE study did not identify any plant changes that would significantly reduce the risk from external events.

Duke Power subsequently responded to an NRC Request for Additional Information (RAI) on the IPEEE submittal November 17, 1995 (Ref. 15). Duke Power also submitted a Supplemental IPEEE Fire Analysis Report to the NRC July 30, 1996 (Ref. 16) in response to a request for supplemental fire investigations as described below.

1. Develop fire accident sequences for those areas that were previously screened from further review because they were considered to be subsumed by other initiators.

2. Perform sensitivity studies on fire detection and suppression parameters.

3. Re-review cable routing to confirm that potential plant trip initiators have been considered in all areas.

The supplemental fire investigations in the report produced a more complete quantification of the fire induced core damage sequences but the conclusions and results were not significantly different from those reported in the original IPEEE report.

By letter dated April 12, 1999 (Ref. 17), the NRC provided an evaluation of the IPEEE submittal. The conclusion of the NRC letter [page 6] states:

> *"The staff finds the licensee's IPEEE submittal is complete with regard to the information requested by Supplement 4 to GL 88-20 (and associated guidance in NUREG-1407), and the IPEEE results are reasonable given the Catawba design, operation, and history. Therefore, the staff concludes that the licensee's IPEEE process is capable of identifying the most likely severe accidents and severe accident vulnerabilities, and therefore, that the Catawba IPEEE has met the intent of Supplement 4 to GL 88-20."*

While the IPEEE Program was a one-time review of external hazard risk and was limited in its purpose to the identification of potential plant vulnerabilities and the understanding of associated severe accident risks, there have not been significant numbers of plant changes made since the initial NRC review that would invalidate the methodologies used in the existing external events models of record.

In 1996, Catawba initiated Revision 2 of the 1992 IPE and provided the results to the NRC in 1998 (Ref. 18). Revision 3 of the Catawba PRA was completed in December 2004 and Revision 3a was completed in August 2006. Revision 3 was a major comprehensive revision to the PRA models and associated documentation. Revision 3a was a change to implement the turbine building 4160 VAC transformer flood wall modifications and other various model enhancements. Revision 3a is the current model of record. Work is currently underway on Revision 4 of the Catawba PRA, which is a major revision to the PRA, and includes a planned revision to the fire PRA model (discussed in Section 2.4.3.1).

2.3 PRA Technical Adequacy Consistent With RG 1.200, Section 4.2

This section addresses Condition 1 of the NRC SER for Initiative 5b.

2.3.1 PRA Model Adequately Represents the As-Built, As-Operated Plant

The basis to conclude that the PRA model to be used adequately represents the as-built, as-operated plant is as follows.

The existing PRA Configuration Control Program at Catawba was assessed against Section 5 of the ASME PRA Standard to meet the requirements necessary to support risk-informed decisions. The results of the self-assessment concluded that the PRA fully meets the requirements for configuration control of a PRA to be used with the ASME PRA Standard to support risk-informed decisions for nuclear power plants. A summary of the program and the basis to conclude that the PRA model adequately represents the as-build, as-operated plant is provided below.

The PRA Configuration Control Program at Catawba is governed by the following workplace procedures.

- XSAA-101, Risk-Impact Review of Nuclear Plant Changes Including Nuclear Station Modification, and Emergency or Abnormal Procedure Changes
- XSAA-106 , PRA Maintenance, Update and Application

XSAA-101 addresses the process for review of plant design changes, plant emergency and abnormal procedure changes, and Technical Specification (TS) changes that have been made for PRA impacts. It also describes in detail the process used to review the impact of potentially significant changes that could impact the PRA before the changes have been made.

XSAA-106 addresses the conditions when a PRA update may be required (e.g., cumulative risk impact of unincorporated PRA changes exceeds a threshold such that the as-built as-operated plant is not adequately represented by the PRA). It addresses a process to assess the risk of a change to the plant and a method to prioritize the implementation of a plant change based on the risk impact to the PRA. It describes a process to ensure that an annual assessment is made of the cumulative impact of PRA changes that have not yet been incorporated into the PRA and provides guidance as to when a PRA update is needed based on the results of the annual assessment. Finally, it describes the electronic tracking tool that is used to track changes that impact the PRA till they are incorporated into the PRA.

The process requires notification of any completed (and planned changes that could significantly impact the PRA model) plant modifications, TS changes, or Emergency Procedure changes are sent to the PRA Section for a review of any PRA impacts. This review is documented. If a plant change is determined to impact the PRA then is it entered into the electronic tracking tool where a risk assessment is performed on the change. The outcome of the risk assessment will "bucket" the plant change into a Low, Medium, or High risk change category based on the estimated delta Core Damage Frequency (CDF) or delta Large Early Release Frequency (LERF) results. Plant changes that are determined to be of a Low risk impact are tracked to completion in the electronic tracking tool and are annually assessed for their cumulative impact on the PRA model. Plant changes that are determined to be of Medium or High risk impact are entered into the site corrective action program for further analysis as to their impact on current applications. They also are tracked to completion in the electronic tracking tool and are annually assessed for their cumulative impact on the baseline PRA model.

For any application that requires a PRA analysis (e.g., License Amendment Request (LAR) or Notice of Enforcement Discretion (NOED)) workplace procedures require that all of the outstanding PRA model changes listed in the electronic tracking tool are individually reviewed for their impact on the application. A justification is made as to why each item does not impact the PRA results used to support the application. This review is documented. If it is determined that an unincorporated change might impact an application then steps are taken to either perform sensitivity studies to demonstrate that the contributors significant to the application were not impacted or the PRA model is revised to address the impact of the change on the application. This analysis will also be performed and documented for every application of Initiative 5b.

The outstanding items in the electronic tracking tool are ultimately incorporated into a major PRA revision which is performed periodically to ensure that the overall number of items being tracked remains manageable. This robust process, governed by written procedures, is sufficient to ensure the PRA model represents the as-built, as-operated plant.

2.3.2 Unincorporated Changes to the Plant

The justification of how unincorporated changes to the plant will be addressed is provided in the response in Section 2.3.1.

2.3.3 Departures from ASME Requirements

The justification for departures from the ASME Standard Capability Category II requirements, including any unresolved findings/observations is as follows.

In March 2002, the PRA at Duke Energy's Catawba Nuclear Station received a peer review by an industry team of knowledgeable PRA practitioners (Ref. 19). Since the performance of this peer review, the industry has utilized the American Society of Mechanical Engineers (ASME) process to develop a standard identifying the requirements associated with PRA. RG 1.200 endorses the ASME PRA Standard as an acceptable method for demonstrating the technical adequacy of a PRA – provided various clarifications are made as identified in the regulatory guide.

Subsequently in 2008, as noted earlier, Duke Energy conducted a self-assessment of the Catawba PRA (Ref. 6) against the ASME PRA Standard through addenda RA-Sc-2007.

The Catawba PRA self-assessment included the Risk Assessment Technical Requirements listed in Section 4 of the ASME PRA Standard. This self-assessment evaluated the PRA with respect to Capability Category II. For the purposes of Initiative 5b, deviations from the Capability Category II supporting requirements were identified and dispositioned to ensure that these issues do not negatively impact Initiative 5b. For those requirements of the standard that have not been met, a justification of why it is acceptable that the requirement has not been met has been provided. A summary of these items is shown in Table 2-1 for Catawba (Ref. 6). Of the 29 items, 26 are either documentation or have no expected impact on Initiative 5b applications. The remaining three could have an impact based on the specific Initiative 5b application.

Because of the broad scope of potential Initiative 5b applications, and the fact that the impact of assumptions may differ for each surveillance requirement being evaluated, Duke Energy will address each of the deviations from Capability Category II listed in Table 2-1 for the Catawba PRA respectively for each application of Initiative 5b on an application specific basis. Again, if a requirement is not met a justification of why it is acceptable that the requirement has not been met will be provided. These results will be with the documentation package for the specific Initiative 5b application.

2.3.4 Methodology to be Used for Initiative 5b

NEI 04-10 Revision 1 provides the detailed process requirements for controlling surveillance frequencies (SF) of the TS Surveillance Requirements (SRs) that have been relocated from the TSs to the SFCP. The methodology described in NEI 04-10 Revision 1 provides a risk-informed process to support a plant expert panel (called an Integrated Decisionmaking Panel or IDP) assessment of proposed changes to SF, assuring appropriate consideration of risk insights and other deterministic factors, which may impact SF, along with appropriate performance monitoring of changes and documentation requirements.

The Duke Energy SFCP, including the methodology of assessing SF changes utilized at Catawba, is consistent with NEI 04-10, Revision 1 and the supporting background document TSTF-425-A, Rev. 3 (Ref. 20).

2.3.5 Identification of Key Assumptions

Identification of Key Assumptions related to SF (if any) and how they will be addressed is given below.

The overall Initiative 5b process is a risk-informed process with the PRA model results providing one of the inputs to the IDP to determine if a SF change is warranted. The methodology recognizes that a key area of uncertainty for this application is the standby failure rate utilized in the determination of the SF change impact. Therefore, the methodology requires the performance of selected sensitivity studies on the standby failure rate of the component(s) of interest for the SF change assessment.

Because of the broad scope of potential Initiative 5b applications, any key assumptions and approximations relevant to the results obtained for an application of Initiative 5b will be addressed and documented on an application specific basis. This includes not only the results of the standby failure rate sensitivity study, but the results of any additional sensitivity studies identified during the performance of the reviews as outlined in Sections 2.3.1, 2.3.2, and 2.3.3.

2.3.6 Resolution of Relevant Peer Review/Self-Assessment Findings and Observations

Section 2.3.3 discusses departures from the ASME PRA Standard Capability Category II requirements and summarizes them on Table 2-1 for Catawba. However as previously noted, because of the broad scope of potential Initiative 5b applications, and the fact that the impact of assumptions may differ for each surveillance requirement being evaluated, Duke Energy will address each of the deviations from Capability Category II listed in Table 2-1 for Catawba for each application of Initiative 5b on an application specific basis. If a requirement is not met a justification of why it is acceptable that the requirement has not been met will be provided. If the PRA model is changed for a specific application of Initiative 5b to address self-assessment findings or if a sensitivity study is performed to demonstrate contributors significant to the application were not impacted by a self-assessment finding, a discussion of the results and conclusions for resolution will be included in the documentation package. Duke Energy will maintain a current listing of deviations from ASME PRA Standard Capability Category II requirements for Catawba for review and resolution against each application of Initiative 5b.

2.3.7 Applicable Capability Category for Initiative 5b

In accordance with NEI 04-10 Revision 1, the PRA must meet Capability Category II to be used for Initiative 5b applications. Duke Energy will ensure the Catawba PRA used for Initiative 5b applications either fully meets Capability Category II or departures from Capability Category II are justified to show insignificant impact on the results of the analysis. This will be done by performing a review of all outstanding departures from Capability Category II against the specific Initiative 5b application being addressed. The results of this review will be in the documentation package for the specific Initiative 5b application.

2.4 External Events Considerations

This section addresses Condition 2 of the NRC Safety Evaluation for Initiative 5b.

Specifically it identifies quality characteristics for PRA models for which NRC-endorsed Standards do not exist, consistent with RG 1.200, Sections 1.2 and 1.3, and justifies the methods to be applied for assessing the risk contribution for those sources of risk not addressed by PRA models.

NRC endorsed standards currently exist for external hazards including seismic and fire PRAs. Revision 2 of Regulatory Guide (RG) 1.200 (Ref. 21), references the ASME/ANS PRA standard RA-Sa-2009, Addendum A to RA-S-2008 (Ref. 7) for internal and external hazards. An NRC endorsed standard does not currently exist for shutdown PRAs. NEI 04-10 Revision 1 references RG 1.200 Revision 1 and ASME PRA Standard RA-Sb-2005b as the governing documents for Initiative 5b.

The NEI 04-10 Revision 1 methodology allows for SF change evaluations to be performed in the absence of quantifiable PRA models for all external hazards. For those cases where the SF cannot be modeled in the plant PRA (or where a particular PRA model does not exist for a given hazard group), a qualitative or bounding analysis is performed to provide justification for the acceptability of the proposed test interval change. In general, it is not expected that seismic, fire, or other external hazards will play a significant role in the impact of a given surveillance frequency change.

This section discusses the Catawba overall external hazards analysis methodology, the Catawba specific seismic and fire PRAs, and describes the methodology to be used to address shutdown risk impacts for Initiative 5b consistent with the requirements of the NEI 04-10 Revision 1 methodology.

2.4.1 Overall External Hazards Analysis Methodology

The general approach used to develop the external event PRA at Catawba is as follows:

1) Identify all natural and man-made credible external events that may affect the site using many reference sources.

2) A screening analysis was conducted using defined bounding criteria in order to select those events that may require further review.

3) A scoping analysis was performed on the remaining non-screened events to determine those that warranted a detailed site and plant-specific analysis.

This approach is consistent with that previously submitted to the NRC in Section 2.3 of Reference 14 and Volume 1, Section 3.0 of Reference 11. These references provide a greater level of detail of the approach if needed.

2.4.2 Catawba Seismic PRA Model

As noted in the IPEEE submittal (Ref. 14), Catawba Unit 2 was selected for a trial assessment of the EPRI developed Seismic Margin Methodology, the methodology for assessing the ability of nuclear plants to withstand earthquakes beyond design basis. The assessment established that Catawba would survive earthquake loads up to approximately twice its design basis. This work is documented in EPRI NP-6359 (Ref. 22).

The current Catawba seismic PRA model of record was last updated as part of Revision 3 of the PRA model (Ref. 23). However, the current methodology used is the same as that described in detail in the IPE submittal (Ref. 11) and Section 3 of the IPEEE submittal (Ref. 14) both of which have already been reviewed by the NRC. The reader is referred to those references for additional details of the seismic analysis.

The plant-specific seismic PRA analysis consists of four steps each of which are described below:

1) The Catawba site was evaluated to obtain the seismic hazard in terms of the frequency of occurrence of ground motions of various magnitudes. The site-specific hazard analysis (Ref. 24) was performed using the Seismicity Owners Group (SOG) methodology developed by EPRI for seismic hazard analysis of nuclear power plant sites in the Central and Eastern United States (CEUS). Uncertainties were addressed in the hazard analysis.

2) From the site-specific seismic hazard curve, the capacities of important plant structures and equipment to withstand seismic events were evaluated to determine conditional probabilities of failure as a function of ground acceleration for significant contributors (i.e., SSCs). These are commonly referred to as 'fragilities' or the site-specific fragility curves. Plant walkdowns were conducted, the most recent ones consistent with the guidelines of EPRI NP-6041 (Ref. 25).

3) An event tree was developed along with supporting top logic and system fault trees to reflect plant response to seismic events. These modified logic models were then solved to obtain Boolean expressions for the seismic event sequences of interest.

4) The Boolean expressions were quantified by convolving the probabilistic site

seismicity and the fragilities for the plant structures and equipment obtained in steps 1 and 2. The resulting sequence frequencies are then integrated into the overall Catawba PRA risk results, resulting in final quantitative results.

The major changes to the current seismic analysis that have been made since the IPEEE submittal are as follows:

1. Comprehensive review and revision of the seismic analysis documentation write-up.
2. Added component/structure fragility information to support values used in analysis.
3. Updated model with new Human Reliability Analysis (HRA) data.
4. Updated model with new common cause data.
5. Changes made to the fault tree are listed below.
   - Made a new top gate for the model to address containment safeguards responses. Added all supporting logic for new containment safeguards responses gate. This change was made to aid in accident "binning" in the seismic analysis.
   - After review reinstated components and structures back into the model that had previously been screened out. These include:
     - ➢ DC Charger
     - ➢ Refueling Water Storage Tank
     - ➢ 125V dc Battery Rack
     - ➢ Emergency Diesel Generator (EDG) Control Panel
     - ➢ Residual Heat Removal (RHR) Heat Exchanger
     - ➢ EDG Engine Control Panel
     - ➢ Neutral Ground Resistor
     - ➢ EDG Load Sequencer
     - ➢ Solid State Protection System (SSPS) Cabinets
     - ➢ 4160V ac Switchgear
     - ➢ 125V dc Panelboard
     - ➢ 120V ac Panelboard
     - ➢ Inverter
     - ➢ Auctioneering Diode Assembly
     - ➢ EDG building
     - ➢ Control Rod Drive Mechanism Seismic Supports
     - ➢ Auxiliary Building Shear Wall
     - ➢ 600V ac Motor Control Center
   - Added logic to include EDG seismic failures, loss of 120V ac, and EDG load sequencer failures.
   - Added Loss of Reactor Coolant (NC) Pump Seal Support logic to reflect the addition of a redesigned seal package. New seals are qualified for higher temperatures that limit the amount of leakage should failure occur.
   - Miscellaneous logic additions to several systems to account for failures of component cooling as a support system.
6. Updated the seismic event tree.
7. Updated the seismic analysis quantitative results table.

As noted previously, Duke Energy is planning to perform a self-assessment against the supporting requirements for seismic events of ASME/ANS PRA standard RA-Sa-2009, Addendum A to RA-S-2008 for the Catawba seismic PRA in 2010. The method as described in Section 2.3.3 of this attachment will be used to justify any departures from the ASME Standard Capability Category II requirements for each application of Initiative 5b. However, in accordance with the discussion in this section above, Duke considers the current seismic model of record as meeting the required quality characteristics of RG 1.200 Sections 1.2 and 1.3 and is therefore sufficient for use as is in the application of Initiative 5b SF changes.

2.4.3 Catawba Fire PRA Model

The current Catawba fire PRA model analysis and methodology (Ref. 26) used in the model of record is the same analysis and methodology as described in the IPE submittal (Ref. 11); Section 4 and Appendix B of the IPEEE submittal (Ref. 14); and as discussed in the Supplemental IPEEE Fire Analysis Report (Ref. 16), all of which have already been reviewed by the NRC. The reader is referred to those references for additional details of the fire analysis.

The plant-specific fire PRA analysis consists of four steps each of which are described below:

a. The Catawba site and plant areas were analyzed to determine critical fire areas and possible scenarios for the possibility of a fire causing one or more of a predetermined set of initiating events. Screening criteria were defined for those fire areas excluded from the fire analysis.

b. If there was a potential for an initiating event to be caused by a fire in an area, then the area was analyzed for the possibility of a fire causing other events which would impact the ability to shutdown the plant. These were identified by reviewing the impact on the internal event analysis models.

c. Each area was examined with an event tree fire model to quantify fire damage probabilities. The event tree related fire initiation, detection suppression, and propagation probabilities to equipment damage states.

d. Fire sequences were derived and quantified based on the fire damage probabilities and the additional failures necessary for a sequence to lead to a core melt. The additional failures were quantified by the models used in the internal events analysis.

The major changes to the current fire analysis that have been made since the IPEEE submittal deal with implementation of changes from the Supplemental IPEEE Fire Analysis Report (Ref. 16) and revised base case fire initiating event frequencies.

Since the Catawba fire PRA model is integrated into the overall PRA model, quantitative fire risk insights will be obtained each time when the PRA model is exercised. When the integrated PRA model is not utilized for a quantitative assessment and modeling of the affected equipment is not feasible, the fire risk insights will be assessed qualitatively. This approach is consistent with the accepted NEI 04-10 Revision 1 methodology.

Duke Energy is planning to perform a self-assessment against the supporting requirements for fire events of ASME/ANS PRA standard RA-Sa-2009, Addendum A to RA-S-2008 for the Catawba fire PRA in 2010. The method as described in Section 2.3.3 of this attachment will be used to justify any departures from the ASME Standard Capability Category II requirements for each application of Initiative 5b. However, in accordance with the discussion in this section above, Duke Energy considers the current fire model of record as meeting the required quality characteristics of RG 1.200 Sections 1.2 and 1.3 and is therefore sufficient for use as is in the application of Initiative 5b SF changes.

2.4.3.1 Catawba Future State Fire PRA Model Initiative

In February 2005, Duke Energy notified the NRC (Ref. 27) of its intent to adopt National Fire Protection Association (NFPA) Standard 805, "NFPA 805, Performance-Based Standard for Fire Protection for Light-Water Reactor Electric Generation Plants," 2001 edition, pursuant to Section 50.48(c) of Part 50 of Title 10 of the *Code of Federal Regulations* (10 CFR 50.48(c)), at all of its nuclear stations.

In a letter dated June 8, 2005, the NRC accepted Duke Energy's intent to adopt 10 CFR 50.48(c) (NFPA 805 Rule) for all three sites with Oconee Nuclear Station beginning the transition as a pilot plant on June 1, 2005 (Ref. 28). Duke Energy was requested to inform the NRC when the transition would begin at Catawba.

Subsequently, Duke Energy informed the NRC in 2007 (Ref. 29) that the transition to NFPA 805 at Catawba Nuclear Station had begun. The NRC response on January 4, 2008 (Ref. 30) acknowledged the transition to the performance-based standard for fire protection had begun at Catawba Units 1 and 2.

The Catawba Fire PRA model being developed uses guidance contained in NUREG/CR-6850/EPRI TR-1011989, EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities (Ref. 31). This is the same methodology and approach as that being used for the Oconee pilot. The Catawba Fire PRA model is to receive an industry peer review against the requirements of Part 4 of ASME/ANS RA-Sa-2009, Addendum to RA-S-2008 (Ref. 7) in April 2010. When the peer review report is received the departures from Capability Category II requirements and other findings will be addressed. In September 2010, Duke Energy is planning to submit a License Amendment Request (LAR) to the NRC to adopt the new fire protection licensing basis which complies with the requirements in 10 CFR 50.48(a), 10 CFR 50.48(c), and the guidance in Regulatory Guide (RG) 1.205. A discussion of the peer review open items and their disposition is expected to be part of that submittal.

2.4.4 Catawba Shutdown Risk Impact Analysis

Since no approved quantitative shutdown risk PRA model for shutdown events currently exists at Duke Energy, Catawba will either 1) utilize the plant shutdown safety assessment tool developed to support implementation of NUMARC 91-06 (Ref. 32) as described in Duke Energy Nuclear Station Directive (NSD) 403 (Ref. 33) or 2) perform an alternate qualitative risk evaluation process to assess the proposed surveillance frequency change that utilizes Initiative 5b. These are acceptable options to not having a quantitative shutdown PRA model in accordance with Section 4 Step 10 (and other places) of NEI 04-10 Revision 1. In either case, the guidance of NEI 04-10 Revision 1 will be followed.

2.5 Summary

In Section 2.3 of this document the Catawba PRA technical adequacy was evaluated in accordance with the requirements of RG 1.200, Section 4.2. Section 2.4 of this document submitted quality characteristics of the seismic and fire PRA models in accordance with the requirements of RG 1.200, Sections 1.2 and 1.3. A discussion of the qualitative method to address shutdown risk was also discussed in Section 2.4.

Because of the broad scope of potential Initiative 5b applications and the fact that the risk assessment details will differ from application to application, for each individual SF interval request, a review of the unincorporated changes to the plant and remaining gaps to specific requirements in the PRA standard will be made to determine which, if any, would merit additional application-specific sensitivity studies in the final analysis.

The results of the discussions above provide a basis for concluding that the current Catawba Units 1 and 2 PRA model is sufficiently robust and suitable for use in risk-informed processes such as that proposed for the implementation of a Surveillance Frequency Control Program.

2.6 References

1. Regulatory Guide 1.200, "*An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities*", Revision 1, US Nuclear Regulatory Commission, January 2007.
2. ASME RA-S-2002, "*Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications*", with Addenda ASME RA-Sa-2003 and ASME RA-Sb-2005, December 2005.
3. NEI 00-02, "*Probabilistic Risk Assessment Peer Review Process Guidance,*" Revision A3, Nuclear Energy Institute, March 20, 2000.
4. Letter, USNRC to Nuclear Energy Institute, "Final Safety Evaluation for Nuclear Energy Institute (NEI) Industry Guidance Document NEI 04-10, Revision 0, "Risk-Informed Technical Specifications Initiative 5B, Risk-Informed Method for Control of Surveillance Frequencies"", September 28, 2006.
5. NEI 04-10, Revision 1, "*Risk-Informed Technical Specifications Initiative 5b, Risk-Informed Method for Control of Surveillance Frequencies,*" April 2007.
6. DPC-1535.00-00-0013 (Cross references: CNC-1535.00-00-0094, MCC-1535.00-00-0089, OSC-9380), "*PRA Quality Self-Assessment, Catawba Units 1 & 2, McGuire Units 1 & 2, Oconee Units 1, 2 & 3*", Revision 2, November 2009.
7. ASME/ANS RA-Sa-2009, "*Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications,*" Addendum A to RA-S-2008, ASME, New York, NY, American Nuclear Society, La Grange Park, Illinois, February 2009.

8. *"Catawba Nuclear Station, Unit 1 Probabilistic Risk Assessment,"* Volume 1, Preface, Duke Power Company, August 1987.
9. "Catawba *Nuclear Station Unit 1 Probabilistic Risk Assessment*," Volumes 1-3, Duke Power Company, August 18, 1987.
10. NRC Generic Letter 88-20, "*Individual Plant Examination for Severe Accident Vulnerabilities*", US Nuclear Regulatory Commission, November 23, 1988.
11. Letter Duke Power Company to Document Control Desk (USNRC), Catawba Units 1 and 2, "*Individual Plant Examination (IPE) Submittal in Response to Generic Letter 88-20,*" September 10, 1992.
12. Letter USNRC to Duke Power Company, "*Safety Evaluation of Catawba Nuclear Station, Units 1 and 2 Individual Plant Examination (IPE) Submittal*," June 7, 1994.
13. NRC Generic Letter 88-20, *"Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities 10 CFR 50.54(f), Supplement 4,"* June 28, 1991.
14. Letter Duke Power Company to Document Control Desk (USNRC), Catawba Units 1 and 2, "*Individual Plant Examination of External Events (IPEEE) Submittal*," June 21, 1994.
15. Letter Duke Power Company to Document Control Desk (USNRC), Catawba Nuclear Station, *"Request for Additional Information- Individual Plant Examinations for External Events; Response,"* November 17, 1995.
16. Letter Duke Power Company to Document Control Desk (USNRC), *"Supplemental IPEEE Report,"* Duke Power Company, McGuire Nuclear Station, Catawba Nuclear Station, July 30, 1996.
17. Letter USNRC to Duke Power Company, "*Catawba Nuclear Station - Review of Individual Plant Examination of External Events (IPEEE)*," April 12, 1999.
18. Letter Duke Energy Corporation to Document Control Desk (USNRC), Catawba Units 1 and 2, "*Probabilistic Risk Assessment (PRA), Revision 2 Summary Report, January 1998.*
19. *"Catawba Nuclear Station Probabilistic Safety Assessment Peer Review Report"*, Westinghouse Electric Co. for the Westinghouse Owners Group, December 2002.
20. Technical Specification Task Force Traveler number TSTF-425, Revision 3, "*Relocate Surveillance Frequencies to Licensee Control – RITSTF Initiative 5*," July 2009.
21. Regulatory Guide 1.200, "*An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities*", Revision 2, US Nuclear Regulatory Commission, March 2009.
22. EPRI NP-6359, "*Seismic Margin Assessment of the Catawba Nuclear Station*," April 1989.
23. CNC-1535.00-00-0059, Catawba Nuclear Station, External Events -Seismic Analysis, December 2003.
24. EPRI NP-4726-A, *"Seismic Hazard Methodology for the Central and Eastern United States,"* July 1986.
25. EPRI NP-6041, Revision 1, *"A Methodology of Assessment of Nuclear Power Plant Seismic Margin,"* August 1991.
26. CNC-1535.00-00.0057, Catawba Nuclear Station, Fire Analysis Notebook, September 1997.
27. Letter Duke Energy Corporation to Document Control Desk (USNRC), *"Letter of Intent to Adopt NFPA 805 Performance-Based Standard for Fire Protection for*

Light Water Reactor Generating Plants, 2001 Edition," February 28, 2005 (Adams Accession Number ML050670305).

28. Letter USNRC to Duke Energy Corporation, "NRC Response to Duke's Letter of Intent to Adopt 10 CFR 50.48(c) (NFPA 805 Rule)," June 8, 2005 (Adams Accession Number ML051080005).
29. Letter Duke Energy Corporation to Document Control Desk (USNRC), *"Letter of Intent to Start the Transition to NFPA 805 Performance-Based Standard for Fire Protection for Light Water Reactor Generating Plants, 2001 Edition," June 7, 2007* (Adams Accession Number ML072260422).
30. Letter USNRC to Duke Energy Corporation, *"Response to Letter of Intent to Adopt National Fire Protection Association Standard 805 for Duke Power Company's Catawba Nuclear Station, Units 1 and 2,"* January 4, 2008 (Adams Accession Number ML072780045).
31. EPRI/NRC-RES, *"Fire PRA Methodology for Nuclear Power Facilities,"* NUREG/CR-6850, EPRI TR-1011989, Final Report, September 2005.
32. NUMARC 91-06, "*Guidelines for Industry Actions to Address Shutdown Management*," December 1991.
33. NSD-403, *"Shutdown Risk Management (Modes 4, 5, 6, and No-Mode) per 10CFR 50.65(a)(4)*," Revision 19, April 2009.

TABLE 2-1
STATUS OF IDENTIFIED GAPS TO CAPABILITY CATEGORY II
OF THE ASME PRA STANDARD
THROUGH ADDENDA RA-Sc-2007

Title	Description of Gap	Applicable SRs	Current Status / Comment	Importance to 5b Application
Gap #1	Accident sequence notebooks and system model notebooks should document the environmental effects of the initiating event and the impact on mitigation systems.	AS-B3	Open. Phenomenological effects are considered in the model, although these considerations are not always documented.	None – documentation issue.
Gap #2	Revise the data calc. to discuss component boundaries definitions.	DA-A1a	Open. SSC and unavailability boundaries, SSC failure modes and success criteria are used consistently across analyses; however, these need to be formally documented.	None – documentation issue.
Gap #3	Revise the data calc. to segregate standby and operating component data. Segregate components by service condition to the extent supported by the data.	DA-B1	Open. Previously, generic data sources often did not provide standby and operating failure rates. NUREG/CR-6928 does provided more of this data, and will be used going forward.	Partitioning the failure rates represents a refinement to the data analysis process, but is not expected to impact the 5b analysis.
Gap #4	Enhance the documentation to include a discussion of the specific checks performed on the Bayesian-updated data, as required by this SR.	DA-D4	Open. As part of the Bayesian update process, checks are performed to assure that the posterior distribution is	None – documentation issue.

Title	Description of Gap	Applicable SRs	Current Status / Comment	Importance to 5b Application
			reasonable given the prior distribution and plant experience. These checks need to be formally documented.	
Gap #5	Provide documentation of the comparison of the component boundaries assumed for the generic CCF estimates to those assumed in the PRA to ensure that these boundaries are consistent.	DA-D6	Open. Generic common cause failure (CCF) probabilities are considered for applicability to the plant. CCF probabilities are consistent with plant experience and component boundaries, although the CCF documentation needs to be enhanced to discuss component boundaries.	None – documentation issue.
Gap #6	Enhance the Human Reliability Analysis (HRA) to consider the potential for calibration errors.	HR-A2	Open. Based on evaluations using the EPRI HRA calculator, calibration errors that result in failure of a single channel are expected to fall in the low 10^{-3} range.	Relative to post-initiator human error probabilities (HEPs), equipment random failure rates and maintenance unavailability, calibration HEPs are not expected to contribute significantly to overall equipment unavailability. Additionally, the next

Title	Description of Gap	Applicable SRs	Current Status / Comment	Importance to 5b Application
				revision of the PRA will incorporate the potential for calibration errors in the HRA. Thus there is no impact on the 5b analysis.
Gap #7	Identify maintenance and calibration activities that could simultaneously affect equipment in either different trains of a redundant system or diverse systems.	HR-A3	Open. Based on evaluations using the EPRI HRA calculator, calibration errors that result in failure of multiple channels are expected to fall in the low 10^{-5} range.	Relative to post-initiator HEPs, latent human error probabilities, equipment random failure rates and maintenance unavailability, calibration HEPs and misalignment of multiple trains of equipment are not expected to contribute significantly to overall equipment unavailability. Thus there is no impact on the 5b analysis.
Gap #8	Develop mean values for pre-initiator	HR-D6	Open. Pre-initiator HEPs are generally set to relatively high	The suggested data refinement is not

Title	Description of Gap	Applicable SRs	Current Status / Comment	Importance to 5b Application
	HEPs.		screening values, which bound the mean values. Even so, pre-initiator HEPs are not significant contributors to risk.	expected to have a significant impact on the results. Thus there is no impact on the 5b analysis.
Gap #9	Document in more detail the influence of performance shaping factors on execution human error probabilities.	HR-G3	Open. Performance shaping factors are accounted for in the development of human error probabilities, although detailed documentation is not always available for every HRA input.	None – documentation issue.
Gap #10	Enhance HRA documentation accordingly.	HR-G4	Open. Thermal Hydraulic (T/H) analyses, simulator runs and operator interviews are used in developing the time available to complete operator actions. The time at which the cue to take action is received is specified in the HEP quantification. However, the HRA documentation needs to be enhanced to provide a traceable path to all analysis inputs.	None – documentation issue.
Gap #11	Document a review of the human failure events (HFEs) and their final	HR-G6	Open. HFEs are reviewed by knowledgeable site personnel	None – documentation

Title	Description of Gap	Applicable SRs	Current Status / Comment	Importance to 5b Application
	HEPs relative to each other to confirm their reasonableness given the scenario context, plant history, procedures, operational practices, and experience.		to assure high quality. However, this review needs to be better documented.	issue.
Gap #12	Develop mean values for post-initiator HEPs.	HR-G9	Open. The use of mean values for HEPs instead of lower probability median values can affect the PRA results.	**The 5b analysis will include a sensitivity study to evaluate the use of different HEPs if the calculated risk is close to the threshold.**
Gap #13	Develop more detailed documentation of operator cues, relevant performance shaping factors, and availability of sufficient manpower to perform the action.	HR-H2	Open. Operator recovery actions are credited only if they are feasible, as determined by the procedural guidance, cues, performance shaping factors and available manpower. As noted for HR-G3, -G4, and -G6 above, the documentation of these considerations needs to be enhanced.	None – documentation issue.
Gap #14	Various enhancements to the initiating events analysis documentation.	IE-A1 IE-A3a	Open. No technical issues are identified, just a need to enhance the documentation.	None – documentation issues.

Title	Description of Gap	Applicable SRs	Current Status / Comment	Importance to 5b Application
		IE-A4 IE-A4a IE-A5 IE-A6 IE-A7 IE-B1 IE-B2 IE-B3 IE-D3		
Gap #15	Various enhancements to the internal flood analysis: • Discuss flood mitigative features. • Address the potential for spray, jet impingement, and pipe whip failures. • Provide more analysis of flood propagation flowpaths. Address potential structural failure of doors or walls due to flooding loads and the potential for barrier unavailability. Address potential	IF-B3 IF-C2c IF-C3 IF-C3b IF-E6b IF-F2	Open.	**Until the flooding analysis is upgraded, the potential for flood-induced failures of SSCs will be assessed on a case-by-case basis.**

Title	Description of Gap	Applicable SRs	Current Status / Comment	Importance to 5b Application
	indirect effects. • Enhance the documentation to address all of the SR details.			
Gap #16	Explicitly model Reactor Coolant System (RCS) depressurization for small Loss of Coolant Accidents (LOCAs) and perform the dependency analysis on the HEPs.	LE-C6	Open. This issue affects certain small LOCAs. However, since the small LOCA contribution to LERF is small, there is no significant impact on the PRA results.	No impact on the 5b analysis.
Gap #17	Various enhancements to the LERF documentation.	LE-G3 LE-G4 LE-G5 LE-G6	Open.	None – documentation issue.
Gap #18	Perform and document a comparison of PRA results with similar plants.	LE-F3 QU-D3	Open. Since Catawba and McGuire are sister plants, in practice, their results are often compared. Also, comparisons performed for MSPI and other programs help identify causes for significant differences. However, to fully meet this SR, the model quantification documentation needs to be enhanced to provide a results	None – documentation issues.

Title	Description of Gap	Applicable SRs	Current Status / Comment	Importance to 5b Application
			comparison.	
Gap #19	Perform and document sensitivity analyses to determine the impact of the assumptions and sources of model uncertainty on the results.	LE-F2 QU-E4	Open.	**Perform and document sensitivity analyses to determine the impact of the assumptions and sources of model uncertainty on the 5b analysis results.**
Gap #20	Expand the documentation of the PRA model results to address all required items.	QU-F2 QU-F6	Open. These SRs pertain to the model quantification documentation.	None – documentation issues.
Gap #21	Improve the documentation on the T/H bases for all safety function success criteria for all initiators.	SC-A4	Open. Success criteria are developed to address all of the modeled initiating events. However, the documentation of success criteria needs to be improved to include initiator information.	None – documentation issue.
Gap #22	Provide evidence that an acceptability review of the T/H analyses is performed.	SC-B5	Open. Catawba success criteria are consistent with those of sister plants included in the Pressurized Water Reactor Owners Group (PWROG) Probabilistic Safety Assessment (PSA) database.	None – documentation issue.

Title	Description of Gap	Applicable SRs	Current Status / Comment	Importance to 5b Application
			However, to fully meet this SR, the success criteria documentation needs to be enhanced to include a results comparison.	
Gap #23	Expand the documentation of the success criteria development to address all required items.	SC-C1 SC-C2	Open. These SRs pertain to the success criteria documentation.	None – documentation issues.
Gap #24	Enhance the system documentation to include an up-to-date system walkdown checklist and system engineer review for each system.	SY-A4	Open. To support system model development, walkdowns and plant personnel interviews were performed. However, documentation of an up-to-date system walkdown is not included with each system notebook.	None – documentation issue.
Gap #25	Enhance systems analysis documentation to discuss component boundaries.	SY-A8	Open. Basic event component boundaries utilized in the systems analysis are consistent with those in the data analysis. In addition, component boundaries are consistent with those defined in the generic failure rate source documents, such as NUREG/CR-6928. Dependencies among components, such as interlocks, are explicitly	None – documentation issue.

Title	Description of Gap	Applicable SRs	Current Status / Comment	Importance to 5b Application
			modeled, consistent with the PRA Modeling Guidelines workplace procedure. There is no evidence of a technical problem with component boundaries, just a need to improve the documentation.	
Gap #26	Provide quantitative evaluations for screening.	SY-A14	Open. It is expected that conversion to a more quantitative approach would not change decisions about whether or not to exclude components or failure modes. A review of our qualitative screening process confirms this expectation. For example, transfer failure events for motor-operated valves (MOVs) with 24 hr exposure times may not be modeled unless probabilistically significant with respect to logically equivalent basic events. For Catawba, the MOV transfers failure probability is less than 1% of the MOV fails to open on demand failure rate. In cases like this, not including the	There is no evidence of a technical problem associated with the screening of components or component failure modes, just a need to document a quantitative screening. Thus there is no impact on the 5b analysis.

Title	Description of Gap	Applicable SRs	Current Status / Comment	Importance to 5b Application
			relatively low probability failure mode in the PRA model does not have an appreciable impact on the results.	
Gap #27	Per Duke Energy's PRA modeling guidelines, ensure that a walkdown/system engineer interview checklist is included in each system notebook. Based on the results of the system walkdown, summarize in the system write-up any possible spatial dependencies or environmental hazards that may impact system operation.	SY-B8	Open. As noted for SY-A4, walkdowns (which look for spatial and environmental hazards) have been performed, although up-to-date walkdown documentation is not included with each system notebook.	None – documentation issue.
Gap #28	Document a consideration of potential SSC failure due to adverse environmental conditions.	SY-B15	Open. The impact of adverse environmental conditions on SSC reliability is considered but is not always documented. However, there is no evidence of a technical problem associated with components that may be required to operate in conditions beyond their environmental qualification, just a need to improve the documentation.	None – documentation issue.

Title	Description of Gap	Applicable SRs	Current Status / Comment	Importance to 5b Application
Gap #29	Enhance system model documentation to comply with all ASME PRA Standard requirements.	SY-C2	Open. This SR pertains to the systems analysis documentation.	None – documentation issue.

ATTACHMENT 3

PROPOSED TECHNICAL SPECIFICATION CHANGES

INSERTS

Insert 1

In accordance with the Surveillance Frequency Control Program

REVIEWER'S NOTE: Text deleted and replaced by Insert 1 will be relocated to the Surveillance Frequency Control Program (SFCP) document(s) per TSTF-425.

Insert 3

5.5.17 Surveillance Frequency Control Program

This Program provides controls for Surveillance Frequencies. The program shall ensure that the Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operations are met.

a. The Surveillance Frequency Control Program shall contain a list of Frequencies of those Surveillance Requirements for which the Frequency is controlled by the program.

b. Changes to the Frequencies listed in the Surveillance Frequency Control Program shall be made in accordance with NEI 04-10, "Risk-Informed Method for Control of Surveillance Frequencies," Revision 1.

c. The provisions of Surveillance Requirements 3.0.2 and 3.0.3 are applicable to the Frequencies established in the Surveillance Frequency Control Program.

Note: Insert 2 is included on Attachment 4.

1.1 Definitions (continued)

RATED THERMAL POWER (RTP)

RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3411 MWt.

REACTOR TRIP SYSTEM (RTS) RESPONSE TIME

The RTS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RTS trip setpoint at the channel sensor until loss of stationary gripper coil voltage. 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 the methodology for verification have been previously reviewed and approved by the NRC.

SHUTDOWN MARGIN (SDM)

SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming:

a. All rod cluster control assemblies (RCCAs) are fully inserted except for the single RCCA of highest reactivity worth, which is assumed to be fully withdrawn. 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.

SLAVE RELAY TEST

A SLAVE RELAY TEST shall consist of energizing each slave relay and verifying the OPERABILITY of each slave relay. The SLAVE RELAY TEST shall include, as a minimum, a continuity check of associated testable actuation devices.

~~STAGGERED TEST BASIS~~

~~A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function.~~

THERMAL POWER

THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

(continued)

3.1 REACTIVITY CONTROL SYSTEMS

3.1.1 SHUTDOWN MARGIN (SDM)

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

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

ACTIONS

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.1.1 Verify SDM is within the limit specified in the COLR.	~~24 hours~~ Insert 1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.2.1	---------NOTE--------- The predicted reactivity values may be adjusted (normalized) to correspond to the measured core reactivity prior to exceeding a fuel burnup of 60 effective full power days (EFPD) after each fuel loading. -------------------- Verify measured core reactivity is within ± 1% Δk/k of predicted values.	Once prior to entering MODE 1 after each refueling <u>AND</u> ~~31 EFPD thereafter~~ Insert 1

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition B not met.	C.1 Be in MODE 3.	6 hours
D. More than one rod not within alignment limit.	D.1.1 Verify SDM is within the limit specified in the COLR. OR D.1.2 Initiate boration to restore required 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 rod positions within alignment limit.	~~12 hours~~ INSERT 1 AND Once within 4 hours and every 4 hours thereafter when the rod position deviation monitor is inoperable

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.4.2	Verify rod freedom of movement (trippability) by moving each rod not fully inserted in the core ≥ 10 steps in either direction.	~~92 days~~ INSERT 1
SR 3.1.4.3	Verify rod drop time of each rod, from the fully withdrawn position, is ≤ 2.2 seconds from the beginning of decay of stationary gripper coil voltage to dashpot entry, with: a. $T_{avg} \geq 551°F$; and b. All reactor coolant pumps operating.	Prior to reactor criticality after each removal of the reactor head

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.5.1	Verify each shutdown bank is within the limits specified in the COLR.	~~12 hours~~ Insert 1

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.6.2	Verify each control bank insertion is within the limits specified in the COLR.	~~12 hours~~ INSERT 1 AND Once within 4 hours and every 4 hours thereafter when the rod insertion limit monitor is inoperable
SR 3.1.6.3	Verify sequence and overlap limits specified in the COLR are met for control banks not fully withdrawn from the core.	~~12 hours~~ Insert 1

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition C not met.	D.1 Be in MODE 3.	15 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.8.1	Perform a CHANNEL OPERATIONAL TEST on power range and intermediate range channels per SR 3.3.1.7, SR 3.3.1.8, and Table 3.3.1-1.	Prior to initiation of PHYSICS TESTS
SR 3.1.8.2	Verify the RCS lowest loop average temperature is ≥ 541°F.	~~30 minutes~~ Insert 1
SR 3.1.8.3	Verify THERMAL POWER is ≤ 5% RTP.	~~1 hour~~ Insert 1
SR 3.1.8.4	Verify SDM is within the limit specified in the COLR.	~~24 hours~~ Insert 1

SURVEILLANCE REQUIREMENTS

---NOTE---

During power escalation at the beginning of each cycle, THERMAL POWER may be increased until an equilibrium power level has been achieved, at which a power distribution map is obtained.

