Enclosure to TSTF-507, Revision 0

Part 2 of 3

Technical Specification Task Force
Improved Standard Technical Specifications Change Traveler

Revise Shutdown Margin definition for stuck rod exception

NUREGs Affected: ☑ 1430 ☑ 1431 ☐ 1432 ☐ 1433 ☐ 1434

Classification: 1) Correct Specifications Recommended for CLIIP?: (Unassigned)

Correction or Improvement: (Unassigned)

Industry Contact: Paul Infanger, (352) 563-4796, paul.infanger@pgnmail.com

Revise the definition of SHUTDOWN MARGIN (SDM) item a to include the following as a new sentence: "However, with all CONTROL RODS verified as fully inserted by two independent means, it is not necessary to account for a stuck CONTROL ROD in the SDM calculation."
The consideration of a stuck rod is provided only to allow for a single failure of one rod to not insert when a scram is initiated. However, with positive indication that all rods are already fully inserted, such a provision is overly conservative. This change is consistent with the definition of SDM provided in NUREG-1432 for CE plants.

Revision History

OG Revision 0 Revision Status: Active

Revision Proposed by: ANO-1

Revision Description:
Original Issue

Owners Group Review Information

Date Originated by OG: 15-Jul-97

Owners Group Comments:
(No Comments)

Owners Group Resolution: Approved Date: 01-Feb-96

TSTF Review Information

TSTF Received Date: 06-Nov-97 Date Distributed for Review: 15-Dec-97

OG Review Completed: ☑ BWOG ☑ WOG ☑ CEOG ☑ BWROG

TSTF Comments:
Applicable to WOG and BWOG.

TSTF Resolution: Approved Date: 05-Feb-98

NRC Review Information

NRC Received Date: 10-Mar-98

Date of NRC Letter: 31-Oct-00

Final Resolution: NRC Approves Final Resolution Date: 31-Oct-00

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Affected Technical Specifications

1.1 Definitions - Shutdown Margin

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TSTF-248

1.1 Definitions

SHUTDOWN MARGIN (SDM) (continued)	a. All rod cluster control assemblies (RCCAs) are fully inserted except for the single RCCA of highest reactivity worth, which is assumed to be fully withdrawn. However, with all RCCAs verified fully inserted by two independent means, it is not necessary to account for a stuck RCCA in the SDM calculation. With any RCCA not capable of being fully inserted, the reactivity worth of the RCCA must be accounted for in the determination of SDM; and b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the [nominal zero power design level].
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.
TRIP ACTUATING DEVICE OPERATIONAL TEST (TADOT)	A TADOT shall consist of operating the trip actuating device and verifying the OPERABILITY of required alarm, interlock, display, and trip functions. The TADOT shall include adjustment, as necessary, of the trip actuating device so that it actuates at the required setpoint within the required accuracy.

Technical Specification Task Force
Improved Standard Technical Specifications Change Traveler

Changes to Section 5.0, Administrative Controls

NUREGs Affected: ☑ 1430 ☑ 1431 ☑ 1432 ☑ 1433 ☑ 1434

Classification: 3) Improve Specifications Recommended for CLIIP?: (Unassigned)

Correction or Improvement: (Unassigned)

Industry Contact: Steve Wideman, (620) 364-4037, stwidem@wcnoc.com

This proposed traveler supersedes travelers TSTF-86 (rejected by NRC and TSTF accepted), TSTF-121, TSTF-167 (rejected by NRC) and WOG-108 (Action Item 147). This traveler is based on the recommendations (with some exceptions noted below) in the April 9, 1997 letter from C. Grimes (NRC) to J. Davis (NEI). This traveler proposes the following changes:
1) Revise Administrative Control 5.2.2, Unit Staff, to delete item b, revises item e eliminating specific details for working hour limits, and revises item g to clarify the requirements for the Shift Technical Advisor function,
2) Inserts brackets around entire second sentence in 5.3.1 and adds 5.3.2, Unit Staff Qualifications to retain elements required in TS by regulations,
3) Revises Section 5.5.4 to be consistent with the intent of 10 CFR 20
4) Revises Section 5.6.4 to be consistent with Generic Letter 97-04, and
5) Revises Section 5.7 in accordance with 10 CFR 20.1601(c)

See attached justification.