SURVEILLANCE		FREQUENCY
SR 3.2.1.1	Verify $F^M_Q(X,Y,Z)$ is within steady state limit.	Once within 12 hours after achieving equilibrium conditions after exceeding, by ≥ 10% RTP, the THERMAL POWER at which $F^M_Q(X,Y,Z)$ was last verified AND ~~31 EFPD thereafter~~ INSERT 1

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.2.1.2	-------------------------------NOTE------------------------------- 1. Extrapolate $F^M_Q(X,Y,Z)$ using at least two measurements to 31 EFPD beyond the most recent measurement. If $F^M_Q(X,Y,Z)$ is within limits and the 31 EFPD extrapolation indicates: $F^M_Q(X,Y,Z)_{EXTRAPOLATED} \geq F^L_Q(X,Y,Z)^{OP}{}_{EXTRAPOLATED}$, and $\frac{F^M_Q(X,Y,Z)_{EXTRAPOLATED}}{F^L_Q(X,Y,Z)^{OP}{}_{EXTRAPOLATED}} > \frac{F^M_Q(X,Y,Z)}{F^L_Q(X,Y,Z)^{OP}}$ then: a. Increase $F^M_Q(X,Y,Z)$ by the appropriate factor specified in the COLR and reverify $F^M_Q(X,Y,Z) \leq F^L_Q(X,Y,Z)^{OP}$; or b. Repeat SR 3.2.1.2 prior to the time at which $F^M_Q(X,Y,Z) \leq F^L_Q(X,Y,Z)^{OP}$ is extrapolated to not be met. 2. Extrapolation of $F^M_Q(X,Y,Z)$ is not required for the initial flux map taken after reaching equilibrium conditions. -- Verify $F^M_Q(X,Y,Z) \leq F^L_Q(X,Y,Z)^{OP}$.	Once within 12 hours after achieving equilibrium conditions after exceeding, by $\geq$ 10% RTP, the THERMAL POWER at which $F^M_Q(X,Y,Z)$ was last verified <u>AND</u> ~~31 EFPD thereafter~~ INSERT 1

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.2.1.3	-------------------------------NOTES------------------------------- 1. Extrapolate $F^M_Q(X,Y,Z)$ using at least two measurements to 31 EFPD beyond the most recent measurement. If $F^M_Q(X,Y,Z)$ is within limits and the 31 EFPD extrapolation indicates: $F^M_Q(X,Y,Z)_{EXTRAPOLATED} \geq F^L_Q(X,Y,Z)^{RPS}{}_{EXTRAPOLATED}$, and $\frac{F^M_Q(X,Y,Z)_{EXTRAPOLATED}}{F^L_Q(X,Y,Z)^{RPS}{}_{EXTRAPOLATED}} > \frac{F^M_Q(X,Y,Z)}{F^L_Q(X,Y,Z)^{RPS}}$ then: a. Increase $F^M_Q(X,Y,Z)$ by the appropriate factor specified in the COLR and reverify $F^M_Q(X,Y,Z) \leq F^L_Q(X,Y,Z)^{RPS}$; or b. Repeat SR 3.2.1.3 prior to the time at which $F^M_Q(X,Y,Z) \leq F^L_Q(X,Y,Z)^{RPS}$ is extrapolated to not be met. 2. Extrapolation of $F^M_Q(X,Y,Z)$ is not required for the initial flux map taken after reaching equilibrium conditions. --- Verify $F^M_Q(X,Y,Z) \leq F^L_Q(X,Y,Z)^{RPS}$.	Once within 12 hours after achieving equilibrium conditions after exceeding, by $\geq$ 10% RTP, the THERMAL POWER at which $F^M_Q(X,Y,Z)$ was last verified AND ~~31 EFPD thereafter~~ INSERT 1

SURVEILLANCE REQUIREMENTS

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

During power escalation at the beginning of each cycle, THERMAL POWER may be increased until an equilibrium power level has been achieved, at which a power distribution map is obtained.

SURVEILLANCE		FREQUENCY
SR 3.2.2.1	Verify $F^M_{\Delta H}(X,Y)$ is within steady state limit.	Once within 12 hours after achieving equilibrium conditions after exceeding, by ≥ 10% RTP, the THERMAL POWER at which $F^M_{\Delta H}(X,Y)$ was last verified AND ~~31 EFPD thereafter~~ INSERT 1

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.2.2.2	-----NOTES----- 1. Extrapolate $F^M_{\Delta H}(X,Y)$ using at least two measurements to 31 EFPD beyond the most recent measurement. If $F^M_{\Delta H}(X,Y)$ is within limits and the 31 EFPD extrapolation indicates: $F^M_{\Delta H}(X,Y)_{EXTRAPOLATED} \geq F^L_{\Delta H}(X,Y)^{SURV}_{EXTRAPOLATED}$ and $\frac{F^M_{\Delta H}(X,Y)_{EXTRAPOLATED}}{F^L_{\Delta H}(X,Y)^{SURV}_{EXTRAPOLATED}} > \frac{F^M_{\Delta H}(X,Y)}{F^L_{\Delta H}(X,Y)^{SURV}}$ then: a. Increase $F^M_{\Delta H}(X,Y)$ by the appropriate factor specified in the COLR and reverify $F^M_{\Delta H}(X,Y) \leq F^L_{\Delta H}(X,Y)^{SURV}$; or b. Repeat SR 3.2.2.2 prior to the time at which $F^M_{\Delta H}(X,Y) \leq F^L_{\Delta H}(X,Y)^{SURV}$ is extrapolated to not be met. 2. Extrapolation of $F^M_{\Delta H}(X,Y)$ is not required for the initial flux map taken after reaching equilibrium conditions. ---------- Verify $F^M_{\Delta H}(X,Y) \leq F^L_{\Delta H}(X,Y)^{SURV}$.	Once within 12 hours after achieving equilibrium conditions after exceeding, by $\geq$ 10% RTP, the THERMAL POWER at which $F^M_{\Delta H}(X,Y)$ was last verified AND ~~31 EFPD thereafter~~ INSERT 1

3.2 POWER DISTRIBUTION LIMITS

3.2.3 AXIAL FLUX DIFFERENCE (AFD)

LCO 3.2.3 The AFD in % flux difference units shall be maintained within the limits specified in the COLR.

--NOTE--
The AFD shall be considered outside limits when two or more OPERABLE excore channels indicate AFD to be outside limits.

APPLICABILITY: MODE 1 with THERMAL POWER $\geq$ 50% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. AFD not within limits.	A.1 Reduce THERMAL POWER to < 50% RTP.	30 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.3.1 Verify AFD within limits for each OPERABLE excore channel.	~~7 days~~ Insert 1 AND Once within 1 hour and every 1 hour thereafter with the AFD monitor alarm inoperable

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.4.1	------NOTES------ 1. With input from one Power Range Neutron Flux channel inoperable and THERMAL POWER <75% RTP, the remaining three power range channels can be used for calculating QPTR. 2. SR 3.2.4.2 may be performed in lieu of this Surveillance. 3. This SR is not required to be performed until 12 hours after exceeding 50% RTP. ------------ Verify QPTR is within limit by calculation.	~~7 days~~ Insert 1 AND Once within 12 hours and every 12 hours thereafter with the QPTR alarm inoperable
SR 3.2.4.2	------NOTES------ Only required to be performed if input from one or more Power Range Neutron Flux channels are inoperable with THERMAL POWER $\geq$ 75% RTP. ------------ Verify QPTR is within limit using the movable incore detectors.	~~12 hours~~ Insert 1

SURVEILLANCE REQUIREMENTS

-----NOTE-----
Refer to Table 3.3.1-1 to determine which SRs apply for each RTS Function.

SURVEILLANCE		FREQUENCY
SR 3.3.1.1	Perform CHANNEL CHECK.	~~12 hours~~ Insert 1
SR 3.3.1.2	-----NOTES----- 1. Adjust NIS channel if absolute difference is > 2%. 2. Not required to be performed until 12 hours after THERMAL POWER is ≥ 15% RTP. ---------- Compare results of calorimetric heat balance calculation to Nuclear Instrumentation System (NIS) channel output.	~~24 hours~~ Insert 1
SR 3.3.1.3	-----NOTES----- 1. Adjust NIS channel if absolute difference is ≥ 3%. 2. Not required to be performed until 24 hours after THERMAL POWER is ≥ 15% RTP. ---------- Compare results of the incore detector measurements to NIS AFD.	~~31 effective full power days (EFPD)~~ Insert 1

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.4	-----NOTE----- This Surveillance must be performed on the reactor trip bypass breaker prior to placing the bypass breaker in service. ----- Perform TADOT.	~~62 days on a STAGGERED TEST BASIS*~~ Insert 1
SR 3.3.1.5	Perform ACTUATION LOGIC TEST.	~~92 days on a STAGGERED TEST BASIS~~ Insert 1
SR 3.3.1.6	-----NOTE----- Not required to be performed until 24 hours after THERMAL POWER is $\geq$ 75% RTP. ----- Calibrate excore channels to agree with incore detector measurements.	~~92 EFPD~~ Insert 1
SR 3.3.1.7	-----NOTE----- Not required to be performed for source range instrumentation prior to entering MODE 3 from MODE 2 until 4 hours after entry into MODE 3. ----- Perform COT.	~~184 days~~ Insert 1

(continued)

~~* The SR 3.3.1.4 Frequency of "62 days on a STAGGERED TEST BASIS" as it applies to Unit 2 Train 2A and Train 2B reactor trip breaker testing may be extended on a one-time basis to March 10, 2009 at 0500 hours, upon which Unit 2 shall be in Mode 3 with reactor trip breakers open for the End of Cycle 16 Refueling Outage. Upon entry into Mode 3 with reactor trip breakers open for this refueling outage, this extension shall expire. The provisions of SR 3.0.2 are not applicable to this extension.~~

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.3.1.8	------NOTE------ This Surveillance shall include verification that interlocks P-6 (for the Intermediate Range channels) and P-10 (for the Power Range channels) are in their required state for existing unit conditions. ------ Perform COT. the frequency specified in the Surveillance Frequency Control Program OR the previous 184 days INSERT 1	------NOTE------ Only required when not performed within ~~previous 184 days~~ ------ Prior to reactor startup AND Four hours after reducing power below P-10 for power and intermediate range instrumentation AND Four hours after reducing power below P-6 for source range instrumentation AND ~~Every 184 days thereafter~~

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.3.1.9	-------------------------------NOTE------------------------------- Verification of setpoint is not required. -- Perform TADOT.	~~92 days~~ Insert 1
SR 3.3.1.10	-------------------------------NOTE------------------------------- This Surveillance shall include verification that the time constants are adjusted to the prescribed values. -- Perform CHANNEL CALIBRATION.	~~18 months~~ Insert 1
SR 3.3.1.11	-------------------------------NOTE------------------------------- 1. Neutron detectors are excluded from CHANNEL CALIBRATION. 2. Power and Intermediate Range Neutron Flux detector plateau voltage verification is not required to be performed prior to entry into MODE 1 or 2. -- Perform CHANNEL CALIBRATION.	~~18 months~~ Insert 1
SR 3.3.1.12	Perform CHANNEL CALIBRATION.	~~18 months~~ Insert 1
SR 3.3.1.13	Perform COT.	~~18 months~~ Insert 1

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.14	-----NOTE----- Verification of setpoint is not required. ----- Perform TADOT.	~~18 months~~ Insert 1
SR 3.3.1.15	-----NOTE----- Verification of setpoint is not required. ----- Perform TADOT.	-----NOTE----- Only required when not performed within previous 31 days ----- Prior to reactor startup
SR 3.3.1.16	-----NOTE----- Neutron detectors are excluded from response time testing. ----- Verify RTS RESPONSE TIME is within limits.	~~18 months on a STAGGERED TEST BASIS~~ Insert 1
SR 3.3.1.17	Verify RTS RESPONSE TIME for RTDs is within limits.	~~18 months~~ INSERT 1

SURVEILLANCE REQUIREMENTS

-----NOTE-----
Refer to Table 3.3.2-1 to determine which SRs apply for each ESFAS Function.

SURVEILLANCE		FREQUENCY
SR 3.3.2.1	Perform CHANNEL CHECK.	12 hours Insert 1
SR 3.3.2.2	Perform ACTUATION LOGIC TEST.	92 days on a STAGGERED TEST BASIS Insert 1
SR 3.3.2.3	-----NOTE----- Final actuation of pumps or valves not required. ---------- Perform TADOT.	31 days INSERT 1
SR 3.3.2.4	Perform MASTER RELAY TEST.	92 days on a STAGGERED TEST BASIS Insert 1
SR 3.3.2.5	Perform COT.	184 days Insert 1
SR 3.3.2.6	Perform SLAVE RELAY TEST.	~~92 days~~ Insert 1 ~~OR~~ ~~18 months for only Westinghouse AR and Potter & Brumfield MDR relay types~~
SR 3.3.2.7	Perform COT.	31 days INSERT 1

(continued)

SURVEILLANCE REQUIREMENTS

------------------------------NOTE------------------------------
Refer to Table 3.3.2-1 to determine which SRs apply for each ESFAS Function.

	SURVEILLANCE	FREQUENCY
SR 3.3.2.1	Perform CHANNEL CHECK.	~~12 hours~~
SR 3.3.2.2	Perform ACTUATION LOGIC TEST.	~~92 days on a STAGGERED TEST BASIS~~ INSERT 1
SR 3.3.2.3	----------NOTE---------- Final actuation of pumps or valves not required. ------------------------ Perform TADOT.	~~31 days~~
SR 3.3.2.4	Perform MASTER RELAY TEST.	~~92 days on a STAGGERED TEST BASIS~~
SR 3.3.2.5	Perform COT.	~~184 days~~ INSERT 1
SR 3.3.2.6	Perform SLAVE RELAY TEST.	~~92 days OR 18 months for only Westinghouse AR and Potter & Brumfield MDR relay types~~
SR 3.3.2.7	Perform COT.	~~31 days~~

(continued)

SURVEILLANCE		FREQUENCY
SR 3.3.2.8	------NOTE------ Verification of setpoint not required for manual initiation functions. ------ Perform TADOT.	~~18 months~~ Insert 1
SR 3.3.2.9	------NOTE------ This Surveillance shall include verification that the time constants are adjusted to the prescribed values. ------ Perform CHANNEL CALIBRATION.	~~18 months~~ Insert 1
SR 3.3.2.10	------NOTE------ Not required to be performed for the turbine driven AFW pump until 24 hours after SG pressure is $\geq$ 600 psig. ------ Verify ESFAS RESPONSE TIMES are within limit.	~~18 months on a STAGGERED TEST BASIS~~ Insert 1
SR 3.3.2.11	Perform COT.	~~18 months~~ INSERT 1
SR 3.3.2.12	Perform ACTUATION LOGIC TEST.	~~18 months~~ INSERT 1

SURVEILLANCE REQUIREMENTS

--NOTE--
SR 3.3.3.1 and SR 3.3.3.3 apply to each PAM instrumentation Function in Table 3.3.3-1.

	SURVEILLANCE	FREQUENCY
SR 3.3.3.1	Perform CHANNEL CHECK for each required instrumentation channel that is normally energized.	~~31 days~~ INSERT 1
SR 3.3.3.2	Not Used	
SR 3.3.3.3	----------------------------NOTES---------------------------- 1. Neutron detectors are excluded from CHANNEL CALIBRATION. 2. CHANNEL CALIBRATION may consist of an electronic calibration of the Containment Area - High Range Radiation Monitor, not including the detector, for range decades above 10 R/h and a one point calibration check of the detector below 10 R/h with an installed or portable gamma source. --- Perform CHANNEL CALIBRATION.	INSERT 1 ~~18 months~~

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.4.1	Perform CHANNEL CHECK for each required instrumentation channel that is normally energized.	~~31 days~~ Insert 1
SR 3.3.4.2	-------------------------------NOTE------------------------------ Not applicable to Reactor Trip Breaker Position. -- Perform CHANNEL CALIBRATION for each required instrumentation channel.	~~18 months~~ Insert 1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.5.1	-----NOTE----- Testing shall consist of voltage sensor relay testing excluding actuation of load shedding diesel start, and time delay times. ----- Perform TADOT.	~~31 days~~ Insert 1
SR 3.3.5.2	Perform CHANNEL CALIBRATION with NOMINAL TRIP SETPOINT and Allowable Value as follows: a. Loss of voltage Allowable Value ≥ 3242 V. Loss of voltage NOMINAL TRIP SETPOINT = 3500 V. b. Degraded voltage Allowable Value ≥ 3738 V. Degraded voltage NOMINAL TRIP SETPOINT = 3766 V.	~~18 months~~ Insert 1

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.5.2

SR 3.3.5.2 is the performance of a CHANNEL CALIBRATION.

The setpoints, as well as the response to a loss of voltage and a degraded voltage test, shall include a single point verification that the trip occurs within the required time delay, as shown in Reference 1.

~~A CHANNEL CALIBRATION is performed every 18 months, or approximately at every refueling.~~ CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.

~~The Frequency of 18 months is based on operating experience and consistency with the typical industry refueling cycle and is justified by the assumption of an 18 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.~~ Insert 2

REFERENCES

1. UFSAR, Section 8.3.

2. UFSAR, Chapter 15.

3. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).

SURVEILLANCE REQUIREMENTS

---NOTE---
Refer to Table 3.3.6-1 to determine which SRs apply for each Containment Air Release and Addition Isolation Function.

	SURVEILLANCE	FREQUENCY
SR 3.3.6.1	Perform ACTUATION LOGIC TEST.	~~92 days on a STAGGERED TEST BASIS~~ Insert 1
SR 3.3.6.2	Perform MASTER RELAY TEST.	~~92 days on a STAGGERED TEST BASIS~~ Insert 1
SR 3.3.6.3	Perform SLAVE RELAY TEST.	~~92 days~~ Insert 1 ~~OR~~ ~~18 months for only Westinghouse AR and Potter & Brumfield MDR relay types~~
SR 3.3.6.4	---------------------------NOTE--------------------------- Verification of setpoint is not required. --- Perform TADOT.	~~18 months~~ INSERT 1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.9.1	Perform CHANNEL CHECK.	~~12 hours~~ Insert 1
SR 3.3.9.2	Perform COT.	~~31 days~~
SR 3.3.9.3	Verify each automatic valve moves to the correct position and Reactor Makeup Water pumps stop upon receipt of an actual or simulated actuation signal.	~~18 months~~
SR 3.3.9.4	------------------------------NOTE------------------------------ Only required to be performed when used to satisfy Required Action A.3 or B.3. --- Perform CHANNEL CHECK on the Source Range Neutron Flux Monitors.	INSERT 1 ~~12 hours~~
SR 3.3.9.5	------------------------------NOTE------------------------------ Only required to be performed when used to satisfy Required Action A.3 or B.3. --- Verify combined flowrates from both Reactor Makeup Water Pumps are ≤ the value in the COLR.	~~31 days~~
SR 3.3.9.6	------------------------------NOTE------------------------------ Only required to be performed when used to satisfy Required Action A.3 or B.3. --- Perform COT on the Source Range Neutron Flux Monitors.	INSERT 1 ~~184 days~~

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.1.1	Verify pressurizer pressure is within limits.	~~12 hours~~ INSERT 1
SR 3.4.1.2	Verify RCS average temperature is within limits.	~~12 hours~~ INSERT 1
SR 3.4.1.3	Verify RCS total flow rate is within limits.	~~12 hours~~ Insert 1
SR 3.4.1.4	Perform CHANNEL CALIBRATION for each RCS total flow indicator.	~~18 months~~ INSERT 1

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. -------------NOTE----------- Required Action C.2 shall be completed whenever this Condition is entered. --------------------------------	C.1 Initiate action to restore parameter(s) to within limits.	Immediately
	AND	
Requirements of LCO not met any time in other than MODE 1, 2, 3, or 4.	C.2 Determine RCS is acceptable for continued operation.	Prior to entering MODE 4

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.3.1	-------------------------------NOTE------------------------------- Only required to be performed during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing. ---	
	Verify RCS pressure, RCS temperature, and RCS heatup and cooldown rates are within limits.	~~30 minutes~~ Insert 1

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.4 RCS Loops — MODES 1 and 2

LCO 3.4.4 Four RCS loops shall be OPERABLE and in operation.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of LCO not met.	A.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.4.1 Verify each RCS loop is in operation.	~~12 hours~~ Insert 1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.5.1	Verify required RCS loops are in operation.	~~12 hours~~ Insert 1
SR 3.4.5.2	Verify steam generator secondary side water levels are ≥ 12% narrow range for required RCS loops.	~~12 hours~~ Insert 1
SR 3.4.5.3	Verify correct breaker alignment and indicated power are available to the required pumps that are not in operation.	~~7 days~~ Insert 1

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One RHR loop OPERABLE. AND ALL RCS loops inoperable.	B.1 Be in MODE 5.	24 hours
C. Both required RCS or RHR loops inoperable. OR No RCS or RHR loop in operation.	C.1 Suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1 and maintain $k_{eff} < 0.99$. AND C.2 Initiate action to restore one loop to OPERABLE status and operation.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.6.1	Verify one RHR or RCS loop is in operation.	~~12 hours~~ INSERT 1
SR 3.4.6.2	Verify SG secondary side water levels are ≥ 12% narrow range for required RCS loops.	~~12 hours~~ INSERT 1
SR 3.4.6.3	Verify correct breaker alignment and indicated power are available to the required pump that is not in operation.	~~7 days~~ INSERT 1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RHR loop inoperable. AND Required SGs secondary side water levels not within limits.	A.1 Initiate action to restore a second RHR loop to OPERABLE status. OR A.2 Initiate action to restore required SG secondary side water levels to within limits.	Immediately Immediately
B. Required RHR loops inoperable. OR No RHR loop in operation.	B.1 Suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1. AND B.2 Initiate action to restore one RHR loop to OPERABLE status and operation.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.7.1 Verify one RHR loop is in operation.	~~12 hours~~ Insert 1
SR 3.4.7.2 Verify SG secondary side water level is ≥ 12% narrow range in required SGs.	~~12 hours~~ Insert 1
SR 3.4.7.3 Verify correct breaker alignment and indicated power are available to the required RHR pump that is not in operation.	~~7 days~~ Insert 1

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required RHR loops inoperable. OR No RHR loop in operation.	B.1 Suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1.	Immediately
	AND	
	B.2 Initiate action to restore one RHR loop to OPERABLE status and operation.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.8.1	Verify one RHR loop is in operation.	~~12 hours~~ Insert 1
SR 3.4.8.2	Verify correct breaker alignment and indicated power are available to the required RHR pump that is not in operation.	~~7 days~~ Insert 1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.9.1	Verify pressurizer water level is $\leq$ 92% (1656 ft^3).	~~12 hours~~ Insert 1
SR 3.4.9.2	Verify capacity of each required group of pressurizer heaters is $\geq$ 150 kW.	~~92 days~~ Insert 1
SR 3.4.9.3	Verify required pressurizer heaters are capable of being powered from an emergency power supply.	~~18 months~~ Insert 1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. (continued)	F.2 Restore one block valve to OPERABLE status if three block valves are inoperable. AND F.3 Restore remaining block valve(s) to OPERABLE status.	2 hours 72 hours
G. Required Action and associated Completion Time of Condition F not met.	G.1 Be in MODE 3. AND G.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.11.1	-------------------------------NOTE------------------------------- Not required to be met with block valve closed in accordance with the Required Action of Condition B or E. -- Perform a complete cycle of each block valve.	~~92 days~~ Insert 1

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.11.2	-------------------------------NOTE------------------------------- Required to be performed in MODE 3 or MODE 4 when the temperature of all RCS cold legs is > 200°F. -- Perform a complete cycle of each PORV.	~~18 months~~ Insert 1
SR 3.4.11.3	------------------------------NOTE------------------------------- This SR is not applicable to valve NC-36B. -- Verify the nitrogen supply for each PORV is OPERABLE by: a. Manually transferring motive power from the air supply to the nitrogen supply, b. Isolating and venting the air supply, and c. Operating the PORV through one complete cycle.	~~18 months~~ INSERT 1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.12.1	Verify a maximum of two pumps (charging, safety injection, or charging and safety injection) are capable of injecting into the RCS.	~~12 hours~~ Insert 1
SR 3.4.12.2	Verify each accumulator is isolated.	~~12 hours~~ Insert 1
SR 3.4.12.3	Verify RHR suction isolation valves are open for each required RHR suction relief valve.	~~12 hours~~ Insert 1
SR 3.4.12.4	Verify PORV block valve is open for each required PORV.	~~72 hours~~ Insert 1
SR 3.4.12.5	-------------------------------NOTE------------------------------- Not required to be met until 12 hours after decreasing RCS cold leg temperature to ≤ 210°F. --- Perform a COT on each required PORV, excluding actuation.	~~31 days~~ Insert 1
SR 3.4.12.6	Perform CHANNEL CALIBRATION for each required PORV actuation channel.	~~18 months~~ Insert 1
SR 3.4.12.7	Verify associated RHR suction isolation valves are open, with operator power removed and locked in removed position, for each required RHR suction relief valve.	~~31 days~~ Insert 1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.13.1	-----NOTES----- 1. Not required to be performed until 12 hours after establishment of steady state operation. 2. Not applicable to primary to secondary LEAKAGE. ---------- Verify RCS Operational LEAKAGE within limits by performance of RCS water inventory balance.	------NOTE------ Only required to be performed during steady state operation ---------- ~~72 hours~~ Insert 1
SR 3.4.13.2	-----NOTE----- Not required to be performed until 12 hours after establishment of steady state operation. ---------- Verify primary to secondary LEAKAGE is ≤ 150 gallons per day through any one SG.	--------NOTE-------- Only required to be performed during steady state operation ---------- ~~72 hours~~ Insert 1

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.4.14.1	-----NOTES----- 1. Not required to be performed in MODES 3 and 4. 2. Not required to be performed on the RCS PIVs located in the RHR flow path when in the shutdown cooling mode of operation. 3. RCS PIVs actuated during the performance of this Surveillance are not required to be tested more than once if a repetitive testing loop cannot be avoided. ---------- Verify leakage from each RCS PIV is equivalent to $\leq$ 0.5 gpm per nominal inch of valve size up to a maximum of 5 gpm at an RCS pressure $\geq$ 2215 psig and $\leq$ 2255 psig.	In accordance with the Inservice Testing ~~Program~~, and ~~18 months~~ Insert 1 AND Prior to entering MODE 2 whenever the unit has been in MODE 5 for 7 days or more, if leakage testing has not been performed in the previous 9 months AND Within 24 hours following valve actuation due to automatic or manual action or flow through the valve

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.14.2	Verify RHR system interlock prevents the valves from being opened with a simulated or actual RCS pressure signal ≥ 425 psig.	~~18 months~~ Insert 1

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.4.15.1	Perform CHANNEL CHECK of the containment atmosphere particulate radioactivity monitor.	~~12 hours~~ Insert 1
SR 3.4.15.2	Perform COT of the containment atmosphere particulate radioactivity monitor.	~~92 days~~ Insert 1
SR 3.4.15.3	Perform CHANNEL CALIBRATION of the containment floor and equipment sump level monitors.	~~18 months~~ Insert 1
SR 3.4.15.4	Perform CHANNEL CALIBRATION of the containment atmosphere particulate radioactivity monitor.	~~18 months~~ Insert 1
SR 3.4.15.5	Perform CHANNEL CALIBRATION of the containment ventilation unit condensate drain tank level monitor.	~~18 months~~ Insert 1
SR 3.4.15.6	Perform CHANNEL CALIBRATION of the incore instrument sump level alarm.	~~18 months~~ INSERT 1

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A not met. OR DOSE EQUIVALENT I-131 in the unacceptable region of Figure 3.4.16-1.	C.1 Be in MODE 3 with $T_{avg} < 500°F$.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.16.1	Verify reactor coolant gross specific activity $\leq 100/\bar{E}$ μCi/gm.	~~7 days~~ Insert 1
SR 3.4.16.2	-------------------------------NOTE------------------------------- Only required to be performed in MODE 1. --- Verify reactor coolant DOSE EQUIVALENT I-131 specific activity $\leq$ 1.0 μCi/gm.	~~14 days~~ Insert 1 AND Between 2 and 6 hours after a THERMAL POWER change of $\geq$ 15% RTP within a 1 hour period

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.16.3	-------------------------------NOTE------------------------------- Not required to be performed until 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours. --- Determine $\bar{E}$ from a sample taken in MODE 1 after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours.	~~184 days~~ INSERT 1

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.17 RCS Loops — Test Exceptions

LCO 3.4.17 The requirements of LCO 3.4.4, "RCS Loops — MODES 1 and 2," may be suspended, with THERMAL POWER < P-7.

APPLICABILITY: MODES 1 and 2 during startup and PHYSICS TESTS.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. THERMAL POWER ≥ P-7.	A.1 Open reactor trip breakers.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.17.1 Verify THERMAL POWER is < P-7.	~~1 hour~~ Insert 1
SR 3.4.17.2 Perform a COT for each power range neutron flux-low and intermediate range neutron flux channel, P-10, and P-13.	Prior to initiation of startup and PHYSICS TESTS

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.1.1	Verify each accumulator isolation valve is fully open.	~~12 hours~~ Insert 1
SR 3.5.1.2	Verify borated water volume in each accumulator is ≥ 7630 gallons and ≤ 8079 gallons.	~~12 hours~~ Insert 1
SR 3.5.1.3	Verify nitrogen cover pressure in each accumulator is ≥ 585 psig and ≤ 678 psig.	~~12 hours~~ Insert 1
SR 3.5.1.4	Verify boron concentration in each accumulator is within the limits specified in the COLR.	~~31 days~~ Insert 1 AND -------NOTE------ Only required to be performed for affected accumulators ------------------ Once within 6 hours after each solution volume increase of ≥ 75 gallons that is not the result of addition from the refueling water storage tank
SR 3.5.1.5	Verify power is removed from each accumulator isolation valve operator when RCS pressure is > 1000 psig.	~~31 days~~ Insert 1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.2.1	Verify the following valves are in the listed position with power to the valve operator removed. Number / Position / Function NI162A / Open / SI Cold Leg Injection NI121A / Closed / SI Hot Leg Injection NI152B / Closed / SI Hot Leg Injection NI183B / Closed / RHR Hot Leg Injection NI173A / Open / RHR Cold Leg Injection NI178B / Open / RHR Cold Leg Injection NI100B / Open / SI Pump Suction from RWST NI147B / Open / SI Pump Mini-Flow	~~12 hours~~ Insert 1
SR 3.5.2.2	Verify each ECCS manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	~~31 days~~ Insert 1
SR 3.5.2.3	Verify ECCS piping is full of water.	~~31 days~~ Insert 1
SR 3.5.2.4	Verify each ECCS pump's developed head at the test flow point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.2.5	Verify each ECCS automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	~~18 months~~ Insert 1
SR 3.5.2.6	Verify each ECCS pump starts automatically on an actual or simulated actuation signal.	~~18 months~~ Insert 1
SR 3.5.2.7	Verify, for each ECCS throttle valve listed below, each position stop is in the correct position. Centrifugal Charging Pump Injection Throttle Valve Number: NI14, NI16, NI18, NI20 Safety Injection Pump Throttle Valve Number: NI164, NI166, NI168, NI170	~~18 months~~ Insert 1
SR 3.5.2.8	Verify, by visual inspection, that the ECCS containment sump strainer assembly is not restricted by debris and shows no evidence of structural distress or abnormal corrosion.	~~18 months~~ Insert 1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.4.1	Verify RWST borated water temperature is ≥ 70°F and ≤ 100°F.	~~24 hours~~ Insert 1
SR 3.5.4.2	Verify RWST borated water volume is ≥ 363,513 gallons.	~~7 days~~ Insert 1
SR 3.5.4.3	Verify RWST boron concentration is within the limits specified in the COLR.	~~7 days~~ Insert 1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.5.1	-----------------------------NOTE--------------------------------- Not required to be performed until 4 hours after the Reactor Coolant System pressure stabilizes at $\geq$ 2215 psig and $\leq$ 2255 psig. --- Verify manual seal injection throttle valves are adjusted to give a flow within limit with centrifugal charging pump operating and the charging flow control valve full open.	~~31 days~~ Insert 1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.2.1	-----NOTES----- 1. An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. 2. Results shall be evaluated against acceptance criteria applicable to SR 3.6.1.1. ---------- Perform required air lock leakage rate testing in accordance with the Containment Leakage Rate Testing Program.	In accordance with the Containment Leakage Rate Testing Program
SR 3.6.2.2	Perform a pressure test on each inflatable air lock door seal and verify door seal leakage is < 15 sccm.	~~6 months~~ INSERT 1
SR 3.6.2.3	Verify only one door in the air lock can be opened at a time.	~~18 months~~

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.3.1	Verify each containment purge supply and exhaust isolation valves for the lower compartment and the upper compartment, instrument room, and the Hydrogen Purge System is sealed closed, except for one purge valve in a penetration flow path while in Condition E of this LCO.	~~31 days~~ Insert 1
SR 3.6.3.2	Verify each Containment Air Release and Addition System isolation valve is closed, except when the valves are open for pressure control, ALARA or air quality considerations for personnel entry, or for Surveillances that require the valves to be open.	~~31 days~~ Insert 1
SR 3.6.3.3	-----NOTE----- Valves and blind flanges in high radiation areas may be verified by use of administrative controls. ----- Verify each containment isolation manual valve and blind flange that is located outside containment or annulus and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative controls.	~~31 days~~ Insert 1

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.6.3.4	------------------------------NOTE------------------------------ Valves and blind flanges in high radiation areas may be verified by use of administrative means. -- Verify each containment isolation manual valve and blind flange that is located inside containment or annulus and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative controls.	Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days
SR 3.6.3.5	Verify the isolation time of automatic power operated containment isolation valve is within limits.	In accordance with the Inservice Testing Program
SR 3.6.3.6	Perform leakage rate testing for Containment Purge System, Hydrogen Purge System, and Containment Air Release and Addition System valves with resilient seals.	In accordance with the Containment Leakage Rate Testing Program
SR 3.6.3.7	Verify each automatic containment isolation valve that is not locked, sealed or otherwise secured in position, actuates to the isolation position on an actual or simulated actuation signal.	~~18 months~~ Insert 1

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.3.8	Verify the combined leakage rate for all reactor building bypass leakage paths is $\leq 0.07\ L_a$ when pressurized to $\geq$ 14.68 psig.	In accordance with the Containment Leakage Rate Testing Program

3.6 CONTAINMENT SYSTEMS

3.6.4 Containment Pressure

LCO 3.6.4 Containment pressure shall be ≥ -0.1 psig and ≤ +0.3 psig.

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. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.6.4.1 Verify containment pressure is within limits.	~~12 hours~~ Insert 1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.5.1	Verify containment upper compartment average air temperature is within limits.	~~24 hours~~ Insert 1
SR 3.6.5.2	Verify containment lower compartment average air temperature is within limits.	~~24 hours~~ Insert 1

3.6 CONTAINMENT SYSTEMS

3.6.6 Containment Spray System

LCO 3.6.6 Two containment spray trains shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One containment spray train inoperable.	A.1 Restore containment spray train to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 84 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.6.1	Verify each containment spray manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position.	~~31 days~~ INSERT1

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.6.2	Verify each containment spray pump's developed head at the flow test point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program
SR 3.6.6.3	Verify each automatic containment spray valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	~~18 months~~ Insert 1
SR 3.6.6.4	Verify each containment spray pump starts automatically on an actual or simulated actuation signal.	~~18 months~~ INSERT 1
SR 3.6.6.5	Verify that each spray pump is de-energized and prevented from starting upon receipt of a terminate signal and is allowed to start upon receipt of a start permissive from the Containment Pressure Control System (CPCS).	~~18 months~~ INSERT 1
SR 3.6.6.6	Verify that each spray pump discharge valve closes or is prevented from opening upon receipt of a terminate signal and is allowed to open upon receipt of a start permissive from the Containment Pressure Control System (CPCS).	~~18 months~~ INSERT 1
SR 3.6.6.7	Verify each spray nozzle is unobstructed.	~~10 years~~ INSERT 1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.8.1	Operate each HSS train for ≥ 15 minutes.	~~92 days~~ INSERT 1
SR 3.6.8.2	Verify the fan motor current is ≤ 69 amps when the fan speed is ≥ 3560 rpm and ≤ 3600 rpm with the hydrogen skimmer fan operating and the motor operated suction valve closed.	~~92 days~~ INSERT 1
SR 3.6.8.3	Verify the motor operated suction valve opens automatically and the fans receive a start permissive signal.	~~92 days~~ INSERT 1
SR 3.6.8.4	Verify each HSS train starts on an actual or simulated actuation signal after a delay of ≥ 8 minutes and ≤ 10 minutes.	~~92 days~~ INSERT 1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.9.1	Energize each HIS train power supply breaker and verify ≥ 34 ignitors are energized in each train.	~~92 days~~ INSERT 1
SR 3.6.9.2	Verify at least one hydrogen ignitor is OPERABLE in each containment region.	~~92 days~~ INSERT1
SR 3.6.9.3	Energize each hydrogen ignitor and verify temperature is ≥ 1700°F.	~~18 months~~ INSERT 1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.10.1	Operate each AVS train for ≥ 10 continuous hours with heaters operating.	~~31 days~~ INSERT 1
SR 3.6.10.2	Perform required AVS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.6.10.3	Verify each AVS train actuates on an actual or simulated actuation signal.	~~18 months~~ INSERT 1
SR 3.6.10.4	Verify each AVS filter cooling bypass valve can be opened.	~~18 months~~ INSERT 1
SR 3.6.10.5	Verify each AVS train flow rate is ≥ 8100 cfm and ≤ 9900 cfm.	~~18 months~~ INSERT 1
SR 3.6.10.6	Verify each AVS train produces a pressure equal to or more negative than -0.88 inch water gauge when corrected to elevation 564 feet.	~~18 months~~ INSERT 1

3.6 CONTAINMENT SYSTEMS

3.6.11 Air Return System (ARS)

LCO 3.6.11 Two ARS trains shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One ARS train inoperable.	A.1 Restore ARS train to OPERABLE status.	72 hours
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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.11.1	Verify each ARS fan starts on an actual or simulated actuation signal, after a delay of ≥ 8.0 minutes and ≤ 10.0 minutes, and operates for ≥ 15 minutes.	~~92 days~~ INSERT 1

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.11.2	Verify, with the ARS air return fan damper closed and with the bypass dampers open, each ARS fan motor current is ≤ 59.0 amps when the fan speed is ≥ 1174 rpm and ≤ 1200 rpm.	~~92 days~~ Insert 1
SR 3.6.11.3	Verify, with the ARS fan not operating, each ARS motor operated damper opens automatically on an actual or simulated actuation signal after a delay of ≥ 9 seconds and ≤ 11 seconds.	~~92 days~~ Insert 1
SR 3.6.11.4	Verify the check damper is open with the ARS fan operating.	~~92 days~~ INSERT 1
SR 3.6.11.5	Verify the check damper is closed with the ARS fan not operating.	~~92 days~~ INSERT 1
SR 3.6.11.6	Verify that each ARS fan is de-energized or is prevented from starting upon receipt of a terminate signal from the Containment Pressure Control System (CPCS) and is allowed to start upon receipt of a start permissive from the CPCS.	~~18 months~~ INSERT 1
SR 3.6.11.7	Verify that each ARS fan motor-operated damper is prevented from opening in the absence of a start permissive from the Containment Pressure Control System (CPCS) and is allowed to open upon receipt of a start permissive from the CPCS.	~~18 months~~ INSERT1

3.6 CONTAINMENT SYSTEMS

3.6.12 Ice Bed

LCO 3.6.12 The ice bed shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Ice bed inoperable.	A.1 Restore ice bed to OPERABLE status.	48 hours
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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.6.12.1 Verify maximum ice bed temperature is ≤ 27°F.	~~12 hours~~ Insert 1

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.6.12.2	-----NOTE----- The chemical analysis may be performed on either the liquid solution or on the resulting ice. ----- Verify, by chemical analysis, that ice added to the ice condenser meets the boron concentration and pH requirements of SR 3.6.12.7.	Each ice addition
SR 3.6.12.3	Verify, by visual inspection, accumulation of ice on structural members comprising flow channels through the ice bed is ≤ 15 percent blockage of the total flow area for each safety analysis section.	~~18 months~~ INSERT 1
SR 3.6.12.4	Verify total mass of stored ice is ≥ 2,132,000 lbs by calculating the mass of stored ice, at a 95 percent confidence, in each of three Radial Zones as defined below, by selecting a random sample of ≥ 30 ice baskets in each Radial Zone, and Verify: 1. Zone A (radial rows 8, 9), has a total mass of ≥ 324,000 lbs 2. Zone B (radial rows 4, 5, 6, 7), has a total mass of ≥ 1,033,100 lbs 3. Zone C (radial rows 1, 2, 3), has a total mass of ≥ 774,900 lbs	~~18 months~~ INSERT 1
SR 3.6.12.5	Verify that the ice mass of each basket sampled in SR 3.6.12.4 is ≥ 600 lbs.	~~18 months~~ INSERT 1

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.12.6	Visually inspect, for detrimental structural wear, cracks, corrosion, or other damage, two ice baskets from each group of bays as defined below: a. Group 1 – bays 1 through 8; b. Group 2 – bays 9 through 16; and c. Group 3 – bays 17 through 24.	~~40 months~~ Insert 1
SR 3.6.12.7	------------------------------ NOTE ------------------------------ The requirements of this SR are satisfied if the boron concentration and pH values obtained from averaging the individual sample results are within the limits specified below. --- Verify, by chemical analysis of the stored ice in at least one randomly selected ice basket from each ice condenser bay, that ice bed: a. Boron concentration is ≥ 1800 ppm and ≤ 2330 ppm; and b. pH is ≥ 9.0 and ≤ 9.5.	~~54 months~~ Insert 1

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition B not met.	C.1 Restore ice condenser door to OPERABLE status and closed positions.	48 hours
D. Required Action and associated Completion Time of Condition A or C not met.	D.1 Be in MODE 3. AND D.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.13.1	Verify all inlet doors indicate closed by the Inlet Door Position Monitoring System.	~~12 hours~~ INSERT 1
SR 3.6.13.2	Verify, by visual inspection, each intermediate deck door is closed and not impaired by ice, frost, or debris.	~~7 days~~ INSERT 1
SR 3.6.13.3	Verify, by visual inspection, each top deck door: a. Is in place; and b. Has no condensation, frost, or ice formed on the door that would restrict its opening.	~~92 days~~ INSERT 1

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.13.4	Verify, by visual inspection, each inlet door is not impaired by ice, frost, or debris.	~~18 months~~ Insert 1
SR 3.6.13.5	Verify torque required to cause each inlet door to begin to open is $\leq$ 675 in-lb.	~~18 months~~ Insert 1
SR 3.6.13.6	Perform a torque test on inlet doors.	~~18 months~~ INSERT 1
SR 3.6.13.7	Verify for each intermediate deck door: a. No visual evidence of structural deterioration; b. Free movement of the vent assemblies; and c. Free movement of the door.	~~18 months~~

ACTIONS (continued)

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.6.14.1 Verify, by visual inspection, all personnel access doors and equipment hatches between upper and lower containment compartments are closed.	Prior to entering MODE 4 from MODE 5
SR 3.6.14.2 Verify, by visual inspection, that the seals and sealing surfaces of each personnel access door and equipment hatch have: a. No detrimental misalignments; b. No cracks or defects in the sealing surfaces; and c. No apparent deterioration of the seal material.	Prior to final closure after each opening AND ---------NOTE------- Only required for seals made of resilient materials ------------------------ ~~10 years~~ Insert 1
SR 3.6.14.3 Verify, by visual inspection, each personnel access door or equipment hatch that has been opened for personnel transit entry is closed.	After each opening

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.14.4	Remove two divider barrier seal test coupons and verify both test coupons' tensile strength is ≥ 39.7 psi.	~~18 months~~ INsert 1
SR 3.6.14.5	Visually inspect ≥ 95% of the divider barrier seal length, and verify: a. Seal and seal mounting bolts are properly installed; and b. Seal material shows no evidence of deterioration due to holes, ruptures, chemical attack, abrasion, radiation damage, or changes in physical appearance.	~~18 months~~ INsert 1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE REQUIREMENTS	FREQUENCY
SR 3.6.15.1 Verify, by visual inspection, that: a. Each refueling canal drain valve is locked open; and b. Each refueling canal drain is not obstructed by debris.	Prior to entering MODE 4 from MODE 5 after each partial or complete fill of the canal
SR 3.6.15.2 Verify, by visual inspection that no debris is present in the upper compartment or refueling canal that could obstruct the refueling canal drain.	~~92 days~~ INSERT 1
SR 3.6.15.3 Verify for each ice condenser floor drain that the: a. Valve opening is not impaired by ice, frost, or debris; b. Valve seat shows no evidence of damage; c. Valve opening force is $\leq$ 66 lb; and d. Drain line from the ice condenser floor to the lower compartment is unrestricted.	~~18 months~~ INSERT 1

3.6 CONTAINMENT SYSTEMS

3.6.16 Reactor Building

LCO 3.6.16 The reactor building shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Reactor building inoperable.	A.1 Restore reactor building to OPERABLE status.	24 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	AND	
	B.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.6.16.1 Verify the door in each access opening is closed, except when the access opening is being used for normal transit entry and exit.	~~31 days~~ INSERT 1

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.16.2	Verify that during the annulus vacuum decay test, the vacuum decay time is ≥ 87 seconds.	~~18 months~~ Insert 1
SR 3.6.16.3	Verify reactor building structural integrity by performing a visual inspection of the exposed interior and exterior surfaces of the reactor building.	~~3 times every 10 years, coinciding with containment visual examinations required by SR 3.6.1.1~~ INSERT 1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.4.1	Verify one of the nitrogen bottles on each SG PORV is pressurized $\geq$ 2100 psig.	~~24 hours~~ Insert 1
SR 3.7.4.2	Verify one complete cycle of each SG PORV.	~~18 months~~ Insert 1
SR 3.7.4.3	Verify one complete cycle of each SG PORV block valve.	~~18 months~~ Insert 1

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.7.5.1	------------------------------NOTE------------------------------ Not applicable to automatic valves when THERMAL POWER is ≤ 10% RTP. -- Verify each AFW manual, power operated, and automatic valve in each water flow path, and in both steam supply flow paths to the steam turbine driven pump, that is not locked, sealed, or otherwise secured in position, is in the correct position.	~~31 days~~ Insert 1
SR 3.7.5.2	------------------------------NOTE------------------------------ Not required to be performed for the turbine driven AFW pump until 24 hours after ≥ 600 psig in the steam generator. -- Verify the developed head of each AFW pump at the flow test point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program
SR 3.7.5.3	------------------------------NOTE------------------------------ Not applicable in MODE 4 when steam generator is relied upon for heat removal. -- Verify each AFW automatic valve that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	~~18 months~~ Insert 1

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.5.4	-----NOTES----- 1. Not required to be performed for the turbine driven AFW pump until 24 hours after ≥ 600 psig in the steam generator. 2. Not applicable in MODE 4 when steam generator is relied upon for heat removal. ----- Verify each AFW pump starts automatically on an actual or simulated actuation signal.	~~18 months~~ INSERT 1
SR 3.7.5.5	Verify proper alignment of the required AFW flow paths by verifying flow from the condensate storage system to each steam generator.	Prior to entering MODE 2, whenever unit has been in MODE 5 or 6 for > 30 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.6.1	Verify the CSS inventory is ≥ 225,000 gal.	~~12 hours~~ Insert 1

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.7.7.1	---NOTE--- Isolation of CCW flow to individual components does not render the CCW System inoperable. --- Verify each CCW manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.	~~31 days~~ Insert 1
SR 3.7.7.2	Verify each CCW automatic valve in the flow path servicing safety related equipment that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	~~18 months~~ Insert 1
SR 3.7.7.3	Verify each CCW pump starts automatically on an actual or simulated actuation signal.	~~18 months~~ Insert 1

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.7.8.1	-----NOTE----- Isolation of NSWS flow to individual components does not render the NSWS inoperable. ----- Verify each NSWS manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.	~~31 days~~ Insert 1
SR 3.7.8.2	-----NOTE----- Not required to be met for valves that are maintained in position to support NSWS single supply header operation. ----- Verify each NSWS automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	~~18 months~~ Insert 1
SR 3.7.8.3	Verify each NSWS pump starts automatically on an actual or simulated actuation signal.	~~18 months~~ Insert 1

3.7 PLANT SYSTEMS

3.7.9 Standby Nuclear Service Water Pond (SNSWP)

LCO 3.7.9 The SNSWP shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SNSWP inoperable.	A.1 Be in MODE 3.	6 hours
	AND	
	A.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.9.1	Verify water level of SNSWP is ≥ 571 ft mean sea level.	~~24 hours~~ Insert 1
SR 3.7.9.2	---NOTE--- Only required to be performed during the months of July, August, and September.	
	Verify average water temperature of SNSWP is ≤ 95°F at an elevation of 568 ft. in SNSWP.	~~24 hours~~ Insert 1
SR 3.7.9.3	Verify, by visual inspection, no abnormal degradation, erosion, or excessive seepage of the SNSWP dam.	~~12 months~~ INSERT 1

REQUIRED ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
G. One or more CRAVS train(s) heater inoperable.	G.1 Restore CRAVS train(s) heater to OPERABLE status. OR G.2 Initiate action in accordance with Specification 5.6.6.	7 days 7 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.10.1	Operate each CRAVS train for $\geq$ 10 continuous hours with the heaters operating.	~~31 days~~ INSERT 1
SR 3.7.10.2	Perform required CRAVS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with VFTP
SR 3.7.10.3	Verify each CRAVS train actuates on an actual or simulated actuation signal.	~~18 months~~ INSERT 1
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

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Two CRACWS trains inoperable in MODE 5 or 6, or during movement of recently irradiated fuel assemblies.	D.1 Suspend movement of recently irradiated fuel assemblies.	Immediately
E. Two CRACWS trains inoperable in MODE 1, 2, 3, or 4.	E.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.11.1 Verify the control room temperature is ≤ 90°F.	~~12 hours~~ Insert 1

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.7.12.1	Operate each ABFVES train for ≥ 10 continuous hours with the heaters operating.	~~31 days~~ INSERT 1
SR 3.7.12.2	Perform required ABFVES filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.7.12.3	Verify each ABFVES train actuates on an actual or simulated actuation signal.	~~18 months~~ INSERT 1
SR 3.7.12.4	Verify one ABFVES train can maintain the ECCS pump rooms at negative pressure relative to adjacent areas.	~~18 months on a STAGGERED TEST BASIS~~ INSERT 1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.13.1	Verify required FHVES train in operation.	~~12 hours~~ INSERT 1
SR 3.7.13.2	Operate required FHVES train for $\geq$ 10 continuous hours with the heaters operating.	~~31 days~~ INSERT 1
SR 3.7.13.3	Perform required FHVES filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.7.13.4	Verify one FHVES train can maintain a pressure $\leq$ -0.25 inches water gauge with respect to atmospheric pressure during operation at a flow rate $\leq$ 36,443 cfm.	~~18 months on a STAGGERED TEST BASIS~~ INSERT 1
SR 3.7.13.5	Verify each FHVES filter bypass damper can be closed.	~~18 months~~ INSERT 1

3.7 PLANT SYSTEMS

3.7.14 Spent Fuel Pool Water Level

LCO 3.7.14 The spent fuel pool water level shall be ≥ 23 ft over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: During movement of irradiated fuel assemblies in the spent fuel pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spent fuel pool water level not within limit.	A.1 -------------NOTE------------- LCO 3.0.3 is not applicable. ---------------------------------- Suspend movement of irradiated fuel assemblies in the spent fuel pool.	 Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.14.1 Verify the spent fuel pool water level is ≥ 23 ft above the top of the irradiated fuel assemblies seated in the storage racks.	~~7 days~~ INSERT 1

3.7 PLANT SYSTEMS

3.7.15 Spent Fuel Pool Boron Concentration

LCO 3.7.15 The spent fuel pool boron concentration shall be within the limit specified in the COLR.

APPLICABILITY: When fuel assemblies are stored in the spent fuel pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spent fuel pool boron concentration not within limit.	------------------NOTE------------------ LCO 3.0.3 is not applicable. ---	
	A.1 Suspend movement of fuel assemblies in the spent fuel pool.	Immediately
	<u>AND</u>	
	A.2 Initiate action to restore spent fuel pool boron concentration to within limit.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.15.1 Verify the spent fuel pool boron concentration is within limit.	~~7 days~~ INSERT 1

3.7 PLANT SYSTEMS

3.7.17 Secondary Specific Activity

LCO 3.7.17 The specific activity of the secondary coolant shall be ≤ 0.10 μCi/gm DOSE EQUIVALENT I-131.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Specific activity not within limit.	A.1 Be in MODE 3. AND A.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.17.1 Verify the specific activity of the secondary coolant is ≤ 0.10 μCi/gm DOSE EQUIVALENT I-131.	~~31 days~~ INSERT 1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.1.1	Verify correct breaker alignment and indicated power availability for each offsite circuit.	~~7 days~~ INSERT 1
SR 3.8.1.2	-------------------------------NOTES------------------------------ 1. Performance of SR 3.8.1.7 satisfies this SR. 2. All DG starts may be preceded by an engine prelube period and followed by a warmup period prior to loading. 3. A modified DG start involving idling and gradual acceleration to synchronous speed may be used for this SR as recommended by the manufacturer. When modified start procedures are not used, the time, voltage, and frequency tolerances of SR 3.8.1.7 must be met. --- Verify each DG starts from standby conditions and achieves steady state voltage $\geq$ 3740 V and $\leq$ 4580 V, and frequency $\geq$ 58.8 Hz and $\leq$ 61.2 Hz.	~~31 days~~ INSERT 1

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.8.1.3	-----NOTES----- 1. DG loadings may include gradual loading as recommended by the manufacturer. 2. Momentary transients outside the load range do not invalidate this test. 3. This Surveillance shall be conducted on only one DG at a time. 4. This SR shall be preceded by and immediately follow without shutdown a successful performance of SR 3.8.1.2 or SR 3.8.1.7. ----- Verify each DG is synchronized and loaded and operates for ≥ 60 minutes at a load ≥ 5600 kW and ≤ 5750 kW.	~~31 days~~ INSERT 1
SR 3.8.1.4	Verify each day tank contains ≥ 470 gal of fuel oil.	~~31 days~~ INSERT 1
SR 3.8.1.5	Check for and remove accumulated water from each day tank.	~~31 days~~ INSERT 1
SR 3.8.1.6	Verify the fuel oil transfer system operates to transfer fuel oil from storage system to the day tank.	~~31 days~~ INSERT 1

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.7	-----NOTE----- All DG starts may be preceded by an engine prelube period. ----- Verify each DG starts from standby condition and achieves in ≤ 11 seconds voltage of ≥ 3740 V and frequency of ≥ 57 Hz and maintains steady-state voltage ≥ 3740 V and ≤ 4580 V, and frequency ≥ 58.8 Hz and ≤ 61.2 Hz.	~~184 days~~ INSERT 1
SR 3.8.1.8	Verify automatic and manual transfer of AC power sources from the normal offsite circuit to each alternate offsite circuit.	~~18 months~~ INSERT 1

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.9	-----NOTE----- If performed with the DG synchronized with offsite power, it shall be performed at a power factor ≤ 0.9. ---------- Verify each DG rejects a load greater than or equal to its associated single largest post-accident load, and: a. Following load rejection, the frequency is ≤ 63 Hz; b. Within 3 seconds following load rejection, the voltage is ≥ 3740 V and ≤ 4580 V; and c. Within 3 seconds following load rejection, the frequency is ≥ 58.8 Hz and ≤ 61.2 Hz.	~~18 months~~ Insert 1
SR 3.8.1.10	Verify each DG does not trip and generator speed is maintained ≤ 500 rpm during and following a load rejection of ≥ 5600 kW and ≤ 5750 kW.	~~18 months~~ Insert 1

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.11	-----NOTES----- 1. All DG starts may be preceded by an engine prelube period. 2. This Surveillance shall not be performed in MODE 1, 2, 3, or 4. ----- Verify on an actual or simulated loss of offsite power signal: a. De-energization of emergency buses; b. Load shedding from emergency buses; c. DG auto-starts from standby condition and: 1. energizes the emergency bus in ≤ 11 seconds, 2. energizes auto-connected shutdown loads through automatic load sequencer, 3. maintains steady state voltage ≥ 3740 V and ≤ 4580 V, 4. maintains steady state frequency ≥ 58.8 Hz and ≤ 61.2 Hz, and 5. supplies auto-connected shutdown loads for ≥ 5 minutes.	~~18 months~~ Insert 1

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.12	-----NOTE----- All DG starts may be preceded by prelube period. ---------- Verify on an actual or simulated Engineered Safety Feature (ESF) actuation signal each DG auto-starts from standby condition and: a. In $\leq$ 11 seconds after auto-start and during tests, achieves voltage $\geq$ 3740 V and $\leq$ 4580 V; b. In $\leq$ 11 seconds after auto-start and during tests, achieves frequency $\geq$ 58.8 Hz and $\leq$ 61.2 Hz; c. Operates for $\geq$ 5 minutes; and d. The emergency bus remains energized from the offsite power system.	~~18 months~~ INSERT 1

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.13	Verify each DG's non-emergency automatic trips are bypassed on actual or simulated loss of voltage signal on the emergency bus concurrent with an actual or simulated ESF actuation signal.	~~18 months~~ Insert 1
SR 3.8.1.14	-----NOTE----- Momentary transients outside the load and power factor ranges do not invalidate this test. ----- Verify each DG operating at a power factor ≤ 0.9 operates for ≥ 24 hours loaded ≥ 5600 kW and ≤ 5750 kW.	~~18 months~~ Insert 1

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.15	---NOTES--- 1. This Surveillance shall be performed within 5 minutes of shutting down the DG after the DG has operated ≥ 1 hour loaded ≥ 5600 kW and ≤ 5750 kW or until operating temperature is stabilized. Momentary transients outside of load range do not invalidate this test. 2. All DG starts may be preceded by an engine prelube period. --- Verify each DG starts and achieves, in ≤ 11 seconds, voltage ≥ 3740 V, and frequency ≥ 57 Hz and maintains steady state voltage ≥ 3740 V and ≤ 4580 V and frequency ≥ 58.8 Hz and ≤ 61.2 Hz.	~~18 months~~ Insert 1
SR 3.8.1.16	---NOTE--- This Surveillance shall not be performed in MODE 1, 2, 3, or 4. --- Verify each DG: a. Synchronizes with offsite power source while loaded with emergency loads upon a simulated restoration of offsite power; b. Transfers loads to offsite power source; and c. Returns to standby operation.	~~18 months~~ Insert 1

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.17	-----NOTE----- This Surveillance shall not be performed in MODE 1, 2, 3, or 4. ---------- Verify, with a DG operating in test mode and connected to its bus, an actual or simulated ESF actuation signal overrides the test mode by: a. Returning DG to standby operation; and b. Automatically energizing the emergency load from offsite power.	~~18 months~~ INSERT 1
SR 3.8.1.18	Verify interval between each sequenced load block is within the design interval for each automatic load sequencer.	~~18 months~~ INSERT 1

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.19	------------------------------NOTES------------------------------ 1. All DG starts may be preceded by an engine prelube period. 2. This Surveillance shall not be performed in MODE 1, 2, 3, or 4. --- Verify on an actual or simulated loss of offsite power signal in conjunction with an actual or simulated ESF actuation signal: a. De-energization of emergency buses; b. Load shedding from emergency buses; and c. DG auto-starts from standby condition and: 1. energizes the emergency bus in $\leq$ 11 seconds, 2. energizes auto-connected emergency loads through load sequencer, 3. achieves steady state voltage $\geq$ 3740 V and $\leq$ 4580 V, 4. achieves steady state frequency $\geq$ 58.8 Hz and $\leq$ 61.2 Hz, and 5. supplies auto-connected emergency loads for $\geq$ 5 minutes.	~~18 months~~ INSERT 1

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.20	-----NOTE----- All DG starts may be preceded by an engine prelube period. ---------- Verify when started simultaneously from standby condition, each DG achieves, in ≤ 11 seconds, voltage of ≥ 3740 V and frequency of ≥ 57 Hz and maintains steady state voltage ≥ 3740 V and ≤ 4580 V, and frequency ≥ 58.8 Hz and ≤ 61.2 Hz.	~~10 years~~ INSERT 1

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One or more DGs with new fuel oil properties not within limits.	D.1 Restore stored fuel oil properties to within limits.	30 days
E. One or more DGs with starting air receiver pressure < 210 psig and ≥ 150 psig.	E.1 Restore starting air receiver pressure to ≥ 210 psig.	48 hours
F. Required Action and associated Completion Time not met. OR One or more DGs diesel fuel oil, lube oil, or starting air subsystem not within limits for reasons other than Condition A, B, C, D, or E.	F.1 Declare associated DG inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.3.1	Verify the fuel oil storage system contains ≥ 77,100 gal of fuel for each DG.	~~31 days~~ Insert 1

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.8.3.2	Verify lubricating oil inventory is ≥ 400 gal.	~~31 days~~ INSERT #1
SR 3.8.3.3	Verify fuel oil properties of new and stored fuel oil are tested in accordance with, and maintained within the limits of, the Diesel Fuel Oil Testing Program.	In accordance with the Diesel Fuel Oil Testing Program
SR 3.8.3.4	Verify each DG air start receiver pressure is ≥ 210 psig.	~~31 days~~ INSERT #1
SR 3.8.3.5	Check for and remove accumulated water from each fuel oil storage tank.	~~31 days~~ INSERT #1

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. A and/or D channel of DC electrical power subsystem inoperable. AND Associated train of DG DC electrical power subsystem inoperable.	D.1 Enter applicable Condition(s) and Required Action(s) of LCO 3.8.9, "Distribution Systems-Operating", for the associated train of DC electrical power distribution subsystem made inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.4.1	Verify DC channel and DG battery terminal voltage is ≥ 125 V on float charge.	~~7 days~~ INSERT 1
SR 3.8.4.2	Not used.	
SR 3.8.4.3	Verify no visible corrosion at the DC channel and DG battery terminals and connectors. OR Verify battery connection resistance of these items is ≤ 1.5 E-4 ohm.	~~92 days~~ INSERT 1

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.4.4	Verify DC channel and DG battery cells, cell plates, and racks show no visual indication of physical damage or abnormal deterioration that could degrade battery performance.	~~18 months~~
SR 3.8.4.5	Remove visible terminal corrosion, verify DC channel and DG battery cell to cell and terminal connections are clean and tight, and are coated with anti-corrosion material.	~~18 months~~
SR 3.8.4.6	Verify DC channel and DG battery connection resistance is ≤ 1.5 E-4 ohm.	~~18 months~~
SR 3.8.4.7	Verify each DC channel battery charger supplies ≥ 200 amps and the DG battery charger supplies ≥ 75 amps with each charger at ≥ 125 V for ≥ 8 hours.	~~18 months~~ INSERT 1
SR 3.8.4.8	-------------------------------NOTES------------------------------ 1. The modified performance discharge test in SR 3.8.4.9 may be performed in lieu of the service test in SR 3.8.4.8. 2. This Surveillance shall not be performed for the DG batteries in MODE 1, 2, 3, or 4. --- Verify DC channel and DG battery capacity is adequate to supply, and maintain in OPERABLE status, the required emergency loads for the design duty cycle when subjected to a battery service test.	~~18 months~~

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.4.9	------------------------------NOTE------------------------------ This Surveillance shall not be performed for the DG batteries in MODE 1, 2, 3, or 4. -- Verify DC channel and DG battery capacity is ≥ 80% of the manufacturer's rating when subjected to a performance discharge test or a modified performance discharge test.	~~60 months~~ INSERT 2 <u>AND</u> 18 months when battery shows degradation or has reached 85% of expected life with capacity < 100% of manufacturer's rating <u>AND</u> --------NOTE-------- Not applicable to DG batteries ---------------------- 24 months when battery has reached 85% of the expected life with capacity ≥ 100% of manufacturer's rating

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.6.1	Verify battery cell parameters of the channels of DC and DG batteries meet Table 3.8.6-1 Category A limits.	~~7 days~~
SR 3.8.6.2	Not used.	
SR 3.8.6.3	Verify battery cell parameters of the channels of DC and DG batteries meet Table 3.8.6-1 Category B limits.	~~92 days~~ INSERT 1 AND Once within 7 days after a battery discharge < 110 V AND Once within 7 days after a battery overcharge > 150 V
SR 3.8.6.4	Verify average electrolyte temperature for the channels of DC and DG batteries of representative cells is $\geq$ 60°F.	~~92 days~~ INSERT 1

NO CHANGES THIS PAGE.
FOR INFORMATION ONLY

Table 3.8.6-1 (page 1 of 1)
Battery Cell Parameters Requirements

PARAMETER	CATEGORY A: LIMITS FOR EACH DESIGNATED PILOT CELL	CATEGORY B: LIMITS FOR EACH CONNECTED CELL	CATEGORY C: ALLOWABLE LIMITS FOR EACH CONNECTED CELL
Electrolyte Level	> Minimum level indication mark, and $\leq$ ¼ inch above maximum level indication mark[(a)]	> Minimum level indication mark, and $\leq$ ¼ inch above maximum level indication mark[(a)]	Above top of plates, and not overflowing
Float Voltage	$\geq$ 2.13 V	$\geq$ 2.13 V	> 2.07 V
Specific Gravity[(b)(c)]	$\geq$ 1.200	$\geq$ 1.195 AND Average of all connected cells $\geq$ 1.205	Not more than 0.020 below average of all connected cells or $\geq$ 1.195 AND Average of all connected cells $\geq$ 1.195

(a) It is acceptable for the electrolyte level to temporarily increase above the specified maximum during equalizing charges provided it is not overflowing.

(b) Corrected for electrolyte temperature and level. Level correction is not required, however, when battery charging is < 2 amps when on float charge.

(c) A battery charging current of < 2 amps when on float charge is acceptable for meeting specific gravity limits following a battery recharge, for a maximum of 7 days. When charging current is used to satisfy specific gravity requirements, specific gravity of each connected cell shall be measured prior to expiration of the 7 day allowance.

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.8.7.1	Verify correct inverter voltage and alignment to required AC vital buses.	~~7 days~~ INSERT 1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.3 Suspend operations involving positive reactivity additions that could result in loss of required SDM or required boron concentration. AND A.2.4 Initiate action to restore required inverters to OPERABLE status.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.8.1	Verify correct voltage and alignment to required AC vital bus.	~~7 days~~ INSERT 1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.9.1	Verify correct breaker alignments and voltage to required AC, DC channel, DC train, and AC vital bus electrical power distribution subsystems.	~~7 days~~ INSERT 1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.4 Initiate actions to restore required AC, channels of DC, DC trains, and AC vital bus electrical power distribution subsystems to OPERABLE status. AND A.2.5 Declare associated required residual heat removal subsystem(s) inoperable and not in operation. AND A.2.6 Declare affected Low Temperature Overpressure Protection feature(s) inoperable.	Immediately Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.8.10.1 Verify correct breaker alignments and voltage to required AC, DC channel, DC train, and AC vital bus electrical power distribution subsystems.	~~7 days~~ INSERT #1

3.9 REFUELING OPERATIONS

3.9.1 Boron Concentration

LCO 3.9.1 Boron concentrations of the Reactor Coolant System, the refueling canal, and the refueling cavity shall be maintained within the limit specified in the COLR.

-----NOTE-----
Only applicable to the refueling canal and refueling cavity when connected to the RCS.

APPLICABILITY: MODE 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Boron concentration not within limit.	A.1 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	A.2 Suspend positive reactivity additions.	Immediately
	<u>AND</u>	
	A.3 Initiate action to restore boron concentration to within limit.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.9.1.1 Verify boron concentration is within the limit specified in COLR.	~~72 hours~~ INSERT 1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.2.1	Perform CHANNEL CHECK.	~~12 hours~~ INSERT 1
SR 3.9.2.2	-------------------------------NOTE------------------------------- Neutron detectors are excluded from CHANNEL CALIBRATION. --- Perform CHANNEL CALIBRATION.	~~18 months~~ INSERT 1

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One or more CPES train(s) heater inoperable.	B.1 Restore CPES train(s) heater to OPERABLE status.	7 days
	OR	
	B.2 Initiate action in accordance with Specification 5.6.6.	7 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.3.1	Verify each required containment penetration is in the required status.	~~7 days~~ INSERT 1
SR 3.9.3.2	Operate each CPES for $\geq$ 10 continuous hours with the heaters operating.	~~31 days~~
SR 3.9.3.3	Perform required CPES filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.4 Close all containment penetrations providing direct access from containment atmosphere to outside atmosphere.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.9.4.1 Verify one RHR loop is in operation and circulating reactor coolant at a flow rate of ≥ 1000 gpm and RCS temperature is ≤ 140°F.	~~12 hours~~ INSERT 1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.2 Initiate action to restore one RHR loop to operation.	Immediately
	<u>AND</u>	
	B.3 Close all containment penetrations providing direct access from containment atmosphere to outside atmosphere.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.5.1	Verify one RHR loop is in operation and circulating reactor coolant at a flow rate of ≥ 1000 gpm and RCS temperature is ≤ 140°F.	~~12 hours~~ INSERT 1
SR 3.9.5.2	Verify correct breaker alignment and indicated power available to the required RHR pump that is not in operation.	~~7 days~~ INSERT 1

3.9 REFUELING OPERATIONS

3.9.6 Refueling Cavity Water Level

LCO 3.9.6 Refueling cavity water level shall be maintained $\geq$ 23 ft above the top of reactor vessel flange.

APPLICABILITY: During CORE ALTERATIONS, except during latching and unlatching of control rod drive shafts,
During movement of irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Refueling cavity water level not within limit.	A.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	A.2 Suspend movement of irradiated fuel assemblies within containment.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.9.6.1 Verify refueling cavity water level is $\geq$ 23 ft above the top of reactor vessel flange.	~~24 hours~~ INSERT 1

3.9 REFUELING OPERATIONS

3.9.7 Unborated Water Source Isolation Valves

LCO 3.9.7 Each valve used to isolate unborated water sources shall be secured in the closed position.

APPLICABILITY: MODE 6.

ACTIONS

---NOTE---
Separate Condition entry is allowed for each unborated water source isolation valve.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. -----------NOTE------------ Required Action A.3 must be completed whenever Condition A is entered. -------------------------------- One or more valves not secured in closed position.	A.1 Suspend CORE ALTERATIONS. AND A.2 Initiate actions to secure valve in closed position. AND A.3 Perform SR 3.9.1.1.	Immediately Immediately 4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.9.7.1 Verify each valve that isolates unborated water sources is secured in the closed position.	~~31 days~~ INSERT 1

5.5 Programs and Manuals (continued)

5.5.16 Control Room Envelope Habitability Program

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Area Ventilation System (CRAVS), CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem total effective dose equivalent (TEDE) for the duration of the accident. The program shall include the following elements:

a. The definition of the CRE and the CRE boundary.

b. Requirements for maintaining the CRE boundary in its design condition including configuration control and preventive maintenance.

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 of operation by one train of the CRAVS, operating at a makeup flow rate of $\leq$ 4000 cfm, at a Frequency of 18 months on a STAGGERED TEST BASIS. The results shall be trended and used as part of the 18 month assessment of the CRE boundary.

e. The quantitative limits on unfiltered air inleakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by the testing described in paragraph c. The unfiltered air inleakage limit for radiological challenges is the inleakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air inleakage limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis.

f. The provisions of SR 3.0.2 are applicable to the Frequencies for assessing CRE habitability, determining CRE unfiltered inleakage, and measuring CRE pressure and assessing the CRE boundary as required by paragraphs c and d, respectively.

INSERT 3

ATTACHMENT 4

PROPOSED TECHNICAL SPECIFICATION BASES CHANGES

INSERTS

INSERT 2

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

REVIEWER'S NOTE: Text deleted and replaced by Insert 2 will be relocated to the Surveillance Frequency Control Program (SFCP) document(s) per TSTF-425. Thus, there are instances in these mark-ups where deleted text is edited for future use in the SFCP. The words "For SFCP addition only" will accompany inserted text that will be relocated to the SFCP. This inserted text will be cross-hatched to indicate it is not to be inserted on the Bases page.

SURVEILLANCE REQUIREMENTS (continued)

In MODE 2 with $k_{eff} < 1.0$ and MODES 3, 4, and 5, SDM is verified by performing a reactivity balance calculation, considering the listed reactivity effects:

a. RCS boron concentration;

b. Control bank position;

c. RCS average temperature;

d. Fuel burnup based on gross thermal energy generation;

e. Xenon concentration;

f. Samarium concentration; and

g. Isothermal temperature coefficient (ITC).

Using the ITC accounts for Doppler reactivity in this calculation because the reactor is subcritical, 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 the low probability of an accident occurring without the required SDM. This allows time for the operator to collect the required data, which includes performing a boron concentration analysis, and complete the calculation.~~ INSERT 2

REFERENCES

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

2. UFSAR, Section 15.1.5.

3. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).

4. UFSAR, Section 15.4.6.

5. 10 CFR 50.67.

BASES

ACTIONS (continued)

B.1

If the core reactivity cannot be restored to within the 1% Δk/k limit, 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. If the SDM for MODE 3 is not met, then the boration required by SR 3.1.1.1 would occur. 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 3.1.2.1

Core reactivity is verified by periodic comparisons of measured and predicted RCS boron concentrations. The comparison is made, considering that other core conditions are fixed or stable, including control rod position, moderator temperature, fuel temperature, fuel depletion, xenon concentration, and samarium concentration. The Surveillance is performed prior to entering MODE 1 as an initial check on core conditions and design calculations at BOC. The SR is modified by a Note. The Note indicates that the normalization of predicted core reactivity to the measured value must take place within the first 60 effective full power days (EFPD) after each fuel loading. This allows sufficient time for core conditions to reach steady state, but prevents operation for a large fraction of the fuel cycle without establishing a benchmark for the design calculations. ~~The required subsequent Frequency of 31 EFPD is acceptable, based on the slow rate of core changes due to fuel depletion and the presence of other indicators (QPTR, AFD, etc.) for prompt indication of an anomaly.~~

Insert 2

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26, GDC 28, and GDC 29.
2. UFSAR, Chapter 15.
3. 10 CFR 50.36, Technical Specification, (c)(2)(ii).

ACTIONS (continued)

D.1.1 and D.1.2

More than one control rod becoming misaligned from its group average position is not expected, and has the potential to reduce SDM. Therefore, SDM must be evaluated. One hour allows the operator adequate time to determine SDM. Restoration of the required SDM, if necessary, requires increasing the RCS boron concentration to provide negative reactivity, as described in the Bases or LCO 3.1.1. The required Completion Time of 1 hour for initiating boration is reasonable, based on the time required for potential xenon redistribution, the low probability of an accident occurring, and the steps required to complete the action. This allows the operator sufficient time to align the required valves and start the boric acid pumps. Boration will continue until the required SDM is restored.

D.2

If more than one rod is found to be misaligned or becomes misaligned because of bank movement, the unit conditions fall outside of the accident analysis assumptions. 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 at least 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 rod positions are within alignment limits at a Frequency of 12 hours provides a history that allows the operator to detect a rod that is beginning to deviate from its expected position. If the rod position deviation monitor is inoperable, a Frequency of 4 hours accomplishes the same goal. The specified Frequency takes into account other rod position information that is continuously available to the operator in the control room, so that during actual rod motion, deviations can immediately be detected.~~

INSERT 2

SR 3.1.4.2

Verifying each control rod is OPERABLE would require that each rod be tripped. However, in MODES 1 and 2, tripping each control rod would result in radial or axial power tilts, or oscillations. Exercising each

BASES

SURVEILLANCE REQUIREMENTS (continued)

individual control rod ~~every 92 days~~ provides increased confidence that all rods continue to be OPERABLE without exceeding the alignment limit, even if they are not regularly tripped. Moving each control rod by 10 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 rods.~~ Insert 2 Between required performances of SR 3.1.4.2 (determination of control rod OPERABILITY by movement), if a control rod(s) is discovered to be immovable, but remains trippable and aligned, the control rod(s) is considered to be OPERABLE. At any time, if a control rod(s) is immovable, a determination of the trippability (OPERABILITY) of the control rod(s) must be made, and appropriate action taken. This may be by verification of a control system failure, usually electrical in nature, or that the failure is associated with the control rod stepping mechanism. During performance of the Control Rod Movement periodic test, there have been some "Control Malfunctions" that prohibited a control rod bank or group from moving when selected, as evidenced by the demand counters and DRPI. In all cases, when the control malfunctions were corrected, the rods moved freely (no excessive friction or mechanical interference) and were trippable.

SR 3.1.4.3

Verification of rod drop times allows the operator to determine that the maximum rod drop time permitted is consistent with the assumed rod drop time used in the safety analysis. Since a removal of the reactor vessel head has the potential to change component alignments affecting rod drop times, measuring drop times prior to the next criticality following any such removal ensures that the reactor internals and rod drive mechanism will not interfere with rod motion or rod drop time, and that no degradation in these systems has occurred that would adversely affect control rod motion or drop time. This testing is performed with all RCPs operating and the average moderator temperature ≥ 551°F to simulate a reactor trip under actual conditions.

This Surveillance is performed during a plant outage, 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.

ACTIONS (continued)

B.1

If the shutdown banks cannot be restored to within their insertion limits within 2 hours, the unit must be brought to a MODE where the LCO is not applicable. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.5.1

Verification that the shutdown banks are within their insertion limits prior to an approach to criticality ensures that when the reactor is critical, or being taken critical, the shutdown banks will be available to shut down the reactor, and the required SDM will be maintained following a reactor trip. This SR and Frequency ensure that the shutdown banks are withdrawn before the control banks are withdrawn during a unit startup.

~~Since the shutdown banks are positioned manually by the control room operator, a verification of shutdown bank position at a Frequency of 12 hours, after the reactor is taken critical, is adequate to ensure that they are within their insertion limits. Also, the 12 hour Frequency takes into account other information available in the control room for the purpose of monitoring the status of shutdown rods.~~ Insert 2

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10, GDC 26, and GDC 28.
2. 10 CFR 50.46.
3. UFSAR, Section 15.4.
4. 10 CFR 50.36, Technical Specification, (c)(2)(ii).

ACTIONS (continued)

required MODE from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

This Surveillance is required to ensure that the reactor does not achieve criticality with the control banks below their insertion limits.

The estimated critical position (ECP) depends upon a number of factors, one of which is xenon concentration. If the ECP was calculated long before criticality, xenon concentration could change to make the ECP substantially in error. Conversely, determining the ECP immediately before criticality could be an unnecessary burden. There are a number of unit parameters requiring operator attention at that point. Verifying the ECP calculation within 4 hours prior to criticality avoids a large error from changes in xenon concentration, but allows the operator some flexibility to schedule the ECP calculation with other startup activities.

SR 3.1.6.2

With an OPERABLE bank insertion limit monitor, ~~verification of the control bank insertion limits at a Frequency of 12 hours is sufficient to ensure OPERABILITY of the bank insertion limit monitor and to detect control banks that may be approaching the insertion limits since normally, very little rod motion occurs in 12 hours.~~ If the insertion limit monitor becomes inoperable, verification of the control bank position at a Frequency of 4 hours is sufficient to detect control banks that may be approaching the insertion limits.

INSERT 2

SR 3.1.6.3

When control banks are maintained within their insertion limits as checked by SR 3.1.6.2 above, it is unlikely that their sequence and overlap will not be in accordance with requirements provided in the COLR. ~~A Frequency of 12 hours is consistent with the insertion limit check above in SR 3.1.6.2.~~

INSERT 2

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.8.2

Verification that the RCS lowest loop T_{avg} is ≥ 541°F will ensure that the unit is not operating in a condition that could invalidate the safety analyses. ~~Verification of the RCS temperature at a Frequency of 30 minutes during the performance of the PHYSICS TESTS will ensure that the initial conditions of the safety analyses are not violated.~~

INSERT 2

SR 3.1.8.3

Verification that THERMAL POWER is ≤ 5% RTP will ensure that the plant is not operating in the condition that could invalidate the safety analyses. ~~Verification of the THERMAL POWER at a Frequency of 1 hour during the performance of the PHYSICS TESTS will ensure that the initial conditions of the safety analyses are not violated.~~

INSERT 2

SR 3.1.8.4

The SDM is verified by performing a reactivity balance calculation, considering the following reactivity effects:

a. RCS boron concentration;

b. Control bank position;

c. RCS average temperature;

d. Fuel burnup based on gross thermal energy generation;

e. Xenon concentration;

f. Samarium concentration; and

g. Isothermal temperature coefficient (ITC).

Using the ITC accounts for Doppler reactivity in this calculation because the reactor is subcritical, 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.~~

INSERT 2

BASES

SURVEILLANCE REQUIREMENTS (continued)

verification. It only requires verification after a power level is achieved for extended operation that is 10% higher than that power at which F_Q was last measured.

SR 3.2.1.1

Verification that $F^M_Q(X,Y,Z)$ is within its specified steady state limits involves either increasing $F^M_Q(X,Y,Z)$ to allow for manufacturing tolerance, K(BU), and measurement uncertainties for the case where these factors are not included in the F_Q limit. For the case where these factors are included, a direct comparison of $F^M_Q(X,Y,Z)$ to the F_Q limit can be performed. Specifically, $F^M_Q(X,Y,Z)$ is the measured value of $F_Q(X,Y,Z)$ obtained from incore flux map results. Values for the manufacturing tolerance, K(BU), and measurement uncertainty are specified in the COLR.

The limit with which $F^M_Q(X,Y,Z)$ is compared varies inversely with power above 50% RTP and directly with functions called K(Z) and K(BU) provided in the COLR.

If THERMAL POWER has been increased by ≥ 10% RTP since the last determination of $F^M_Q(X,Y,Z)$, another evaluation of this factor is required 12 hours after achieving equilibrium conditions at this higher power level (to ensure that $F^M_Q(X,Y,Z)$ values have decreased sufficiently with power increase to stay within the LCO limits).

~~The Frequency of 31 EFPD is adequate to monitor the change of power distribution with core burnup because such changes are slow and well controlled when the plant is operated in accordance with the Technical Specifications (TS).~~

INSERT 2

SR 3.2.1.2 and 3.2.1.3

The nuclear design process includes calculations performed to determine that the core can be operated within the $F_Q(X,Y,Z)$ limits. Because flux maps are taken in steady state conditions, the variations in power distribution resulting from normal operational maneuvers are not present in the flux map data. These variations are, however, conservatively calculated by considering a wide range of unit maneuvers in normal operation. The maximum peaking factor increase over steady state values, is determined by a maneuvering analysis (Ref. 5).

SURVEILLANCE REQUIREMENTS (continued)

The limit with which $F^M_Q(X,Y,Z)$ is compared varies and is provided in the COLR. No additional uncertainties are applied to the measured $F_Q(X,Y,Z)$ because the limits already include uncertainties.

$F^L_Q(X,Y,Z)^{OP}$ and $F^L_Q(X,Y,Z)^{RPS}$ limits are not applicable for the following axial core regions, measured in percent of core height:

a. Lower core region, from 0 to 15% inclusive; and

b. Upper core region, from 85 to 100% inclusive.

The top and bottom 15% of the core are excluded from the evaluation because of the low probability that these regions would be more limiting in the safety analyses and because of the difficulty of making a precise measurement in these regions.

This Surveillance has been modified by a Note that may require that more frequent surveillances be performed. If $F^M_Q(X,Y,Z)$ is evaluated and found to be within the applicable transient limit, an evaluation is required to account for any increase to $F^M_Q(X,Y,Z)$ that may occur and cause the $F_Q(X,Y,Z)$ limit to be exceeded before the next required $F_Q(X,Y,Z)$ evaluation.

In addition to ensuring via surveillance that the heat flux hot channel factor is within its limits when a measurement is taken, there are also requirements to extrapolate trends in both the measured hot channel factor and in its operational and RPS limits. Two extrapolations are performed for each of these two limits:

1. The first extrapolation determines whether the measured heat flux hot channel factor is likely to exceed its limit prior to the next performance of the SR.

2. The second extrapolation determines whether, prior to the next performance of the SR, the ratio of the measured heat flux hot channel factor to the limit is likely to decrease below the value of that ratio when the measurement was taken.

Each of these extrapolations is applied separately to each of the operational and RPS heat flux hot channel factor limits. If both of the extrapolations for a given limit are unfavorable, i.e., if the extrapolated factor is expected to exceed the extrapolated limit and the extrapolated factor is expected to become a larger fraction of the extrapolated limit

BASES

SURVEILLANCE REQUIREMENTS (continued)

than the measured factor is of the current limit, additional actions must be taken. These actions are to meet the $F_Q(X,Y,Z)$ limit with the last $F^M_Q(X,Y,Z)$ increased by the appropriate factor specified in the COLR or to evaluate $F_Q(X,Y,Z)$ prior to the projected point in time when the extrapolated values are expected to exceed the extrapolated limits. These alternative requirements attempt to prevent $F_Q(X,Y,Z)$ from exceeding its limit for any significant period of time without detection using the best available data. $F^M_Q(X,Y,Z)$ is not required to be extrapolated for the initial flux map taken after reaching equilibrium conditions since the initial flux map establishes the baseline measurement for future trending. Also, extrapolation of $F^M_Q(X,Y,Z)$ limits are not valid for core locations that were previously rodded, or for core locations that were previously within ±2% of the core height about the demand position of the rod tip.

$F_Q(X,Y,Z)$ is verified at power levels ≥ 10% RTP above the THERMAL POWER of its last verification, 12 hours after achieving equilibrium conditions to ensure that $F_Q(X,Y,Z)$ is within its limit at higher power levels.

~~The Surveillance Frequency of 31 EFPD is adequate to monitor the change of power distribution with core burnup. The Surveillance may be done more frequently if required by the results of $F_Q(X,Y,Z)$ evaluations.~~

~~The Frequency of 31 EFPD is adequate to monitor the change of power distribution because such a change is sufficiently slow, when the plant is operated in accordance with the TS, to preclude adverse peaking factors between 31 day surveillances.~~

INSERT 2

REFERENCES

1. 10 CFR 50.46.
2. UFSAR Section 15.4.8.
3. 10 CFR 50, Appendix A, GDC 26.
4. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
5. DPC-NE-2011PA "Duke Power Company Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors".

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.2.2.1

The value of $F^M_{\Delta H}(X,Y)$ is determined by using the movable incore detector system to obtain a flux distribution map at any THERMAL POWER greater than 5% RTP. A computer program is used to process the measured 3-D power distribution to calculate the steady state $F^L_{\Delta H}(X,Y)^{LCO}$ limit which is compared against $F^M_{\Delta H}(X,Y)$.

$F^M_{\Delta H}(X,Y)$ is verified at power levels $\geq$ 10% RTP above the THERMAL POWER of its last verification, 12 hours after achieving equilibrium conditions to ensure that $F^M_{\Delta H}(X,Y)$ is within its limit at high power levels.

~~The 31 EFPD Frequency is acceptable because the power distribution changes relatively slowly over this amount of fuel burnup. Accordingly, this Frequency is short enough that the $F_{\Delta H}(X,Y)$ limit cannot be exceeded for any significant period of operation.~~ INSERT 2

SR 3.2.2.2

The nuclear design process includes calculations performed to determine that the core can be operated within the $F_{\Delta H}(X,Y)$ limits. Because flux maps are taken in steady state conditions, the variations in power distribution resulting from normal operational maneuvers are not present in the flux map data. These variations are, however, conservatively calculated by considering a wide range of unit maneuvers in normal operation. The maximum peaking factor increase over steady state values is a limit called $F^L_{\Delta H}(X,Y)^{SURV}$. This Surveillance compares the measured $F^M_{\Delta H}(X,Y)$ to the Surveillance limit to ensure that safety analysis limits are maintained.

This Surveillance has been modified by a Note that may require that more frequent surveillances be performed. If $F^M_{\Delta H}(X,Y)$ is evaluated and found to be within its surveillance limit, an evaluation is required to account for any increase to $F^M_{\Delta H}(X,Y)$ that may occur and cause the $F_{\Delta H}(X,Y)^{SURV}$ limit to be exceeded before the next required $F_{\Delta H}(X,Y)^{SURV}$ evaluation.

In addition to ensuring via surveillance that the enthalpy rise hot channel factor is within its steady state and surveillance limits when a measurement is taken, there are also requirements to extrapolate trends in both the measured hot channel factor and in its surveillance limit. Two extrapolations are performed for this limit:

SURVEILLANCE REQUIREMENTS (continued)

1. The first extrapolation determines whether the measured enthalpy rise hot channel factor is likely to exceed its surveillance limit prior to the next performance of the SR.

2. The second extrapolation determines whether, prior to the next performance of the SR, the ratio of the measured enthalpy rise hot channel factor to the surveillance limit is likely to decrease below the value of that ratio when the measurement was taken.

Each of these extrapolations is applied separately to the enthalpy rise hot channel factor surveillance limit. If both of the extrapolations are unfavorable, i.e., if the extrapolated factor is expected to exceed the extrapolated limit and the extrapolated factor is expected to become a larger fraction of the extrapolated limit than the measured factor is of the current limit, additional actions must be taken. These actions are to meet the $F^M_{\Delta H}(X,Y)$ limit with the last $F^M_{\Delta H}(X,Y)$ increased by a factor of 1.02, or to evaluate $F^M_{\Delta H}(X,Y)$ prior to the point in time when the extrapolated values are expected to exceed the extrapolated limits. These alternative requirements attempt to prevent $F^M_{\Delta H}(X,Y)$ from exceeding its limit for any significant period of time without detection using the best available data. $F^M_{\Delta H}(X,Y)$ is not required to be extrapolated for the initial flux map taken after reaching equilibrium conditions since the initial flux map establishes the baseline measurement for future trending.

$F^M_{\Delta H}(X,Y)$ is verified at power levels 10% RTP above the THERMAL POWER of its last verification, 12 hours after achieving equilibrium conditions to ensure that $F^M_{\Delta H}(X,Y)$ is within its limit at high power levels.

~~The Surveillance Frequency of 31 EFPD is adequate to monitor the change of power distribution with core burnup. The Surveillance may be done more frequently if required by the results of~~ ~~$F^M_{\Delta H}(X,Y)$~~ ~~evaluations.~~

~~The Frequency of 31 EFPD is adequate to monitor the change of power distribution because such a change is sufficiently slow, when the plant is operated in accordance with the TS, to preclude adverse peaking factors between 31 day surveillances.~~

INSERT 2

REFERENCES

1. UFSAR Section 15.4.8

2. 10 CFR 50, Appendix A, GDC 26.

LCO (continued)

Signals are available to the operator from the Nuclear Instrumentation System (NIS) excore neutron detectors (Ref. 3). Separate signals are taken from the top and bottom detectors. The AFD is defined as the difference in normalized flux signals between the top and bottom excore detectors in each detector well. For convenience, this flux difference is converted to provide flux difference units expressed as a percentage and labeled as %Δ flux or %ΔI.

The AFD limits are provided in the COLR. The AFD limits do not depend on the target flux difference. However, the target flux difference may be used to minimize changes in the axial power distribution.

Violating this LCO on the AFD could produce unacceptable consequences if a Condition 2, 3, or 4 event occurs while the AFD is outside its specified limits.

APPLICABILITY

The AFD requirements are applicable in MODE 1 greater than or equal to 50% RTP when the combination of THERMAL POWER and core peaking factors are of primary importance in safety analysis.

For AFD limits developed using maneuvering analysis methodology, the value of the AFD does not affect the limiting accident consequences with THERMAL POWER < 50% RTP and for lower operating power MODES.

ACTIONS

A.1

As an alternative to restoring the AFD to within its specified limits, Required Action A.1 requires a THERMAL POWER reduction to < 50% RTP. This places the core in a condition for which the value of the AFD is not important in the applicable safety analyses. A Completion Time of 30 minutes is reasonable, based on operating experience, to reach 50% RTP without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1

The AFD is monitored on an automatic basis using the unit process computer, which has an AFD monitor alarm. The computer determines the 1 minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for two or more OPERABLE excore channels is outside its specified limits.

BASES

SURVEILLANCE REQUIREMENTS (continued)

This Surveillance verifies that the AFD, as indicated by the NIS excore channel, is within its specified limits and is consistent with the status of the AFD monitor alarm. With the AFD monitor alarm inoperable, the AFD is monitored every hour to detect operation outside its limit. The Frequency of 1 hour is based on operating experience regarding the amount of time required to vary the AFD, and the fact that the AFD is closely monitored. With the AFD monitor alarm OPERABLE, ~~the Surveillance Frequency of 7 days is adequate considering that the AFD is monitored by a computer and any deviation from requirements is alarmed.~~ INSERT 2

REFERENCES

1. DPC-NE-2011PA, "Duke Power Company Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors".
2. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
3. UFSAR, Chapter 7.

BASES

ACTIONS (continued)

reaching RTP. As an added precaution, if the core power does not reach RTP within 24 hours, but is increased slowly, then the peaking factor surveillances must be performed within 48 hours of the time when the more restrictive of the power level limit determined by Required Action A.1 or A.2 is exceeded. These Completion Times are intended to allow adequate time to increase THERMAL POWER to above the more restrictive limit of Required Action A.1 or A.2, while not permitting the core to remain with unconfirmed power distributions for extended periods of time.

Required Action A.7 is modified by a Note that states that the peaking factor surveillances must be done after the excore detectors have been calibrated to show zero tilt (i.e., Required Action A.6). The intent of this Note is to have the peaking factor surveillances performed at operating power levels, which can only be accomplished after the excore detectors are calibrated to show zero tilt and the core returned to power.

B.1

If Required Actions A.1 through A.7 are not completed within their associated Completion Times, the unit must be brought to a MODE or condition in which the requirements do not apply. To achieve this status, THERMAL POWER must be reduced to ≤ 50% RTP within 4 hours. The allowed Completion Time of 4 hours is reasonable, based on operating experience regarding the amount of time required to reach the reduced power level without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1

SR 3.2.4.1 is modified by three Notes. Note 1 allows QPTR to be calculated with three power range channels if THERMAL POWER is < 75% RTP and the input from one Power Range Neutron Flux channel is inoperable. Note 2 allows performance of SR 3.2.4.2 in lieu of SR 3.2.4.1. Note 3 states that the SR is not required to be performed until 12 hours after exceeding 50% RTP. This is necessary to establish core conditions necessary to provide meaningful calculation.

This Surveillance verifies that the QPTR, as indicated by the Nuclear Instrumentation System (NIS) excore channels, is within its limits. The

BASES

SURVEILLANCE REQUIREMENTS (continued) INSERT 2

Frequency of 7 days when the QPTR alarm is OPERABLE is acceptable because of the low probability that this alarm can remain inoperable without detection.