Revision History

OG Revision 0 Revision Status: Closed

Revision Proposed by: Wolf Creek

Revision Description:
Original Issue

Owners Group Review Information

Date Originated by OG: 20-Nov-97

Owners Group Comments:
(No Comments)

Owners Group Resolution: Approved Date: 20-Nov-97

TSTF Review Information

TSTF Received Date: 20-Nov-97 Date Distributed for Review: 06-Jan-98

OG Review Completed: ☑ BWOG ☑ WOG ☑ CEOG ☑ BWROG

TSTF Comments:

Look at CEOG change to titles, Make indicated changes,
Add SR 3.0.3 to 5.5.4,
Clearly indicate or annotate the differences of the proposed TSTF versus the 4/9/97 letter.

TSTF Resolution: Approved Date: 05-Feb-98

NRC Review Information

NRC Received Date: 19-Mar-98

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OG Revision 0 Revision Status: Closed

NRC Comments: Date of NRC Letter: 16-Jul-98

4/21/98 - Under NRC review.
6/11/98 - NRC Comments:
5.7.2d.2 and 5.7.2d.3(ii)
The proposed change is not acceptable for high radiation areas with dose rates in excess of 1 R/hr. The STS provides several options for licensees to use and provides adequate flexibility while still maintaining an adequate level of control over workers in high radiation areas. The proposed changes will reduce the level of control maintained by the TS.

5.7.le and 5.7.2e
The proposed change to "..when the knowledge of the dose rates must be made to the worker." is not accepted. The STS provides appropriate controls to ensure workers are adequately controlled and protected while working in high radiation areas. The STS allows time for the RP technician to evaluate the radiological hazard and brief the workers about the radiological conditions in the work area prior to the workers entering the high radiation area. The control is required to ensure that workers do not focus solely on the work to be performed but remain informed about radiological conditions. However, there has been alternate wording to the STS proposed by a licensee that was accepted by the staff in a licensee amendment. The approved alternate wording follows: "Except for individuals qualified in radiation protection procedures or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination and knowledge does not require documentation prior to initial entry." This alternate wording may be proposed by licensees in lieu of the STS. However, the wording of the STS will remain as issued.

We agree with the comment in 5.7.2.a.1, however, we do not fully understand it since the proposed words are the same as the standard TS (perhaps the standard TS was missing the [ ] for the designated positions).

5.7.2a
The proposed change to substitute the word "inadvertent" for "unauthorized" is not accepted. High radiation area controls are divided into distinct modes of control; areas below 1 R/hr where barricades are acceptable to prevent inadvertent entry and areas above 1 R/hr where the radiological [hazard] is significantly greater and thus requires the use of locked doors which are not only intended to prevent inadvertent but to ensure that unauthorized entry is prevented. The use of the word "unauthorized" is expressly used to denote the extra controls that are required for high radiation areas greater than 1 R/hr. Regulatory Guide 8.38 does discuss the use of physical barriers to prevent unauthorized entry. The word "unauthorized" will continue to be used.

5.7.2a.2
The comment is not accepted. The STS does not imply that an area would be locked so as to prevent personnel from exiting the area. The STS is designed to be very clear and literal. The NRC and the licensee will recognize that a locked door will be "unlocked" when workers enter or exit the area, and this action would not result in a NOV. The standard TS wording will not be changed.

5.7.2f
The comment in not clear. The STS control is offered to licensees as a "relief" from 10 CFR Part 20. If a licensee has a special need; a custom TS can be proposed and justified for the staff's consideration. The standard TS wording will not be changed.

Final Resolution: Superceded by Revision Final Resolution Date: 11-Jun-98

TSTF Revision 1 Revision Status: Closed

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TSTF Revision 1 Revision Status: Closed

Revision Proposed by: WOG

Revision Description:
Revised to Address NRC Comments:
5.7.2d.2 and 5.7.2d.3(ii) - Comment accepted.