When the QPTR alarm is inoperable, the Frequency is increased to 12 hours. This Frequency is adequate to detect any relatively slow changes in QPTR, because for those causes of QPT that occur quickly (e.g., a dropped rod), there typically are other indications of abnormality that prompt a verification of core power tilt.

The QPTR alarm is inoperable for the duration of excore channel calibrations performed for agreement with incore detector measurements.

SR 3.2.4.2

This Surveillance is modified by a Note, which states that it is required only when the input from one or more Power Range Neutron Flux channels are inoperable and the THERMAL POWER is ≥ 75% RTP.

With an NIS power range channel inoperable, tilt monitoring for a portion of the reactor core becomes degraded. Large tilts are likely detected with the remaining channels, but the capability for detection of small power tilts in some quadrants is decreased. Performing SR 3.2.4.2 at a Frequency of 12 hours provides an accurate alternative means for ensuring that any tilt remains within its limits. INSERT 2

For purposes of monitoring the QPTR when one power range channel is inoperable, the moveable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the indicated QPTR and any previous data indicating a tilt. The incore detector monitoring is performed with a full incore flux map or two sets of four thimble locations with quarter core symmetry. The two sets of four symmetric thimbles is a set of eight unique detector locations. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, and N-8.

The symmetric thimble flux map can be used to generate symmetric thimble "tilt." This can be compared to a reference symmetric thimble tilt, from the most recent full core flux map, to generate an incore tilt. Therefore, incore tilt can be used to confirm that QPTR is within limits.

With one or more NIS channel inputs to QPTR inoperable, the indicated tilt may be changed from the value indicated with all four channels OPERABLE. To confirm that no change in tilt has actually occurred,

SURVEILLANCE REQUIREMENTS (continued)

which might cause the QPTR limit to be exceeded, the incore result may be compared against previous flux maps either using the symmetric thimbles as described above or a complete flux map. Nominally, quadrant tilt from the Surveillance should be within 2% of the tilt shown by the most recent flux map data.

REFERENCES

1. 10 CFR 50.46.
2. UFSAR Section 15.4.8.
3. 10 CFR 50, Appendix A, GDC 26.
4. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).

BASES

ACTIONS (continued)

U.1

With two RTS trains inoperable, no automatic capability is available to shut down the reactor, and immediate plant shutdown in accordance with LCO 3.0.3 is required.

SURVEILLANCE REQUIREMENTS

The SRs for each RTS Function are identified by the SRs column of Table 3.3.1-1 for that Function.

A Note has been added to the SR Table stating that Table 3.3.1-1 determines which SRs apply to which RTS Functions.

Note that each channel of process protection supplies both trains of the RTS. When testing Channel I, Train A and Train B must be examined. Similarly, Train A and Train B must be examined when testing Channel II, Channel III, and Channel IV (if applicable). The CHANNEL CALIBRATION and COTs are performed in a manner that is consistent with the assumptions used in analytically calculating the required channel accuracies.

Performing the Neutron Flux Instrumentation surveillances meets the License Renewal Commitments for License Renewal Program for High-Range Radiation and Neutron Flux Instrumentation Circuits per UFSAR Chapter 18, Table 18-1 and License Renewal Commitments specification CNS-1274.00-00-0016.

SR 3.3.1.1

Performance of the CHANNEL CHECK ~~once every 12 hours~~ ensures that gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff based on a combination of the channel instrument uncertainties, including indication

BASES

SURVEILLANCE REQUIREMENTS (continued)

and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

For SFCP Addition only

of 12 hours

~~The Frequency is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.~~

Insert 2

SR 3.3.1.2

SR 3.3.1.2 compares the calorimetric heat balance calculation to the NIS channel output every 24 hours. If the calorimetric exceeds the NIS channel output by > 2% RTP, the NIS is not declared inoperable, but must be adjusted. If the NIS channel output cannot be properly adjusted, the channel is declared inoperable.

Two Notes modify SR 3.3.1.2. The first Note indicates that the NIS channel output shall be adjusted consistent with the calorimetric results if the absolute difference between the NIS channel output and the calorimetric is > 2% RTP. The second Note clarifies that this Surveillance is required only if reactor power is ≥ 15% RTP and that 12 hours is allowed for completing the first Surveillance after reaching 15% RTP. At lower power levels, calorimetric data are inaccurate.

~~The Frequency of every 24 hours is adequate. It is based on unit operating experience, considering instrument reliability and operating history data for instrument drift. Together these factors demonstrate the change in the absolute difference between NIS and heat balance calculated powers rarely exceeds 2% in any 24 hour period. Maintaining the 2% agreement is only applicable during equilibrium conditions.~~

~~In addition, control room operators periodically monitor redundant indications and alarms to detect deviations in channel outputs.~~

Insert 2

SR 3.3.1.3

SR 3.3.1.3 compares the incore system to the NIS channel output every 31 EFPD. If the absolute difference is ≥ 3%, the NIS channel is still OPERABLE, but must be readjusted.

BASES

SURVEILLANCE REQUIREMENTS (continued)

If the NIS channel cannot be properly readjusted, the channel is declared inoperable. This Surveillance is performed to verify the f(ΔI) input to the overtemperature ΔT Function and overpower ΔT Function.

Two Notes modify SR 3.3.1.3. Note 1 indicates that the excore NIS channel shall be adjusted if the absolute difference between the incore and excore AFD is ≥ 3%. Note 2 clarifies that the Surveillance is required only if reactor power is ≥ 15% RTP and that 24 hours is allowed for completing the first Surveillance after reaching 15% RTP.

~~The Frequency of every 31 EFPD is adequate. It is based on unit operating experience, considering instrument reliability and operating history data for instrument drift. Also, the slow changes in neutron flux during the fuel cycle can be detected during this interval.~~ Insert 2

SR 3.3.1.4

SR 3.3.1.4 is the performance of a TADOT ~~every 62 days on a STAGGERED TEST BASIS~~. This test shall verify OPERABILITY by actuation of the end devices.

The RTB test shall include separate verification of the undervoltage and shunt trip mechanisms. Independent verification of RTB undervoltage and shunt trip Function is not required for the bypass breakers. No capability is provided for performing such a test at power. The independent test for bypass breakers is included in SR 3.3.1.14. The bypass breaker test shall include a local shunt trip. A Note has been added to indicate that this test must be performed on the bypass breaker prior to placing it in service.

~~The Frequency of every 62 days on a STAGGERED TEST BASIS is justified in Reference 12.~~ Insert 2

WCAP-15376-P-A, Rev. 1, March 2003
For SSPS Addition only

SR 3.3.1.5

SR 3.3.1.5 is the performance of an ACTUATION LOGIC TEST. The SSPS is tested ~~every 92 days on a STAGGERED TEST BASIS~~, using the semiautomatic tester. The train being tested is placed in the bypass

BASES

SURVEILLANCE REQUIREMENTS (continued)

condition, thus preventing inadvertent actuation. Through the semiautomatic tester, all possible logic combinations, with and without applicable permissives, are tested for each protection function. ~~The Frequency of every 92 days on a STAGGERED TEST BASIS is justified in Reference 12.~~ Insert 2

~~WCAP-15376-P-A, Rev. 1, March 2003~~
For SFCP Addition Only

SR 3.3.1.6

SR 3.3.1.6 is a calibration of the excore channels to the incore channels. If the measurements do not agree, the excore channels are not declared inoperable but must be calibrated to agree with the incore detector measurements. If the excore channels cannot be adjusted, the channels are declared inoperable. This Surveillance is performed to verify the f(ΔI) input to the overtemperature ΔT Function and overpower ΔT Function.

At Beginning of Cycle (BOC), the excore channels are compared to the incore detector measurements prior to exceeding 75% power. Excore detectors are adjusted as necessary. This low power surveillance satisfies the initial performance of SR 3.3.1.6 with subsequent surveillances conducted at least every 92 EFPD.

At BOC, after reaching full power steady state conditions, additional incore and excore measurements are taken at various ΔI conditions to determine the M_j factors. The M_j factors are normally only determined at BOC, but they may be changed at other points in the fuel cycle if the relationship between excore and incore measurements changes significantly.

A Note modifies SR 3.3.1.6. The Note states that this Surveillance is required only if reactor power is > 75% RTP and that 24 hours is allowed for completing the first surveillance after reaching 75% RTP.

~~The Frequency of 92 EFPD is adequate. It is based on industry operating experience, considering instrument reliability and operating history data for instrument drift.~~ Insert 2

SR 3.3.1.7

SR 3.3.1.7 is the performance of a COT ~~every 184 days.~~

A COT is performed on each required channel to ensure the channel will

BASES

SURVEILLANCE REQUIREMENTS (continued)

perform the intended Function.

The tested portion of the loop must trip within the Allowable Values specified in Table 3.3.1-1.

The setpoint shall be left set consistent with the assumptions of the setpoint methodology.

SR 3.3.1.7 is modified by a Note that provides a 4 hour delay in the requirement to perform this Surveillance for source range instrumentation when entering MODE 3 from MODE 2. This Note allows a normal shutdown to proceed without a delay for testing in MODE 2 and for a short time in MODE 3 until the RTBs are open and SR 3.3.1.7 is no longer required to be performed. If the unit is to be in MODE 3 with the RTBs closed for > 4 hours this Surveillance must be completed within 4 hours after entry into MODE 3.

WCAP-15376-P-A, Rev. 1, March 2003
For SFCP Addition only

~~The Frequency of 184 days is justified in Reference 12.~~

Insert 2

SR 3.3.1.8

[the Frequency specified in the Surveillance Frequency Control Program OR

SR 3.3.1.8 is the performance of a COT as described in SR 3.3.1.7, except it is modified by a Note that this test shall include verification that the P-6, during the Intermediate Range COT, and P-10, during the Power Range COT, interlocks are in their required state for the existing unit condition. The verification is performed by visual observation of the permissive status light in the unit control room. The Frequency is modified by a Note that allows this surveillance to be satisfied if it has been performed within 184 days] of the Frequencies prior to reactor startup and four hours after reducing power below P-10 and P-6. The Frequency of "prior to startup" ensures this surveillance is performed prior to critical operations and applies to the source, intermediate and power range low instrument channels. The Frequency of "4 hours after reducing power below P-10" (applicable to intermediate and power range low channels) and "4 hours after reducing power below P-6" (applicable to source range channels) allows a normal shutdown to be completed and the unit removed from the MODE of Applicability for this surveillance without a delay to perform the testing required by this surveillance. The Frequency of every 184 days thereafter applies if the plant remains in the MODE of Applicability after the initial performances of prior to reactor startup and four hours after reducing power below P-10 or P-6. The MODE of Applicability for this surveillance is < P-10 for the power range low and intermediate range channels and < P-6 for the source range channels.

BASES

SURVEILLANCE REQUIREMENTS (continued)

Once the unit is in MODE 3, this surveillance is no longer required. If power is to be maintained < P-10 or < P-6 for more than 4 hours, then the testing required by this surveillance must be performed prior to the expiration of the 4 hour limit. Four hours is a reasonable time to complete the required testing or place the unit in a MODE where this surveillance is no longer required. This test ensures that the NIS source, intermediate, and power range low channels are OPERABLE prior to taking the reactor critical and after reducing power into the applicable MODE (< P-10 or < P-6) for periods > 4 hours. ~~The Frequency of 184 days is justified in Reference 12.~~ Insert 2

For SFCP Addition only

WCAP-15376-P-A, Rev. 1, March 2003

SR 3.3.1.9

SR 3.3.1.9 is the performance of a TADOT and ~~is performed every 92 days, as justified in Reference 7.~~

Insert 2

The SR is modified by a Note that excludes verification of setpoints from the TADOT. Since this SR applies to RCP undervoltage and underfrequency relays, setpoint verification is accomplished during the CHANNEL CALIBRATION.

WCAP-10271-P-A, Supplement 2, Rev. 1, June 1990,

For SFCP Addition only

SR 3.3.1.10

~~A CHANNEL CALIBRATION is performed every 18 months, or approximately at every refueling.~~ CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.

CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the setpoint methodology.

~~The Frequency of 18 months is based on the assumption of an 18 month calibration interval in the determination of the magnitude of equipment drift in the setpoint methodology.~~ Insert 2

SR 3.3.1.10 is modified by a Note stating that this test shall include verification that the time constants are adjusted to the prescribed values where applicable. The applicable time constants are shown in Table 3.3.1-1.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.11

SR 3.3.1.11 is the performance of a CHANNEL CALIBRATION, as described in SR 3.3.1.10, ~~every 18 months.~~ This SR is modified by two notes. Note 1 states that neutron detectors are excluded from the CHANNEL CALIBRATION. The CHANNEL CALIBRATION for the power range neutron detectors consists of a normalization of the detectors based on a power calorimetric and flux map performed above 15% RTP. The CHANNEL CALIBRATION for the source range and intermediate range neutron detectors consists of obtaining the high voltage detector plateau and discriminator curves for source range, and the high voltage detector plateau for intermediate range, evaluating those curves, and comparing the curves to the manufacturer's data. Note 2 states that this Surveillance is not required for the NIS power range detectors for entry into MODE 2 or 1, and is not required for the NIS intermediate range detectors for entry into MODE 2, because the unit must be in at least MODE 2 to perform the test for the intermediate range detectors and MODE 1 for the power range detectors. ~~The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed on the 18 month Frequency.~~ Insert 2

SR 3.3.1.12

SR 3.3.1.12 is the performance of a CHANNEL CALIBRATION, as described in SR 3.3.1.10, ~~every 18 months.~~

~~The Frequency is justified by the assumption of an 18 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.~~ Insert 2

SR 3.3.1.13

SR 3.3.1.13 is the performance of a COT of RTS interlocks ~~every 18 months.~~

FOR SFCP addition only

of ~~18~~ months

~~The Frequency is based on the known reliability of the interlocks and the multichannel redundancy available, and has been shown to be acceptable through operating experience.~~ Insert 2

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.14

SR 3.3.1.14 is the performance of a TADOT of the Manual Reactor Trip and the SI Input from ESFAS. ~~This TADOT is performed every 18 months~~. The test shall independently verify the OPERABILITY of the undervoltage and shunt trip mechanisms for the Manual Reactor Trip Function for the Reactor Trip Breakers and Reactor Trip Bypass Breakers. The Reactor Trip Bypass Breaker test shall include testing of the automatic undervoltage trip.

~~The Frequency~~ ~~of 18 months~~ ~~is based on the known reliability of the Functions and the multichannel redundancy available, and has been shown to be acceptable through operating experience.~~ Insert 2

For SFCP Addition only

The SR is modified by a Note that excludes verification of setpoints from the TADOT. The Functions affected have no setpoints associated with them.

SR 3.3.1.15

SR 3.3.1.15 is the performance of a TADOT of Turbine Trip Functions. This TADOT is as described in SR 3.3.1.4, except that this test is performed prior to reactor startup. A Note states that this Surveillance is not required if it has been performed within the previous 31 days. Verification of the Trip Setpoint does not have to be performed for this Surveillance. Performance of this test will ensure that the turbine trip Function is OPERABLE prior to taking the reactor critical.

SR 3.3.1.16 and SR 3.3.1.17

SR 3.3.1.16 and SR 3.3.1.17 verify that the individual channel/train actuation response times are less than or equal to the maximum values assumed in the accident analysis. Response time testing acceptance criteria are included in the UFSAR (Ref. 1). Individual component response times are not modeled in the analyses.

SURVEILLANCE REQUIREMENTS (continued)

The analyses model the overall or total elapsed time, from the point at which the parameter exceeds the trip setpoint value at the sensor to the point at which the equipment reaches the required functional state (i.e., control and shutdown rods fully inserted in the reactor core).

For channels that include dynamic transfer Functions (e.g., lag, lead/lag, rate/lag, etc.), the response time test may be performed with the transfer Function set to one, with the resulting measured response time compared to the appropriate UFSAR response time. Alternately, the response time test can be performed with the time constants set to their nominal value, provided the required response time is analytically calculated assuming the time constants are set at their nominal values. The response time may be measured by a series of overlapping tests such that the entire response time is measured.

Response time may be verified by actual response time tests in any series of sequential, overlapping or total channel measurements, or by the summation of allocated sensor, signal processing and actuation logic response times with actual response time tests on the remainder of the channel. Allocations for sensor response times may be obtained from: (1) historical records based on acceptable response time tests (hydraulic, noise, or power interrupt tests), (2) in place, onsite, or offsite (e.g. vendor) test measurements, or (3) utilizing vendor engineering specifications. WCAP-13632-P-A Revision 2, "Elimination of Pressure Sensor Response Time Testing Requirements" provides the basis and methodology for using allocated sensor response times in the overall verification of the channel response time for specific sensors identified in the WCAP. In addition, while not specifically identified in the WCAP, ITT Barton 386A and 580A-0 sensors were compared to sensors which were identified. It was concluded that the WCAP results could be applied to these two sensor types as well. Response time verification for other sensor types must be demonstrated by test.

WCAP-14036-P-A Revision 1, "Elimination of Periodic Protection Channel Response Time Tests" provides the basis and methodology for using allocated signal processing and actuation logic response times in the overall verification of the protection system channel response time. The allocations for sensor, signal conditioning and actuation logic response times must be verified prior to placing the component in operational service and re-verified following maintenance that may adversely affect response time. In general, electrical repair work does not impact response time provided the parts used for repair are of the same type and value. Specific components identified in the WCAP may be replaced without verification testing. One example where response

BASES

SURVEILLANCE REQUIREMENTS (continued)

time could be affected is replacing the sensing assembly of a transmitter.

~~As appropriate, each channel's response must be verified every 18 months on a STAGGERED TEST BASIS. Testing of the final actuation devices is included in the testing. Testing of the RTS RTDs is performed on an 18 month frequency. Response times cannot be determined during unit operation because equipment operation is required to measure response times. Experience has shown that these components usually pass this surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.~~ Insert 2

SR 3.3.1.16 is modified by a Note stating that neutron detectors are excluded from RTS RESPONSE TIME testing. This Note is necessary because of the difficulty in generating an appropriate detector input signal. Excluding the detectors is acceptable because the principles of detector operation ensure a virtually instantaneous response. The response time of the neutron flux signal portion of the channel shall be measured from detector output or input of the first electronic component in the channel.

REFERENCES

1. UFSAR, Chapter 7.

2. UFSAR, Chapter 6.

3. UFSAR, Chapter 15.

4. IEEE-279-1971.

5. 10 CFR 50.49.

6. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).

7. WCAP-10271-P-A, Supplement 2, Rev. 1, June 1990.

8. WCAP-13632-P-A Revision 2, "Elimination of Pressure Sensor Response Time Testing Requirements" Sep., 1995.

9. WCAP-14036-P-A Revision 1, "Elimination of Periodic Protection Channel Response Time Tests" Oct., 1998.

10. 10 CFR 50.67.

SURVEILLANCE REQUIREMENTS

The SRs for each ESFAS Function are identified by the SRs column of Table 3.3.2-1.

A Note has been added to the SR Table to clarify that Table 3.3.2-1 determines which SRs apply to which ESFAS Functions.

Note that each channel of process protection supplies both trains of the ESFAS. When testing channel I, train A and train B must be examined. Similarly, train A and train B must be examined when testing channel II, channel III, and channel IV (if applicable). The CHANNEL CALIBRATION and COTs are performed in a manner that is consistent with the assumptions used in analytically calculating the required channel accuracies.

SR 3.3.2.1

Performance of the CHANNEL CHECK ~~once every 12 hours~~ ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff, based on a combination of the channel instrument uncertainties, including indication and reliability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

~~of 12 hours~~

For SFCP Addition only

~~The Frequency is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.~~ Insert 2

SR 3.3.2.2

SR 3.3.2.2 is the performance of an ACTUATION LOGIC TEST. ~~The SSPS is tested every 92 days on a STAGGERED TEST BASIS,~~ using the semiautomatic tester. The train being tested is placed in the bypass condition, thus preventing inadvertent actuation. Through the

BASES

SURVEILLANCE REQUIREMENTS (continued)

semiautomatic tester, all possible logic combinations, with and without applicable permissives, are tested for each protection function. In addition, the master relay coil is pulse tested for continuity. This verifies that the logic modules are OPERABLE and that there is an intact voltage signal path to the master relay coils. ~~The Frequency of every 92 days on a STAGGERED TEST BASIS is justified in Reference 14.~~ INSERT 2

WCAP-15376-P-A, Revision 1, March 2003

For SFCP Addition only

SR 3.3.2.3

SR 3.3.2.3 is the performance of a TADOT ~~every 31 days~~. This test is a check of the Loss of Offsite Power Function. Each Function is tested up to, and including, the master transfer relay coils.

This test also includes trip devices that provide actuation signals directly to the SSPS. The SR is modified by a Note that excludes final actuation of pumps and valves to minimize plant upsets that would occur. ~~The 31 day Frequency is adequate based on operating experience, considering instrument reliability and operating history data.~~ INSERT 2

FOR SFCP Addition only

SR 3.3.2.4

SR 3.3.2.4 is the performance of a MASTER RELAY TEST. The MASTER RELAY TEST is the energizing of the master relay, verifying contact operation and a low voltage continuity check of the slave relay coil. Upon master relay contact operation, a low voltage is injected to the slave relay coil. This voltage is insufficient to pick up the slave relay, but large enough to demonstrate signal path continuity. ~~This test is performed every 92 days on a STAGGERED TEST BASIS.~~ The time allowed for the testing (4 hours) is justified in Reference 7. ~~The Frequency of 92 days is justified in Reference 14.~~ INSERT 2

~~on a STAGGERED TEST BASIS~~

FOR SFCP Addition Only

~~WCAP-15376-P-A Revision 1, March 2003~~

SR 3.3.2.5

SR 3.3.2.5 is the performance of a COT.

A COT is performed on each required channel to ensure the channel will perform the intended Function. The tested portion of the loop must trip within the Allowable Values specified in Table 3.3.2-1.

SURVEILLANCE REQUIREMENTS (continued)

The setpoint shall be left set consistent with the assumptions of the setpoint methodology.

The Frequency of 184 days is justified in ~~Reference 14~~.

INSERT 2

WCAP-15376-A, Revision 1, March 2003.

For SFCP Addition only

SR 3.3.2.6

SR 3.3.2.6 is the performance of a SLAVE RELAY TEST. The SLAVE RELAY TEST is the energizing of the slave relays. Contact operation is verified in one of two ways. Actuation equipment that may be operated in the design mitigation MODE is either allowed to function, or is placed in a condition where the relay contact operation can be verified without operation of the equipment. Actuation equipment that may not be operated in the design mitigation MODE is prevented from operation by the SLAVE RELAY TEST circuit. For this latter case, contact operation is verified by a continuity check of the circuit containing the slave relay. This test is performed every 92 days. The Frequency is adequate, based on industry operating experience, considering instrument ~~reliability and~~ operating history data.

of 92 days

For SFCP addition only

For slave relays or any auxiliary relays in the ESFAS circuit that are of the type Westinghouse AR or Potter & Brumfield MDR, the SLAVE RELAY TEST is performed every 18 months. This ~~test frequency~~ is based on the relay reliability assessments presented in References ~~10, 11, and 12~~. These reliability assessments are relay specific and apply only to the Westinghouse AR and Potter & Brumfield MDR type relays. SSPS slave relays or any auxiliary relays not addressed by ~~Reference 10~~ do not qualify for extended surveillance intervals and will continue to be tested at a 92 day Frequency.

: 1) WCAP-13900, "Extension of Slave Relay Surveillance Test Intervals," April 1994; 2) WCAP-13877 Revision 2-P-A, "Reliability Assessment of Westinghouse Type AR Relays Used As SSPS Slave Relays," August 2000; 3) WCAP-13878-P-A Revision 2, "Reliability Assessment of Potter & Brumfield MDR Series Relays," August 2000.

For SFCP addition only

For slave relays or any auxiliary relays in the ESFAS circuit that are of the type Westinghouse AR or Potter & Brumfield MDR, the SLAVE RELAY TEST frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

WCAP-13900, "Extension of Slave Relay Surveillance Test Intervals," April 1994,

For SFCP Addition only

SR 3.3.2.7

SR 3.3.2.7 is the performance of a COT on the RWST level and Containment Pressure Control Start and Terminate Permissives.

A COT is performed on each required channel to ensure the entire channel will perform the intended Function. Setpoints must be found within the Allowable Values specified in Table 3.3.1-1. This test is performed every 31 days. The Frequency is adequate, based on operating experience, considering instrument reliability and operating history data.

INSERT 2

31 day

For SFCP addition only

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.8

SR 3.3.2.8 is the performance of a TADOT. This test is a check of the Manual Actuation Functions, AFW pump start on trip of all MFW pumps, AFW low suction pressure, Reactor Trip (P-4) Interlock, and Doghouse Water Level - High High Feedwater Isolation. ~~It is performed every 18 months.~~ Each Manual Actuation Function is tested up to, and including, the master relay coils. In some instances, the test includes actuation of the end device (i.e., pump starts, valve cycles, etc.). ~~The Frequency is adequate, based on industry operating experience and is consistent with the typical refueling cycle.~~ The SR is modified by a Note that excludes verification of setpoints during the TADOT for manual initiation Functions. The manual initiation Functions have no associated setpoints.

of 18 months

For SFCP Addition only

INSERT 2

SR 3.3.2.9

SR 3.3.2.9 is the performance of a CHANNEL CALIBRATION.

~~A CHANNEL CALIBRATION is performed every 18 months, or approximately at every refueling.~~ CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to measured parameter within the necessary range and accuracy.

CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the unit specific setpoint methodology.

~~The Frequency of 18 months is based on the assumption of an 18 month calibration interval in the determination of the magnitude of equipment drift in the setpoint methodology.~~ INSERT 2

This SR is modified by a Note stating that this test should include verification that the time constants are adjusted to the prescribed values where applicable. The applicable time constants are shown in Table 3.3.2-1.

SR 3.3.2.10

This SR ensures the individual channel ESF RESPONSE TIMES are less than or equal to the maximum values assumed in the accident analysis. Response Time testing acceptance criteria are included in the UFSAR (Ref. 2). Individual component response times are not modeled in the

SURVEILLANCE REQUIREMENTS (continued)

time could be affected is replacing the sensing assembly of a transmitter.

~~ESF RESPONSE TIME tests are conducted on an 18 month STAGGERED TEST BASIS. Testing of the final actuation devices, which make up the bulk of the response time, is included in the testing of each channel. The final actuation device in one train is tested with each channel. Therefore, staggered testing results in response time verification of these devices every 18 months. The 18 month Frequency is consistent with the typical refueling cycle and is based on unit operating experience, which shows that random failures of instrumentation components causing serious response time degradation, but not channel failure, are infrequent occurrences.~~ INSERT 2

This SR is modified by a Note that clarifies that the turbine driven AFW pump is tested within 24 hours after reaching 600 psig in the SGs.

SR 3.3.2.11

SR 3.3.2.11 is the performance of a COT on the NSWS Suction Transfer - Low Pit Level.

A COT is performed on each required channel to ensure the entire channel will perform the intended Function. Setpoints must be found within the Allowable Values specified in Table 3.3.2-1. ~~This test is performed every 18 months.~~ The ~~Frequency is adequate based on operating experience.~~ 18 month INSERT 2

FOR SFCP Addition only

SR 3.3.2.12

SR 3.3.2.12 is the performance of an ACTUATION LOGIC TEST on the Doghouse Water Level-High High and NSWS Suction Transfer-Emergency Low Pit Level Functions.

An ACTUATION LOGIC TEST to satisfy the requirements of GL 96-01 is performed on each instrumentation to ensure all logic combinations will initiate the appropriate Function. ~~This test is performed every 18 months.~~ ~~The Frequency is adequate based on operating experience.~~ INSERT 2

18 month

FOR SFCP Addition only

BASES

REFERENCES

1. UFSAR, Chapter 6.

2. UFSAR, Chapter 7.

3. UFSAR, Chapter 15.

4. IEEE-279-1971.

5. 10 CFR 50.49.

6. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).

7. WCAP-10271-P-A, Supplement 1 and Supplement 2, Rev. 1, May 1986 and June 1990.

8. WCAP-13632-P-A Revision 2, "Elimination of Pressure Sensor Response Time Testing Requirements" Sep., 1995.

9. WCAP-14036-P-A Revision 1, "Elimination of Periodic Protection Channel Response Time Tests" Oct., 1998.

10. ~~WCAP-13900, "Extension of Slave Relay Surveillance Test Intervals," April 1994.~~

11. ~~WCAP-13877 Revision 2-P-A, "Reliability Assessment of Westinghouse Type AR Relays Used As SSPS Slave Relays," August 2000.~~

12. ~~WCAP-13878-P-A Revision 2, "Reliability Assessment of Potter & Brumfield MDR Series Relays," August 2000~~

13. WCAP-14333-P-A, Revision 1, October 1998.

14. WCAP-15376-P-A, Revision 1, March 2003.

SURVEILLANCE REQUIREMENTS

A Note has been added to the SR Table to clarify that SR 3.3.3.1 and SR 3.3.3.3 apply to each PAM instrumentation Function in Table 3.3.3-1.

SR 3.3.3.1

Performance of the CHANNEL CHECK ~~once every 31 days~~ ensures that a gross instrumentation failure has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION. The high radiation instrumentation should be compared to similar unit instruments located throughout the unit.

Agreement criteria are determined by the unit staff, based on a combination of the channel instrument uncertainties, including isolation, indication, and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit. If the channels are within the criteria, it is an indication that the channels are OPERABLE.

As specified in the SR, a CHANNEL CHECK is only required for those channels that are normally energized.

~~The Frequency of 31 days is based on operating experience that demonstrates that channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.~~ INSERT 2

SR 3.3.3.2

Not Used

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.3.3

~~A CHANNEL CALIBRATION is performed every 18 months, or approximately at every refueling.~~ CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to measured parameter with the necessary range and accuracy. This SR is modified by two Notes. Note 1 excludes neutron detectors. The calibration method for neutron detectors is specified in the Bases of LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation." Note 2 describes the calibration methods for the Containment Area - High Range monitor. ~~The Frequency is based on operating experience and consistency with the typical industry refueling cycle.~~ Insert 2

of 18 months

For SCFP Addition only

REFERENCES

1. UFSAR Section 1.8.
2. Regulatory Guide 1.97, Rev. 2.
3. NUREG-0737, Supplement 1, "TMI Action Items."
4. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).

BASES

ACTIONS (continued)

B.1 and B.2

If the Required Action and associated Completion Time of Condition A is not met, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 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 3.3.4.1

Performance of the CHANNEL CHECK ~~once every 31 days~~ ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff, based on a combination of the channel instrument uncertainties, including indication and readability. If the channels are within the criteria, it is an indication that the channels are OPERABLE. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

As specified in the Surveillance, a CHANNEL CHECK is only required for those channels which are normally energized.

~~The Frequency of 31 days is based upon operating experience which demonstrates that channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.~~

Insert 2

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.4.2

CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.

The surveillance is modified by a Note that excepts the RTB position indication from a CHANNEL CALIBRATION. The RTB position is indicated by a mechanical "flag" on the breaker.

~~The Frequency of 18 months is based upon operating experience and consistency with the typical industry refueling cycle.~~ Insert 2

REFERENCES

1. 10 CFR 50, Appendix A, GDC 19.

ACTIONS (continued)

B.1

Condition B applies when more than one loss of voltage or more than one degraded voltage channel on a single bus is inoperable.

Required Action B.1 requires restoring all but one channel to OPERABLE status. The 1 hour Completion Time should allow ample time to repair most failures and takes into account the low probability of an event requiring an LOP start occurring during this interval.

C.1

Condition C applies to each of the LOP DG start Functions when the Required Action and associated Completion Time for Condition A or B are not met.

In these circumstances the Conditions specified in LCO 3.8.1, "AC Sources—Operating," or LCO 3.8.2, "AC Sources—Shutdown," for the DG made inoperable by failure of the LOP DG start instrumentation are required to be entered immediately. The actions of those LCOs provide for adequate compensatory actions to assure unit safety.

SURVEILLANCE REQUIREMENTS

SR 3.3.5.1

SR 3.3.5.1 is the performance of a TADOT. ~~This test is performed every 31 days.~~ The test checks trip devices that provide actuation signals directly, bypassing the analog process control equipment. For these tests, the relay Trip Setpoints are verified and adjusted as necessary. ~~The Frequency is based on the known reliability of the relays and controls and the multichannel redundancy available, and has been shown to be acceptable through operating experience.~~ INSERT 2

OF 31 DAYS

Testing consists of voltage sensor relay testing only. Actuation of load shedding and time delay timers is not required.

FOR SFCP Addition only

SURVEILLANCE REQUIREMENTS

A Note has been added to the SR Table to clarify that Table 3.3.6-1 determines which SRs apply to which containment air release and addition isolation Functions.

SR 3.3.6.1

SR 3.3.6.1 is the performance of an ACTUATION LOGIC TEST. The train being tested is placed in the bypass condition, thus preventing inadvertent actuation. Through the semiautomatic tester, all possible logic combinations, with and without applicable permissives, are tested for each protection function. In addition, the master relay coil is pulse tested for continuity. This verifies that the logic modules are OPERABLE and there is an intact voltage signal path to the master relay coils. ~~This test is performed every 92 days on a STAGGERED TEST BASIS. The Surveillance interval is justified in Reference 6.~~ Insert 2

FOR SFCP Addition only WCAP-15376-P-A, Revision 1, March 2003

SR 3.3.6.2

SR 3.3.6.2 is the performance of a MASTER RELAY TEST. The MASTER RELAY TEST is the energizing of the master relay, verifying contact operation and a low voltage continuity check of the slave relay coil. Upon master relay contact operation, a low voltage is injected to the slave relay coil. This voltage is insufficient to pick up the slave relay, but large enough to demonstrate signal path continuity. ~~This test is performed every 92 days on a STAGGERED TEST BASIS. The Surveillance interval is justified in Reference 6.~~ Insert 2

SR 3.3.6.3

SR 3.3.6.3 is the performance of a SLAVE RELAY TEST. The SLAVE RELAY TEST is the energizing of the slave relays. Contact operation is verified in one of two ways. Actuation equipment that may be operated in the design mitigation mode is either allowed to function or is placed in a condition where the relay contact operation can be verified without operation of the equipment. Actuation equipment that may not be operated in the design mitigation mode is prevented from operation by the SLAVE RELAY TEST circuit. For this latter case, contact operation is verified by a continuity check of the circuit containing the slave relay. ~~This test is performed every 92 days. The Frequency is acceptable based on instrument reliability and industry operating experience.~~ INSERT 2

of ~~92~~ DAYS
FOR SFCP Addition only

SURVEILLANCE REQUIREMENTS (continued)

For SFCP Addition only

: 1) WCAP-13900, "Extension of Slave Relay Surveillance Test Intervals," April 1994;2) WCAP-13877 Revision 2-P-A, "Reliability Assessment of Westinghouse Type AR Relays Used as SSPS Slave Relays," August 2000; 3) WCAP-13878-P-A Revision 2, "Reliability Assessment of Potter & Brumfield MDR Series Relays," August 2000.

~~For slave relays or any auxiliary relays in the circuit that are of the type Westinghouse AR or Potter & Brumfield MDR, the SLAVE RELAY TEST is performed every 18 months. This test frequency is based on the relay reliability assessments presented in References 3, 4, and 5. These reliability assessments are relay specific and apply only to the Westinghouse AR and Potter & Brumfield MDR type relays. SSPS slave relays or any auxiliary relays not addressed by Reference 3 do not qualify for extended surveillance intervals and will continue to be tested at a 92 day Frequency.~~

For SFCP Addition only

WCAP-13900, "Extension of Slave Relay Surveillance Test Intervals," April 1994,

For slave relays or any auxiliary relays in the circuit that are of the type Westinghouse AR or Potter & Brumfield MDR, the SLAVE RELAY TEST frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.3.6.4

SR 3.3.6.4 is the performance of a TADOT. This test is a check of the Manual Actuation Functions and is performed every 18 months. Each Manual Actuation Function is tested up to, and including, the master relay coils. In some instances, the test includes actuation of the end device (i.e., pump starts, valve cycles, etc.).

The test also includes trip devices that provide actuation signals directly to the SSPS, bypassing the analog process control equipment. The SR is modified by a Note that excludes verification of setpoints during the TADOT. The Functions tested have no setpoints associated with them.

For SFCP Addition only

of 18 months

~~The Frequency is based on the known reliability of the Function and the redundancy available, and has been shown to be acceptable through operating experience.~~ INSERT 2

REFERENCES

1. 10 CFR 50.67.

2. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).

3. ~~WCAP-13900, "Extension of Slave Relay Surveillance Test Intervals," April 1994.~~

4. ~~WCAP-13877 Revision 2-P-A, "Reliability Assessment of Westinghouse Type AR Relays Used as SSPS Slave Relays," August 2000.~~

5. ~~WCAP-13878-P-A Revision 2, "Reliability Assessment of Potter & Brumfield MDR Series Relays," August 2000.~~

6. ~~WCAP-15376-P-A, Revision 1, March 2003.~~

ACTIONS (continued)

The Completion Times are based on the remaining OPERABLE BDMS train and the low probability of an event occurring during this time.

B.1, B.2.1, B.2.2, B.3.1, and B.3.2

With both BDMS trains inoperable, the automatic capability for mitigation of dilution events is no longer available. In this case, one BDMS train is required to be restored to OPERABLE status within 12 hours. As an alternative (Required Actions B.2.1 and B.2.2), operations involving positive reactivity additions must be suspended and valve NV-230 must be closed and secured within the following hour to isolate the unborated water sources. A third alternative (Required Actions B.3.1 and B.3.2) is to provide alternate methods of monitoring core reactivity conditions and controlling boron dilution incidents. Alternative monitoring may be provided by the two Source Range Neutron Flux monitors. These monitors must be verified to operate with alarm setpoints less than or equal to one-half decade (Square root of 10) above the steady-state count rate. In addition, the combined flowrate from both reactor makeup water pumps must be verified within the next hour to be within the limits specified in the COLR. Required Action B.2.1 is modified by a Note, which permits plant temperature changes provided the temperature change is accounted for in the calculated SDM and that k_{eff} remains < 0.99. Introduction of temperature changes, including temperature increases when a positive MTC exists, must be evaluated to ensure they do not result in a loss of required SDM or adequate margin to criticality.

The Completion Times are based on the low probability of an event occurring during this time.

SURVEILLANCE REQUIREMENTS

SR 3.3.9.1

SR 3.3.9.1 is the performance of a CHANNEL CHECK on the BDMS, which is a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that the two indication channels should be consistent with core conditions. Changes in fuel loading and core geometry can result in significant differences, but each channel should be consistent with its local conditions. ~~The Frequency of 12 hours is consistent with the CHANNEL CHECK Frequency specified for similar instruments in LCO 3.3.1.~~

INSERT 2

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.9.2

SR 3.3.9.2 is the performance of a COT for the BDMS, which is the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify the OPERABILITY of required alarm, interlock, display, and trip functions. The COT also includes adjustments, as necessary, of the required alarm, interlock, and trip setpoints so that the setpoints are within the required range and accuracy. INSERT 2

~~This surveillance must be performed once per 31 days. The frequency is based on operating experience, which has shown to be adequate.~~

SR 3.3.9.3

SR 3.3.9.3 is performed on the BDMS to verify the actuation signal causes the appropriate valves to move to their correct position and the Reactor Makeup Water Pumps stop to mitigate a boron dilution accident.

~~The 18 month frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage. Operating experience has shown these components usually pass the Surveillance when performed at the 18 month Frequency.~~

SR 3.3.9.4

SR 3.3.9.4 is the performance of a CHANNEL CHECK on the Source Range Neutron Flux monitors, which is a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that the two indication channels should be consistent with core conditions. Changes in fuel loading and core geometry can result in significant differences, but each channel should be consistent with its local conditions.

INSERT 2

A note is provided to clarify that the CHANNEL CHECK only needs to be performed on the Source Range Neutron Flux Monitors when used to satisfy Required Action A.3 or B.3.

~~The Frequency of 12 hours is consistent with the CHANNEL CHECK Frequency specified for the same instruments in LCO 3.3.1.~~

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.9.5

SR 3.3.9.5 verifies the combined flow rates from both Reactor Makeup Water Pumps are within the value specified in the COLR. This surveillance is only required when implementing Required Action A.3 or B.3. It ensures the assumptions in the analysis for the boron dilution event under these conditions are satisfied. INSERT 2

~~This surveillance must be performed in conjunction with Required Action A.3 or B.3 and once per 31 days and is based on engineering judgement and the unlikely event that a boron dilution will occur during this time.~~

SR 3.3.9.6

SR 3.3.9.6 is the performance of a COT for the Source Range Neutron Flux monitors, which is the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify the OPERABILITY of required alarm, interlock, display, and trip functions. The COT also includes adjustments, as necessary, of the required alarm, interlock, and trip setpoints so that the setpoints are within the required range and accuracy. These monitors must be verified to operate with alarm setpoints less than or equal to 0.5 decade above the steady-state count rate. This SR is only required when the Source Range Neutron Flux Monitors are used to satisfy Required Action A.3 or B.3. This surveillance must be performed prior to placing the monitors in service for Required Action A.3 or B.3 and ~~once per 184 days~~ at the specified frequency thereafter. ~~The 184 day Frequency is justified in~~ Reference 3. Insert 2

REFERENCES

1. UFSAR, Chapter 15.
2. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
3. ~~WCAP-15376-P-A, Rev 1, March 2003.~~

~~WCAP-15376-P-A, Rev. 1, March 2003~~

For addition to SFCP only.

BASES

ACTIONS (continued)

RTP, the Power Range Neutron Flux - High Trip Setpoint must also be reduced to ≤ 55% RTP. The Completion Time of 6 hours to reset the trip setpoints is consistent with Required Action B.2. This is a sensitive operation that may inadvertently trip the Reactor Protection System. Operation is permitted to continue provided the RCS total flow is restored to ≥ 99% of the value specified in the COLR within 24 hours. The Completion Time of 24 hours is reasonable considering the increased margin to DNB at power levels below 50% and the fact that power increases associated with a transient are limited by the reduced trip setpoint.

D.1

If the Required Actions are not met within the associated 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 2 within 6 hours. The Completion Time of 6 hours is reasonable to reach the required plant conditions in an orderly manner.

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1

This surveillance demonstrates that the pressurizer pressure remains within the required limits. Alarms and other indications are available to alert operators if this limit is approached or exceeded. ~~The frequency of 12 hours is sufficient, considering the other indications available to the operator in the control room for monitoring the RCS pressure and related equipment status. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess for potential degradation and to verify operation is within safety analysis assumptions.~~ Insert 2

SR 3.4.1.2

This surveillance demonstrates that the average RCS temperature remains within the required limits. Alarms and other indications are available to alert operators if this limit is approached or exceeded. ~~The frequency of 12 hours is sufficient, considering the other indications available to the operator in the control room for monitoring the RCS. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess for potential degradation and to verify operation is within safety analysis assumptions.~~ Insert 2

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.1.3

This surveillance demonstrates that the RCS total flow rate remains within the required limits. Alarms and other indications are available to alert operators if this limit is approached or exceeded. ~~The frequency of 12 hours is sufficient, considering the other indications available to the operator in the control room for monitoring the RCS flow rate and related equipment status (e.g. RCP voltage and frequency, etc.). The 12 hour interval has been shown by operating practice to be sufficient to regularly assess potential degradation and to verify operation within safety analysis assumptions.~~ Insert 2

SR 3.4.1.4

Calibration of the installed RCS flow instrumentation permits verification that the actual RCS flow rate is greater than or equal to the minimum required RCS flow rate. INSERT 2

~~The Frequency of 18 months is consistent with operating experience.~~

REFERENCES

1. UFSAR, Section 15.
2. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).

BASES

ACTIONS (continued)

Condition C is modified by a Note requiring Required Action C.2 to be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone per Required Action C.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

SURVEILLANCE REQUIREMENTS

SR 3.4.3.1

Verification that operation is within the specified limits is required ~~every 30 minutes~~ when RCS pressure and temperature conditions are undergoing planned changes. ~~This Frequency is considered reasonable in view of the control room indication available to monitor RCS status. Also, since temperature rate of change limits are specified in hourly increments, 30 minutes permits assessment and correction for minor deviations within a reasonable time.~~ Insert 2

of 30 minutes
FOR SFCP addition only

Surveillance for heatup, cooldown, or ISLH testing may be discontinued when the definition given in the relevant plant procedure for ending the activity is satisfied.

This SR is modified by a Note that only requires this SR to be performed during system heatup, cooldown, and ISLH testing. No SR is given for criticality operations because LCO 3.4.2 contains a more restrictive requirement.

REFERENCES

1. 10 CFR 50, Appendix G.
2. ASME, Boiler and Pressure Vessel Code, Section III, Appendix G.
3. ASTM E 185-73, 1973 (Unit 1), E 185-82, 1982 (Unit 2).
4. 10 CFR 50, Appendix H.
5. Regulatory Guide 1.99, Revision 2, May 1988.
6. ASME, Boiler and Pressure Vessel Code, Section XI, Appendix E.
7. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).

BASES

APPLICABILITY (continued)

LCO 3.4.5, "RCS Loops—MODE 3";
LCO 3.4.6, "RCS Loops—MODE 4";
LCO 3.4.7, "RCS Loops—MODE 5, Loops Filled";
LCO 3.4.8, "RCS Loops—MODE 5, Loops Not Filled";
LCO 3.4.17, "RCS Loops—Test Exceptions";
LCO 3.9.4, "Residual Heat Removal (RHR) and Coolant Circulation—High Water Level" (MODE 6); and
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation—Low Water Level" (MODE 6).

ACTIONS

A.1

If the requirements of the LCO are not met, the Required Action is to reduce power and bring the plant to MODE 3. This lowers power level and thus reduces the core heat removal needs and minimizes the possibility of violating DNB limits.

The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging safety systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.4.1

This SR requires verification ~~every 12 hours~~ that each RCS loop is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal while maintaining the margin to DNB. ~~The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RCS loop performance.~~ Insert 2

REFERENCES

1. UFSAR, Section 15.

2. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).

BASES

ACTIONS (continued)

CRDMs must be de-energized by opening the RTBs or de-energizing the MG sets. All operations involving introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1 must be suspended, and action to restore one of the RCS loops to OPERABLE status and operation must be initiated. RCP seal injection flow is not considered to be an operation involving a reduction in RCS boron concentration. Boron dilution requires forced circulation for proper mixing, and opening the RTBs or de-energizing the MG sets removes the possibility of an inadvertent rod withdrawal. Suspending the introduction of coolant into the RCS of coolant with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however, coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to criticality. The immediate Completion Time reflects the importance of maintaining operation for heat removal. The action to restore must be continued until one loop is restored to OPERABLE status and operation. Once the CRDMs have been de-energized by opening the RTBs or de-energizing the MG sets, other methods to keep the CRDMs de-energized may be used. These methods are pulling fuses or opening sliding links in the rod control cabinets. This allows the flexibility for closing the RTBs or energizing the MG sets, while still preventing rod motion.