5.7.le and 5.7.2e - Comment accepted, except did not incorporate the following sentence: "These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination and knowledge does not require documentation prior to initial entry." This information was considered to be of a level of detail below that normally included in the Administrative Controls.

5.7.2.a.1 - Brackets removed.

5.7.2a - Comment accepted.

5.7.2a.2 - Comment accepted.

5.7.2f - Information added to the justification to address comment.

TSTF Review Information

TSTF Received Date: 22-Sep-98 Date Distributed for Review: 23-Sep-98

OG Review Completed: ☑ BWOG ☑ WOG ☑ CEOG ☑ BWROG

TSTF Comments:
(No Comments)

TSTF Resolution: Approved Date: 23-Sep-98

NRC Review Information

NRC Received Date: 25-Sep-98

Final Resolution: Superceded by Revision

TSTF Revision 2 Revision Status: Closed

Revision Proposed by: NRC

Revision Description:
Revised changes to 5.7.1e and 5.7.2e per NRC comments to include the sentences, "These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry."

TSTF Review Information

TSTF Received Date: 26-Oct-98 Date Distributed for Review: 26-Oct-98

OG Review Completed: ☑ BWOG ☑ WOG ☑ CEOG ☑ BWROG

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TSTF Revision 2 Revision Status: Closed

TSTF Comments:

(No Comments)

TSTF Resolution: Approved Date: 26-Oct-98

NRC Review Information

NRC Received Date: 28-Oct-98

NRC Comments:

11/12/98 - Rad Protection Branch agrees with Rad Areas position of TSTF-258. Human Factors Branch insists that a sentence requiring the Plant Manager approve overtime limits on a routine basis be reinstated in the TS requirement. B. Tjader to setup a meeting for 12/16 or 12/17 to discuss and resolve.
12/16/98 - NRC requested revision.

Final Resolution: Superceded by Revision

TSTF Revision 3 Revision Status: Closed

Revision Proposed by: TSTF

Revision Description:
Revised changes to 5.2.2.e to address NRC concerns regarding independent review of procedures to ensure overtime limits are maintained. The deleted sentence is replaced with the statement "Controls shall be included in the procedures to require a periodic independent review be conducted to ensure that excessive hours have not been assigned." As used in this application, the term independent is to only ensure the review is performed by individual (s) different than the individual (s) actually authorizing the over time. As used in this application, the term "periodic frequency" shall be based on plant experience, outage frequencies, and other management review practices (i.e., more frequent reviews conducted during shutdown / refueling when a lot of overtime can be expected, versus less frequent reviews at RTP with little anticipated overtime during which the reviews would not exceed the quarterly frequency). While no changes were made to the pages, there was discussion of and agreement by the NRC and the Industry that the meaning of the term "designee" as it is used in this section permits the designee to either be a permanent position or a temporary position to satisfy the requirements.

TSTF Review Information

TSTF Received Date: 17-Dec-98 Date Distributed for Review: 17-Dec-98

OG Review Completed: ☑ BWOG ☑ WOG ☑ CEOG ☑ BWROG

TSTF Comments:

(No Comments)

TSTF Resolution: Approved Date: 18-Dec-98

NRC Review Information

NRC Received Date: 24-Dec-98

NRC Comments:

12/16/98 - TSTF / NRC meeting.

Final Resolution: Superceded by Revision Final Resolution Date: 03-Feb-99

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TSTF Revision 4 Revision Status: Active

Revision Proposed by: NRC

Revision Description:
Revision 4 revised the Revision Description for Revision 3 to included a parenthetical explanation of the meaning of "periodic frequency" requested by the NRC. No other changes were made to the package.

TSTF Review Information

TSTF Received Date: 20-Jan-99 Date Distributed for Review: 09-Mar-99

OG Review Completed: ☑ BWOG ☐ WOG ☑ CEOG ☑ BWROG

TSTF Comments:
(No Comments)

TSTF Resolution: Approved Date: 09-Apr-99

NRC Review Information

NRC Received Date: 28-Apr-99

NRC Comments: Date of NRC Letter: 29-Jun-99
6/16/99 - NRC stated they believed this change was approved.

Final Resolution: NRC Approves Final Resolution Date: 29-Jun-99

Affected Technical Specifications

5.2.2	Administrative Controls, Unit Staff
5.3.1	Administrative Controls, Unit Staff Qualifications
5.5.4	Administrative Controls, Radioactive Effluent Controls Program
5.6.4	Administrative Controls, Monthly Operating Reports
5.7	Administrative Controls, High Radiation Area

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Justification:

1. Changes to Section 5.2, Organization

a. The requirements of 10 CFR 50.54(m)(2)(iii) and 50.54(k) adequately provide for shift manning. These regulations, 50.54(m)(2)(iii), require "when a nuclear power unit is in an operational mode other than cold shutdown or refueling, as defined by the unit's technical specifications, each licensee shall have a person holding a senior operator license for the nuclear power unit in the control room at all times. In addition to this senior operator, for each fueled nuclear power unit, a licensed operator or senior operator shall be present at the controls at all times." Further, 50.54(k) requires "An operator or senior operator licensed pursuant to part 55 of this chapter shall be present at the controls at all times during the operation of the facility." The ISTS 5.2.2.b requirements will be met through compliance with these regulations and is not required to be reiterated in the ISTS.

In the April 9, 1997 letter from C. Grimes to J. Davis, the staff proposed revising 5.2.2.c by adding “Shift crew composition shall meet the requirements stipulated herein and in 10 CFR 50.54(m).” Adding this sentence is a duplicative of the code of federal regulations since all licensees are required to meet 10 CFR 50.54.

b. Section 5.2.2.e. is revised from specific working hour limits to administrative procedures to control working hours. The proposed changes will provide reasonable assurance that impaired performance caused by excessive working hours will not jeopardize safe plant operation. Specific working hour limits are not otherwise required to be in the technical specifications under 10 CFR 50.36(c)(5). Specific controls for working hours of reactor plant staff are described in procedures that require a deliberate decision making process to minimize the potential for impaired personnel performance, and that established procedure control processes will provide sufficient control for changes to that procedure. These changes are consistent with the recommendations in the April 9, 1997 letter from C. Grimes to J. Davis. Additionally, the statement “Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the [Plant Superintendent] or his designee to ensure that excessive hours have not been assigned.” is being deleted. There is no guidance in Generic Letter 82-12 that discusses these additional controls. The additional requirement to have the Plant Manager (or his designee) review individual overtime on a monthly bases is unnecessary since sufficient administrative controls and policies exist, as well as the role of the individuals supervisors in supervising personnel prevent excessive or abuse of overtime.

c. Section 5.2.2.g is revised to eliminate the title of "Shift Technical Advisor (STA)." STAs are not used at all plants (the function may be fulfilled by one of the other on-shift individuals). Therefore, 5.2.2.g is revised so that it does not imply that the STA and the Shift Supervisor must be different individuals. Option 1 of the Commission Policy Statement on Engineering Expertise on Shift is satisfied by assigning an individual with specified educational qualifications to each operating crew as one of the SROs (preferably the shift supervisor) required by 10 CFR 50.54(m)(2)(i) to provide the technical expertise on shift. However, the 5.2.2.g wording of, "the STA shall provide ... support to the Shift Supervisor...", is considered to be easily misinterpreted to require separate individuals. Therefore, the wording is revised so that the STA function may be provided by either a separate individual or the individual who also fulfills another role in the shift command structure.

2. **Changes to 5.3, Unit Staff Qualifications**.

   a. In Section 5.3.1, the second sentence is bracketed in its entirety. There may be cases were the entire unit staff are covered by the standard specified in the first sentence or there may be specific exceptions for specific positions that could then be specified by bracketing the entire sentence.

   b. Definitions in 10 CFR 55.4 state: "Actively performing the functions of an operator or senior operator means that an individual has a position on the shift crew that requires the individual to be licensed as defined in the facility's technical specifications, and that ...." Adding paragraph 5.3.2 ensures that there is no misunderstanding when complying with 10 CFR 55.4 requirements. Adding this paragraph is consistent with the recommendations in the April 9, 1997 letter from C. Grimes to J. Davis.