SURVEILLANCE REQUIREMENTS

SR 3.4.5.1

This SR requires verification ~~every 12 hours~~ that the required loops are in operation. Verification includes flow rate, temperature, and pump status monitoring, which help ensure that forced flow is providing heat removal. ~~The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RCS loop performance.~~ Insert 2

SR 3.4.5.2

SR 3.4.5.2 requires verification of SG OPERABILITY. SG OPERABILITY is verified by ensuring that the secondary side narrow range water level is ≥ 12% for required RCS loops. If the SG secondary side narrow range water level is < 12%, the tubes may become uncovered and the associated loop may not be capable of providing the heat sink for removal of the decay heat. ~~The 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to a loss of SG level.~~ Insert 2

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.5.3

Verification that the required RCPs are OPERABLE ensures that safety analyses limits are met. The requirement also ensures that an additional RCP can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power availability to the required RCPs. INSERT 2

REFERENCES 1. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).

~~The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.~~

For SFCP addition only

BASES

ACTIONS (continued)

C.1 and C.2

If no loop is OPERABLE or in operation, except during conditions permitted by Note 1 in the LCO section, all operations involving introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1 and maintain $k_{eff} < 0.99$ must be suspended and action to restore one RCS or RHR loop to OPERABLE status and operation must be initiated. RCP seal injection flow is not considered to be an operation involving a reduction in RCS boron concentration. The required margin to criticality must not be reduced in this type of operation. Suspending the introduction of coolant into the RCS of coolant with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 and maintain $k_{eff} < 0.99$ is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however, coolant added with boron concentration meeting the minimum SDM and k_{eff} requirements maintains acceptable margin to criticality. The immediate Completion Times reflect the importance of maintaining operation for decay heat removal. The action to restore must be continued until one loop is restored to OPERABLE status and operation.

SURVEILLANCE REQUIREMENTS

SR 3.4.6.1

This SR requires verification ~~every 12 hours~~ that one RCS or RHR loop is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. ~~The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RCS and RHR loop performance.~~ Insert 2

SR 3.4.6.2

SR 3.4.6.2 requires verification of SG OPERABILITY. SG OPERABILITY is verified by ensuring that the secondary side narrow range water level is ≥ 12%. If the SG secondary side narrow range water level is < 12%, the tubes may become uncovered and the associated loop may not be capable of providing the heat sink necessary for removal of decay heat. ~~The 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to the loss of SG level.~~ Insert 2

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.6.3

Verification that the required pump is OPERABLE ensures that an additional RCS or RHR pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the required pump. ~~The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.~~ Insert 7

REFERENCES

1. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).

BASES

ACTIONS

A.1 and A.2

If one RHR loop is inoperable and the required SGs have secondary side narrow range water levels < 12%, redundancy for heat removal is lost. Action must be initiated immediately to restore a second RHR loop to OPERABLE status or to restore the required SG secondary side water levels. Either Required Action A.1 or Required Action A.2 will restore redundant heat removal paths. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

B.1 and B.2

If no RHR loop is in operation, except during conditions permitted by Note 1, or if no loop is OPERABLE, all operations involving introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1 must be suspended and action to restore one RHR loop to OPERABLE status and operation must be initiated. RCP seal injection flow is not considered to be an operation involving a reduction in RCS boron concentration. Suspending the introduction of coolant into the RCS of coolant with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however, coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to criticality. The immediate Completion Times reflect the importance of maintaining operation for heat removal.

SURVEILLANCE REQUIREMENTS

SR 3.4.7.1

This SR requires verification ~~every 12 hours~~ that the required loop is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. ~~The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RHR loop performance.~~ INSERT 2

SR 3.4.7.2

Verifying that at least two SGs are OPERABLE by ensuring their secondary side narrow range water levels are ≥ 12% ensures an alternate decay heat removal method in the event that the second RHR loop is not OPERABLE. If both RHR loops are OPERABLE, this Surveillance is not needed. ~~The 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to the loss of SG level.~~ INSERT 2

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.7.3

Verification that a second RHR pump is OPERABLE ensures that an additional pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the RHR pump. If secondary side narrow range water level is ≥ 12% in at least two SGs, this Surveillance is not needed. ~~The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.~~ Insert 2

REFERENCES

1. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).

BASES

ACTIONS (continued)

immediately to restore an RHR loop to OPERABLE status and operation.

RCP seal injection flow is not considered to be an operation involving a reduction in RCS boron concentration. The required margin to criticality must not be reduced in this type of operation. Suspending the introduction of coolant into the RCS of coolant with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however, coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to criticality. The immediate Completion Time reflects the importance of maintaining operation for heat removal. The action to restore must continue until one loop is restored to OPERABLE status and operation.

SURVEILLANCE REQUIREMENTS

SR 3.4.8.1

This SR requires verification ~~every 12 hours~~ that one loop is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. ~~The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RHR loop performance.~~ Insert 2

SR 3.4.8.2

Verification that the required number of pumps are OPERABLE ensures that an additional pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the required pumps. ~~The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.~~ Insert 2

REFERENCES

1. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).

BASES

ACTIONS (continued)

MODE 4 within 12 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 3.4.9.1

This SR requires that during steady state operation, pressurizer level is maintained below the nominal upper limit to provide a minimum space for a steam bubble. The Surveillance is performed by observing the indicated level. ~~The Frequency of 12 hours corresponds to verifying the parameter each shift. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess level for any deviation and verify that operation is within safety analyses assumptions. Alarms are also available for early detection of abnormal level indications.~~ Insert 2

SR 3.4.9.2

The SR is satisfied when the power supplies are demonstrated to be capable of producing the minimum power and the associated pressurizer heaters are verified to be at their design rating. This SR may be verified by energizing the heaters and measuring circuit current. ~~The Frequency of 92 days is considered adequate to detect heater degradation and has been shown by operating experience to be acceptable.~~ Insert 2

SR 3.4.9.3

This Surveillance demonstrates that the heaters can be automatically transferred from the normal to the emergency power supply. ~~The Frequency of 18 months is based on a typical fuel cycle and is consistent with similar verifications of emergency power supplies.~~ Insert 2

REFERENCES

1. UFSAR, Section 15.

2. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).

3. NUREG-0737, November 1980.

BASES

ACTIONS (continued)

MODE 4 within 12 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. In MODES 4 and 5, maintaining PORV OPERABILITY may be required. See LCO 3.4.12.

SURVEILLANCE REQUIREMENTS

SR 3.4.11.1

INSERT 2

Block valve cycling verifies that the valve(s) can be closed if needed. ~~The basis for the Frequency of 92 days is the ASME Code (Ref. 4).~~ If the block valve is closed to isolate a PORV that is capable of being manually cycled, the OPERABILITY of the block valve is of importance, because opening the block valve is necessary to permit the PORV to be used for manual control of reactor pressure. If the block valve is closed to isolate an otherwise inoperable PORV, the maximum Completion Time to restore the PORV and open the block valve is 72 hours, which is well within the allowable limits (25%) to extend the block valve Frequency ~~of 92 days~~. Furthermore, these test requirements would be completed by the reopening of a recently closed block valve upon restoration of the PORV to OPERABLE status (i.e., completion of the Required Actions fulfills the SR).

The Note modifies this SR by stating that it is not required to be met with the block valve closed, in accordance with the Required Action of this LCO.

SR 3.4.11.2

SR 3.4.11.2 requires a complete cycle of each PORV. Operating a PORV through one complete cycle ensures that the PORV can be manually actuated for mitigation of an SGTR. ~~The Frequency of 18 months is based on a typical refueling cycle and industry accepted practice.~~ INSERT 2

The SR is modified by a Note which states that the SR is required to be performed in MODE 3 or 4 when the temperature of the RCS cold legs is > 200°F consistent with Generic Letter 90-06 (Ref. 5).

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.11.3

The Surveillance demonstrates that the emergency nitrogen supply can be provided and is performed by transferring power from normal air supply to emergency nitrogen supply and cycling the valves. ~~The Frequency of 18 months is based on a typical refueling cycle and industry accepted practice.~~ INSERT 2

This SR is modified by a Note which states the SR is not applicable to PORV NC-36B. This PORV does not have a nitrogen supply.

REFERENCES

1. Regulatory Guide 1.32, February 1977.
2. UFSAR, Section 15.4.
3. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
4. ASME Code for Operations and Maintenance of Nuclear Power Plants.
5. Resolution of Generic Issue 70, "Power-Operated Relief Valve and Block Valve Reliability," and Generic Issue 94, "Additional Low-Temperature Overpressure Protection for Light Water Reactors," Pursuant to 10 CFR 50.54(f) (Generic Letter 90-06).

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.12.1 and SR 3.4.12.2

To minimize the potential for a low temperature overpressure event by limiting the mass input capability, a maximum of two pumps (charging and/or safety injection) are verified capable of injecting into the RCS and the accumulator discharge isolation valves are verified closed and power removed.

The pumps are rendered incapable of injecting into the RCS through removing the power from the pumps by racking the breakers out under administrative control. An alternate method of LTOP control may be employed using at least two independent means to prevent a pump start such that a single failure or single action will not result in an injection into the RCS. This may be accomplished through two valves in the discharge flow path being closed.

~~The Frequency of 12 hours is sufficient, considering other indications and alarms available to the operator in the control room, to verify the required status of the equipment.~~ INSERT 2

SR 3.4.12.3

Each required RHR suction relief valve shall be demonstrated OPERABLE by verifying its RHR suction isolation valves are open and by testing it in accordance with the Inservice Testing Program. This Surveillance is only required to be performed if the RHR suction relief valve is being used to meet this LCO.

of 12 hours

For addition to SFCP only

The RHR suction isolation valves are verified to be opened ~~every 12 hours~~. ~~The Frequency is considered adequate in view of other administrative controls such as valve status indications available to the operator in the control room that verify the RHR suction isolation valves remain open.~~ INSERT 2

The ASME Code (Ref. 9), test per Inservice Testing Program verifies OPERABILITY by proving relief valve mechanical motion and by measuring and, if required, adjusting the lift setpoint.

SR 3.4.12.4

The PORV block valve must be verified open ~~every 72 hours~~ to provide the flow path for each required PORV to perform its function when actuated. The valve must be remotely verified open in the main control room. This Surveillance is performed if the PORV satisfies the LCO.

The block valve is a remotely controlled, motor operated valve. The

BASES

SURVEILLANCE REQUIREMENTS (continued)

power to the valve operator is not required removed, and the manual operator is not required locked in the inactive position. Thus, the block valve can be closed in the event the PORV develops excessive leakage or does not close (sticks open) after relieving an overpressure situation. INSERT 2

~~The 72 hour Frequency is considered adequate in view of other administrative controls available to the operator in the control room, such as valve position indication, that verify that the PORV block valve remains open.~~

SR 3.4.12.5

~~The 31 day frequency considers experience and equipment reliability.~~

Performance of a COT is required periodically within 12 hours after decreasing RCS temperature to ≤ 210°F and ~~every 31 days~~ on each required PORV to verify and, as necessary, adjust its lift setpoint. The COT will verify the setpoint is within the allowed maximum limits. PORV actuation could depressurize the RCS and is not required. INSERT 2

The 12 hour Frequency considers the unlikelihood of a low temperature overpressure event during this time.

For SFCP addition only

A Note has been added indicating that this SR is required to be met 12 hours after decreasing RCS cold leg temperature to ≤ 210°F. The COT cannot be performed until in the LTOP MODES when the PORV lift setpoint can be reduced to the LTOP setting. The test must be performed within 12 hours after entering the LTOP MODES.

SR 3.4.12.6

~~The 18 month frequency considers operating experience with equipment reliability and matches the typical refueling outage schedule.~~

Performance of a CHANNEL CALIBRATION on each required PORV actuation channel is required ~~every 18 months~~ to adjust the whole channel so that it responds and the valve opens within the required range and accuracy to known input. INSERT 2

SR 3.4.12.7

Each required RHR suction relief valve shall be demonstrated OPERABLE by verifying its RHR suction isolation valves are open and by testing it in accordance with the Inservice Testing Program. (Refer to SR 3.4.12.3 for the RHR suction isolation valves Surveillance and for a description of the Inservice Testing Program.) This Surveillance is only required to be performed if the RHR suction relief valve is being used to meet this LCO.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~Each 31 days~~ the RHR suction isolation valves are verified open, with power to the valve operator removed and locked in the removed position, to ensure that accidental closure will not occur. The "locked open in the removed position" power supply must be locally verified in its open position with the power supply to the valve locked in its inactive position. ~~The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve position.~~ INSERT 2

REFERENCES

1. 10 CFR 50, Appendix G.
2. Generic Letter 88-11.
3. UFSAR, Section 5.2
4. 10 CFR 50, Section 50.46.
5. 10 CFR 50, Appendix K.
6. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
7. Generic Letter 90-06.
8. ASME, Boiler and Pressure Vessel Code, Section III.
9. ASME Code for Operation and Maintenance of Nuclear Power Plants.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1

Verifying RCS LEAKAGE to be within the LCO limits ensures the integrity of the RCPB is maintained. Pressure boundary LEAKAGE would at first appear as unidentified LEAKAGE and can only be positively identified by inspection. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. Unidentified LEAKAGE and identified LEAKAGE are determined by performance of an RCS water inventory balance. For this SR, the volumetric calculation of unidentified LEAKAGE and identified LEAKAGE is based on a density at room temperature of 77 degrees F.

The Surveillance is modified by two Notes. The RCS water inventory balance must be performed with the reactor at steady state operating conditions and near operating pressure. Therefore, Note 1 indicates that this SR is not required to be completed until 12 hours of steady state operation near operating pressure have been established.

Steady state operation is required to perform a proper inventory balance; calculations during maneuvering are not useful and Note 1 requires the Surveillance to be met when steady state is established. For RCS operational LEAKAGE determination by water inventory balance, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows.

Note 2 states that this SR is not applicable to primary to secondary LEAKAGE because LEAKAGE of 150 gallons per day or lower cannot be measured accurately by an RCS water inventory balance.

An early warning of pressure boundary LEAKAGE or unidentified LEAKAGE is provided by the automatic systems that monitor the containment atmosphere radioactivity and the containment sump level. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. These leakage detection systems are specified in LCO 3.4.15, "RCS Leakage Detection Instrumentation."

~~The 72 hour Frequency is a reasonable interval to trend LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents and reduction of potential consequences.~~ A Note under the Frequency column states that this SR is only required to be performed during steady state operation.

Insert 2

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.13.2

This SR verifies that primary to secondary LEAKAGE is less than or equal to 150 gallons per day through any one SG. Satisfying the primary to secondary LEAKAGE limit ensures that the operational LEAKAGE performance criterion in the Steam Generator Program is met. If this SR is not met, compliance with LCO 3.4.18, "Steam Generator (SG) Tube Integrity," should be evaluated. The 150 gallons per day limit is based on measurements taken at room temperature. The primary to secondary leak rate assumed in the safety analyses is taken also at room temperature.

The Surveillance is modified by a Note which states that this SR is not required to be completed until 12 hours of steady state operation near operating pressure have been established. During normal operation the primary to secondary LEAKAGE is determined using continuous process radiation monitors or radiochemical grab sampling.

~~The 72 hour Frequency is a reasonable interval to trend primary to secondary LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents and reduction of potential consequences.~~ Insert 2 A Note under the Frequency column states that this SR is only required to be performed during steady state operation.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 30.

2. Regulatory Guide 1.45, May 1973.

3. UFSAR, Section 15.

4. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).

5. EPRI TR-104788-R2, “PWR Primary-to-Secondary Leak Guidelines,” Revision 2.

6. NEI 97-06, “Steam Generator Program Guidelines.”

7. UFSAR, Section 18, Table 18-1.

8. Catawba License Renewal Commitments, CNS-1274.00-00-0016, Section 4.27.

9. 10 CFR 50.67.

BASES

ACTIONS (continued)

B.1 and B.2

If leakage cannot be reduced, or the other Required Actions accomplished, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. This Action may reduce the leakage and also reduces the potential for a LOCA outside the containment. 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.

C.1

The RHR interlock prevents the RHR suction isolation valves inadvertent opening at RCS pressures in excess of the RHR systems design pressure. If the RHR interlock is inoperable, operation may continue as long as the affected RHR suction penetration is closed by at least one closed manual or deactivated automatic valve within 4 hours. This Action accomplishes the purpose of the interlock function.

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1

Performance of leakage testing on each RCS PIV or isolation valve used to satisfy Required Action A.1 is required to verify that leakage is below the specified limit and to identify each leaking valve. The leakage limit of 0.5 gpm per inch of nominal valve diameter up to 5 gpm maximum applies to each valve. Leakage testing requires a stable pressure condition.

For the two PIVs in series, the leakage requirement applies to each valve individually and not to the combined leakage across both valves. If the PIVs are not individually leakage tested, one valve may have failed completely and not be detected if the other valve in series meets the leakage requirement. In this situation, the protection provided by redundant valves would be lost.

~~Testing is to be performed every 18 months, a typical refueling cycle, if the plant does not go into MODE 5 for at least 7 days. The 18 month Frequency is consistent with 10 CFR 50.55a(g) (Ref. 9) as contained in the Inservice Testing Program, is within frequency allowed by the American Society of Mechanical Engineers (ASME) Code (Ref. 8), and is based on the need to perform such surveillances under the conditions that~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~apply during an outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.~~ INSERT 2

In addition, testing must be performed once after the valve has been opened by flow or exercised to ensure tight reseating. PIVs disturbed in the performance of this Surveillance should also be tested unless documentation shows that an infinite testing loop cannot practically be avoided. Testing must be performed within 24 hours after the valve has been reseated. Within 24 hours is a reasonable and practical time limit for performing this test after opening or reseating a valve.

The leakage limit is to be met at the RCS pressure associated with MODES 1 and 2. This permits leakage testing at high differential pressures with stable conditions not possible in the MODES with lower pressures.

Entry into MODES 3 and 4 is allowed to establish the necessary differential pressures and stable conditions to allow for performance of this Surveillance. The Note that allows this provision is complementary to the Frequency of prior to entry into MODE 2 whenever the unit has been in MODE 5 for 7 days or more, if leakage testing has not been performed in the previous 9 months. In addition, this Surveillance is not required to be performed on the RHR System when the RHR System is aligned to the RCS in the shutdown cooling mode of operation. PIVs contained in the RHR shutdown cooling flow path must be leakage rate tested after RHR is secured and stable unit conditions and the necessary differential pressures are established.

SR 3.4.14.2

Verifying that the RHR interlock is OPERABLE ensures that RCS pressure will not pressurize the RHR system beyond its design pressure of 600 psig. The interlock setpoint that prevents the valves from being opened is set so the actual RCS pressure must be < 425 psig to open the valves. This setpoint ensures the RHR design pressure will not be exceeded and the RHR relief valves will not lift. ~~The 18 month Frequency is based on the need to perform the Surveillance under conditions that apply during a plant outage. The 18 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment.~~

INSERT 2

BASES

ACTIONS (continued)

G.1

With all required monitors inoperable, no automatic means of monitoring leakage are available, and immediate plant shutdown in accordance with LCO 3.0.3 is required. The required monitors during MODE 1 for LCO 3.0.3 entry are defined as the simultaneous inoperability of one CFAE level monitor, the containment atmosphere particulate radioactivity monitor, and the CVUCDT level monitor. The required monitors during MODES 2, 3, and 4 for LCO 3.0.3 entry are defined as the simultaneous inoperability of one CFAE level monitor and the CVUCDT level monitor. This condition does not apply to the incore instrument sump level alarm.

SURVEILLANCE REQUIREMENTS

SR 3.4.15.1

SR 3.4.15.1 requires the performance of a CHANNEL CHECK of the containment atmosphere particulate radioactivity monitor. The check gives reasonable confidence that the channel is operating properly. ~~The Frequency of 12 hours is based on instrument reliability and is reasonable for detecting off normal conditions.~~

Insert 2

SR 3.4.15.2

SR 3.4.15.2 requires the performance of a COT on the containment atmosphere particulate radioactivity monitor. The test ensures that a signal from the monitor can generate the appropriate alarm associated with the detection of a minimum 1 gpm RCS leak. The desired alarm is derived from a digital database. Database manipulation concurrent with a signal supplied from the detector verifies the OPERABILITY of the required alarm. ~~The Frequency of 92 days considers instrument reliability, and operating experience has shown that it is proper for detecting degradation.~~

Insert 2

SR 3.4.15.3, SR 3.4.15.4, SR 3.4.15.5, and SR 3.4.15.6

These SRs require the performance of a CHANNEL CALIBRATION for each of the RCS leakage detection instrumentation channels. The calibration verifies the accuracy of the instrument string, including the instruments located inside containment. ~~The Frequency of 18 months is a typical refueling cycle and considers channel reliability. Again, operating experience has proven that this Frequency is acceptable.~~

Insert 2

BASES

ACTIONS (continued)

transient specific activity excursions while the plant remains at, or proceeds to power operation.

B.1

With the gross specific activity in excess of the allowed limit, the unit must be placed in a MODE in which the requirement does not apply.

The change within 6 hours to MODE 3 and RCS average temperature < 500°F lowers the saturation pressure of the reactor coolant below the setpoints of the main steam safety valves and prevents venting the SG to the environment in an SGTR event. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 below 500°F from full power conditions in an orderly manner and without challenging plant systems.

C.1

If a Required Action and the associated Completion Time of Condition A is not met or if the DOSE EQUIVALENT I–131 is in the unacceptable region of Figure 3.4.16-1, the reactor must be brought to MODE 3 with RCS average temperature < 500°F within 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 below 500°F from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.16.1

SR 3.4.16.1 requires performing a gamma isotopic analysis as a measure of the gross specific activity of the reactor coolant ~~at least once every 7 days~~. A gross radioactivity analysis shall consist of the quantitative measurement of the total specific activity of the reactor coolant except for radionuclides with half-lives less than 10 minutes and all radioiodines. The total specific activity shall be the sum of the beta-gamma activity in the sample within 2 hours after the sample is taken and extrapolated back to when the sample was taken. Determination of the contributors to the gross specific activity shall be based upon those energy peaks identifiable with a 95% confidence level. The latest available data may be used for pure beta-emitting radionuclides. This Surveillance provides an indication of any increase in gross specific activity.

BASES

SURVEILLANCE REQUIREMENTS (continued)

Trending the results of this Surveillance allows proper remedial action to be taken before reaching the LCO limit under normal operating conditions. The Surveillance is applicable in MODES 1 and 2, and in MODE 3 with T_{avg} at least 500°F. ~~The 7 day Frequency considers the unlikelihood of a gross fuel failure during the time.~~ INSERT 2

SR 3.4.16.2

This Surveillance is performed in MODE 1 only to ensure iodine remains within limit during normal operation and following fast power changes when fuel failure is more apt to occur. ~~The 14 day Frequency is adequate to trend changes in the iodine activity level, considering gross activity is monitored every 7 days.~~ INSERT 2 The Frequency, between 2 and 6 hours after a power change ≥ 15% RTP within a 1 hour period, is established because the iodine levels peak during this time following fuel failure; samples at other times would provide inaccurate results. If the power excursion is one continuous process spanning over several hours, there is no need to sample every hour, only 2 to 6 hours after the last major power change of ≥ 15% RTP, since this sample will encompass the maximum potential for additional iodine release to have occurred.

SR 3.4.16.3

A radiochemical analysis for $\bar{E}$ determination is required ~~every 184 days (6 months)~~ with the plant operating in MODE 1 equilibrium conditions. The $\bar{E}$ determination directly relates to the LCO and is required to verify plant operation within the specified gross activity LCO limit. The analysis for $\bar{E}$ is a measurement of the average energies per disintegration for isotopes with half lives longer than 10 minutes, excluding iodines. ~~The Frequency of 184 days recognizes $\bar{E}$ does not change rapidly.~~ INSERT 2

This SR has been modified by a Note that indicates sampling is required to be performed within 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for at least 48 hours. This ensures that the radioactive materials are at equilibrium so the analysis for $\bar{E}$ is representative and not skewed by a crud burst or other similar abnormal event.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.17.1

Verification that the power level is < the P–7 interlock setpoint (10%) will ensure that the fuel design criteria are not violated during the performance of the PHYSICS TESTS. ~~The Frequency of once per hour is adequate to ensure that the power level does not exceed the limit. Plant operations are conducted slowly during the performance of PHYSICS TESTS and monitoring the power level once per hour is sufficient to ensure that the power level does not exceed the limit.~~ Insert 2

SR 3.4.17.2

The power range and intermediate range neutron detectors and P–10 and P–13 inputs to the P–7 interlock setpoint must be verified to be OPERABLE and adjusted to the proper value. A COT is performed prior to initiation of the PHYSICS TESTS. This will ensure that the RTS 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.
2. 10 CFR 50, Appendix A, GDC 1, 1988.
3. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.1

Each accumulator valve should be verified to be fully open ~~every 12 hours~~. This verification ensures that the accumulators are available for injection and ensures timely discovery if a valve should be less than fully open. If an isolation valve is not fully open, the rate of injection to the RCS would be reduced. Although a motor operated valve position should not change with power removed, a closed valve could result in not meeting accident analyses assumptions. ~~This Frequency is considered reasonable in view of other administrative controls that ensure a mispositioned isolation valve is unlikely.~~ Insert 2

of 12 hours

For SFCP addition only

SR 3.5.1.2 and SR 3.5.1.3

TKa

~~Every 12 hours,~~ borated water volume and nitrogen cover pressure are verified for each accumulator. This is typically performed using the installed control room indication. ~~This Frequency is sufficient to ensure adequate injection during a LOCA. Because of the static design of the accumulator, a 12 hour Frequency usually allows the operator to identify changes before limits are reached. Operating experience has shown this Frequency to be appropriate for early detection and correction of off normal trends.~~ Insert 2

of 12 hours

SR 3.5.1.4

The boron concentration should be verified to be within required limits for each accumulator ~~every 31 days~~ since the static design of the accumulators limits the ways in which the concentration can be changed. ~~The 31 day Frequency is adequate to identify changes that could occur from mechanisms such as stratification or inleakage.~~ Insert 2 Sampling the affected accumulator within 6 hours after a 75 gallon increase will identify whether inleakage has caused a reduction in boron concentration to below the required limit. It is not necessary to verify boron concentration if the added water inventory is from the refueling water storage tank (RWST), because the water contained in the RWST is within the accumulator boron concentration requirements. This is consistent with the recommendation of NUREG-1366 (Ref. 7).

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.1.5

Verification ~~every 31 days~~ that power is removed from each accumulator isolation valve operators for NI54A, NI65B, NI76A, and NI88B when the RCS pressure is > 1000 psig ensures that an active failure could not result in the undetected closure of an accumulator motor operated isolation valve. If this were to occur, only two accumulators would be available for injection given a single failure coincident with a LOCA. ~~Since power is removed and circuit breakers padlocked under administrative control, the 31 day Frequency will provide adequate assurance that power is removed.~~ Insert 2

This SR allows power to be supplied to the motor operated isolation valves when RCS pressure is ≤ 1000 psig, thus allowing operational flexibility by avoiding unnecessary delays to manipulate the breakers during plant startups or shutdowns. Even with power supplied to the valves, inadvertent closure is prevented by the RCS pressure interlock associated with the valves.

Should closure of a valve occur in spite of the interlock, the SI signal provided to the valves would open a closed valve in the event of a LOCA.

REFERENCES

1. IEEE Standard 279-1971.
2. UFSAR, Chapter 6.
3. 10 CFR 50.46.
4. DPC-NE-3004.
5. 10 CFR 50.36, Technical Specification, (c)(2)(ii).
6. WCAP-15049-A, Rev. 1, April 1999.
7. NUREG-1366, February 1990.

BASES

ACTIONS (continued)

An event accompanied by a loss of offsite power and the failure of an EDG can disable one ECCS train until power is restored. A reliability analysis (Ref. 6) has shown that the impact of having one full ECCS train inoperable is sufficiently small to justify continued operation for 72 hours.

Reference 7 describes situations in which one component, such as an RHR crossover valve, can disable both ECCS trains. With one or more component(s) inoperable such that 100% of the flow equivalent to a single OPERABLE ECCS train is not available, the facility is in a condition outside the accident analysis. Therefore, LCO 3.0.3 must be immediately entered.

B.1 and B.2

If the inoperable trains cannot be returned to OPERABLE status within the associated 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 MODE 3 within 6 hours and MODE 4 within 12 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 3.5.2.1

Verification of proper valve position ensures that the flow path from the ECCS pumps to the RCS is maintained. Misalignment of these valves could render both ECCS trains inoperable. Securing these valves using the power disconnect switches in the correct position ensures that they cannot change position as a result of an active failure or be inadvertently misaligned. These valves are of the type, described in Reference 7, that can disable the function of both ECCS trains and invalidate the accident analyses. ~~A 12 hour Frequency is considered reasonable in view of other administrative controls that will ensure a mispositioned valve is unlikely.~~

INSERT 2

SR 3.5.2.2

Verifying the correct alignment for manual, power operated, and automatic valves in the ECCS flow paths provides assurance that the proper flow paths will exist for ECCS operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these were verified to be in the correct position prior to locking, sealing,

BASES

SURVEILLANCE REQUIREMENTS (continued)

or securing. A valve that receives an actuation signal is allowed to be in a nonaccident position provided the valve will automatically reposition within the proper stroke time. 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 appropriate because the valves are operated under administrative control.~~

~~This Frequency has been shown to be acceptable through operating experience.~~ INSERT 2

SR 3.5.2.3

With the exception of the operating centrifugal charging pump, the ECCS pumps are normally in a standby, nonoperating mode. As such, flow path piping has the potential to develop voids and pockets of entrained gases. ECCS piping is verified to be water filled by venting to remove gas from accessible locations susceptible to gas accumulation. Alternative means may be used to verify water filled conditions (e.g., ultrasonic testing or high point sight glass observation). Maintaining the piping from the ECCS pumps to the RCS full of water ensures that the system will perform properly, injecting its full capacity into the RCS upon demand. This will also prevent water hammer, pump cavitation, and pumping of noncondensible gas (e.g., air, nitrogen, or hydrogen) into the reactor vessel following an SI signal or during shutdown cooling. ~~The 31 day Frequency takes into consideration the gradual nature of gas accumulation in the ECCS piping and the procedural controls governing system operation.~~ INSERT 2

SR 3.5.2.4

Periodic surveillance testing of ECCS pumps to detect gross degradation caused by impeller structural damage or other hydraulic component problems is required by the ASME Code. This type of testing may be accomplished by measuring the pump developed head at only one point of the pump characteristic curve. This verifies both that the measured performance is within an acceptable tolerance of the original pump baseline performance and that the performance at the test flow is greater than or equal to the performance assumed in the plant safety analysis. SRs are specified in the Inservice Testing Program, the ASME Code. The ASME Code provides the activities and Frequencies necessary to satisfy the requirements.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.5 and SR 3.5.2.6

These Surveillances demonstrate that each automatic ECCS valve actuates to the required position on an actual or simulated SI and Containment Sump Recirculation signal and that each ECCS pump starts on receipt of an actual or simulated SI signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. ~~The 18 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a plant outage and the potential for unplanned plant transients if the Surveillances were performed with the reactor at power. The 18 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment. The actuation logic is tested as part of ESF Actuation System testing, and equipment performance is monitored as part of the Inservice Testing Program.~~ INSERT 2

SR 3.5.2.7

The position of throttle valves in the flow path on an SI signal is necessary for proper ECCS performance. These valves have mechanical locks to ensure proper positioning for restricted flow to a ruptured cold leg, ensuring that the other cold legs receive at least the required minimum flow. ~~The 18 month Frequency is based on the same reasons as those stated in SR 3.5.2.5 and SR 3.5.2.6.~~ INSERT 2

SR 3.5.2.8

Periodic inspections of the containment sump suction inlet ensure that it is unrestricted and stays in proper operating condition. ~~The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and on the need to have access to the location. This Frequency has been found to be sufficient to detect abnormal degradation and is confirmed by operating experience.~~ INSERT 2

Upon completion of the ECCS sump strainer assembly modifications during outage 2EOC15 for Unit 2 and 1EOC17 for Unit 1, the following SR Bases will apply:

Periodic inspections of the ECCS containment sump strainer assembly (consisting of modular tophats, grating, plenums, and waterboxes) ensure it is unrestricted and remains in proper operating condition.

SURVEILLANCE REQUIREMENTS (continued)

Inspections will consist of a visual examination of the exterior surfaces of the strainer assembly for any evidence of debris, structural distress or abnormal corrosion. The intent of this surveillance is to ensure the absence of any condition which could adversely affect strainer functionality. Surveillance performance does not require removal of any tophat modules or grating, but the strainer exteriors shall be visually inspected. This surveillance is not a commitment to inspect 100% of the surface area of all tophats, but a sufficiently detailed inspection of exterior strainer surfaces is required to establish a high confidence that no adverse conditions are present. The scope of inspection necessary to provide high confidence includes 100% of the strainer areas that can be accessed and inspected using normal means and tools (i.e., flashlight, extendable mirror, hand held digital camera) without disassembly, and that difficult to access areas will be inspected to the extent possible using these same means.

Any damage detected in the strainer assembly inspection will result in an expansion of the scope of the inspection to include other areas of potential damage. Inspection scope should be expanded, as needed, for degradation of strainer components identified during this inspection that were not considered readily accessible during the inspector's initial evaluation.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 35.
2. 10 CFR 50.46.
3. UFSAR, Section 6.2.1.
4. UFSAR, Chapter 15.
5. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
6. NRC Memorandum to V. Stello, Jr., from R.L. Baer, "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975.
7. IE Information Notice No. 87-01.

BASES

ACTIONS (continued)

C.1 and C.2

If the RWST cannot be returned to OPERABLE status within the associated 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 3.5.4.1

The RWST borated water temperature should be verified ~~every 24 hours~~ to be within the limits assumed in the accident analyses band. ~~This~~ The ~~Frequency is sufficient to identify a temperature change that would approach either limit and has been shown to be acceptable through operating experience.~~ Insert 2

of ~~24~~ hours

For SFCP addition only

For SFCP Addition only

SR 3.5.4.2

The RWST water volume should be verified ~~every 7 days~~ to be above the required minimum level in order to ensure that a sufficient initial supply is available for injection and to support continued ECCS and Containment Spray System pump operation on recirculation. ~~Since the RWST volume is normally stable and is protected by an alarm, a 7 day Frequency is appropriate and has been shown to be acceptable through operating experience.~~ Insert 2

SR 3.5.4.3

The boron concentration of the RWST should be verified ~~every 7 days~~ to be within the required limits. This SR ensures that the reactor will remain subcritical following a LOCA and that the boron content assumed for the injection water in the MSLB analysis is available. Further, it assures that the resulting sump pH will be maintained in an acceptable range so that boron precipitation in the core will not occur and the effect of chloride and caustic stress corrosion on mechanical systems and components will be minimized. ~~Since the RWST volume is normally stable, a 7 day sampling Frequency to verify boron concentration is appropriate and has been shown to be acceptable through operating experience.~~

INSERT 2

BASES

ACTIONS (continued)

operator has 4 hours from the time the flow is known to be above the limit to correctly position the manual valves and thus be in compliance with the accident analysis. The Completion Time minimizes the potential exposure of the plant to a LOCA with insufficient injection flow and provides a reasonable time to restore seal injection flow within limits. This time is conservative with respect to the Completion Times of other ECCS LCOs; it is based on operating experience and is sufficient for taking corrective actions by operations personnel.

B.1 and B.2

When the Required Actions cannot be completed within the required Completion Time, a controlled shutdown must be initiated. The Completion Time of 6 hours for reaching MODE 3 from MODE 1 is a reasonable time for a controlled shutdown, based on operating experience and normal cooldown rates, and does not challenge plant safety systems or operators. Continuing the plant shutdown begun in Required Action B.1, an additional 6 hours is a reasonable time, based on operating experience and normal cooldown rates, to reach MODE 4, where this LCO is no longer applicable.

SURVEILLANCE REQUIREMENTS

SR 3.5.5.1

Verification ~~every 31 days~~ that the manual seal injection throttle valves are adjusted to give a flow within the limit ensures that proper manual seal injection throttle valve position, and hence, proper seal injection flow, is maintained. ~~The Frequency of 31 days is based on engineering judgment and is consistent with other ECCS valve Surveillance Frequencies. The Frequency has proven to be acceptable through operating experience.~~ Insert 2

As noted, the Surveillance is required to be performed within 4 hours after the RCS pressure has stabilized within a ± 20 psig range of normal operating pressure. The RCS pressure requirement is specified since this configuration will produce the required pressure conditions necessary to assure that the manual valves are set correctly. The exception is limited to 4 hours to ensure that the Surveillance is timely.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.2.2

Door seals must be tested ~~every 6 months~~ to verify the integrity of the inflatable door seal. The measured leakage rate must be less than 15 standard cubic centimeters per minute (sccm) per door seal when the seal is inflated to approximately 85 psig. This ensures that the seals will remain inflated for at least 7 days should the instrument air supply to the seals be lost. ~~The Frequency of testing has been demonstrated to be acceptable through operating experience.~~ INSERT 2

6 months

For SFCP Addition only

SR 3.6.2.3

The air lock interlock is designed to prevent simultaneous opening of both doors in a single air lock. Since both the inner and outer doors of an air lock are designed to withstand the maximum expected post accident containment pressure, closure of either door will support containment OPERABILITY. Thus, the door interlock feature supports containment OPERABILITY while the air lock is being used for personnel transit in and out of the containment. Periodic testing of this interlock demonstrates that the interlock will function as designed and that simultaneous opening of the inner and outer doors will not inadvertently occur. ~~Due to the purely mechanical nature of this interlock, and given that the interlock mechanism is not normally challenged when the containment air lock door is used for entry and exit (procedures require strict adherence to single door opening), this test is only required to be performed every 18 months. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage, and the potential for loss of containment OPERABILITY if the surveillance were performed with the reactor at power. The 18 month Frequency for the interlock is justified based on generic operating experience. The Frequency is based on engineering judgment and is considered adequate given that the interlock is not challenged during the use of the interlock.~~

BASES

ACTIONS (continued)

For the isolation devices inside containment, the time period specified as "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is based on engineering judgment and is considered reasonable in view of the inaccessibility of the isolation devices and other administrative controls that will ensure that isolation device misalignment is an unlikely possibility.

For the valve with resilient seal that is isolated in accordance with Required Action E.1, SR 3.6.3.6 must be performed at least once every 92 days. This assures that degradation of the resilient seal is detected and confirms that the leakage rate of the containment purge valve does not increase during the time the penetration is isolated.

F.1 and F.2

If the Required Actions and associated Completion Times are not met, 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 3.6.3.1

Each containment purge supply and exhaust isolation valve for the lower compartment and the upper compartment, instrument room, and the Hydrogen Purge System is required to be verified sealed closed at 31 day intervals. This Surveillance is designed to ensure that a gross breach of containment is not caused by an inadvertent or spurious opening of a containment purge valve. Detailed analysis of these valves to conclusively demonstrate their ability to close during a LOCA in time to limit offsite doses has not been performed. Therefore, these valves are required to be in the sealed closed position during MODES 1, 2, 3, and 4. A valve that is sealed closed must have motive power to the valve operator removed. This can be accomplished by de-energizing the source of electric power or by removing the air supply to the valve operator. In this application, the term "sealed" has no connotation of leak tightness.

BASES

SURVEILLANCE REQUIREMENTS (continued)

of 31 days For SFCP addition only

~~The Frequency is a result of an NRC initiative, Generic Issue B-24 (Ref. 5) related to containment purge valve use during plant operations.~~ INSERT 2 In the event valve leakage requires entry into Condition E, the Surveillance permits opening one valve in a penetration flow path to perform repairs.

SR 3.6.3.2

This SR ensures that the Containment Air Release and Addition System isolation valves are closed as required or, if open, open for an allowable reason. If a valve is open in violation of this SR, the valve is considered inoperable. If the inoperable valve is not otherwise known to have excessive leakage when closed, it is not considered to have leakage outside of limits. The SR is not required to be met when the valves are open for the reasons stated. The valves may be opened for pressure control, ALARA or air quality considerations for personnel entry, or for Surveillances that require the valves to be open. The valves are capable of closing in the environment following a LOCA. Therefore, these valves are allowed to be open for limited periods of time. ~~The 31 day Frequency is consistent with other containment isolation valve requirements discussed in SR 3.6.3.3.~~ INSERT 2

SR 3.6.3.3

This SR requires verification that each containment isolation manual valve and blind flange located outside containment or annulus and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside of the containment boundary is within design limits. This SR does not require any testing or valve manipulation. Rather, it involves verification, through system walkdown or computer status indication, that those containment isolation valves outside containment and capable of being mispositioned are in the correct position. ~~Since verification of valve position for containment isolation valves outside containment is relatively easy, the 31 day Frequency is based on engineering judgment and was chosen to provide added assurance of the correct positions.~~ INSERT 2 The SR specifies that containment isolation valves that are open under administrative controls are not required to meet the SR during the time the valves are open.

SURVEILLANCE REQUIREMENTS (continued)

This SR does not apply to valves that are locked, sealed, or otherwise secured in the closed position, since these were verified to be the correct position upon locking, sealing, or securing.

The Note applies to valves and blind flanges located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, 3 and 4 for ALARA reasons. Therefore, the probability of misalignment of these containment isolation valves, once they have been verified to be in the proper position, is small.

SR 3.6.3.4

This SR requires verification that each containment isolation manual valve and blind flange located inside containment or annulus and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside of the containment boundary is within design limits. For containment isolation valves inside containment, the Frequency of "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is appropriate since these containment isolation valves are operated under administrative controls and the probability of their misalignment is low. The SR specifies that containment isolation valves that are open under administrative controls are not required to meet the SR during the time they are open. This SR does not apply to valves that are locked, sealed, or otherwise secured in the closed position, since these were verified to be the correct position upon locking, sealing, or securing.

This Note allows valves and blind flanges located in high radiation areas to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, 3, and 4, for ALARA reasons. Therefore, the probability of misalignment of these containment isolation valves, once they have been verified to be in their proper position, is small.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.3.5

Verifying that the isolation time of each automatic power operated containment isolation valve is within limits is required to demonstrate OPERABILITY. The isolation time test ensures the valve will isolate in a time period less than or equal to that assumed in the safety analyses. The isolation time is specified in the UFSAR and the Frequency of this SR is in accordance with the Inservice Testing Program.

SR 3.6.3.6

For the Containment Purge System valves with resilient seals, additional leakage rate testing beyond the test requirements of 10 CFR 50, Appendix J, Option B is required to ensure OPERABILITY. The measured leakage rate for the Containment Purge System and Hydrogen Purge System valves must be $\leq 0.05\ L_a$ when pressurized to P_a. Operating experience has demonstrated that this type of seal has the potential to degrade in a shorter time period than other seal types. Based on this observation and the importance of maintaining this penetration leak tight (due to the direct path between containment and the environment), these valves will not be placed on the maximum extended test interval. Therefore, these valves will be tested in accordance with Regulatory Guide 1.163, which allows a maximum test interval of 30 months.

The Containment Air Release and Addition System valves have a demonstrated history of acceptable leakage. The measured leakage rate for containment air release and addition valves must be $\leq 0.01\ L_a$ when pressurized to P_a. These valves will be tested in accordance with Regulatory Guide 1.163, which allows a maximum test interval of 30 months.

SR 3.6.3.7

Automatic containment isolation valves close on a containment isolation signal to prevent leakage of radioactive material from containment following a DBA. This SR ensures that each automatic containment isolation valve will actuate to its isolation position on a containment

BASES

SURVEILLANCE REQUIREMENTS (continued)

isolation signal. The isolation signals involved are Phase A, Phase B, and Safety Injection. This surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. ~~The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass this Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.~~ Insert 2

SR 3.6.3.8

This SR ensures that the combined leakage rate of all reactor building bypass leakage paths is less than or equal to the specified leakage rate. This provides assurance that the assumptions in the safety analysis are met. The Frequency is required by the Containment Leakage Rate Testing Program. This SR simply imposes additional acceptance criteria.

Bypass leakage is considered part of L_a.

REFERENCES

1. UFSAR, Section 15.
2. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
3. UFSAR, Section 6.2.
4. Standard Review Plan 6.2.4.
5. Generic Issue B-24.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1

Verifying that containment pressure is within limits ensures that unit operation remains within the limits assumed in the containment analysis. ~~The 12 hour Frequency of this SR was developed based on operating experience related to trending of containment pressure variations during the applicable MODES. Furthermore, the 12 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal containment pressure condition.~~ Insert 2

REFERENCES

1. UFSAR, Section 6.2.
2. 10 CFR 50, Appendix K.
3. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.5.1 and SR 3.6.5.2

Verifying that containment average air temperature is within the LCO limits ensures that containment operation remains within the limits assumed for the containment analyses. In order to determine the containment average air temperature, an arithmetical average of ambient air temperature monitoring stations is calculated using measurements taken at locations within the containment selected to provide a representative sample of the overall containment atmosphere. The upper compartment measurements should be taken at elevation 653 feet at the inlet of each operating upper containment ventilation unit. The lower compartment measurements should be taken at elevation 570 feet at the inlet of each operating lower containment ventilation unit. ~~The 24 hour Frequency of these SRs is considered acceptable based on observed slow rates of temperature increase within containment as a result of environmental heat sources (due to the large volume of containment). Furthermore, the 24 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal containment temperature condition.~~

Insert 2

REFERENCES

1. UFSAR, Section 6.2.
2. 10 CFR 50.49.
3. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).

BASES

ACTIONS

A.1

With one containment spray train inoperable, the affected train must be restored to OPERABLE status within 72 hours. The components in this degraded condition are capable of providing 100% of the heat removal after an accident. The 72 hour Completion Time was developed taking into account the redundant heat removal and iodine removal capabilities afforded by the OPERABLE train and the low probability of a DBA occurring during this period.

B.1 and B.2

If the affected containment spray train 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 84 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. The extended interval to reach MODE 5 allows additional time and is reasonable when considering that the driving force for a release of radioactive material from the Reactor Coolant System is reduced in MODE 3.

SURVEILLANCE REQUIREMENTS

SR 3.6.6.1

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

For SFCP Addition only.