   c… The April 9, 1997 letter from C. Grimes to J. Davis proposed a Reviewers Note in conjunction with the addition of paragraph 5.3.2. The Reviewer's Note stated: "The minimum staffing requirements stipulated in 10 CFR 50.54(m), for unit members actively performing the functions of an operator or senior operator, can be exceeded by stipulating the enhanced staffing requirements in paragraph 5.3.2." This Reviewer's Note is not required based on the discussions in Generic Letter 87-16 (Transmittal of NUREG-1262) which indicated that facilities can take credit for more than the minimum number of watchstanders required by Technical Specifications provided that there are administrative controls which assure that functions and duties are divided and rotated in a manner which provides each watchstander meaningful and significant opportunity to maintain proficiency in the performance of the functions of an operator and/or senior operator as appropriate. By stipulating enhanced staffing requirements in paragraph 5.3.2, when a licensee decided to change its staffing requirements, a license amendment request would have to be submitted, reviewed and approved before the staffing requirements could be made which may not be timely and is an unnecessary burden on the licensee's and NRC's resources.

3. **Changes to 5.5.4, Radioactive Effluent Controls Program.**

   - After issuance of Generic Letter 89-01, 10 CFR 20 was updated. The NRC issued a draft Generic Letter, 93-XX, on proposed changes to STS NUREGS base on the new 10 CFR 20. The proposed changes are consistent with the draft generic letter and the April 9, 1997 letter from C. Grimes to J. Davis with some exceptions noted below. The proposed changes maintain the same overall level of effluent control while retaining the operational flexibility that exists with current TS under the previous 10 CFR 20. This limitation (i.e., less than 10 times the concentration values...) provides reasonable assurance that the levels of radioactive materials in bodies of water in Unrestricted Areas will result in exposures within (1) the Section II.A design objectives of appendix I to 10 CFR Part 50 and (2) restrictions authorized by 10 CFR 20.1301(e). These changes are intended to eliminate possible confusion or improper implementation of the revised 10 CFR 20 requirements. The recommendations in the April 9, 1997 letter uses the term "total body" in reference to the noble gas dose rate. This limit is based on the dosimetry of ICRP 2, and the correct term is "whole body" as shown in NUREG-1301, Specification 3.11.2.1, page 45. Additionally, some minor editorial changes were made from the recommendations in the April 9, 1997 letter.
   - The provisions of SR 3.0.2 are applied to the Radioactive Effluent Controls Program surveillance frequencies (5.5.4e.) to allow for scheduling flexibility. SR 3.0.2 permits a 25%

extension of the interval specified in the Frequency (31 days). Allowing a 25% extension in the frequency of performing the monthly cumulative dose and projected dose calculation for the current quarter/year will have no affect on outcome of the calculations.

3. **Changes to 5.6.4, Monthly Operating Reports.** The reporting of pressurizer safety and relief valve failures and challenges is based on the guidance in NUREG-0694, "TMI-Related Requirements for New Operating Licensees." The guidance of NUREG-0694 states: "Assure that any failure of a PORV or safety valve to close will be reported to the NRC promptly. All challenges to the PORVs or safety valves should be documented in the annual report." NRC Generic Letter 97-02, "Revised Contents of the Monthly Operating Report" requests the submittal of less information in the monthly operating report. The generic letter identifies what needs to be reported to support the NRC Performance Indicator Program, and availability and capacity statistics. The generic letter does not specifically identify the need to report challenges to the pressurizer safety and relief valves. Mr. Marcel Harper, NRC (AEOD) was contacted and he indicated that this information was not required for the Performance Indicator Program and therefore would not need to be reported. Based on this information, it is acceptable to delete the requirement to provide documentation of all challenges to the pressurizer power operated relief valves or pressurizer safety valves.

5. **Changes to 5.7, High Radiation Area.** Section 5.7 is revised in accordance with 10 CFR 20.1601(c) and updates the acceptable alternate controls to those given in 10 CFR 20.1601. These changes are consistent (with the exception provided below) with the draft Generic Letter (93-XX) on proposed changes to STS NUREGs based on the new 10 CFR 20 and the letter from C. Grimes, NRC, to J. Davis, NEI dated April 9, 1997. (The NRC proposed version of Section 5.7 provided in the April 9, 1997 letter is included in this traveler with the recommended changes marked.)

- Changes to 5.7.1d.4.(ii): In the event that communications are lost between an individual worker, and the Radiation Protection staff providing the remote surveillance, the worker should be able to continue to work in the area provided that the worker can communicate with other workers in the same area who are working on the same job and under the same RWP, and provided that the communications remain satisfactory between these workers and the RP staff providing the remote surveillance..

- Changes to 5.7.1.e and 5.7.2.e: Revised to allow any individual or group of individuals to enter a high-high radiation area (dose rates > 1 Rem/hr at 30 cm) when accompanied by an individual qualified in radiation protection procedures with a radiation dose rate monitoring device. The qualified individual is responsible for providing positive control and shall perform periodic radiation surveillances at the frequency specified in the RWP. Furthermore, these continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry. Many plant's CTS requirements allow this option, which compliments the plant's practices of requiring qualified individual escort at all times during the work in a high-high radiation area. This option would provide adequate protection while (keeping with ALARA practices) minimizing exposure to the qualified individual.

- Changes to 5.7.2a: Section 5.7.2a is revised to state “Each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked or continuously guarded door or gate ...” This change is consistent with RG 8.38 Section 2.5 which indicates that the use of a locked door or one control point where positive control over personnel entry is exercised. Posting an individual to monitor a door provides positive controls over a high radiation area.

- Changes to 5.7.2.a.1: The Shift Foreman is only one of the many possible operations shift management positions who may be designated for the key control function. This change is similar to the wording of the NRC 7-28-95 letter to the Owner's Group Chairmen which identifies key control responsibility with the "shift supervisor, radiation protection manager, or his or her designee."

- Changes to 5.7.2f. (deleting “that is controlled as a high radiation area”): The 5.7.2.f provision has applied (in previous STS as well as ISTS NUREGs) without the added constraint of having the larger area controlled as a high radiation area. It is not always practical to control such areas as a High Radiation Area (outside of these High-High Radiation Areas). The proposed change to the NRC proposed Model Specification would restore the requirement as it exists in ISTS NUREG Rev1.

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5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

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

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

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

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

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

d. Each individual or group entering such an area shall possess:

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

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

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

(continued)

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5.7 High Radiation Area

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

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

   (i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

   (ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with individuals in the area who are covered by such surveillance.

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

INSERT F

5.7 High Radiation Area

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation

a. Each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked or continuously guarded door or gate that prevents unauthorized entry, and, in addition:

   1. All such door and gate keys shall be maintained under the administrative control of the shift supervisor, radiation protection manager, or his or her designee.

   2. Doors and gates shall remain locked except during periods of personnel or equipment entry or exit.

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

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

d. Each individual or group entering such an area shall possess:

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

(continued)

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation (continued)

2. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area with the means to communicate with and control every individual in the area, or

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

   (i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

   (ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with and control every individual in the area.

4. In those cases where options (2) and (3), above, are impractical or determined to be inconsistent with the "As Low As is Reasonably Achievable" principle, a radiation monitoring device that continuously displays radiation dose rates in the area.

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. . These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

(continued)

INSERT F

5.7 High Radiation Area

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation (continued)

f. Such individual areas that are within a larger area where no enclosure exists for the purpose of locking and where no enclosure can reasonably be constructed around the individual area need not be controlled by a locked door or gate, nor continuously guarded, but shall be barricaded, conspicuously posted, and a clearly visible flashing light shall be activated at the area as a warning device.

INSERT A

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

INSERT B

5.3.2 For the purpose of 10 CFR 55.4, a licensed Senior Reactor Operator (SRO) and a licensed reactor operator (RO) are those individuals who, in addition to meeting the requirements of TS 5.3.1, perform the functions described in 10 CFR 50.54(m).

INSERT C

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

INSERT D

shall be in accordance with the following:

1. For noble gases: a dose rate ≤ 500 mrem/yr to the whole body and a dose rate ≤ 3000 mrem/yr to the skin, and

2. For iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days: a dose rate ≤ 1500 mrem/yr to any organ;

INSERT E

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluent Controls Program surveillance frequency.