Verifying the correct alignment of manual, power operated, and automatic valves, excluding check valves, in the Containment Spray System provides assurance that the proper flow path exists for Containment Spray System operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position since they were verified in the correct position prior to being secured. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown or computer status indication, that those valves outside containment and capable of potentially being mispositioned, are in the correct position. INSERT 2

SR 3.6.6.2

Verifying that each containment spray pump's developed head at the flow test point is greater than or equal to the required developed head

BASES

SURVEILLANCE REQUIREMENTS (continued)

ensures that spray pump performance has not degraded during the cycle. Flow and differential head are normal tests of centrifugal pump performance required by the ASME Code (Ref. 6). Since the containment spray pumps cannot be tested with flow through the spray headers, they are tested on bypass flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice inspections confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this SR is in accordance with the Inservice Testing Program.

SR 3.6.6.3 and SR 3.6.6.4

These SRs require verification that each automatic containment spray valve actuates to its correct position and each containment spray pump starts upon receipt of an actual or simulated Containment Phase B Isolation signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. ~~The 18 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillances were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillances when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.~~ INSERT 2

The surveillance of containment sump isolation valves is also required by SR 3.6.6.3. A single surveillance may be used to satisfy both requirements.

SR 3.6.6.5 and SR 3.6.6.6

These SRs require verification of proper interaction between the CPCS system and the Containment Spray System.

SR 3.6.6.5 deals solely with the containment spray pumps. It must be shown through testing that: (1) the containment spray pumps are prevented from starting in the absence of a CPCS permissive, (2) the containment spray pumps start when given a CPCS permissive, and (3) when running, the containment spray pumps stop when the CPCS permissive is removed. The "inhibit", "permit", and "terminate" parts of the CPCS interface with the containment spray pumps are verified by

BASES

SURVEILLANCE REQUIREMENTS (continued)

testing in this fashion.

SR 3.6.6.6 deals solely with containment spray header containment isolation valves NS12B, NS15B, NS29A, and NS32A. It must be shown through testing that: (1) each valve closes when the CPCS permissive is removed, OR (2) each valve is prevented from opening in the absence of a CPCS permissive. In addition to one of the above, it must also be shown that each valve opens when given a CPCS permissive.

~~The 18 month Frequency is appropriate based on the reliability of the components.~~ INSERT 2

SR 3.6.6.7

With the containment spray inlet valves closed and the spray header drained of any solution, low pressure air or smoke can be blown through test connections. The spray nozzles can also be periodically tested using a vacuum blower to induce air flow through each nozzle to verify unobstructed flow. This SR ensures that each spray nozzle is unobstructed and that spray coverage of the containment during an accident is not degraded. ~~Because of the passive design of the nozzle, a test at 10 year intervals is considered adequate to detect obstruction of the spray nozzles.~~ INSERT 2

REFERENCES

1. 10 CFR 50, Appendix A, GDC 38, GDC 39, GDC 40, GDC 41, GDC 42, and GDC 43.
2. UFSAR, Section 6.2.
3. 10 CFR 50.49.
4. 10 CFR 50, Appendix K.
5. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
6. ASME Code for Operation and Maintenance of Nuclear Power Plants.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.8.1

Operating each HSS train for ≥ 15 minutes ensures that each train is OPERABLE and that all associated controls are functioning properly. It also ensures that blockage, fan and/or motor failure, or excessive vibration can be detected for corrective action. ~~The 92 day Frequency is consistent with Inservice Testing Program Surveillance Frequencies, operating experience, the known reliability of the fan motors and controls, and the two train redundancy available.~~ INSERT 2

SR 3.6.8.2

Verifying HSS fan motor current at rated speed with the motor operated suction valves closed is indicative of overall fan motor performance. Since these fans are required to function during post-accident situations, the air density that the fans experience during surveillance testing will be different than the air density following a LOCA. An air density adjustment will be made to the average fan motor current test data before it is compared to the Technical Specification SR acceptance criteria. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. ~~The Frequency of 92 days was based on operating experience which has shown this Frequency to be acceptable.~~ INSERT 2

SR 3.6.8.3

This SR verifies the motor operated suction valves open upon receipt of a Containment Pressure – High High signal and associated time delay and that the HSS fans receive a start permissive when the valves start to open. ~~The Frequency of 92 days was based on operating experience, which has shown this Frequency to be acceptable.~~

INSERT 2

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.8.4

This SR ensures that each HSS train responds properly to a containment pressure high-high actuation signal. The Surveillance verifies that each fan starts after a delay of ≥ 8 minutes and ≤ 10 minutes. ~~The Frequency of 92 days conforms with the testing requirements for similar ESF equipment and considers the known reliability of fan motors and controls and the two train redundancy available. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.~~ INSERT 2

REFERENCES

1. 10 CFR 50.44.
2. 10 CFR 50, Appendix A, GDC 41, 42, and 43.
3. Regulatory Guide 1.7, Revision 2.
4. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).

BASES

ACTIONS (continued)

length of time after the event that operator action would be required to prevent hydrogen accumulation from exceeding this limit, and the low probability of failure of the OPERABLE HIS train. Alternative Required Action A.2, by frequent surveillances, provides assurance that the OPERABLE train continues to be OPERABLE.

B.1

Condition B is one containment region with no OPERABLE hydrogen ignitor. Thus, while in Condition B, or in Conditions A and B simultaneously, there would always be ignition capability in the adjacent containment regions that would provide redundant capability by flame propagation to the region with no OPERABLE ignitors.

Required Action B.1 calls for the restoration of one hydrogen ignitor in each region to OPERABLE status within 7 days. The 7 day Completion Time is based on the same reasons given under Required Action A.1.

C.1

The unit must be placed in a MODE in which the LCO does not apply if the HIS subsystem(s) cannot be restored to OPERABLE status within the associated Completion Time. This is done by placing the unit in at least MODE 3 within 6 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.9.1

This SR confirms that ≥ 34 of 35 hydrogen ignitors can be successfully energized in each train. The ignitors are simple resistance elements. Therefore, energizing provides assurance of OPERABILITY. The allowance of one inoperable hydrogen ignitor is acceptable because, although one inoperable hydrogen ignitor in a region would compromise redundancy in that region, the containment regions are interconnected so that ignition in one region would cause burning to progress to the others (i.e., there is overlap in each hydrogen ignitor's effectiveness between regions). ~~The Frequency of 92 days has been shown to be acceptable through operating experience.~~ INSERT 2

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.9.2

This SR confirms that the two inoperable hydrogen ignitors allowed by SR 3.6.9.1 (i.e., one in each train) are not in the same containment region. ~~The Frequency of 92 days is acceptable based on the Frequency of SR 3.6.9.1, which provides the information for performing this SR.~~

INSERT 2

SR 3.6.9.3

A more detailed functional test is performed ~~every 18 months~~ to verify system OPERABILITY. Each ignitor is visually examined to ensure that it is clean and that the electrical circuitry is energized. All ignitors, including normally inaccessible ignitors, are visually checked for a glow to verify that they are energized. Additionally, the surface temperature of each ignitor is measured in calm, nonturbulent atmospheric conditions to be ≥ 1700°F to demonstrate that a temperature sufficient for ignition is achieved. The 1700°F temperature is a surveillance requirement. "An Analysis of Hydrogen Control Measures at McGuire Nuclear Station" (Ref. 5) section 3.8 identifies that the required normal operation temperature is 1500°F. Therefore, based upon ignitor performance testing conducted at Catawba, the surveillance requirement of 1700°F ensures that sufficient margin is present for continued hydrogen ignition under degraded bus conditions. ~~The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the SR when performed at the 18 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.~~

INSERT 2

REFERENCES

1. 10 CFR 50.44.
2. 10 CFR 50, Appendix A, GDC 41.
3. UFSAR, Section 6.2.
4. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
5. An Analysis of Hydrogen Control Measures at McGuire Nuclear Station.

BASES

ACTIONS (continued)

C.1 and C.2

If the AVS train 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 3.6.10.1

Operating each AVS train from the control room with flow through the HEPA filters and carbon adsorbers ensures that all trains are OPERABLE and that all associated controls are functioning properly. It also ensures that blockage, fan or motor failure, or excessive vibration can be detected for corrective action. Operation with the heaters on for ≥ 10 continuous hours eliminates moisture on the adsorbers and HEPA filters. Experience from filter testing at operating units indicates that the 10 hour period is adequate for moisture elimination on the adsorbers and HEPA filters. ~~The 31 day Frequency was developed in consideration of the known reliability of fan motors and controls, the two train redundancy available, and the iodine removal capability of the Containment Spray System and Ice Condenser.~~ INSERT 2

SR 3.6.10.2

This SR verifies that the required AVS filter testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The AVS filter tests are in accordance with Regulatory Guide 1.52 (Ref. 5). The VFTP includes testing HEPA filter performance, carbon adsorber efficiency, system flow rate, and the physical properties of the activated carbon (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the VFTP.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.10.3

The automatic startup on a safety injection signal ensures that each AVS train responds properly. ~~The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore the Frequency was concluded to be acceptable from a reliability standpoint. Furthermore, the SR interval was developed considering that the AVS equipment OPERABILITY is demonstrated at a 31 day Frequency by SR 3.6.10.1.~~ INSERT 2

SR 3.6.10.4

The AVS filter cooling electric motor-operated bypass valves are tested to verify OPERABILITY. The valves are normally closed and may need to be opened from the control room to initiate miniflow cooling through a filter unit that has been shutdown following a DBA LOCA. Miniflow cooling may be necessary to limit temperature increases in the idle filter train due to decay heat from captured fission products. ~~The 18 month Frequency is considered to be acceptable based on valve reliability and design, and the fact that operating experience has shown that the valves usually pass the Surveillance when performed at the 18 month Frequency.~~ INSERT 2

SR 3.6.10.5

The proper functioning of the fans, dampers, filters, adsorbers, etc., as a system is verified by the ability of each train to produce the required system flow rate. ~~The 18 month Frequency is consistent with Regulatory Guide 1.52 (Ref. 5) guidance for functional testing.~~ INSERT 2

SR 3.6.10.6

The ability of the AVS train to produce the required negative pressure of at least -0.88 inch water gauge when corrected to elevation 564 feet ensures that the annulus negative pressure is at least -0.25 inch water gauge everywhere in the annulus. The -0.88 inch water gauge annulus pressure includes a correction for an outside air temperature induced hydrostatic pressure gradient of -0.63 inch water gauge. The negative

BASES

SURVEILLANCE REQUIREMENTS (continued)

pressure prevents unfiltered leakage from the reactor building, since outside air will be drawn into the annulus by the negative pressure differential.

The CANVENT computer code is used to model the thermal effects of a LOCA on the annulus and the ability of the AVS to develop and maintain a negative pressure in the annulus after a design basis accident. The annulus pressure drawdown time during normal plant conditions is not an input to any dose analyses. Therefore, the annulus pressure drawdown time during normal plant conditions is insignificant.

The AVS trains are tested ~~every 18 months~~ to ensure each train will function as required. ~~Operating experience has shown that each train usually passes the surveillance when performed at the 18 month Frequency.~~ INSERT 2 Furthermore, the SR interval was developed considering that the AVS equipment OPERABILITY is demonstrated at a 31 day Frequency by SR 3.6.10.1. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 41.
2. UFSAR, Sections 6.2.3 and 9.4.9.
3. UFSAR, Chapter 15.
4. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
5. Regulatory Guide 1.52, Revision 2.
6. Catawba Nuclear Station License Amendments 90/84 for Units 1/2, August 23, 1991.
7. NUREG-0800, Sections 6.2.3 and 6.5.3, Rev. 2, July 1981.

BASES

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause an increase in containment pressure and temperature requiring the operation of the ARS. Therefore, the LCO is applicable in MODES 1, 2, 3, and 4.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the ARS is not required to be OPERABLE in these MODES.

ACTIONS

A.1

If one of the required trains of the ARS is inoperable, it must be restored to OPERABLE status within 72 hours. The 72 hour Completion Time was developed taking into account the redundant flow of the OPERABLE ARS train and the low probability of a DBA occurring in this period.

B.1 and B.2

If the ARS train 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 3.6.11.1

Verifying that each ARS fan starts on an actual or simulated actuation signal, after a delay ≥ 8 minutes and ≤ 10 minutes, and operates for ≥ 15 minutes is sufficient to ensure that all fans are OPERABLE and that all associated controls and time delays are functioning properly. It also ensures that blockage, fan and/or motor failure, or excessive vibration can be detected for corrective action. ~~The 92 day Frequency was developed considering the known reliability of fan motors and controls and the two train redundancy available.~~ INSERT 2

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.11.2

Verifying ARS fan motor current at rated speed with the return air dampers closed confirms one operating condition of the fan. This test is indicative of overall fan motor performance. Since these fans are required to function during post-accident situations, the air density that the fans experience during surveillance testing will be different than the air density following a LOCA. An air density adjustment will be made to the average fan motor current test data before it is compared to the Technical Specification SR acceptance criteria. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. ~~The Frequency of 92 days conforms with the testing requirements for similar ESF equipment and considers the known reliability of fan motors and controls and the two train redundancy available.~~ INSERT 2

SR 3.6.11.3

Verifying the OPERABILITY of the return air damper provides assurance that the proper flow path will exist when the fan is started. This Surveillance also tests the circuitry, including time delays to ensure the system operates properly. ~~The Frequency of 92 days was developed considering the importance of the dampers, their location, physical environment, and probability of failure. Operating experience has also shown this Frequency to be acceptable.~~ INSERT 2

SR 3.6.11.4 and SR 3.6.11.5

Verifying the OPERABILITY of the check damper in the air return fan discharge line to the containment lower compartment provides assurance that the proper flow path will exist when the fan is started and that reverse flow can not occur when the fan is not operating. ~~The Frequency of 92 days was developed considering the importance of the dampers, their location, physical environment, and probability of failure. Operating experience has also shown this Frequency to be acceptable.~~ INSERT 2

~~the~~ 92 day
For SFCP Addition only

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.11.6 and SR 3.6.11.7

These SRs require verification that each ARS motor operated damper is allowed to open or is prevented from opening and each ARS fan is allowed to start or is de-energized or prevented from starting based on the presence or absence of Containment Pressure Control System start permissive and terminate signals. The CPCS is described in the Bases for LCO 3.3.2, "ESFAS." ~~The 18 month Frequency is based on operating experience which has shown it to be acceptable.~~ INSERT 2

REFERENCES

1. UFSAR, Section 6.2.

2. 10 CFR 50, Appendix K.

3. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).

4. NRC Bulletin 2003-01, "Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized Water Reactors."

BASES

ACTIONS

A.1

If the ice bed is inoperable, it must be restored to OPERABLE status within 48 hours. The Completion Time was developed based on operating experience, which confirms that due to the very large mass of stored ice, the parameters comprising OPERABILITY do not change appreciably in this time period. Because of this fact, the Surveillance Frequencies are long (months), except for the ice bed temperature, which is checked every 12 hours. If a degraded condition is identified, even for temperature, with such a large mass of ice it is not possible for the degraded condition to significantly degrade further in a 48 hour period. Therefore, 48 hours is a reasonable amount of time to correct a degraded condition before initiating a shutdown.

B.1 and B.2

If the ice bed 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 3.6.12.1

Verifying that the maximum temperature of the ice bed is ≤ 27°F ensures that the ice is kept well below the melting point. ~~The 12 hour Frequency was based on operating experience, which confirmed that, due to the large mass of stored ice, it is not possible for the ice bed temperature to degrade significantly within a 12 hour period and was also based on assessing the proximity of the LCO limit to the melting temperature.~~

~~Furthermore, the 12 hour Frequency is considered adequate in view of indications in the control room, including the alarm, to alert the operator to an abnormal ice bed temperature condition.~~ INSERT 2 This SR may be satisfied by use of the Ice Bed Temperature Monitoring System.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.12.2

This SR ensures that initial ice fill and any subsequent ice additions meet the boron concentration and pH requirements of SR 3.6.12.7. The SR is modified by a NOTE that allows the chemical analysis to be performed on either the liquid or resulting ice of each sodium tetraborate solution prepared. If ice is obtained from offsite sources, then chemical analysis data must be obtained for the ice supplied.

SR 3.6.12.3

This SR ensures that the air/steam flow channels through the ice bed have not accumulated ice blockage that exceeds 15 percent of the total flow area through the ice bed region. The allowable 15 percent buildup of ice is based on the analysis of the sub-compartment response to a design basis LOCA with partial blockage of the ice condenser flow channels. The analysis did not perform detailed flow area modeling, but rather lumped the ice condenser bays into six sections ranging from 2.75 bays to 6.5 bays. Individual bays are acceptable with greater than 15 percent blockage, as long as 15 percent blockage is not exceeded for any analysis section.

To provide a 95 percent confidence that flow blockage does not exceed the allowed 15 percent, the visual inspection must be made for at least 54 (33 percent) of the 162 flow channels per ice condenser bay. The visual inspection of the ice bed flow channels is to inspect the flow area, by looking down from the top of the ice bed, and where view is achievable up from the bottom of the ice bed. Flow channels to be inspected are determined by random sample. As the most restrictive ice bed flow passage is found at a lattice frame elevation, the 15 percent blockage criteria only applies to "flow channels" that comprise the area:

a. between ice baskets, and

b. past lattice frames and wall panels.

Due to a significantly larger flow area in the regions of the upper deck grating and the lower inlet plenum support structures and turning vanes, it would require a gross buildup of ice on these structures to obtain a degradation in air/steam flow. Therefore, these structures are excluded as part of a flow channel for application of the 15 percent blockage criteria. Plant and industry experience have shown that removal of ice from the excluded structures during the refueling outage is sufficient to

BASES

SURVEILLANCE REQUIREMENTS (continued)

ensure they remain operable throughout the operating cycle. Thus, removal of any gross ice buildup on the excluded structures is performed following outage maintenance activities.

Operating experience has demonstrated that the ice bed is the region that is the most flow restrictive, due to the normal presence of ice accumulation on lattice frames and wall panels. The flow area through the ice basket support platform is not a more restrictive flow area because it is easily accessible from the lower plenum and is maintained clear of ice accumulation. There is not a mechanistically credible method for ice to accumulate on the ice basket support platform during plant operation.

Plant and industry experience has shown that the vertical flow area through the ice basket support platform remains clear of ice accumulation that could produce blockage. Normally only a glaze may develop or exist on the ice basket support platform which is not significant to blockage of flow area. Additionally, outage maintenance practices provide measures to clear the ice basket support platform following maintenance activities of any accumulation of ice that could block flow areas.

Activities that have a potential for significant degradation of flow channels should be limited to outage periods. Performance of this SR following completion of these activities assures the ice bed is in an acceptable condition for the duration of the operating cycle.

FOR SFCP Addition only (SR 3.6.12.3)

Frost buildup or loose ice is not to be considered as flow channel blockage, whereas attached ice is considered blockage of a flow channel. Frost is the solid form of water that is loosely adherent, and can be brushed off with the open hand. INSERT 2

~~Operating and maintenance experience has shown that, with the 18 month frequency, accumulation of ice on structural members will not compromise the Safety Analysis.~~

SR 3.6.12.4

Ice mass determination methodology is designed to verify the total as-found (pre-maintenance) mass of ice in the ice bed, and the appropriate distribution of that mass, using a random sampling of individual baskets. The random sample will include at least 30 baskets from each of three defined Radial Zones (at least 90 baskets total). Radial Zone A consists of baskets located in rows 8, and 9 (innermost rows adjacent to the Crane Wall), Radial Zone B consists of baskets located in rows 4, 5, 6, and 7 (middle rows of the ice bed), and Radial Zone C consists of baskets located in rows 1, 2, and 3 (outermost rows adjacent to the Containment Vessel).

BASES

SURVEILLANCE REQUIREMENTS (continued)

The Radial Zones chosen include the row groupings nearest the inside and outside walls of the ice bed and the middle rows of the ice bed. These groupings facilitate the statistical sampling plan by creating sub-populations of ice baskets that have similar mean mass and sublimation characteristics.

Methodology for determining sample ice basket mass will be either by direct lifting or by alternative techniques. Any method chosen will include procedural allowances for the accuracy of the method used. The number of sample baskets in any Radial Zone may be increased once by adding 20 or more randomly selected baskets to verify the total mass of that Radial Zone.

In the event the mass of a selected basket in a sample population (initial or expanded) cannot be determined by any available means (e.g., due to surface ice accumulation or obstruction), a randomly selected representative alternate basket may be used to replace the original selection in that sample population. If employed, the representative alternate must meet the following criteria:

a. Alternate selection must be from the same bay-Zone (i.e., same bay, same Radial Zone) as the original selection, and

b. Alternate selection cannot be a repeated selection (original or alternate) in the current Surveillance, and cannot have been used as an analyzed alternate selection in the three most recent Surveillances.

The complete basis for the methodology used in establishing the 95% confidence level in the total ice bed mass is documented in Ref. 5.

The total ice mass and individual Radial Zone ice mass requirements defined in this Surveillance, and the minimum ice mass per basket requirement defined by SR 3.6.12.5, are the minimum requirements for OPERABILITY. Additional ice mass beyond the SRs is maintained to address sublimation. This sublimation allowance is generally applied to baskets in each Radial Zone, as appropriate, at the beginning of an operating cycle to ensure sufficient ice is available at the end of the operating cycle for the ice condenser to perform its intended design function.

~~The Frequency of 18 months was based on ice storage tests, and the typical sublimation allowance maintained in the ice mass over and above the minimum ice mass assumed in the safety analyses. Operating and~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~maintenance experience has verified that, with the 18 month Frequency, the minimum mass and distribution requirements in the ice bed are maintained.~~ ~~INSERT 2~~

SR 3.6.12.5

Verifying that each selected sample basket from SR 3.6.12.4 contains at least 600 lbs of ice in the as-found (pre-maintenance) condition ensures that a significant localized degraded mass condition is avoided.

This SR establishes a per basket limit to ensure any ice mass degradation is consistent with the initial conditions of the DBA by not significantly affecting the containment pressure response. Ref. 5 provides insights through sensitivity runs that demonstrate that the containment peak pressure during a DBA is not significantly affected by the ice mass in a large localized region of baskets being degraded below the required safety analysis mean, when the Radial Zone and total ice mass requirements of SR 3.6.12.4 are satisfied. Any basket identified as containing less than 600 lbs of ice requires appropriately entering the TS Required Action for an inoperable ice bed due to the potential that it may represent a significant condition adverse to quality.

(SR 3.6.12.5) For addition to the SFCP only.

~~The frequency of 18 months was based on ice storage tests and the typical sublimation allowance maintained in the ice mass assumed in the safety analyses. Operating and Maintenance experience has verified that, with the 18 month frequency, the minimum mass and distribution requirements of the ice bed are maintained.~~

As documented in Ref. 5, maintenance practices actively manage individual ice basket mass above the required safety analysis mean for each Radial Zone. Specifically, each basket is serviced to keep its ice mass above 750 lbs for Radial Zone A, 1196 lbs for Radial Zone B, and 1196 lbs for Radial Zone C. If a basket sublimates below the safety analysis mean value, this instance is identified within the plant's corrective action program, including evaluating maintenance practices to identify the cause and correct any deficiencies. These maintenance practices provide defense in depth beyond compliance with the ice bed surveillance requirements by limiting the occurrence of individual baskets with ice mass less than the required safety analysis mean.

INSERT 2

SR 3.6.12.6

This SR ensures that a representative sampling of accessible portions of ice baskets, which are relatively thin walled, perforated cylinders, have not been degraded by wear, cracks, corrosion, or other damage. The SR is designed around a full-length inspection of a sample of baskets, and is intended to monitor the effect of the ice condenser environment on ice baskets. The groupings defined in the SR (two baskets in each

BASES

SURVEILLANCE REQUIREMENTS (continued)

azimuthal third of the ice bed) ensure that the sampling of baskets is reasonably distributed. The Frequency of 40 months for a visual inspection of the structural soundness of the ice baskets is based on engineering judgment and considers such factors as the thickness of the basket walls relative to corrosion rates expected in their service environment and the results of the long term ice storage testing.

INSERT 2

SR 3.6.12.7

Verifying the chemical composition of the stored ice ensures that the stored ice has a boron concentration ≥ 1800 ppm and ≤ 2330 ppm as sodium tetraborate and a high pH, ≥ 9.0 and ≤ 9.5 at 25°C, in order to meet the requirement for borated water when the melted ice is used in the ECCS recirculation mode of operation. Additionally, the minimum boron concentration setpoint is used to assure reactor subcriticality in a post LOCA environment, while the maximum boron concentration is used as the bounding value in the hot leg switchover timing calculation (Ref. 4). This is accomplished by obtaining at least 24 ice samples. Each sample is taken approximately one foot from the top of the ice of each randomly selected ice basket in each ice condenser bay. The SR is modified by a NOTE that allows the boron concentration and pH value obtained from averaging the individual samples' analysis results to satisfy the requirements of the SR. If either the average boron concentration or average pH value is outside their prescribed limit, then entry into ACTION Condition A is required. Sodium tetraborate has been proven effective in maintaining the boron content for long storage periods, and it also enhances the ability of the solution to remove and retain fission product iodine. The high pH is required to enhance the effectiveness of the ice and the melted ice in removing iodine from the containment atmosphere. This pH range also minimizes the occurrence of chloride and caustic stress corrosion on mechanical systems and components exposed to ECCS and Containment Spray System fluids in the recirculation mode of operation. The Frequency of 54 months is intended to be consistent with the expected length of three fuel cycles, and was developed considering these facts:

a. Long term ice storage tests have determined that the chemical composition of the stored ice is extremely stable;

b. There are no normal operating mechanisms that significantly change the boron concentration of the stored ice, and pH remains within a 9.0 – 9.5 range when boron concentrations are above approximately 1200 ppm; and

BASES

SURVEILLANCE REQUIREMENTS (continued)

c. ~~Operating experience has demonstrated that meeting the boron concentration and pH requirements has not been a problem.~~ INSERT 2

REFERENCES

1. UFSAR, Section 6.2.
2. 10 CFR 50, Appendix K.
3. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
4. UFSAR, Section 6.3.3.
5. Topical Report ICUG-001, Application of the Active Ice Mass Management Concept to the Ice Condenser Ice Mass Technical Specification, Revision 2.
6. UFSAR, Section 18, Table 18-1.
7. Catawba License Renewal Commitments, CNS-1274.00-00-0016, Section 4.17.

BASES

ACTIONS (continued)

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 3.6.13.1

Verifying, by means of the Inlet Door Position Monitoring System, that the inlet doors are in their closed positions makes the operator aware of an inadvertent opening of one or more doors. ~~The Frequency of 12 hours ensures that operators on each shift are aware of the status of the doors.~~ INSERT 2

SR 3.6.13.2

Verifying, by visual inspection, that each intermediate deck door is closed and not impaired by ice, frost, or debris provides assurance that the intermediate deck doors (which form the floor of the upper plenum where frequent maintenance on the ice bed is performed) have not been left open or obstructed. In determining if a door is impaired by ice, the frost accumulation on the doors, joints, and hinges are to be considered in conjunction with the lifting force limits of SR 3.6.13.7. ~~The Frequency of 7 days is based on engineering judgment and takes into consideration such factors as the frequency of entry into the intermediate ice condenser deck, the time required for significant frost buildup, and the probability that a DBA will occur.~~ INSERT 2

SR 3.6.13.3

Verifying, by visual inspection, that the top deck doors are in place and not obstructed provides assurance that the doors are performing their function of keeping warm air out of the ice condenser during normal operation, and would not be obstructed if called upon to open in response to a DBA. ~~The Frequency of 92 days is based on engineering judgment, which considered such factors as the following:~~

~~a. The relative inaccessibility and lack of traffic in the vicinity of the doors make it unlikely that a door would be inadvertently left open;~~

~~b. Excessive air leakage would be detected by temperature monitoring in the ice condenser; and~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~c. The light construction of the doors would ensure that, in the event of a DBA, air and gases passing through the ice condenser would find a flow path, even if a door were obstructed.~~

INSERT 2

SR 3.6.13.4

Verifying, by visual inspection, that the ice condenser inlet doors are not impaired by ice, frost, or debris provides assurance that the doors are free to open in the event of a DBA. ~~For this unit, the Frequency of 18 months is based on door design, which does not allow water condensation to freeze, and operating experience, which indicates that the inlet doors very rarely fail to meet their SR acceptance criteria.~~ Because of high radiation in the vicinity of the inlet doors during power operation, this Surveillance is normally performed during a shutdown.

INSERT 2

SR 3.6.13.5

Verifying the opening torque of the inlet doors provides assurance that no doors have become stuck in the closed position. The value of 675 in-lb is based on the design opening pressure on the doors of 1.0 lb/ft^2. ~~For this unit, the Frequency of 18 months is based on the passive nature of the closing mechanism (i.e., once adjusted, there are no known factors that would change the setting, except possibly a buildup of ice; ice buildup is not likely, however, because of the door design, which does not allow water condensation to freeze). Operating experience indicates that the inlet doors usually meet their SR acceptance criteria.~~ Because of high radiation in the vicinity of the inlet doors during power operation, this Surveillance is normally performed during a shutdown.

SR 3.6.13.6

The torque test Surveillance ensures that the inlet doors have not developed excessive friction and that the return springs are producing a door return torque within limits. The torque test consists of the following:

BASES

SURVEILLANCE REQUIREMENTS (continued)

1. Verify that the torque, T(OPEN), required to cause opening motion at the 40° open position is ≤ 195 in-lb;

2. Verify that the torque, T(CLOSE), required to hold the door stationary (i.e., keep it from closing) at the 40° open position is ≥ 78 in-lb but ≤ 250.6 in-lb; and

3. Calculate the frictional torque, T(FRICT) = 0.5 {T(OPEN) - T(CLOSE)}, and verify that the T(FRICT) is ≥ - 40 in-lb but ≤ + 40 in-lb.

The purpose of the friction and return torque Specifications is to ensure that, in the event of a small break LOCA or SLB, all of the 24 door pairs open uniformly. This assures that, during the initial blowdown phase, the steam and water mixture entering the lower compartment does not pass through part of the ice condenser, depleting the ice there, while bypassing the ice in other bays. ~~The Frequency of 18 months is based on the passive nature of the closing mechanism (i.e., once adjusted, there are no known factors that would change the setting, except possibly a buildup of ice; ice buildup is not likely, however, because of the door design, which does not allow water condensation to freeze). Operating experience indicates that the inlet doors very rarely fail to meet their SR acceptance criteria.~~ Because of high radiation in the vicinity of the inlet doors during power operation, this Surveillance is normally performed during a shutdown.

INSERT 2

SR 3.6.13.7

Verifying the OPERABILITY of the intermediate deck doors provides assurance that the intermediate deck doors are free to open in the event of a DBA. The verification consists of visually inspecting the intermediate doors for structural deterioration, verifying free movement of the vent assemblies, and ascertaining free movement of each door when lifted with the applicable force shown below:

	Door	Lifting Force
a.	Adjacent to crane wall	≤ 37.4 lb
b.	Paired with door adjacent to crane wall	≤ 33.8 lb
c.	Adjacent to containment wall	≤ 31.8 lb
d.	Paired with door adjacent to containment wall	≤ 31.0 lb

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~The 18 month Frequency is based on the passive design of the intermediate deck doors, the frequency of personnel entry into the intermediate deck, and the fact that SR 3.6.13.2 confirms on a 7 day Frequency that the doors are not impaired by ice, frost, or debris, which are ways a door would fail the opening force test (i.e., by sticking or from increased door weight).~~ INSERT 2

REFERENCES

1. UFSAR, Chapter 6.
2. 10 CFR 50, Appendix K.
3. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
4. DPC-1201.17-00-0006, "Design and Licensing Basis for Ice Condenser Lower Inlet Doors Technical Specification Surveillance Requirements, 40° Opening, Closing and Frictional Torques."

ACTIONS (continued)

inspections in the pressurizer compartment during power operation and analysis performed that shows an open hatch (7.5 ft^2 bypass area) during a DBA does not impact the design pressure or temperature of the containment.

C.1

If the divider barrier seal is inoperable, 1 hour is allowed to restore the seal to OPERABLE status. The 1 hour Completion Time is consistent with LCO 3.6.1, which requires that containment be restored to OPERABLE status within 1 hour.

D.1 and D.2

If divider barrier integrity 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 3.6.14.1

Verification, by visual inspection, that all personnel access doors and equipment hatches between the upper and lower containment compartments are closed provides assurance that divider barrier integrity is maintained prior to the reactor being taken from MODE 5 to MODE 4. This SR is necessary because many of the doors and hatches may have been opened for maintenance during the shutdown.

SR 3.6.14.2

Verification, by visual inspection, that the personnel access door and equipment hatch seals, sealing surfaces, and alignments are acceptable provides assurance that divider barrier integrity is maintained. This

BASES

SURVEILLANCE REQUIREMENTS (continued)

inspection cannot be made when the door or hatch is closed. Therefore, SR 3.6.14.2 is required for each door or hatch that has been opened, prior to the final closure. Some doors and hatches may not be opened for long periods of time. ~~Those that use resilient materials in the seals must be opened and inspected at least once every 10 years to provide assurance that the seal material has not aged to the point of degraded performance. The Frequency of 10 years is based on the known resiliency of the materials used for seals, the fact that the openings have not been opened (to cause wear), and operating experience that confirms that the seals inspected at this Frequency have been found to be acceptable.~~

INSERT 2

SR 3.6.14.3

Verification, by visual inspection, after each opening of a personnel access door or equipment hatch that it has been closed makes the operator aware of the importance of closing it and thereby provides additional assurance that divider barrier integrity is maintained while in applicable MODES.

SR 3.6.14.4

Conducting periodic physical property tests on divider barrier seal test coupons provides assurance that the seal material has not degraded in the containment environment, including the effects of irradiation with the reactor at power. The required tests include a tensile strength test. ~~The Frequency of 18 months was developed considering such factors as the known resiliency of the seal material used, the inaccessibility of the seals and absence of traffic in their vicinity, and the unit conditions needed to perform the SR. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.~~ INSERT 2

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.14.5

Visual inspection of the seal around the perimeter provides assurance that the seal is properly secured in place. ~~The Frequency of 18 months was developed considering such factors as the inaccessibility of the seals and absence of traffic in their vicinity, the strength of the bolts and mechanisms used to secure the seal, and the unit conditions needed to perform the SR. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.~~ Insert 2

REFERENCES

1. UFSAR, Section 6.2.
2. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.15.1 and SR 3.6.15.2

Verifying the OPERABILITY of the refueling canal drains ensures that they will be able to perform their functions in the event of a DBA. SR 3.6.15.1 confirms that the refueling canal drain valves have been locked open and that the drains are clear of any obstructions that could impair their functioning. In addition to debris near the drains, SR 3.6.15.2 requires attention be given to any debris that is located where it could be moved to the drains in the event that the Containment Spray System is in operation and water is flowing to the drains. SR 3.6.15.1 must be performed before entering MODE 4 from MODE 5 after every filling of the canal to ensure that the valves have been locked open and that no debris that could impair the drains was deposited during the time the canal was filled. ~~SR 3.6.15.2 is performed every 92 days for the upper compartment and refuel canal areas. The 92 day Frequency was developed considering such factors as the inaccessibility of the drains, the absence of traffic in the vicinity of the drains, and the redundancy of the drains.~~

INSERT 2

SR 3.6.15.3

Verifying the OPERABILITY of the ice condenser floor drains ensures that they will be able to perform their functions in the event of a DBA. Inspecting the drain valve disk ensures that the valve is performing its function of sealing the drain line from warm air leakage into the ice condenser during normal operation, yet will open if melted ice fills the line following a DBA. Verifying that the drain lines are not obstructed ensures their readiness to drain water from the ice condenser. ~~The 18 month Frequency developed considering such factors as the inaccessibility of the drains during power operation; the design of the ice condenser, which precludes melting and refreezing of the ice; and operating experience that has confirmed that the drains are found to be acceptable when the Surveillance is performed at an 18 month Frequency.~~ Because of high radiation in the vicinity of the drains during power operation, this Surveillance is normally done during a shutdown.

Insert 2

REFERENCES

1. UFSAR, Section 6.2.
2. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).

BASES

APPLICABILITY

Maintaining reactor building OPERABILITY prevents leakage of radioactive material from the reactor building. Radioactive material may enter the reactor building from the containment following a LOCA. Therefore, reactor building OPERABILITY is required in MODES 1, 2, 3, and 4 when a 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, reactor building OPERABILITY is not required in MODE 5 or 6.

ACTIONS

A.1

In the event reactor building OPERABILITY is not maintained, reactor 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 reactor 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 3.6.16.1

Maintaining reactor building OPERABILITY requires maintaining the door in the access opening closed, except when the access opening is being used for normal transit entry and exit. ~~The 31 day Frequency of this SR is based on engineering judgment and is considered adequate in view of the other indications of door status that are available.~~ INSERT 2

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.16.2

The annulus vacuum decay test is performed to verify the reactor building is OPERABLE. A minimum annulus vacuum decay time of 87 seconds ensures that the reactor building design outside air inleakage rate is ≤ 2000 cfm at an annulus differential pressure of -1.0 inch water gauge. Higher reactor building annulus outside air inleakage rates correlate to less holdup, mixing, and filtration of radiological effluents which increase offsite and operator doses.

The vacuum decay test is performed by isolating the pressure transmitter and starting the AVS fan to draw down the annulus pressure to a significant vacuum. Isolating the transmitter enables the fan to reduce the annulus pressure below the normal setpoint. The fan is then secured and the time it takes for the annulus pressure to decay or increase from -3.5 inches water gauge to -0.5 inch water gauge is measured. The time required for the pressure in the annulus to increase from -3.5 inches water gauge to -0.5 inch water gauge is known as the vacuum decay time.

The reactor building annulus outside air inleakage is an input to the CANVENT computer code, which provides input to the dose analyses. The CANVENT computer code is used to model the thermal effects of a LOCA on the annulus and the ability of the AVS to develop and maintain a negative pressure in the annulus after a design basis accident. The code also determines AVS exhaust and recirculation airflow rates following a LOCA. The results of the CANVENT analysis for annulus conditions and AVS response to the LOCA also are used for the rod ejection accident.

The 2000 cfm at -1.0 inch water gauge reactor building annulus outside air inleakage rate is conservatively corrected for ambient temperature and pressure as well as annulus differential pressure conditions prior to use as an input to the CANVENT computer code. The CANVENT results are then used as an input to the dose analyses.

~~The reactor building pressure boundary is tested every 18 months. The 18 month Frequency is consistent with the guidance provided in NUREG-0800.~~

INSERT 2

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.16.3

This SR would give advance indication of gross deterioration of the concrete structural integrity of the reactor building. ~~The Frequency is based on engineering judgment, and is the same as that for containment visual inspections performed in accordance with SR 3.6.1.1.~~

INSERT 2

REFERENCES

1. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
2. UFSAR, Sections 6.2.3 and 6.2.6.5.
3. NUREG-0800, Sections 6.2.3 and 6.5.3, Rev. 2, July 1981.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.4.1

Verification of the nitrogen supply pressure on at least one tank for each SG PORV ensures that the SG PORVs will be available to mitigate the consequences of a steam generator tube rupture concurrent with the loss of offsite power. ~~The 24 hour frequency is consistent with operating experience and has shown to be acceptable.~~ INSERT 2

SR 3.7.4.2

To perform a controlled cooldown of the RCS, the SG PORVs must be able to be opened remotely and throttled through their full range using the safety-related nitrogen gas supply. This SR ensures that the SG PORVs are tested through a full control cycle at least once per fuel cycle. Performance of inservice testing or use of an SG PORV during a unit cooldown may satisfy this requirement. ~~Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. The Frequency is acceptable from a reliability standpoint.~~ Insert 2

SR 3.7.4.3

The function of the block valve is to isolate a failed open SG PORV. Cycling the block valve both closed and open demonstrates its capability to perform this function. Performance of inservice testing or use of the block valve during unit cooldown may satisfy this requirement. ~~Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. The Frequency is acceptable from a reliability standpoint.~~ Insert 2

REFERENCES

1. UFSAR, Section 10.3.
2. UFSAR, Chapter 15.
3. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).

BASES

ACTIONS (continued)

E.1

In MODE 4, either the reactor coolant pumps or the RHR loops can be used to provide forced circulation. This is addressed in LCO 3.4.6, "RCS Loops—MODE 4." With one required AFW train inoperable, action must be taken to immediately restore the inoperable train to OPERABLE status. The immediate Completion Time is consistent with LCO 3.4.6.

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1

Verifying the correct alignment for manual, power operated, and automatic valves in the AFW System water and steam supply flow paths provides assurance that the proper flow paths will exist for AFW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they 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 SR is also modified by a note that excludes automatic valves when THERMAL POWER is ≤ 10% RTP. Some automatic valves may be in a throttled position to support low power operation.

~~The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.~~ INSERT 2

SR 3.7.5.2

Verifying that each AFW pump's developed head at the flow test point is greater than or equal to the required developed head ensures that AFW pump performance has not degraded during the cycle. Flow and differential head are normal tests of centrifugal pump performance required by the ASME Code (Ref. 3). Because it is undesirable to introduce cold AFW into the steam generators while they are operating, this testing is performed on recirculation flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. Performance of inservice testing discussed in the ASME Code (Ref. 3) (only required at 3 month intervals) satisfies this requirement.

BASES

SURVEILLANCE REQUIREMENTS (continued)

This SR is modified by a Note indicating that the SR should be deferred until suitable test conditions are established. This deferral is required because there is insufficient steam pressure to perform the test.

SR 3.7.5.3

This SR verifies that AFW can be delivered to the appropriate steam generator in the event of any accident or transient that generates an ESFAS, by demonstrating that each automatic valve in the flow path actuates to its correct position on an actual or simulated actuation signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. ~~The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 18 month Frequency is acceptable based on operating experience and the design reliability of the equipment.~~ INSERT 2

This SR is modified by a Note that states the SR is not required in MODE 4. In MODE 4, the required AFW train may already be aligned and operating.

SR 3.7.5.4

This SR verifies that the AFW pumps will start in the event of any accident or transient that generates an ESFAS by demonstrating that each AFW pump starts automatically on an actual or simulated actuation signal in MODES 1, 2, and 3. In MODE 4, the required pump may already be operating and the autostart function is not required. ~~The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.~~ INSERT 2

This SR is modified by two Notes. Note 1 indicates that the SR can be deferred until suitable test conditions are established. This deferral is required because there is insufficient steam pressure to perform the test. Note 2 states that the SR is not required in MODE 4. In MODE 4, the required pump may already be operating and the autostart function is not required. In MODE 4, the heat removal requirements would be less providing more time for operator action to manually start the required AFW pump if it were not in operation.

BASES

ACTIONS (continued)

MODE 3 within 6 hours, and in MODE 4, without reliance on the steam generator for heat removal, within 12 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 3.7.6.1

This SR verifies that the CSS contains the required inventory of cooling water. ~~The 12 hour Frequency is based on operating experience and the need for operator awareness of unit evolutions that may affect the CSS inventory between checks. Also, the 12 hour Frequency is considered adequate in view of other indications in the control room, including alarms, to alert the operator to abnormal deviations in the CSS level.~~ Insert 2

REFERENCES

1. UFSAR, Section 10.4.
2. UFSAR, Chapter 6.
3. UFSAR, Chapter 15.

BASES

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 RHR heat exchanger.

In MODE 5 or 6, the requirements of the CCW System are determined by the systems it supports.

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that the applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops—MODE 4," be entered if an inoperable CCW train results in an inoperable RHR loop. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.

If one CCW train is inoperable, action must be taken to restore OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE CCW train is adequate to perform the heat removal function. The 72 hour Completion Time is reasonable, based on the redundant capabilities afforded by the OPERABLE train, and the low probability of a DBA occurring during this period.

B.1 and B.2

If the 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 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 to safety related equipment provides assurance that the proper flow paths exist for CCW operation.

BASES

SURVEILLANCE REQUIREMENTS (continued)

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.~~ INSERT 2

SR 3.7.7.2

This SR verifies proper automatic operation of the CCW valves on an actual or simulated actuation safety injection, Phase 'A' Isolation, or Phase 'B' Isolation 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 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.~~ INSERT 2

SR 3.7.7.3

This SR verifies proper automatic operation of the CCW pumps 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. ~~The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.~~

INSERT 2

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.8.1

This SR is modified by a Note indicating that the isolation of the NSWS components or systems may render those components inoperable, but does not affect the OPERABILITY of the NSWS.

Verifying the correct alignment for manual, power operated, and automatic valves in the NSWS flow path provides assurance that the proper flow paths exist for NSWS operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to being locked, sealed, or secured. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves. ~~The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.~~

INSERT 2

SR 3.7.8.2

This SR verifies proper automatic operation of the NSWS valves on an actual or simulated actuation signal. The signals that cause the actuation are from Safety Injection and Phase 'B' isolation. The NSWS is a normally operating system that cannot be fully actuated as part of normal testing. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. ~~The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.~~

INSERT 2

This SR is modified by a Note that states that the SR is not required to be met for valves that are maintained in position to support NSWS single supply header operation. When the NSWS is placed in this alignment, certain automatic valves in the system are maintained in position and will not automatically reposition in response to an actuation signal while the NSWS is in this alignment.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.8.3

This SR verifies proper automatic operation of the NSWS pumps on an actual or simulated actuation signal. The signals that cause the actuation are from Safety Injection and Loss of Offsite Power. The NSWS is a normally operating system that cannot be fully actuated as part of normal testing during normal operation. ~~The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.~~ Insert 2

REFERENCES

1. UFSAR, Section 9.2.
2. UFSAR, Section 6.2.
3. UFSAR, Section 5.4.
4. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).

BASES

LCO

The SNSWP is required to be OPERABLE and is considered OPERABLE if it contains a sufficient volume of water at or below the maximum temperature that would allow the NSWS to operate for at least 30 days following the design basis accident without the loss of net positive suction head (NPSH), and without exceeding the maximum design temperature of the equipment served by the NSWS. To meet this condition, the SNSWP temperature should not exceed 95°F at 568 ft mean sea level and the level should not fall below 571 ft mean sea level during normal unit operation.

APPLICABILITY

In MODES 1, 2, 3, and 4, the SNSWP is required to support the OPERABILITY of the equipment serviced by the SNSWP and required to be OPERABLE in these MODES.

In MODE 5 or 6, the requirements of the SNSWP are determined by the systems it supports.