INSERT G

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

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5.2 Organization

5.2.2 Unit Staff (continued)

shall be assigned for each control room from which a reactor is operating in MODES 1, 2, 3, or 4.

[Two unit sites with both units shutdown or defueled require a total of three non-licensed operators for the two units.]

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

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

c d. A [Health Physics Technician] shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

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

e Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating personnel work an [8 or 12] hour day, nominal 40 hour week while the unit is operating. However, in the event that unforeseen problems require substantial amounts of overtime to be used, or during extended periods of shutdown for refueling, major maintenance, or major plant modification, on a temporary basis the following guidelines shall be followed:

1. An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time;

(continued)

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5.2 Organization

5.2.2 Unit Staff (continued)

2. An individual should not be permitted to work more than 16 hours in any 24 hour period, nor more than 24 hours in any 48 hour period, nor more than 72 hours in any 7 day period, all excluding shift turnover time;

3. A break of at least 8 hours should be allowed between work periods, including shift turnover time;

4. Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.

Insert A

Any deviation from the above guidelines shall be authorized in advance by the [Plant Superintendent] or his designee, in accordance with approved administrative procedures, or by higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation.

(the [Plant Superintendent's])

Insert G

Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the [Plant Superintendent] or his designee to ensure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized.

OR

working hour

shall not be

The amount of overtime worked by unit staff members performing safety related functions shall be limited and controlled in accordance with the NRC Policy Statement on working hours (Generic Letter 82-12).

e → f. The [Operations Manager or Assistant Operations Manager] shall hold an SRO license.

An individual

f g. The Shift Technical Advisor (STA) shall provide advisory technical support to the Shift Supervisor (SS) in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. In addition, the STA shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift.

unit operations shift crew

This individual

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5.0 ADMINISTRATIVE CONTROLS

5.3 Unit Staff Qualifications

Reviewer's Note: Minimum qualifications for members of the unit staff shall be specified by use of an overall qualification statement referencing an ANSI Standard acceptable to the NRC staff or by specifying individual position qualifications. Generally, the first method is preferable; however, the second method is adaptable to those unit staffs requiring special qualification statements because of unique organizational structures.

5.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of [Regulatory Guide 1.8, Revision 2, 1987, or more recent revisions, or ANSI Standard acceptable to the NRC staff]. [The staff not covered by [Regulatory Guide 1.8] shall meet or exceed the minimum qualifications of [Regulations, Regulatory Guides, or ANSI Standards acceptable to NRC staff].

Insert B

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5.5.4 Radioactive Effluent Controls Program (continued)

be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to ~~10 CFR 20, Appendix B, Table 2, Column 2;~~ Insert C

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

d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each unit to unrestricted areas, conforming to 10 CFR 50, Appendix I;

e. Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days;

f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2% of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I;

g. Limitations on the dose rate resulting from (from the site) radioactive material released in gaseous effluents to areas at or beyond the site boundary ~~conforming to the dose associated with 10 CFR 20, Appendix B, Table 2, Column 1;~~ Insert D

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

(continued)

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5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and

j. Limitations on the annual dose or dose commitment to any member of the public, beyond the site boundary, due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

Insert E

5.5.5 Component Cyclic or Transient Limit

This program provides controls to track the FSAR, Section [ ], cyclic and transient occurrences to ensure that components are maintained within the design limits.

[5.5.6 Pre-Stressed Concrete Containment Tendon Surveillance Program

This program provides controls for monitoring any tendon degradation in pre-stressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial operations. The Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with [Regulatory Guide 1.35, Revision 3, 1989].

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Tendon Surveillance Program inspection frequencies.]

5.5.7 Reactor Coolant Pump Flywheel Inspection Program

This program shall provide for the inspection of each reactor coolant pump flywheel per the recommendations of Regulatory. Position c.4.b of Regulatory Guide 1.14, Revision 1, August 1975.

(continued)

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5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience, ~~including documentation of all challenges to the pressurizer power operated relief valves or pressurizer safety valves,~~ shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

[The individual specifications that address core operating limits must be referenced here.]