ACTIONS

A.1

If the SNSWP is inoperable 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 3.7.9.1

This SR verifies that adequate long term (30 day) cooling can be maintained. The specified level also ensures that sufficient NPSH is available to operate the NSWS pumps. ~~The 24 hour Frequency is based on operating experience related to trending of the parameter variations during the applicable MODES.~~ This SR verifies that the SNSWP water level is ≥ 571 ft mean sea level. INSERT 2

SR 3.7.9.2

This SR verifies that the NSWS is available to cool the CCW System to at least its maximum design temperature with the maximum accident or normal design heat loads for 30 days following a Design Basis Accident.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~The 24 hour Frequency is based on operating experience related to trending of the parameter variations during the applicable MODES.~~ This SR verifies that the average water temperature of the SNSWP is ≤ 95°F. The SR is modified by a note that states the Surveillance is only required to be performed during the months of July, August, and September. During other months, the ambient temperature is below the surveillance limit.

INSERT 2

SR 3.7.9.3

This SR verifies dam integrity by inspection to detect degradation, erosion, or excessive seepage. ~~Operating experience has shown that these components usually pass the Surveillance when performed at the 12 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.~~

REFERENCES

1. UFSAR, Section 9.2.
2. Regulatory Guide 1.27.
3. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.10.1

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not too severe, testing each train once every month provides an adequate check of this system. Monthly heater operations dry out any moisture accumulated in the carbon from humidity in the ambient air. Systems with heaters must be operated from the control room for ≥ 10 continuous hours with the heaters energized and flow through the HEPA filters and carbon adsorbers. ~~The 31 day Frequency is based on the reliability of the equipment and the two train redundancy.~~ INSERT 2

SR 3.7.10.2

This SR verifies that the required CRAVS testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The CRAVS filter tests are in accordance with Regulatory Guide 1.52 (Ref. 5). The VFTP includes testing the performance of the HEPA filter and carbon adsorber efficiencies and the physical properties of the activated carbon. Specific test Frequencies and additional information are discussed in detail in the VFTP.

SR 3.7.10.3

This SR verifies that each CRAVS train starts and operates on an actual or simulated actuation signal. ~~The Frequency of 18 months is based on industry operating experience and is consistent with the typical refueling cycle.~~ INSERT 2

SR 3.7.10.4

This SR verifies the OPERABILITY of the CRE boundary by testing for unfiltered air inleakage past the CRE boundary and into the CRE. The details of the testing are specified in the Control Room Envelope Habitability Program.

The CRE is considered habitable when the radiological dose to CRE occupants calculated in the licensing basis analyses of DBA consequences is no more than 5 rem TEDE and the CRE occupants are protected from hazardous chemicals and smoke. This SR verifies that the unfiltered air inleakage into the CRE is no greater than the flow rate assumed in the licensing basis analyses of DBA consequences. When unfiltered air inleakage is greater than the assumed flow rate, Condition B

SURVEILLANCE REQUIREMENTS (continued)

must be entered. Required Action B.3 allows time to restore the CRE boundary to OPERABLE status provided mitigating actions can ensure that the CRE remains within the licensing basis habitability limits for the occupants following an accident. Compensatory measures are discussed in Regulatory Guide 1.196, Section C.2.7.3 (Ref. 9), which endorses, with exceptions, NEI 99-03, Section 8.4 and Appendix F (Ref. 7). These compensatory measures may also be used as mitigating actions as required by Required Action B.2. Temporary analytical methods may also be used as compensatory measures to restore OPERABILITY (Ref. 8). Options for restoring the CRE boundary to OPERABLE status include changing the licensing basis DBA consequence analysis, repairing the CRE boundary, or a combination of these actions. Depending upon the nature of the problem and the corrective action, a full scope inleakage test may not be necessary to establish that the CRE boundary has been restored to OPERABLE status.

REFERENCES

1. UFSAR, Section 6.4.

2. UFSAR, Section 9.4.1.

3. UFSAR, Chapter 15.

4. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).

5. Regulatory Guide 1.52, Rev. 2.

6. Catawba Nuclear Station License Amendments 90/84 for Units 1/2, August 23, 1991.

7. NEI 99-03, "Control Room Habitability Assessment", June 2001.

8. Letter from Eric J. Leeds (NRC) to James W. Davis (NEI) dated January 30, 2004, "NEI Draft White Paper, Use of Generic Letter 91-18 Process and Alternative Source Terms in the Context of Control Room Habitability", (ADAMS Accession No. ML040300694).

9. Regulatory Guide 1.196, Rev. 1.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.11.1

This SR verifies that the heat removal capability of the system is sufficient to maintain the temperature in the control room at or below 90°F. ~~The 12 hour Frequency is appropriate since significant degradation of the CRACWS is slow and is not expected over this time period.~~

INSERT 2

REFERENCES

1. UFSAR, Section 9.4.
2. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
3. 10 CFR 50.67, Accident source term.
4. Regulatory Guide 1.183, Revision 0.

ACTIONS (continued)

hazards such as radioactive contamination, toxic chemicals, smoke, temperature and relative humidity, and physical security. Preplanned measures should be available to address these concerns for intentional and unintentional entry into the condition. The 24 hour Completion Time is reasonable based on the low probability of a DBA occurring during this time period and the use of compensatory measures. The 24 hour Completion Time is a typically reasonable time to diagnose, plan and possibly repair, and test most problems with the ECCS pump rooms pressure boundary.

C.1 and C.2

If the ABFVES train or ECCS pump rooms pressure boundary 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 experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

D.1 and D.2

With one or more ABFVES heaters inoperable, the heater must be restored to OPERABLE status within 7 days. Alternatively, a report must be initiated per Specification 5.6.6, which details the reason for the heater's inoperability and the corrective action required to return the heater to OPERABLE status.

The heaters do not affect OPERABILITY of the ABFVES filter trains because carbon adsorber efficiency testing is performed at 30°C and 95% relative humidity. The accident analysis shows that site boundary radiation doses are within 10 CFR 50.67 limits during a DBA LOCA under these conditions.

SURVEILLANCE REQUIREMENTS

SR 3.7.12.1

Systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not severe, testing each train once a month provides an adequate check on this system. Monthly heater operations dry out any moisture that may have accumulated in the carbon from humidity in the ambient air. Systems with heaters must be operated from the control room ≥ 10 continuous hours with flow through the HEPA filters and

BASES

SURVEILLANCE REQUIREMENTS (continued)

carbon adsorbers and with the heaters energized. ~~The 31 day Frequency is based on the known reliability of equipment and the two train redundancy available.~~ INSERT 2

SR 3.7.12.2

This SR verifies that the required ABFVES testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The ABFVES filter tests are in accordance with Reference 5. The VFTP includes testing HEPA filter performance, carbon adsorbers efficiency, system flow rate, and the physical properties of the activated carbon (general use and following specific operations). The system flow rate determination and in-place testing of the filter unit components is performed in the normal operating alignment with both trains in operation. Flow through each filter unit in this alignment is approximately 30,000 cfm. The normal operating alignment has been chosen to minimize normal radiological protection concerns that occur when the system is operated in an abnormal alignment for an extended period of time. Operation of the system in other alignments may alter flow rates to the extent that the 30,000 cfm ±10% specified in Technical Specification 5.5.11 will not be met. Flow rates outside the specified band under these operating alignments will not require the system to be considered inoperable.

Certain postulated failures and post accident recovery operational alignments may result in post accident system operation with only one train of ABFVES in a "normal" alignment. Under these conditions system flow rate is expected to increase above the normal flow band specified in Technical Specification 5.5.11. An analysis has been performed which conservatively predicts the maximum flow rate under these conditions is approximately 37,000 cfm. 37,000 cfm corresponds to a face velocity of approximately 48 ft/min that is significantly more than the normal 40 ft/min velocity specified in ASTM D3803-1989 (Ref. 10). Therefore, the laboratory test of the carbon penetration is performed in accordance with ASTM D3803-1989 and Generic Letter 99-02 at a face velocity of 48 ft/min. These test results are to be adjusted for a 2.27 inch bed using the methodology presented in ASTM D3803-1989 prior to comparing them to the Technical Specification 5.5.11 limit. Specific test Frequencies and additional information are discussed in detail in the VFTP.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.12.3

This SR verifies that each ABFVES train starts and operates with flow through the HEPA filters and carbon adsorbers on an actual or simulated actuation signal. ~~The 18 month Frequency is consistent with that specified in Reference 4.~~

INSERT 2

UFSAR, Section 15.6

For SFCP Addition only

SR 3.7.12.4

This SR verifies the pressure boundary integrity of the ECCS pump rooms. The following rooms are considered to be ECCS pump rooms (with respect to the ABFVES): centrifugal charging pump rooms, safety injection pump rooms, residual heat removal pump rooms, and the containment spray pump rooms. Although the containment spray system is not normally considered an ECCS system, it is included in this ventilation boundary because of its accident mitigation function which requires the pumping of post accident containment sump fluid. The Elevation 522 pipe chase area is also maintained at a negative pressure by the ABFVES. Since the Elevation 543 and 560 mechanical penetration rooms communicate directly with the Elevation 522 pipe chase area, these penetration rooms are also maintained at a negative pressure by the ABFVES. The ability of the system to maintain the ECCS pump rooms at a negative pressure, with respect to potentially unfiltered adjacent areas, is periodically tested to verify proper functioning of the ABFVES. Upon receipt of a safety injection signal to initiate LOCA operation, the ABFVES is designed to maintain a slight negative pressure in the ECCS pump rooms, with respect to adjacent areas, to prevent unfiltered LEAKAGE. The ABFVES will continue to operate in this mode until the safety injection signal is reset. ~~The Frequency of 18 months is consistent with the guidance provided in NUREG-0800, Section 6.5.1 (Ref. 8).~~

INSERT 2

~~NUREG 0800~~, Section 6.5.1, Rev. 2, July 1981

For SFCP addition only

BASES

ACTIONS (continued)

With the movement of recently irradiated fuel in the fuel handling building, two trains of FHVES are required to be OPERABLE and one in operation. The movement of recently irradiated fuel must be immediately suspended, if one or more trains of FHVES are inoperable or one is not in operation. This does not preclude the movement of an irradiated fuel assembly to a safe position. This action ensures that a fuel handling accident with unacceptable consequences could not occur.

B.1 and B.2

With one or more FHVES heaters inoperable, the heater must be restored to OPERABLE status within 7 days. Alternatively, a report must be initiated per Specification 5.6.6, which details the reason for the heater's inoperability and the corrective action required to return the heater to OPERABLE status.

The heaters do not affect OPERABILITY of the FHVES filter trains because carbon adsorber efficiency testing is performed at 30°C and 95% relative humidity. The accident analysis shows that site boundary radiation doses are within 10 CFR 50.67 limits during a fuel handling accident under these conditions.

SURVEILLANCE REQUIREMENTS

SR 3.7.13.1

With the FHVES train in service, a periodic monitoring of the system for proper operation should be checked on a routine basis to ensure that the system is functioning properly. The 12 hour Frequency is sufficient to ensure proper operation through the HEPA and carbon filters and is based on the known reliability of the equipment.

INSERT 2

SR 3.7.13.2

Systems should be checked periodically to ensure that they function properly. As the environmental and normal operating conditions on this system are not severe, testing each train once every month provides an adequate check on this system.

BASES

REFERENCES

1. UFSAR, Section 6.5.
2. UFSAR, Section 9.4.
3. UFSAR, Section 15.7.
4. Regulatory Guide 1.25.
5. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
6. Not used.
7. Regulatory Guide 1.52 (Rev. 2).
8. ~~NUREG-0800, Section 6.5.1, Rev. 2, July 1981.~~
9. 10 CFR 50.67, Accident source term.
10. Regulatory Guide 1.183 (Rev. 0).
11. Catawba Nuclear Station License Amendments 90/84 for Units 1/2, August 23, 1991.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~Monthly heater operation dries out any moisture accumulated in the carbon from humidity in the ambient air.~~ Systems with heaters must be operated from the control room for ≥ 10 continuous hours with flow through the HEPA filters and carbon adsorbers and with the heaters energized. ~~The 31 day Frequency is based on the known reliability of the equipment.~~

INSERT 2

SR 3.7.13.3

This SR verifies that the required FHVES testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The FHVES filter tests are in accordance with Regulatory Guide 1.52 (Ref. 7). The VFTP includes testing HEPA filter performance, carbon adsorber efficiency, system flow rate, and the physical properties of the activated carbon (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the VFTP.

SR 3.7.13.4

This SR verifies the integrity of the fuel building enclosure. The ability of the system to maintain the fuel building at a negative pressure with respect to atmospheric pressure is periodically tested to verify proper function of the FHVES. During operation, the FHVES is designed to maintain a slight negative pressure in the fuel building, to prevent unfiltered LEAKAGE. The FHVES is designed to maintain ≤ -0.25 inches water gauge with respect to atmospheric pressure at a flow rate of ≤ 36,443 cfm. ~~The Frequency of 18 months (on a STAGGERED TEST BASIS) is consistent with the guidance provided in NUREG-0800, Section 6.5.1~~ ~~(Ref. 8)~~, Rev. 2, July 1981.

Insert 2

SR 3.7.13.5 FOR SFCP Addition only

Operating the FHVES filter bypass damper is necessary to ensure that the system functions properly. The OPERABILITY of the FHVES filter bypass damper is verified if it can be manually closed. ~~An 18 month Frequency is consistent with Reference 8~~.

INSERT 2

NUREG-0800, Section 6.5.1, Rev. 2, July 1981

FOR SFCP only

BASES

LCO

The spent fuel pool water level is required to be ≥ 23 ft over the top of irradiated fuel assemblies seated in the storage racks. The specified water level preserves the assumptions of the fuel handling accident analysis (Ref. 3). As such, it is the minimum required for fuel storage and movement within the spent fuel pool.

APPLICABILITY

This LCO applies during movement of irradiated fuel assemblies in the spent fuel pool, since the potential for a release of fission products exists.

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply.

When the initial conditions for prevention of an accident cannot be met, steps should be taken to preclude the accident from occurring. When the spent fuel pool water level is lower than the required level, the movement of irradiated fuel assemblies in the spent fuel pool is immediately suspended to a safe position. This action effectively precludes the occurrence of a fuel handling accident. This does not preclude movement of a fuel assembly to a safe position.

If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODES 1, 2, 3, and 4, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.14.1

This SR verifies sufficient spent fuel pool water is available in the event of a fuel handling accident. The water level in the spent fuel pool must be checked periodically. ~~The 7 day Frequency is appropriate because the volume in the pool is normally stable. Water level changes are controlled by plant procedures and are acceptable based on operating experience.~~ Insert 2

During refueling operations, the level in the spent fuel pool is in equilibrium with the refueling canal, and the level in the refueling canal is checked daily in accordance with SR 3.9.6.1.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The concentration of dissolved boron in the spent fuel pool satisfies Criterion 2 of 10 CFR 50.36 (Ref. 7).

LCO

The spent fuel pool boron concentration is required to be within the limits specified in the COLR. The specified concentration of dissolved boron in the spent fuel pool preserves the assumptions used in the analyses of the potential critical accident scenarios as described in Reference 6. This concentration of dissolved boron is the minimum required concentration for fuel assembly storage and movement within the spent fuel pool.

APPLICABILITY

This LCO applies whenever fuel assemblies are stored in the spent fuel pool.

ACTIONS

A.1 and A.2

The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply.

When the concentration of boron in the fuel storage pool is less than required, immediate action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies. The concentration of boron is restored simultaneously with suspending movement of fuel assemblies.

If the LCO is not met while moving irradiated fuel assemblies in MODE 5 or 6, LCO 3.0.3 would not be applicable. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.15.1

This SR verifies that the concentration of boron in the spent fuel pool is within the required limit. As long as this SR is met, the analyzed accidents are fully addressed. ~~The 7 day Frequency is appropriate because no major replenishment of pool water is expected to take place over such a short period of time.~~ INSERT 2

BASES

ACTIONS

A.1 and A.2

DOSE EQUIVALENT I-131 exceeding the allowable value in the secondary coolant, is an indication of a problem in the RCS and contributes to increased post accident doses. If the secondary specific activity cannot be restored to within limits 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 3.7.17.1

This SR verifies that the secondary specific activity is within the limits of the accident analysis. A gamma isotopic analysis of the secondary coolant, which determines DOSE EQUIVALENT I-131, confirms the validity of the safety analysis assumptions as to the source terms in post accident releases. It also serves to identify and trend any unusual isotopic concentrations that might indicate changes in reactor coolant activity or LEAKAGE. ~~The 31 day Frequency is based on the detection of increasing trends of the level of DOSE EQUIVALENT I-131, and allows for appropriate action to be taken to maintain levels below the LCO limit.~~ INSERT 2

REFERENCES

1. 10 CFR 50.67.
2. UFSAR, Section 15.1.5.
3. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
4. Regulatory Guide 1.183, July 2000.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.1

This SR ensures proper circuit continuity for the offsite AC electrical power supply to the onsite distribution network and availability of offsite AC electrical power. The breaker alignment verifies that each breaker is in its correct position to ensure that distribution buses and loads are connected to their preferred power source, and that appropriate independence of offsite circuits is maintained. ~~The 7 day Frequency is adequate since breaker position is not likely to change without the operator being aware of it and because its status is displayed in the control room.~~ INSERT 2

SR 3.8.1.2 and SR 3.8.1.7

These SRs help to ensure the availability of the standby electrical power supply to mitigate DBAs and transients and to maintain the unit in a safe shutdown condition.

To minimize the wear on moving parts that do not get lubricated when the engine is not running, these SRs are modified by a Note (Note 2 for SR 3.8.1.2) to indicate that all DG starts for these Surveillances may be preceded by an engine prelube period and followed by a warmup period prior to loading.

For the purposes of SR 3.8.1.2 and SR 3.8.1.7 testing, the DGs are started from standby conditions using a manual start, loss of offsite power signal, safety injection signal, or loss of offsite power coincident with a safety injection signal. Standby conditions for a DG mean that the diesel engine coolant and oil are being continuously circulated and temperature is being maintained consistent with manufacturer recommendations.

In order to reduce stress and wear on diesel engines, the manufacturer recommends a modified start in which the starting speed of DGs is limited, warmup is limited to this lower speed, and the DGs are gradually accelerated to synchronous speed prior to loading. These start procedures are the intent of Note 3, which is only applicable when such modified start procedures are recommended by the manufacturer.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.7 requires that, ~~at a 184 day Frequency,~~ the DG starts from standby conditions and achieves required voltage and frequency within 11 seconds. The 11 second start requirement supports the assumptions of the design basis LOCA analysis in the UFSAR, Chapter 15 (Ref. 5).

The 11 second start requirement is not applicable to SR 3.8.1.2 (see Note 3) when a modified start procedure as described above is used. If a modified start is not used, the 11 second start requirement of SR 3.8.1.7 applies.

Since SR 3.8.1.7 requires a 11 second start, it is more restrictive than SR 3.8.1.2, and it may be performed in lieu of SR 3.8.1.2. This is the intent of Note 1 of SR 3.8.1.2.

~~The normal 31 day Frequency for SR 3.8.1.2 is consistent with Regulatory Guide 1.9 (Ref. 3). The 184 day Frequency for SR 3.8.1.7 is a reduction in cold testing consistent with Generic Letter 84-15 (Ref. 8). These Frequencies provide adequate assurance of DG OPERABILITY, while minimizing degradation resulting from testing.~~ INSERT 2

SR 3.8.1.3

This Surveillance verifies that the DGs are capable of synchronizing with the offsite electrical system and accepting loads greater than or equal to the equivalent of the maximum expected accident loads. A minimum run time of 60 minutes is required to stabilize engine temperatures, while minimizing the time that the DG is connected to the offsite source.

Although no power factor requirements are established by this SR, the DG is normally operated at a power factor between 0.8 lagging and 1.0. The 0.8 value is the design rating of the machine, while the 1.0 is an operational limitation to ensure circulating currents are minimized. The load band is provided to avoid routine overloading of the DG. Routine overloading may result in more frequent teardown inspections in accordance with vendor recommendations in order to maintain DG OPERABILITY.

~~The 31 day Frequency for this Surveillance is consistent with Regulatory Guide 1.9 (Ref. 3).~~ INSERT 2

This SR is modified by four Notes. Note 1 indicates that diesel engine runs for this Surveillance may include gradual loading, as recommended

BASES

SURVEILLANCE REQUIREMENTS (continued)

by the manufacturer, so that mechanical stress and wear on the diesel engine are minimized. Note 2 states that momentary transients, because of changing bus loads, do not invalidate this test. Similarly, momentary power factor transients above the limit do not invalidate the test. Note 3 indicates that this Surveillance should be conducted on only one DG at a time in order to avoid common cause failures that might result from offsite circuit or grid perturbations. Note 4 stipulates a prerequisite requirement for performance of this SR. A successful DG start must precede this test to credit satisfactory performance.

SR 3.8.1.4

This SR provides verification that the level of fuel oil in the day tank is at or above the level at which fuel oil is automatically added. The level is expressed as an equivalent volume in gallons, and is selected to ensure adequate fuel oil for a minimum of 1 hour of DG operation at full load plus 10%.

~~The 31 day Frequency is adequate to assure that a sufficient supply of fuel oil is available, since low level alarms are provided and facility operators would be aware of any large uses of fuel oil during this period.~~

INSERT 2

SR 3.8.1.5

Microbiological fouling is a major cause of fuel oil degradation. There are numerous bacteria that can grow in fuel oil and cause fouling, but all must have a water environment in order to survive. Removal of water from the fuel oil day tanks ~~once every 31 days~~ eliminates the necessary environment for bacterial survival. This is the most effective means of controlling microbiological fouling. In addition, it eliminates the potential for water entrainment in the fuel oil during DG operation. Water may come from any of several sources, including condensation, ground water, rain water, contaminated fuel oil, and breakdown of the fuel oil by bacteria. Frequent checking for and removal of accumulated water minimizes fouling and provides data regarding the watertight integrity of the fuel oil system. ~~The Surveillance Frequencies are established by Regulatory Guide 1.137 (Ref. 11).~~ This SR is for preventative maintenance. The presence of water does not necessarily represent failure of this SR, provided the accumulated water is removed during the performance of this Surveillance.

INSERT 2

~~Frequency of 31 days~~
for SFCP addition only

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.6

This Surveillance demonstrates that each required fuel oil system operates and transfers fuel oil from its associated storage tanks to its associated day tank. This is required to support continuous operation of standby power sources. This Surveillance provides assurance that the fuel oil valve is OPERABLE, and allows gravity feed of fuel oil to the day tank from underground storage tanks, to ensure the fuel oil piping system is intact, the fuel delivery piping is not obstructed, and the controls and control systems for fuel transfer systems are OPERABLE.

~~The design of fuel transfer systems is such that the transfer valve operates automatically or the transfer valve bypass valve may be opened manually in order to maintain an adequate volume of fuel oil in the day tanks during or following DG testing. Therefore, a 31 day Frequency is appropriate.~~ INSERT 2

SR 3.8.1.7

See SR 3.8.1.2.

SR 3.8.1.8

Transfer of each 4.16 kV ESF bus power supply from the normal offsite circuit to the alternate offsite circuit demonstrates the capability of the alternate circuit distribution network to power the shutdown loads. The alternate circuit distribution network consists of an offsite power source through a 6.9 kV bus incoming breaker, its associated 6.9 kV bus tie breaker and the aligned 6.9/4.16 kV transformer to the essential bus. The requirement of this SR is the transfer from the normal offsite circuit to the alternate offsite circuit via the automatic and manual actuation of the 6.9 kV bus tie breaker and 6.9 kV bus incoming breakers upon loss of the normal offsite source that is being credited. Capability of manually swapping to a standby transformer is not required to satisfy this SR. ~~The 18 month Frequency of the Surveillance is based on engineering judgment, taking into consideration the unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths. Operating experience has shown that these components usually pass the SR when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.~~ INSERT 2

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.9

Each DG is provided with an engine overspeed trip to prevent damage to the engine. Recovery from the transient caused by the loss of a large load could cause diesel engine overspeed, which, if excessive, might result in a trip of the engine. This Surveillance demonstrates the DG load response characteristics and capability to reject the largest single load without exceeding predetermined voltage and frequency and while maintaining a specified margin to the overspeed trip. For this unit, the single load for each DG and its horsepower rating is as follows: Nuclear Service Water pump which is a 1000 H.P. motor. This Surveillance may be accomplished by:

a. Tripping the DG output breaker with the DG carrying greater than or equal to its associated single largest post-accident load while paralleled to offsite power, or while solely supplying the bus; or

b. Tripping its associated single largest post-accident load with the DG solely supplying the bus.

As required by Regulatory Guide 1.9 (Ref. 3), the load rejection test is acceptable if the increase in diesel speed does not exceed 75% of the difference between synchronous speed and the overspeed trip setpoint. The value of 63 Hz has been selected for the frequency limit for the load rejection and it is a more conservative limit than required by Reference 3.

The time, voltage, and frequency tolerances specified in this SR are derived from Regulatory Guide 1.9 (Ref. 3) recommendations for response during load sequence intervals. The 3 seconds specified is equal to 60% of a typical 5 second load sequence interval associated with sequencing of the largest load. The voltage and frequency specified are consistent with the design range of the equipment powered by the DG. SR 3.8.1.9.a corresponds to the maximum frequency excursion, while SR 3.8.1.9.b and SR 3.8.1.9.c are steady state voltage and frequency values to which the system must recover following load rejection. ~~The 18 month Frequency is consistent with the recommendation of Regulatory Guide 1.108 (Ref. 10).~~ Insert 2

This SR is modified by a Note. In order to ensure that the DG is tested under load conditions that are as close to design basis conditions as possible, the Note requires that, if synchronized to offsite power, testing must be performed using a power factor ≤ 0.9. This power factor is chosen to be representative of the actual design basis inductive loading that the DG would experience.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.10

This Surveillance demonstrates the DG capability to reject a full load without overspeed tripping or exceeding the predetermined voltage limits. The DG full load rejection may occur because of a system fault or inadvertent breaker tripping. This Surveillance ensures proper engine generator load response under the simulated test conditions. This test simulates the loss of the total connected load that the DG experiences following a full load rejection and verifies that the DG does not trip upon loss of the load. These acceptance criteria provide for DG damage protection. While the DG is not expected to experience this transient during an event and continues to be available, this response ensures that the DG is not degraded for future application, including reconnection to the bus if the trip initiator can be corrected or isolated.

Although not representative of the design basis inductive loading that the DG would experience, a power factor of approximately unity (1.0) is used for testing. This power factor is chosen in accordance with manufacturer's recommendations to minimize DG overvoltage damage during testing.

~~The 18 month Frequency is consistent with the recommendation of Regulatory Guide 1.108 (Ref. 10) and is intended to be consistent with expected fuel cycle lengths.~~ INSERT 2

SR 3.8.1.11

As required by Regulatory Guide 1.108 (Ref. 10), paragraph 2.a.(1), this Surveillance demonstrates the as designed operation of the standby power sources during loss of the offsite source. This test verifies all actions encountered from the loss of offsite power, including shedding of the nonessential loads and energization of the emergency buses and respective loads from the DG. It further demonstrates the capability of the DG to automatically achieve the required voltage and frequency within the specified time.

The DG autostart time of 11 seconds is derived from requirements of the accident analysis to respond to a design basis large break LOCA. The Surveillance should be continued for a minimum of 5 minutes in order to demonstrate that all starting transients have decayed and stability is achieved.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The requirement to verify the connection and power supply of the emergency bus and autoconnected loads is intended to satisfactorily show the relationship of these loads to the DG loading logic. In certain circumstances, many of these loads cannot actually be connected or loaded without undue hardship or potential for undesired operation. For instance, Emergency Core Cooling Systems (ECCS) injection valves are not desired to be stroked open, or high pressure injection systems are not capable of being operated at full flow, or residual heat removal (RHR) systems performing a decay heat removal function are not desired to be realigned to the ECCS mode of operation. In lieu of actual demonstration of connection and loading of loads, testing that adequately shows the capability of the DG systems to perform these functions is acceptable. This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

~~The Frequency of 18 months is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 10), paragraph 2.a.(1), takes into consideration unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths.~~ INSERT 2

This SR is modified by two Notes. The reason for Note 1 is to minimize wear and tear on the DGs during testing. For the purpose of this testing, the DGs must be started from standby conditions, that is, with the engine coolant and oil continuously circulated and temperature maintained consistent with manufacturer recommendations. The reason for Note 2 is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems.

SR 3.8.1.12

This Surveillance demonstrates that the DG automatically starts and achieves the required voltage and frequency within the specified time (11 seconds) from the design basis actuation signal (LOCA signal) and operates for ≥ 5 minutes. The 5 minute period provides sufficient time to demonstrate stability. SR 3.8.1.12.d ensures that the emergency bus remains energized from the offsite electrical power system on an ESF signal without loss of offsite power.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~The Frequency of 18 months takes into consideration unit conditions required to perform the Surveillance and is intended to be consistent with the expected fuel cycle lengths. Operating experience has shown that these components usually pass the SR when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.~~ This SR is modified by a Note. The reason for the Note is to minimize wear and tear on the DGs during testing. For the purpose of this testing, the DGs must be started from standby conditions, that is, with the engine coolant and oil continuously circulated and temperature maintained consistent with manufacturer recommendations.

Insert 2

SR 3.8.1.13

This Surveillance demonstrates that DG non-emergency protective functions (e.g., high jacket water temperature) are bypassed on a loss of voltage signal concurrent with an ESF actuation test signal. Non-emergency automatic trips are all automatic trips except:

a. Engine overspeed;
b. Generator differential current;
c. Low - low lube oil pressure; and
d. Voltage control overcurrent relay scheme.

The non-emergency trips are bypassed during DBAs and provide an alarm on an abnormal engine condition. This alarm provides the operator with sufficient time to react appropriately. The DG availability to mitigate the DBA is more critical than protecting the engine against minor problems that are not immediately detrimental to emergency operation of the DG. Currently, DG emergency automatic trips are tested periodically per the station periodic maintenance program.

~~The 18 month Frequency is based on engineering judgment, taking into consideration unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths. Operating experience has shown that these components usually pass the SR when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.~~ INSERT 2

SR 3.8.1.14

Regulatory Guide 1.108 (Ref. 10), paragraph 2.a.(3), requires demonstration ~~once per 18 months~~ that the DGs can start and run

SURVEILLANCE REQUIREMENTS (continued)

continuously at full load capability for an interval of not less than 24 hours. The DG starts for this Surveillance can be performed either from standby or hot conditions. The provisions for prelubricating and warmup, discussed in SR 3.8.1.2, and for gradual loading, discussed in SR 3.8.1.3, are applicable to this SR.

In order to ensure that the DG is tested under load conditions that are as close to design conditions as possible, testing must be performed using a power factor of ≤ 0.9. This power factor is chosen to be representative of the actual design basis inductive loading that the DG would experience. The load band is provided to avoid routine overloading of the DG. Routine overloading may result in more frequent teardown inspections in accordance with vendor recommendations in order to maintain DG OPERABILITY.

~~The 18 month Frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 10), paragraph 2.a.(3), takes into consideration unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths.~~ INSERT 2

This Surveillance is modified by a Note. The Note states that momentary transients due to changing bus loads do not invalidate this test. Similarly, momentary power factor transients above the power factor limit will not invalidate the test.

SR 3.8.1.15

This Surveillance demonstrates that the diesel engine can restart from a hot condition, such as subsequent to shutdown from normal Surveillances, and achieve the required voltage and frequency within 11 seconds. The 11 second time is derived from the requirements of the accident analysis to respond to a design basis large break LOCA. ~~The 18 month Frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 10), paragraph 2.a.(5)~~ INSERT 2

This SR is modified by two Notes. Note 1 ensures that the test is performed with the diesel sufficiently hot. The load band is provided to

SURVEILLANCE REQUIREMENTS (continued)

avoid routine overloading of the DG. Routine overloads may result in more frequent teardown inspections in accordance with vendor recommendations in order to maintain DG OPERABILITY. The requirement that the diesel has operated for at least an hour at full load conditions prior to performance of this Surveillance is based on manufacturer recommendations for achieving hot conditions. Momentary transients due to changing bus loads do not invalidate this test. Note 2 allows all DG starts to be preceded by an engine prelube period to minimize wear and tear on the diesel during testing.

SR 3.8.1.16

As required by Regulatory Guide 1.108 (Ref. 10), paragraph 2.a.(6), this Surveillance ensures that the manual synchronization and automatic load transfer from the DG to the offsite source can be made and the DG can be returned to standby operation when offsite power is restored. It also ensures that the autostart logic is reset to allow the DG to reload if a subsequent loss of offsite power occurs. The DG is considered to be in standby operation when the DG is at rated speed and voltage, the output breaker is open and can receive an autoclose signal on bus undervoltage, and the load sequence timers are reset.

~~The Frequency of 18 months is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 10), paragraph 2.a.(6), and takes into consideration unit conditions required to perform the Surveillance.~~

INSERT 2

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems.

SR 3.8.1.17

Demonstration of the test mode override ensures that the DG availability under accident conditions will not be compromised as the result of testing and the DG will automatically reset to standby operation if a LOCA actuation signal is received during operation in the test mode. Standby operation is defined as the DG running at rated speed and voltage with the DG output breaker open. These provisions for automatic switchover are required by Regulatory Guide 1.9 (Ref. 3).

SURVEILLANCE REQUIREMENTS (continued)

The requirement to automatically energize the emergency loads with offsite power is essentially identical to that of SR 3.8.1.12. The intent in the requirement associated with SR 3.8.1.17.b is to show that the emergency loading was not affected by the DG operation in test mode. In lieu of actual demonstration of connection and loading of loads, testing that adequately shows the capability of the emergency loads to perform these functions is acceptable. This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

~~The 18 month Frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 10), paragraph 2.a.(8), takes into consideration unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths.~~ INSERT 2

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems.

SR 3.8.1.18

Under accident and loss of offsite power conditions loads are sequentially connected to the bus by the automatic load sequencer. The sequencing logic controls the permissive and starting signals to motor breakers to prevent overloading of the DGs due to high motor starting currents. The load sequence time interval tolerance in Table 8-6 of Reference 2 ensures that sufficient time exists for the DG to restore frequency and voltage prior to applying the next load and that safety analysis assumptions regarding ESF equipment time delays are not violated. Table 8-6 of Reference 2 provides a summary of the automatic loading of ESF buses.

~~The Frequency of 18 months is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 10), paragraph 2.a.(2), takes into consideration unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths.~~ INSERT 2

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.19

In the event of a DBA coincident with a loss of offsite power, the DGs are required to supply the necessary power to ESF systems so that the fuel, RCS, and containment design limits are not exceeded.

This Surveillance demonstrates the DG operation, as discussed in the Bases for SR 3.8.1.11, during a loss of offsite power actuation test signal in conjunction with an ESF actuation signal. In lieu of actual demonstration of connection and loading of loads, testing that adequately shows the capability of the DG system to perform these functions is acceptable. This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

~~The Frequency of 18 months takes into consideration unit conditions required to perform the Surveillance and is intended to be consistent with an expected fuel cycle length of 18 months.~~ INSERT 2

This SR is modified by two Notes. The reason for Note 1 is to minimize wear and tear on the DGs during testing. For the purpose of this testing, the DGs must be started from standby conditions, that is, with the engine coolant and oil continuously circulated and temperature maintained consistent with manufacturer recommendations for DGs. The reason for Note 2 is that the performance of the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems.

SR 3.8.1.20

This Surveillance demonstrates that the DG starting independence has not been compromised. Also, this Surveillance demonstrates that each engine can achieve proper speed within the specified time when the DGs are started simultaneously.

~~The 10 year Frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 10).~~ INSERT 2

This SR is modified by a Note. The reason for the Note is to minimize wear on the DG during testing. For the purpose of this testing, the DGs

ACTIONS (continued)

for reasons other than addressed by Conditions A through E, the associated DG may be incapable of performing its intended function and must be immediately declared inoperable.

SURVEILLANCE REQUIREMENTS

SR 3.8.3.1

This SR provides verification that there is an adequate inventory of fuel oil in the storage tanks to support each DG's operation for 7 days at full load. The 7 day period is sufficient time to place the unit in a safe shutdown condition and to bring in replenishment fuel from an offsite location.

~~The 31 day Frequency is adequate to ensure that a sufficient supply of fuel oil is available, since low level alarms are provided and unit operators would be aware of any large uses of fuel oil during this period.~~

INSERT 2

SR 3.8.3.2

This Surveillance ensures that sufficient lube oil inventory is available to support at least 7 days of full load operation for each DG. The 400 gal requirement is based on the DG manufacturer consumption values for the run time of the DG. In order to account for the lube oil sump tank inventory decrease that occurs when the DG is started, the 400 gal requirement shall be met with the Surveillance conducted while the DG is running.

~~A 31 day Frequency is adequate to ensure that a sufficient lube oil supply is available, since DG starts and run time are closely monitored by the unit staff.~~

INSERT 2

SR 3.8.3.3

The tests listed below are a means of determining whether new fuel oil is of the appropriate grade and has not been contaminated with substances that would have an immediate, detrimental impact on diesel engine combustion. If results from these tests are within acceptable limits, the fuel oil may be added to the storage tanks without concern for contaminating the entire volume of fuel oil in the storage tanks. These

SURVEILLANCE REQUIREMENTS (continued)

tests are to be conducted prior to adding the new fuel to the storage tank(s). The tests, limits, and applicable ASTM Standards are as follows:

a. Sample the new fuel oil in accordance with ASTM D4057 (Ref. 7);

b. Verify in accordance with the tests specified in ASTM D975 (Ref. 7) that the sample has a kinematic viscosity at 40°C of ≥ 1.9 centistokes and ≤ 4.1 centistokes, and a flash point of ≥ 125°F; and

c. Verify that the new fuel oil has a clear and bright appearance with proper color when tested in accordance with ASTM D4176 (Ref. 7) or a water and sediment content within limits when tested in accordance with ASTM D2709; and

d. Verify that the new fuel oil has an absolute specific gravity at 60/60°F of ≥ 0.83 and ≤ 0.89 when tested in accordance with ASTM D1298 or an API gravity at 60°F of ≥ 27° and ≤ 39° when tested in accordance with ASTM D287 (Ref. 7).

Failure to meet any of the above limits, except for clear and bright, is cause for rejecting the fuel oil, but does not represent a failure to meet the LCO concern since the fuel oil is not added to the storage tanks. If the fuel oil fails on clear and bright, it may be accepted if it passes water and sediment. The specifications for water and sediment recognize that a small amount of water and sediment is acceptable. Thus, this test may be used after a clear and bright test to provide a more quantitative result.

Within 31 days following the initial new fuel oil sample, the fuel oil is analyzed to establish that the other properties specified in Table 1 of ASTM D975 (Ref. 7) are met for new fuel oil when tested in accordance with ASTM D975 (Ref. 7). The 31 day period is acceptable because the fuel oil properties of interest, even if they were not within stated limits, would not have an immediate effect on DG operation. This Surveillance ensures the availability of high quality fuel oil for the DGs.

Fuel oil degradation during long term storage shows up as an increase in particulate, due mostly to oxidation. The presence of particulate does not mean the fuel oil will not burn properly in a diesel engine. The particulate can cause fouling of filters and fuel oil injection equipment, however, which can cause engine failure.

Particulate concentrations should be determined based on ASTM D6217 (Ref. 7). This test method is used for assessing the mass quantity of

SURVEILLANCE REQUIREMENTS (continued)

particulates in middle distillate fuels, which includes 2-D diesel fuel. This method involves a gravimetric determination of total particulate concentration in the fuel oil and has a limit of 10 mg/l. For those designs in which the total stored fuel oil volume is contained in two or more interconnected tanks, each tank must be considered and tested separately.

The Frequency of this test takes into consideration fuel oil degradation trends that indicate that particulate concentration is unlikely to change significantly between Frequency intervals.

SR 3.8.3.4

This Surveillance ensures that, without the aid of the refill compressor, sufficient air start capacity for each DG is available. The system design requirements provide for a minimum of five engine start cycles without recharging. A start cycle is defined by the DG vendor, but usually is measured in terms of time (seconds of cranking) or engine cranking speed. The pressure specified in this SR is intended to reflect the lowest value at which the five starts can be accomplished.

~~The 31 day Frequency takes into account the capacity, capability, redundancy, and diversity of the AC sources and other indications available in the control room, including alarms, to alert the operator to below normal air start pressure.~~ INSERT 2

SR 3.8.3.5

Microbiological fouling is a major cause of fuel oil degradation. There are numerous bacteria that can grow in fuel oil and cause fouling, but all must have a water environment in order to survive. Removal of water from the fuel storage tanks ~~once every 31 days~~ eliminates the necessary environment for bacterial survival. This is the most effective means of controlling microbiological fouling. In addition, it eliminates the potential for water entrainment in the fuel oil during DG operation. Water may come from any of several sources, including condensation, ground water, rain water, and contaminated fuel oil, and from breakdown of the fuel oil by bacteria. Frequent checking for and removal of accumulated water minimizes fouling and provides data regarding the watertight integrity of the fuel oil system. ~~The Surveillance Frequencies are established by~~ 4 ~~of 31 days~~

FOR SFCP addition only

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~Regulatory Guide 1.137 (Ref. 2).~~ This SR is for preventive maintenance. The presence of water does not necessarily represent failure of this SR, provided the accumulated water is removed during performance of the Surveillance. INSERT 2

REFERENCES

1. UFSAR, Section 9.5.4.2.
2. Regulatory Guide 1.137.
3. ANSI N195-1976, Appendix B.
4. UFSAR, Chapter 6.
5. UFSAR, Chapter 15.
6. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
7. ASTM Standards: D4057; D975; D1298; D4176; D2709; D6217; and D287.
8. UFSAR, Section 18.2.4.
9. Catawba License Renewal Commitments, CNS-1274.00-00-0016, Section 4.5.

ACTIONS (continued)

the loss of the channel DC power and the associated DG DC power, the load center power for the train is inoperable and the Condition(s) and Required Action(s) for the Distribution Systems must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.8.4.1

Verifying battery terminal voltage while on float charge for the batteries helps to ensure the effectiveness of the charging system and the ability of the batteries to perform their intended function. Float charge is the condition in which the charger is supplying the continuous charge required to overcome the internal losses of a battery (or battery cell) and maintain the battery (or a battery cell) in a fully charged state. The voltage requirements are based on the nominal design voltage of the battery and are consistent with the initial voltages assumed in the battery sizing calculations. ~~The 7 day Frequency is consistent with manufacturer recommendations and IEEE-450 (Ref. 9).~~ INSERT 2

SR 3.8.4.2

Not used.

SR 3.8.4.3

For the DC channel and DG batteries, visual inspection to detect corrosion of the battery terminals and connections, or measurement of the resistance of each intercell, interrack, intertier, and terminal connection, provides an indication of physical damage or abnormal deterioration that could potentially degrade battery performance. The presence of visible corrosion does not necessarily represent a failure of this SR, provided an evaluation determines that the visible corrosion does not affect the OPERABILITY of the battery.

~~The Surveillance Frequency for these inspections, which can detect conditions that can cause power losses due to resistance heating, is 92 days. This Frequency is considered acceptable based on operating experience related to detecting corrosion trends.~~

INSERT 2

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.4.4

For the DC channel and DG batteries, visual inspection of the battery cells, cell plates, and battery racks provides an indication of physical damage or abnormal deterioration that could potentially degrade battery performance. The presence of physical damage or deterioration does not necessarily represent a failure of this SR, provided an evaluation determines that the physical damage or deterioration does not affect the OPERABILITY of the battery (its ability to perform its design function).

Operating experience has shown that these components usually pass the SR when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

INSERT 2

SR 3.8.4.5 and SR 3.8.4.6

Visual inspection and resistance measurements of intercell, interrack, intertier, and terminal connections provide an indication of physical damage or abnormal deterioration that could indicate degraded battery condition. The anticorrosion material, as recommended by the manufacturer for the batteries, is used to help ensure good electrical connections and to reduce terminal deterioration. The visual inspection for corrosion is not intended to require removal of and inspection under each terminal connection. The removal of visible corrosion is a preventive maintenance SR. The presence of visible corrosion does not necessarily represent a failure of this SR provided visible corrosion is removed during performance of SR 3.8.4.5.

Operating experience has shown that these components usually pass the SR when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

INSERT 2

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.4.7

This SR requires that each battery charger for the DC channel be capable of supplying at least 200 amps and at least 75 amps for the DG chargers. All chargers shall be tested at a voltage of at least 125 V for ≥ 8 hours. These requirements are based on the design capacity of the chargers (Ref. 4). According to Regulatory Guide 1.32 (Ref. 10), the battery charger supply is required to be based on the largest combined demands of the various steady state loads and the charging capacity to restore the battery from the design minimum charge state to the fully charged state, irrespective of the status of the unit during these demand occurrences. The minimum required amperes and duration ensures that these requirements can be satisfied.

~~The Surveillance Frequency is acceptable, given the unit conditions required to perform the test and the other administrative controls existing to ensure adequate charger performance during these 18 month intervals. In addition, this Frequency is intended to be consistent with expected fuel cycle lengths.~~

INSERT 2

SR 3.8.4.8

A battery service test is a special test of battery capability, as found, to satisfy the design requirements (battery duty cycle) of the DC electrical power system. The vital battery's actual duty cycle is identified in calculation CNC-1381.05-00-0011, 125 VDC Vital Instrumentation and Control Power System Battery and Battery Charger Sizing Calculation. The test duty cycle is the actual duty cycle adjusted for the temperature correction factor for 60°F operation, and a design margin of typically 10 to 15% for load addition. The minimum DC battery terminal voltage is determined through Calculation CNC-1381.05-00-0149, 125 VDC Vital I&C Power System (EPL) Voltage Drop Analysis. The DG battery's actual duty cycle is identified in calculation CNC-1381.05-00-0050, 125 VDC Diesel Generator Battery and Battery Charger Sizing Calculation. The test duty cycle is the actual duty cycle adjusted for the temperature correction factor for 60°F operation, and a design margin of typically 10 to 15% for load addition. The minimum DG battery terminal voltage is determined through Calculations CNC-1381.05-00-0235, Unit 1 125 VDC Essential Diesel Power System (EPQ) Voltage Drop Analysis and CNC-1381.05-00-0236, Unit 2 125 VDC Essential Diesel Power System (EPQ) Voltage Drop Analysis. (Note: The duty cycle in the UFSAR is used for battery sizing and includes the temperature factor of 11%, a design margin of 15%, and an aging factor of 25%.)

SURVEILLANCE REQUIREMENTS (continued)

~~Except for performing SR 3.8.4.8 for the DC channel batteries with the unit on line, the Surveillance Frequency of 18 months is consistent with the recommendations of Regulatory Guide 1.32 (Ref. 10) which states that the battery service test should be performed during refueling operations or at some other outage, with intervals between tests, not to exceed 18 months.~~ INSERT 2

This SR is modified by two Notes. Note 1 allows the performance of a modified performance discharge test in lieu of a service test.

The modified performance discharge test is a performance discharge test that is augmented to include the high-rate, short duration discharge loads (during the first minute and 11-to-12 minute discharge periods) of the service test. The duty cycle of the modified performance test must fully envelope the duty cycle of the service test if the modified performance discharge test is to be used in lieu of the service test. Since the ampere-hours removed by the high-rate, short duration discharge periods of the service test represents a very small portion of the battery capacity, the test rate can be changed to that for the modified performance discharge test without compromising the results of the performance discharge test. The battery terminal voltage for the modified performance discharge test should remain above the minimum battery terminal voltage specified in the battery service test for the duration of time equal to that of the service test.

A modified discharge test is a test of the battery capacity and its ability to provide a high rate, short duration load (usually the highest rates of the duty cycle). This will often confirm the battery's ability to meet the critical periods of the load duty cycle, in addition to determining its percentage of rated capacity. Initial conditions for the modified performance discharge test should be identical to those specified for a service test. The reason for Note 2 is that performing the Surveillance would perturb the electrical distribution system and challenge safety systems.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.4.9

A battery performance discharge test is a test of constant current capacity of a battery, normally done in the as found condition, after having been in service, to detect any change in the capacity determined by the acceptance test. The test is intended to determine overall battery degradation due to age and usage.

A battery modified performance discharge test is described in the Bases for SR 3.8.4.8. Either the battery performance discharge test or the modified performance discharge test is acceptable for satisfying SR 3.8.4.9; however, only the modified performance discharge test may be used to satisfy SR 3.8.4.9 while satisfying the requirements of SR 3.8.4.8 at the same time.

The acceptance criteria for this Surveillance are consistent with IEEE-450 (Ref. 9). This reference recommends that the battery be replaced if its capacity is below 80% of the manufacturer's rating. A capacity of 80% shows that the battery rate of deterioration is increasing, even if there is ample capacity to meet the load requirements.

INSERT 2

The Surveillance Frequency ~~for this test is normally~~ 60 months. If the battery shows degradation, or if the battery has reached 85% of its expected life and capacity is < 100% of the manufacturer's rating, the Surveillance Frequency is reduced to 18 months. However (for DC vital batteries only), if the battery shows no degradation but has reached 85% of its expected life, the Surveillance Frequency is only reduced to 24 months for batteries that retain capacity ≥ 100% of the manufacturer's rating. Degradation is indicated, according to IEEE-450 (Ref. 9), when the battery capacity drops by more than 10% relative to its average capacity on the previous performance tests or when it is ≥ 10% below the manufacturer's rating. These Frequencies ~~are~~ consistent with the recommendations ~~in~~ IEEE-450 (Ref. 9). This SR is modified by a Note which is applicable to the DG batteries only. The reason for the Note is that performing the Surveillance would perturb the associated electrical distribution system and challenge safety systems.

ACTIONS (continued)

B.1 and B.2

With one or more batteries (DC batteries, DG batteries, or both) with one or more battery cell parameters outside the Category C limit for any connected cell, sufficient capacity to supply the maximum expected load requirement is not assured and the corresponding DC electrical power subsystem must be declared inoperable. Additionally, other potentially extreme conditions, such as not completing the Required Actions of Condition A within the required Completion Time or average electrolyte temperature of representative cells falling below 60°F, are also cause for immediately declaring the associated DC electrical power subsystem inoperable per Required Action B.1.

In addition, Required Action B.2 mandates that the appropriate LCO(s) must then be entered for the DG supported by the inoperable DC subsystem. If the plant is in MODES 1 through 4, LCO 3.8.1, "AC Sources – Operating" is required to be entered. If the DG is required to support equipment during MODES 5 or 6 or movement of irradiated fuel assemblies, regardless of operating mode, LCO 3.8.2, "AC Sources – Shutdown," is the appropriate LCO.

Required Action B.2 is modified by a Note indicating that it is only applicable for inoperable DG batteries.

SURVEILLANCE REQUIREMENTS

SR 3.8.6.1

~~This SR verifies that Category A battery cell parameters are consistent with IEEE-450 (Ref. 4), which recommends regular battery inspections (at least one per month) including voltage, specific gravity, and electrolyte temperature of pilot cells.~~ This SR is applicable to both DC batteries and DG batteries.

INSERT 2

SR 3.8.6.2

Not used.

BASES

SURVEILLANCE REQUIREMENTS (continued)

INSERT 2

SR 3.8.6.3

~~The quarterly inspection of the channels of DC and DG batteries for specific gravity and voltage is consistent with IEEE-450 (Ref. 4).~~ In addition, within 24 hours of a battery discharge < 110 V or a battery overcharge > 150 V, the battery must be demonstrated to meet Category B limits. Transients, such as motor starting transients, which may momentarily cause battery voltage to drop to ≤ 110 V, do not constitute a battery discharge provided the battery terminal voltage and float current return to pre-transient values. This inspection is also consistent with IEEE-450 (Ref. 4), which recommends special inspections following a severe discharge or overcharge, to ensure that no significant degradation of the battery occurs as a consequence of such discharge or overcharge.

SR 3.8.6.4

~~This Surveillance verification that the average temperature of representative cells is ≥ 60°F, is consistent with a recommendation of IEEE-450 (Ref. 4), that states that the temperature of electrolytes in representative cells should be determined on a quarterly basis.~~ INSERT 2

Lower than normal temperatures act to inhibit or reduce battery capacity. This SR ensures that the operating temperatures remain within an acceptable operating range. This limit is based on manufacturer recommendations.

The term "representative cells" replaces the fixed number of "six connected cells", consistent with the recommendations of IEEE-450 (Ref. 4) to provide a general guidance to the number of cells adequate to monitor the temperature of the battery cells as an indicator of satisfactory performance. For some cases, the number of cells may be less than six, in other conditions, the number may be more.

ACTIONS (continued)

Required Action A.1 allows 24 hours to fix the inoperable inverter and return it to service. The 24 hour limit is based upon engineering judgment, taking into consideration the time required to repair an inverter and the additional risk to which the unit is exposed because of the inverter inoperability. This has to be balanced against the risk of an immediate shutdown, along with the potential challenges to safety systems such a shutdown might entail. When the AC vital bus is powered from its voltage regulated transformer, it is relying upon interruptible AC electrical power sources (offsite and onsite). The uninterruptible inverter source to the AC vital buses is the preferred source for powering instrumentation trip setpoint devices.

If the channel-related inoperable inverter is replaced by its train's swing inverter, the 24 hour limit does not apply (unless the swing inverter is also inoperable).

B.1 and B.2

If the inoperable devices or components cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit 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 unit conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.8.7.1

This Surveillance verifies that the inverters are functioning properly with all required circuit breakers closed and AC vital bus energized from the inverter. The verification of proper indicated voltage output ensures that the required power is readily available for the instrumentation of the RPS and ESFAS connected to the AC vital buses. ~~The 7 day Frequency takes into account the redundant capability of the inverters and other indications available in the control room that alert the operator to inverter malfunctions.~~ INSERT 2

BASES

ACTIONS (continued)

this option may involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend CORE ALTERATIONS, movement of irradiated fuel assemblies, and operations involving positive reactivity additions) that could result in loss of required SDM (MODE 5) or required boron concentration (MODE 6). Suspending positive reactivity additions that could result in failure to meet the minimum SDM or boron concentration limits is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that what would be required in the RCS for minimum SDM or refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes including temperature increases when operating with a positive MTC must also be evaluated to ensure they do not result in a loss of required SDM.

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required inverters and to continue this action until restoration is accomplished in order to provide the necessary inverter power to the unit safety systems.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required inverters should be completed as quickly as possible in order to minimize the time the unit safety systems may be without power or powered from a constant voltage source transformer.

SURVEILLANCE REQUIREMENTS

SR 3.8.8.1

This Surveillance verifies that the power sources are functioning properly with all required circuit breakers closed and AC vital bus energized from the required power source. The verification of proper indicated voltage ensures that required power is readily available for the instrumentation connected to the AC vital bus. ~~The 7 day Frequency takes into account the redundant capability of the power sources and other indications available in the control room that alert the operator to inverter malfunctions.~~ INSERT 2

ACTIONS (continued)

status, the unit 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 unit conditions from full power conditions in an orderly manner and without challenging plant systems.

F.1

Condition F corresponds to a level of degradation in the electrical power distribution system that causes a required safety function to be lost. When more than one inoperable electrical power distribution subsystem results in the loss of a required function, the plant is in a condition outside the accident analysis. Therefore, no additional time is justified for continued operation. LCO 3.0.3 must be entered immediately to commence a controlled shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.8.9.1

This Surveillance verifies that the AC, channels of DC, DC trains, and AC vital bus electrical power distribution systems are functioning properly, with the correct circuit breaker alignment. The correct breaker alignment ensures the appropriate separation and independence of the electrical divisions is maintained, and the appropriate voltage is available to each required bus. The verification of proper indicated voltage availability on the buses ensures that the required voltage is readily available for motive as well as control functions for critical system loads connected to these buses. ~~The 7 day Frequency takes into account the redundant capability of the AC, DC, and AC vital bus electrical power distribution subsystems, and other indications available in the control room that alert the operator to subsystem malfunctions.~~ INSERT 2

REFERENCES

1. UFSAR, Chapter 6.
2. UFSAR, Chapter 15.
3. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
4. Regulatory Guide 1.93, December 1974.

SURVEILLANCE REQUIREMENTS

SR 3.8.10.1

This Surveillance verifies that the AC, channels of DC, DC trains, and AC vital bus electrical power distribution subsystems are functioning properly, with all the buses energized. The verification of proper indicated voltage availability on the buses ensures that the required power is readily available for motive as well as control functions for critical system loads connected to these buses. ~~The 7 day Frequency takes into account the capability of the electrical power distribution subsystems, and other indications available in the control room that alert the operator to subsystem malfunctions.~~ INSERT 2

REFERENCES

1. UFSAR, Chapter 6.
2. UFSAR, Chapter 15.
3. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).

ACTIONS

A.1 and A.2

Continuation of CORE ALTERATIONS or positive reactivity additions (including actions to reduce boron concentration) is contingent upon maintaining the unit in compliance with the LCO. If the boron concentration of any coolant volume in the RCS, the refueling canal, or the refueling cavity is less than its limit, all operations involving CORE ALTERATIONS or positive reactivity additions must be suspended immediately.

Suspension of CORE ALTERATIONS and positive reactivity additions shall not preclude moving a component to a safe position. Operations that individually add limited positive reactivity (e.g., temperature fluctuations from inventory addition or temperature control fluctuations), but when combined with all other operations affecting core reactivity (e.g., intentional boration) result in overall net negative reactivity addition, are not precluded by this Action.

A.3

In addition to immediately suspending CORE ALTERATIONS and positive reactivity additions, boration to restore the concentration must be initiated immediately.

In determining the required combination of boration flow rate and concentration, no unique Design Basis Event must be satisfied. The only requirement is to restore the boron concentration to its required value as soon as possible. In order to raise the boron concentration as soon as possible, the operator should begin boration with the best source available for unit conditions. An acceptable method is to borate at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or its equivalent.

Once actions have been initiated, they must be continued until the boron concentration is restored. The restoration time depends on the amount of boron that must be injected to reach the required concentration.

SURVEILLANCE REQUIREMENTS

SR 3.9.1.1

This SR ensures that the coolant boron concentration in the RCS, and connected portions of the refueling canal and the refueling cavity, is within the COLR limits. The boron concentration of the coolant in each required volume is determined periodically by chemical analysis. Prior to re-connecting portions of the refueling canal or the refueling cavity to the RCS, this SR must be met per SR 3.0.4. If any dilution activity has

BASES

SURVEILLANCE REQUIREMENTS (continued)

occurred while the cavity or canal were disconnected from the RCS, this SR ensures the correct boron concentration prior to communication with the RCS. One sample from the refueling canal or reactor cavity is sufficient to determine the boron concentration in that volume of water. An additional sample is taken from the RCS.

~~A minimum Frequency of once every 72 hours is a reasonable amount of time to verify the boron concentration of representative samples. The Frequency is based on operating experience, which has shown 72 hours to be adequate.~~ INSERT 2

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26.
2. 10 CFR 50.36, Technical Specifications (c)(2)(ii).

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.2.1

SR 3.9.2.1 is the performance of a CHANNEL CHECK which is a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that the two indication channels should be consistent with core conditions. Changes in fuel loading and core geometry can result in significant differences between source range channels, but each channel should be consistent with its local conditions.

~~The Frequency of 12 hours is consistent with the CHANNEL CHECK Frequency specified similarly for the same instruments in LCO 3.3.1 and LCO 3.3.9.~~ INSERT 2

SR 3.9.2.2

SR 3.9.2.2 is the performance of a CHANNEL CALIBRATION ~~every 18 months~~. This SR is modified by a Note stating that neutron detector sensors (NIS and BDMS) are excluded from the CHANNEL CALIBRATION.

The CHANNEL CALIBRATION for the source range neutron flux monitors (NIS) consists of obtaining the detector plateau and pulse height discriminator curves, evaluating those curves, and comparing the curves to the manufacturer's data.

The CHANNEL CALIBRATION for the source range neutron flux monitors (Gamma-Metrics) consists of verifying that the channels respond correctly to test inputs with the necessary range and accuracy.

~~The 18 month Frequency is based on the need to perform this surveillance under the conditions that apply during a plant outage. Operating experience has shown these components usually pass the Surveillance when performed at the 18 month Frequency.~~ INSERT 2

REFERENCES

1. 10 CFR 50, Appendix A, GDC 13, GDC 26, GDC 28, and GDC 29.
2. UFSAR, Sections 4.2, 15.4.6.
3. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).

BASES

APPLICABILITY (continued)

not completely block the penetration or be capable of resisting pressure. The purpose is to enable ventilation systems to draw the release from a postulated fuel handling accident in the proper directions such that it can be treated and monitored.

ACTIONS

A.1 and A.2

If the containment equipment hatch, air locks, or any containment penetration that provides direct access from the containment atmosphere to the outside atmosphere is not in the required status, the unit must be placed in a condition where the isolation function is not needed. This is accomplished by immediately suspending movement of recently irradiated fuel assemblies within containment. Performance of these actions shall not preclude completion of movement of a component to a safe position.

B.1 and B.2

With one or more Containment Purge Exhaust System heaters inoperable, the heater must be restored to OPERABLE status within 7 days. Alternatively, a report must be initiated per Specification 5.6.6, which details the reason for the heater's inoperability and the corrective action required to return the heater to OPERABLE status.

The heaters do not affect OPERABILITY of the Containment Purge Exhaust System filter trains because carbon adsorber efficiency testing is performed at 30°C and 95% relative humidity. The accident analysis shows that site boundary radiation doses are within the limits of 10 CFR 50.67 and Regulatory Guide 1.183 during a DBA LOCA under these conditions.

SURVEILLANCE REQUIREMENTS

SR 3.9.3.1

This Surveillance demonstrates that each of the containment penetrations required to be in its closed position is in that position. The Surveillance on the open purge and exhaust valves will demonstrate that the valves are exhausting through an OPERABLE Containment Purge Exhaust System HEPA Filter and carbon adsorber.

The Surveillance is performed every 7 days during movement of recently irradiated fuel assemblies within containment. The Surveillance interval is selected to be commensurate with the normal duration of time to complete fuel handling operations. As such, this Surveillance ensures that a postulated fuel handling accident involving recently irradiated fuel that

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~releases fission product radioactivity within the containment will not result in a release of significant fission product radioactivity to the environment.~~

INSERT 2

SR 3.9.3.2

Standby systems should be checked periodically to ensure that they function properly. ~~As the environment and normal operating conditions on this system are not severe, testing each train once a month provides an adequate check on this system. Monthly heater operations dry out any moisture that may have accumulated in the carbon from humidity in the ambient air.~~ Systems with heaters must be operated by initiating flow through the HEPA filters and activated carbon adsorbers for ≥ 10 continuous hours with the heaters energized. ~~The 31 day Frequency is based on the known reliability of equipment and the two train redundance available.~~

INSERT 2

SR 3.9.3.3

This SR verifies that the required testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The Containment Purge Exhaust System filter tests are in accordance with Reference 4. The VFTP includes testing HEPA filter performance, carbon adsorbers efficiency, system flow rate, and the physical properties of the activated carbon (general use and following specific operations). Specific test Frequencies and additional information are discussed in detail in the VFTP.

REFERENCES

1. UFSAR, Section 15.7.4.
2. NUREG-0800, Section 15.7.4, Rev. 1, July 1981.
3. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
4. Regulatory Guide 1.52 (Rev. 2).
5. 10 CFR 50.67, Accident source term.
6. Regulatory Guide 1.183 (Rev. 0).
7. Catawba Nuclear Station License Amendments 90/84 for Units 1/2, August 23, 1991.

BASES

ACTIONS (continued)

A.3

If RHR loop requirements are not met, actions shall be initiated and continued in order to satisfy RHR loop requirements. With the unit in MODE 6 and the refueling water level ≥ 23 ft above the top of the reactor vessel flange, corrective actions shall be initiated immediately.

A.4

If RHR loop requirements are not met, all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere must be closed within 4 hours. With the RHR loop requirements not met, the potential exists for the coolant to boil and release radioactive gas to the containment atmosphere. Closing containment penetrations that are open to the outside atmosphere ensures dose limits are not exceeded.

The Completion Time of 4 hours is reasonable, based on the low probability of the coolant boiling in that time.

SURVEILLANCE REQUIREMENTS

SR 3.9.4.1

This Surveillance demonstrates that the RHR loop is in operation and circulating reactor coolant. The flow rate is determined by the flow rate necessary to provide sufficient decay heat removal capability and to prevent thermal and boron stratification in the core. The RCS temperature is determined to ensure the appropriate decay heat removal is maintained. ~~The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control, and alarm indications available to the operator in the control room for monitoring the RHR System.~~ INSERT 2

REFERENCES

1. UFSAR, Section 5.5.7.

2. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).

3. NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management."

BASES

ACTIONS (continued)

concentration greater than that which would be required in the RCS for minimum refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation.

B.2

If no RHR loop is in operation, actions shall be initiated immediately, and continued, to restore one RHR loop to operation. Since the unit is in Conditions A and B concurrently, the restoration of two OPERABLE RHR loops and one operating RHR loop should be accomplished expeditiously.

B.3

If no RHR loop is in operation, all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere must be closed within 4 hours. With the RHR loop requirements not met, the potential exists for the coolant to boil and release radioactive gas to the containment atmosphere. Closing containment penetrations that are open to the outside atmosphere ensures that dose limits are not exceeded. The Completion Time of 4 hours is appropriate for the majority of time during refueling operations, based on time to coolant boiling, since water level is not routinely maintained at low levels.

SURVEILLANCE REQUIREMENTS

SR 3.9.5.1

This Surveillance demonstrates that one RHR loop is in operation and circulating reactor coolant. The flow rate is determined by the flow rate necessary to provide sufficient decay heat removal capability, prevent vortexing in the suction of the RHR pumps, and to prevent thermal and boron stratification in the core. The RCS temperature is determined to ensure the appropriate decay heat removal is maintained. In addition, during operation of the RHR loop with the water level in the vicinity of the reactor vessel nozzles, the RHR pump suction requirements must be met. ~~The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control, and alarm indications available to the operator for monitoring the RHR System in the control room.~~ INSERT 2

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.9.5.2

Verification that the required pump is OPERABLE ensures that an additional RCS or RHR pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the required pump. ~~The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.~~ INSERT 2

REFERENCES

1. UFSAR, Section 5.5.7.
2. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).

BASES

APPLICABILITY

LCO 3.9.6 is applicable during CORE ALTERATIONS, except during latching and unlatching of control rod drive shafts, and is also applicable when moving irradiated fuel assemblies within containment. The LCO minimizes the possibility of a fuel handling accident in containment that is beyond the assumptions of the safety analysis. If irradiated fuel assemblies are not present in containment, there can be no significant radioactivity release as a result of a postulated fuel handling accident. Requirements for fuel handling accidents in the spent fuel pool are covered by LCO 3.7.14, "Fuel Storage Pool Water Level."

ACTIONS

A.1 and A.2

With a water level of < 23 ft above the top of the reactor vessel flange, all operations involving CORE ALTERATIONS or movement of irradiated fuel assemblies within the containment shall be suspended immediately to ensure that a fuel handling accident cannot occur.

The suspension of CORE ALTERATIONS and fuel movement shall not preclude completion of movement of a component to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.9.6.1

Verification of a minimum water level of 23 ft above the top of the reactc vessel flange ensures that the design basis for the analysis of the postulated fuel handling accident during refueling operations is met. Water at the required level above the top of the reactor vessel flange limits the consequences of damaged fuel rods that are postulated to result from a fuel handling accident inside containment (Ref. 2).

~~The Frequency of 24 hours is based on engineering judgment and is considered adequate in view of the large volume of water and the normal procedural controls of valve positions, which make significant unplanned level changes unlikely.~~

INSERT 2

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.7.1

These valves are to be secured closed to isolate possible dilution paths. The likelihood of a significant reduction in the boron concentration during MODE 6 operations is remote due to the large mass of borated water in the refueling cavity and the fact that all unborated water sources are isolated, precluding a dilution. The boron concentration is checked ~~every 72 hours~~ during MODE 6 under SR 3.9.1.1. This Surveillance demonstrates that the valves are closed through a system walkdown. ~~The 31 day Frequency is based on engineering judgment and is considered reasonable in view of other administrative controls that will ensure that the valve opening is an unlikely possibility.~~ INSERT 2

REFERENCES

1. UFSAR, Section 15.4.6.
2. NUREG-0800, Section 15.4.6.

ATTACHMENT 5

TSTF-425, Rev.3 (NUREG-1431) SURVEILLANCES VERSUS CATAWBA SURVEILLANCES CROSS REFERENCE TABLE

ATTACHMENT 5

TSTF-425, Rev. 3 (NUREG-1431) SRs vs. Catawba SRs Cross Reference

TSTF SR	Catawba SR & Bases	COMMENTS
1.1	1.1	Deleted in TS and relocated to the Surveillance Frequency Control Program (SFCP).
3.1.1.1	3.1.1.1	
3.1.2.1	3.1.2.1	
3.1.4.1	3.1.4.1	
3.1.4.2	3.1.4.2	
3.1.5.1	3.1.5.1	
3.1.6.2	3.1.6.2	
3.1.6.3	3.1.6.3	
3.1.8.2	3.1.8.2	
3.1.8.3	3.1.8.3	
3.1.8.4	3.1.8.4	
3.2.1.1	3.2.1.1	Catawba used a modified Option B. (RAOC)
3.2.1.2	3.2.1.2	Catawba used a modified Option B. (RAOC)
3.2.1.2	3.2.1.3	Catawba used a modified Option B (RAOC). (This is an additional SR that encompasses TSTF SR 3.2.1.2.)
3.2.2.1	3.2.2.1	
---------	3.2.2.2	Catawba SR not in TSTF.
3.2.3.1	3.2.3.1	Catawba used a modified Option B. (RAOC)
3.2.4.1	3.2.4.1	
3.2.4.2	3.2.4.2	
3.3.1.1	3.3.1.1	
3.3.1.2	3.3.1.2	
3.3.1.3	3.3.1.3	
3.3.1.4	3.3.1.4	A note concerning an expired one-time extension on a surveillance is deleted here, also.
3.3.1.5	3.3.1.5	
3.3.1.6	3.3.1.6	
3.3.1.7	3.3.1.7	7/1/09 LAR modifies the Bases only for this SR. The Catawba SR will be relocated here, also.
3.3.1.8	3.3.1.8	7/1/09 LAR modifies the Bases only for this SR. The Catawba SR will be relocated here, also.
3.3.1.9	3.3.1.9	
3.3.1.10	3.3.1.10	
3.3.1.11	3.3.1.11	7/1/09 LAR proposes to modify the SR description. SR 3.3.1.11 will be relocated here, also.
3.3.1.12	3.3.1.12	
3.3.1.13	3.3.1.13	
3.3.1.14	3.3.1.14	
3.3.1.16	3.3.1.16	
----------	3.3.1.17	Catawba SR not in TSTF.
3.3.2.1	3.3.2.1	

TSTF SR	Catawba SR & Bases	COMMENTS
3.3.2.2	3.3.2.2	
3.3.2.3	---------	TSTF SR not in Catawba TS.
---------	3.3.2.3	Catawba SR not in TSTF.
3.3.2.4	3.3.2.4	
3.3.2.5	3.3.2.5	
3.3.2.6	3.3.2.6	
3.3.2.7	--------	TSTF SR not in Catawba TS.
----------	3.3.2.7	Catawba SR not in TSTF. Note: 9/2/08 LAR modifies the Bases for this SR.
3.3.2.8	3.3.2.8	
3.3.2.9	3.3.2.9	9/2/08 LAR modifies the Bases for this SR.
3.3.2.10	3.3.2.10	
3.3.2.11	-----------	TSTF SR not in Catawba TS.
--------	3.3.2.11	Catawba SR not in TSTF.
---------	3.3.2.12	Catawba SR not in TSTF.
3.3.3.1	3.3.3.1	
--------	3.3.3.2	Catawba SR 3.3.3.2 is an unused SR.
3.3.3.2	3.3.3.3	
3.3.4.1	3.3.4.1	
3.3.4.2	---------	TSTF SR not in Catawba TS.
3.3.4.3	3.3.4.2	
3.3.4.4	--------	TSTF SR not in Catawba TS.
3.3.5.1	------------	TSTF SR not in Catawba TS.
3.3.5.2	3.3.5.1	
3.3.5.3	3.3.5.2	
3.3.6.1	-----------	TSTF SR not in Catawba TS.
3.3.6.2	-----------	TSTF SR not in Catawba TS.
3.3.6.3	------------	TSTF SR not in Catawba TS.
3.3.6.4	3.3.6.1	
3.3.6.5	3.3.6.2	
3.3.6.6	----------	TSTF SR not in Catawba TS.
3.3.6.7	3.3.6.3	A 9/2/08 LAR modifies the Bases only for this SR. The SR will be relocated here, also.
3.3.6.8	3.3.6.4	A 9/2/08 LAR modifies the TS SR and the Bases. The SR will be relocated here, also.
3.3.6.9	----------	TSTF SR not in Catawba TS.
3.3.7.1	-------	Catawba does not have this TSTF TS.
3.3.7.2	-------	Catawba does not have this TSTF TS.
3.3.8.1	-------	Catawba does not have this TSTF TS.
3.3.8.2	-------	Catawba does not have this TSTF TS.
3.3.8.3	-------	Catawba does not have this TSTF TS.
3.3.9.1	3.3.9.1	
	3.3.9.2	Catawba SR not in TSTF.
	3.3.9.3	Catawba SR not in TSTF.
	3.3.9.4	Catawba SR not in TSTF.
	3.3.9.5	Catawba SR not in TSTF.

TSTF SR	Catawba SR & Bases	COMMENTS
3.3.9.2	3.3.9.6	
3.3.9.3	--------	TSTF SR not in Catawba TS.
3.4.1.1	3.4.1.1	
3.4.1.2	3.4.1.2	
3.4.1.3	3.4.1.3	
3.4.1.4	---------	TSTF SR not in Catawba TS.
--------	3.4.1.4	Catawba SR not in TSTF.
3.4.2.1	----------	Event driven SR at Catawba, retain frequency in TS.
3.4.3.1	3.4.3.1	
3.4.4.1	3.4.4.1	
3.4.5.1	3.4.5.1	
3.4.5.2	3.4.5.2	
3.4.5.3	3.4.5.3	
3.4.6.1	3.4.6.1	
3.4.6.2	3.4.6.2	
3.4.6.3	3.4.6.3	
3.4.7.1	3.4.7.1	
3.4.7.2	3.4.7.2	
3.4.7.3	3.4.7.3	
3.4.8.1	3.4.8.1	
3.4.8.2	3.4.8.2	
3.4.9.1	3.4.9.1	
3.4.9.2	3.4.9.2	
3.4.9.3	3.4.9.3	
3.4.11.1	3.4.11.1	

TSTF SR	Catawba SR & Bases	COMMENTS
3.4.11.2	3.4.11.2	
-----------	3.4.11.3	Catawba SR not in TSTF.
3.4.11.3	-----------	TSTF SR not in Catawba TS.
3.4.11.4	-----------	TSTF SR not in Catawba TS.
3.4.12.1	3.4.12.1	Catawba SR is TSTF SR 3.4.12.1 and 3.4.12.2 combined.
3.4.12.2	--------	See above.
3.4.12.3	3.4.12.2	
3.4.12.4	3.4.12.3	
3.4.12.5	----------	TSTF SR not in Catawba TS.
3.4.12.6	3.4.12.4	
3.4.12.7	3.4.12.7	
3.4.12.8	3.4.12.5	
3.4.12.9	3.4.12.6	
3.4.13.1	3.4.13.1	
3.4.13.2	3.4.13.2	
3.4.14.1	3.4.14.1	
3.4.14.2	3.4.14.2	
3.4.14.3	------------	TSTF SR not in Catawba TS.
3.4.15.1	3.4.15.1	
3.4.15.2	3.4.15.2	
3.4.15.3	3.4.15.3	
3.4.15.4	3.4.15.4	
3.4.15.5	3.4.15.5	
----------	3.4.15.6	Catawba SR not in TSTF.

TSTF SR	Catawba SR & Bases	COMMENTS
3.4.16.1	3.4.16.1	A 12/15/09 LAR proposes to modify the SR description to allow Xe-133 sampling. This SR will be relocated here, also.
3.4.16.2	3.4.16.2	
3.4.16.3	3.4.16.3	A 12/15/09 LAR proposes to modify Catawba SR 3.4.16.3. This SR will be relocated here, also.
3.4.17.2	----------	TSTF SR not in Catawba TS. [RCS Loops Isolation Valves].
3.4.19.1	3.4.17.1	
3.4.19.2	3.4.17.2	TSTF does not revise this SR. Catawba is not relocating this SR.
3.4.19.3	----------	TSTF does not revise this SR.
3.5.1.1	3.5.1.1	
3.5.1.2	3.5.1.2	
3.5.1.3	3.5.1.3	
3.5.1.4	3.5.1.4	
3.5.1.5	3.5.1.5	
3.5.2.1	3.5.2.1	
3.5.2.2	3.5.2.2	
3.5.2.3	3.5.2.3	
3.5.2.5	3.5.2.5	
3.5.2.6	3.5.2.6	
3.5.2.7	3.5.2.7	
3.5.2.8	3.5.2.8	
3.5.4.1	3.5.4.1	
3.5.4.2	3.5.4.2	9/2/08 LAR proposes to modify the minimum RWST volume. SR 3.5.4.2 will be relocated here, also.
3.5.4.3	3.5.4.3	
3.5.5.1	3.5.5.1	
3.5.6	----------	TSTF SR not in Catawba TS .
3.6.2.1	3.6.2.1	TSTF does not revise this SR. Catawba is not relocating this SR.
3.6.2.2	3.6.2.3	
---------	3.6.2.2	Catawba SR not in TSTF.
3.6.3.1	3.6.3.1	
3.6.3.2	3.6.3.2	
3.6.3.3	3.6.3.3	

TSTF SR	Catawba SR & Bases	COMMENTS
3.6.3.4	3.6.3.4	TSTF does not revise this SR. Catawba is not relocating this SR.
3.6.3.5	3.6.3.5	The SR is only based on the In-service Testing Program. Catawba SR 3.6.3.5 is not revised.
3.6.3.6	---------	TSTF SR not in Catawba TS .
----------	3.6.3.6	Catawba SR 3.6.3.6 will not be relocated since it is based on the Containment Leakage Rate Testing Program.
3.6.3.8	3.6.3.7	
3.6.3.9	----------	TSTF SR not in Catawba TS.
3.6.3.10	----------	TSTF SR not in Catawba TS.
3.6.3.11	3.6.3.8	TSTF does not revise this SR. Catawba is not relocating this SR.
3.6.4.1 A	3.6.4.1	3.6.4B from the TSTF is not applicable.
3.6.5.1 B	3.6.5.1	3.6.5A and 3.6.5C from the TSTF are not applicable.
3.6.5.2 B	3.6.5.2	
3.6.6.1 C	3.6.6.1	3.6.6A, B, D and E from the TSTF are not applicable.
3.6.6.2 C	3.6.6.2	TSTF does not revise this SR. Catawba is not relocating this SR.
3.6.6.3 C	3.6.6.3	
3.6.6.4 C	3.6.6.4	9/2/08 LAR proposes to modify this SR description. SR 3.6.6.4 will be relocated here, also.
-----------	3.6.6.5	Catawba SR not in TSTF.
-----------	3.6.6.6	Catawba SR not in TSTF.
3.6.6.5.C	3.6.6.7	A 9/30/09 License Amendment proposes to modify the frequency of this SR.
3.6.7	-----------	Catawba does not have this TSTF TS.
3.6.8.1	------------	TSTF SR not in Catawba TS.

TSTF SR	Catawba SR & Bases	COMMENTS
3.6.8.2	3.6.16.1	
----------	3.6.16.2	Catawba SR not in TSTF.
3.6.8.3	3.6.16.3	TSTF does not modify the SR. However, it is relocated here since Catawba has a frequency of "3 times in 10 years" specified.
3.6.8.4	-----------	TSTF SR not in Catawba TS.
3.6.9.1+	3.6.8.1	
3.6.9.2	----------	TSTF SR not in Catawba TS.
----------	3.6.8.2	Catawba SR not in TSTF.
----------	3.6.8.3	Catawba SR not in TSTF.
3.6.9.3	3.6.8.4	
3.6.10.1	3.6.9.1	
3.6.10.2	3.6.9.2	
3.6.10.3	3.6.9.3	
3.6.11	------	TSTF TS not applicable at Catawba.
3.6.13.1	3.6.10.1	
3.6.13.2	3.6.10.2	TSTF does not revise this SR. Catawba is not relocating this SR.
3.6.13.3	3.6.10.3	
3.6.13.4	3.6.10.4	
3.6.13.5	3.6.10.5	
----------	3.6.10.6	Catawba SR not in TSTF.
3.6.14.1	3.6.11.1	
3.6.14.2	3.6.11.2	
3.6.14.3	-----------	TSTF SR not in Catawba TS.
3.6.14.4	3.6.11.3	
----------	3.6.11.4	Catawba SR not in TSTF.

TSTF SR	Catawba SR & Bases	COMMENTS
----------	3.6.11.5	Catawba SR not in TSTF.
----------	3.6.11.6	Catawba SR not in TSTF.
----------	3.6.11.7	Catawba SR not in TSTF.
3.6.15.1	3.6.12.1	
3.6.15.2	3.6.12.4	
3.6.15.3	3.6.12.5	
3.6.15.4	3.6.12.3	
3.6.15.5	3.6.12.7	
3.6.15.6	3.6.12.6	
3.6.15.7	3.6.12.2	TSTF does not revise this SR. Catawba is not relocating this SR.
3.6.16.1	3.6.13.1	10/2/08 LAR proposes to modify this SR. Catawba SR will be relocated here, also.
3.6.16.2	3.6.13.2	
3.6.16.3	3.6.13.4	10/2/08 LAR proposes to modify this SR. Catawba SR will be relocated here, also.
3.6.16.4	3.6.13.5	10/2/08 LAR proposes to modify this SR. Catawba SR will be relocated here, also.
3.6.16.5	3.6.13.6	10/2/08 LAR proposes to modify Catawba SR 3.6.13.6. This SR will be relocated here, also.
3.6.16.6	3.6.13.7	
3.6.16.7	3.6.13.3	
3.6.17.1	3.6.14.1	TSTF does not modify this SR. Catawba is not relocating this SR.
3.6.17.2	3.6.14.2	
3.6.17.3	3.6.14.3	TSTF does not modify this SR. Catawba is not relocating this SR.
3.6.17.4	3.6.14.4	
3.6.17.5	3.6.14.5	
-----------	3.6.15.1	Event-driven SR and thus, not relocated.
3.6.18.1	3.6.15.2	Catawba SR is TSTF SR 3.6.18.1 c. only.

TSTF SR	Catawba SR & Bases	COMMENTS
3.6.18.2	3.6.15.3	
----------	3.7.1.1	TSTF does not modify this SR. Catawba is not relocating this SR.
3.7.2.2	---------	TSTF SR not in Catawba TS. (SR 3.7.2.1 is not revised by the TSTF)
3.7.3.2	------	TSTF SR not in Catawba TS. (SR 3.7.3.1 is not revised by the TSTF)
--------	3.7.4.1	Catawba SR not in TSTF.
3.7.4.1	3.7.4.2	
3.7.4.2	3.7.4.3	
3.7.5.1	3.7.5.1	
3.7.5.2	3.7.5.2	TSTF does not modify this SR. Catawba is not relocating this SR.
3.7.5.3	3.7.5.3	
3.7.5.4	3.7.5.4	
3.7.5.5	3.7.5.5	TSTF does not modify this SR. Catawba is not relocating this SR.
3.7.6.1	3.7.6.1	
3.7.7.1	3.7.7.1	
3.7.7.2	3.7.7.2	
3.7.7.3	3.7.7.3	
3.7.8.1	3.7.8.1	
3.7.8.2	3.7.8.2	
3.7.8.3	3.7.8.3	
3.7.9.1	3.7.9.1	
3.7.9.2	3.7.9.2	
--------	3.7.9.3	Catawba SR not in TSTF.
3.7.9.3	--------	TSTF SR not in Catawba TS.
3.7.9.4	---------	TSTF SR not in Catawba TS.
3.7.10.1	3.7.10.1	
3.7.10.2	3.7.10.2	TSTF does not modify this SR. Catawba is not relocating this SR.
3.7.10.3	3.7.10.3	
3.7.10.4	3.7.10.4	Catawba SR Frequency is specified by the CRH Program in TS 5.5.16. Catawba SR was not relocated.
3.7.11.1	-----------	TSTF SR not in Catawba TS.
---------	3.7.11.1	Catawba SR not in TSTF.
3.7.12.1	3.7.12.1	
3.7.12.2	3.7.12.2	TSTF does not modify this SR. Catawba is not relocating this SR.
3.7.12.3	3.7.12.3	
3.7.12.4	3.7.12.4	
3.7.12.5	----------	TSTF SR not in Catawba TS.
---------	3.7.13.1	Catawba SR not in TSTF.
3.7.13.1	3.7.13.2	
3.7.13.2	3.7.13.3	TSTF does not modify this SR.
3.7.13.3	---------	TSTF SR not in Catawba TS.
3.7.13.4	3.7.13.4	
3.7.13.5	3.7.13.5	
3.7.14	----------	Catawba does not have this TSTF TS.
3.7.15.1	3.7.14.1	

TSTF SR	Catawba SR & Bases	COMMENTS
3.7.16.1	3.7.15.1	
3.7.18.1	3.7.17.1	
3.8.1.1	3.8.1.1	
3.8.1.2	3.8.1.2	5/28/09 LAR proposes to modify the SR Description to change the steady state voltage values. The Catawba SR will be relocated here, also.
3.8.1.3	3.8.1.3	
3.8.1.4	3.8.1.4	
3.8.1.5	3.8.1.5	
3.8.1.6	3.8.1.6	
3.8.1.7	3.8.1.7	5/28/09 LAR proposes to modify the SR Description to change the steady state voltage values. The Catawba SR will be relocated here, also.
3.8.1.8	3.8.1.8	
3.8.1.9	3.8.1.9	5/28/09 LAR proposes to modify the SR Description to change the steady state voltage values. The Catawba SR will be relocated here, also.
3.8.1.10	3.8.1.10	
3.8.1.11	3.8.1.11	5/28/09 LAR proposes to modify the SR Description to change the steady state voltage values. The Catawba SR will be relocated here, also.
3.8.1.12	3.8.1.12	5/28/09 LAR proposes to modify the SR Description to change the steady state voltage values. The Catawba SR will be relocated here, also.
3.8.1.13	3.8.1.13	
3.8.1.14	3.8.1.14	
3.8.1.15	3.8.1.15	5/28/09 LAR proposes to modify the SR Description to change the steady state voltage values. The Catawba SR will be relocated here, also.
3.8.1.16	3.8.1.16	
3.8.1.17	3.8.1.17	
3.8.1.18	3.8.1.18	
3.8.1.19	3.8.1.19	5/28/09 LAR proposes to modify the SR Description to change the steady state voltage values. The Catawba SR will be relocated here, also.
3.8.1.20	3.8.1.20	5/28/09 LAR proposes to modify the SR Description to change the steady state voltage values. The Catawba SR will be relocated here, also.
3.8.3.1	3.8.3.1	
3.8.3.2	3.8.3.2	
3.8.3.3	3.8.3.3	TSTF does not modify this SR. Catawba is not relocating this SR.
3.8.3.4	3.8.3.4	
3.8.3.5	3.8.3.5	
3.8.4.1	3.8.4.1	
---------	3.8.4.2	Catawba SR is labeled "not used". Catawba SR not in TSTF.
---------	3.8.4.3	Catawba SR not in TSTF. A 12-14-09 LAR proposes to modify this SR to refer to a new Table 3.8.4-1. The Catawba SR will be relocated here, also.
---------	3.8.4.4	Catawba SR not in TSTF.

TSTF SR	Catawba SR & Bases	COMMENTS
---------	3.8.4.5	Catawba SR not in TSTF.
--------	3.8.4.6	Catawba SR not in TSTF. A 12-14-09 LAR proposes to modify this SR to refer to a new Table 3.8.4-1. The Catawba SR will be relocated here, also.
3.8.4.2	3.8.4.7	
3.8.4.3	3.8.4.8	
----------	3.8.4.9	Catawba SR not in TSTF.
3.8.6.1	-----------	TSTF SR not in Catawba TS.
3.8.6.2	-----------	TSTF SR not in Catawba TS.
3.8.6.3	------------	TSTF SR not in Catawba TS.
3.8.6.4	3.8.6.4	
3.8.6.5	------------	TSTF SR not in Catawba TS.
3.8.6.6	-----------	TSTF SR not in Catawba TS.
--------	3.8.6.1	Catawba SR not in TSTF. The Catawba SR is in Table format (TS Table 3.8.6-1).
--------	3.8.6.2	TSTF SR not in Catawba TS and not relocated as part of this LAR. Catawba SR 3.8.6.2 is labeled "not used".
--------	3.8.6.3	Catawba SR not in TSTF. The Catawba SR is in Table format (TS Table 3.8.6-1).
3.8.7.1	3.8.7.1	
3.8.8.1	3.8.8.1	
3.8.9.1	3.8.9.1	
3.8.10.1	3.8.10.1	
3.9.1.1	3.9.1.1	
3.9.2.1	3.9.7.1	
3.9.3.1	3.9.2.1	
3.9.3.2	3.9.2.2	
3.9.4.1	3.9.3.1	
3.9.4.2	----------	TSTF SR not in Catawba TS.
--------	3.9.3.2	Catawba SR not in TSTF.
-------	3.9.3.3	Catawba SR not in TSTF and it will not be relocated.
3.9.5.1	3.9.4.1	
3.9.6.1	3.9.5.1	
3.9.6.2	3.9.5.2	
3.9.7.1	3.9.6.1	
5.5.18	5.5.17	Insert 4 from TSTF is inserted and labeled as Insert 3 here. Numbering was adjusted for the proper section: 5.5.17.

ATTACHMENT 6

PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION

In accordance with the provisions of 10 CFR 50.90, Duke Energy Carolinas (Duke Energy) is submitting a request for an amendment to the Technical Specifications (TS) for Catawba Nuclear Station (Catawba) Units 1 and 2.

The proposed amendment requests the adoption of an approved change to the Standard Technical Specifications (STS) for Westinghouse Plants (NUREG-1431) to allow relocation of specific Technical Specification (TS) surveillance frequencies (SF) to a licensee controlled program. The proposed change is described in Technical Specification Task Force (TSTF) Traveler, TSTF-425, Revision 3 (ADAMS Accession No. ML080280275) related to the Relocation of Surveillance Frequencies to Licensee Control, RITSTF Initiative 5b, and was described in the Notice of Availability published in the Federal Register on July 6, 2009, 74 FR 31996-32006.

The proposed changes are consistent with NRC approved TSTF-425 Revision 3, "Relocate Surveillance Frequencies to Licensee Control – RITSTF Initiative 5b." The proposed change relocates SFs to a licensee controlled program, the Surveillance Frequency Control Program (SFCP). This change is applicable to licensees using probabilistic risk guidelines contained in NRC approved NEI 04-10, "Risk Informed Technical Specifications Initiative 5b, Risk Informed Method for Control of Surveillance Frequencies," (ADAMS Accession No. 071360456).

The basis for the proposed no significant hazards consideration as required by 10 CFR 50.91(a is presented below:

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

   Response: No.

   The proposed change relocates the specified frequencies for periodic SRs to licensee control under a new Surveillance Frequency Control Program. Surveillance frequencies are not an initiator to any accident previously evaluated. As a result, the probability of any accident previously evaluated is not significantly increased. The systems and components required by the TS for which the SFs are relocated are still required to be operable, meet the acceptance criteria for the surveillance requirements (SRs), and be capable of performing any mitigation function assumed in the accident analysis. As a result, the consequences of any accident previously evaluated are not significantly increased.

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

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

   Response: No.

   No new or different accidents result from utilizing the proposed changes. The changes do not involve a physical alteration of the plant (*i.e.,* no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. In addition, the changes do not impose any new or different requirements.

The changes do not alter assumptions made in the safety analysis. The proposed changes are consistent with the safety analysis assumptions and current plant operating practice.

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

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

   Response: No.

   The design, operation, testing methods, and acceptance criteria for systems, structures, and components (SSCs), specified in applicable codes and standards (or alternatives approved for use by the NRC) will continue to be met as described in the plant licensing basis (including the Updated Final Safety Analysis Report and Bases to the Technical Specifications), since these are not affected by changes to the SFs. Similarly, there is no impact to safety analysis acceptance criteria as described in the plant licensing basis. To evaluate a change in the relocated SF, Duke Energy will perform a probabilistic risk evaluation using the guidance contained in NRC approved NEI 04–10, Rev. 1 in accordance with the TS SFCP. NEI 04–10, Rev. 1, methodology provides reasonable acceptance guidelines and methods for evaluating the risk increase of proposed changes to SFs consistent with Regulatory Guide 1.177.

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

Based upon the reasoning presented above, the licensee concludes that the requested change does not involve a significant hazards consideration as set forth in 10 CFR 50.92(c), Issuance of Amendment.