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

[Identify the Topical Report(s) by number, title, date, and NRC staff approval document, or identify the staff Safety Evaluation Report for a plant specific methodology by NRC letter and date.]

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

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

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heat up, cooldown, low temperature operation, criticality, and hydrostatic

(continued)

Insert F

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[5.7 High Radiation Area]

5.7.1 Pursuant to 10 CFR 20, paragraph 20.1601(c), in lieu of the requirements of 10 CFR 20.1601, each high radiation area, as defined in 10 CFR 20, in which the intensity of radiation is > 100 mrem/hr but < 1000 mrem/hr, shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP). Individuals qualified in radiation protection procedures (e.g., [Health Physics Technicians]) or personnel continuously escorted by such individuals may be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas with exposure rates ≤ 1000 mrem/hr, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas.

Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device that continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device that continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel are aware of them.

c. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the [Radiation Protection Manager] in the RWP.

5.7.2 In addition to the requirements of Specification 5.7.1, areas with radiation levels ≥ 1000 mrem/hr shall be provided with locked or continuously guarded doors to prevent unauthorized entry and the keys shall be maintained under the administrative control of the Shift Foreman on duty or health physics supervision. Doors shall remain locked except during periods of access by personnel under an approved RWP that shall specify the dose rate levels in

(continued)

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[5.7 High Radiation Area]

5.7.2 (continued)

the immediate work areas and the maximum allowable stay times for individuals in those areas. In lieu of the stay time specification of the RWP, direct or remote (such as closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area.

5.7.3 For individual high radiation areas with radiation levels of > 1000 mrem/hr, accessible to personnel, that are located within large areas such as reactor containment, where no enclosure exists for purposes of locking, or that cannot be continuously guarded, and where no enclosure can be reasonably constructed around the individual area, that individual area shall be barricaded and conspicuously posted, and a flashing light shall be activated as a warning device.

5.2.2 Unit Staff (continued)

shall be assigned for each control room from which a reactor is operating in MODES 1, 2, 3, or 4.

[Two unit sites with both units shutdown or defueled require a total of three non-licensed operators for the two units.]

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

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

c d. A [Health Physics Technician] shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

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

e Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating personnel work an [8 or 12] hour day, nominal 40 hour week while the unit is operating. However, in the event that unforeseen problems require substantial amounts of overtime to be used, or during extended periods of shutdown for refueling, major maintenance, or major plant modification, on a temporary basis the following guidelines shall be followed:]

e

(continued)

5.2.2 Unit Staff (continued)

1. An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time;

2. An individual should not be permitted to work more than 16 hours in any 24 hour period, nor more than 24 hours in any 48 hour period, nor more than 72 hours in any 7 day period, all excluding shift turnover time;

3. A break of at least 8 hours should be allowed between work periods, including shift turnover time;

4. Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.

Insert A

Any deviation from the above guidelines shall be authorized in advance by the [Plant Superintendent] or his designee, in accordance with approved administrative procedures, or by higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation.

Insert G

Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the [Plant Superintendent] or his designee to ensure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized.

OR

The amount of overtime worked by unit staff members performing safety related functions shall be limited and controlled in accordance with the NRC Policy Statement on working hours (Generic Letter 82-12).

e f. The [Operations Manager or Assistant Operations Manager] shall hold an SRO license.

An individual

f g. The Shift Technical Advisor (STA) shall provide advisory technical support to the Shift Supervisor (SS) in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. In addition, the STA shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift.

This individual

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5.0 ADMINISTRATIVE CONTROLS

5.3 Unit Staff Qualifications

Reviewer's Note: Minimum qualifications for members of the unit staff shall be specified by use of an overall qualification statement referencing an ANSI Standard acceptable to the NRC staff or by specifying individual position qualifications. Generally, the first method is preferable; however, the second method is adaptable to those unit staffs requiring special qualification statements because of unique organizational structures.

5.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of [Regulatory Guide 1.8, Revision 2, 1987, or more recent revisions, or ANSI Standard acceptable to the NRC staff]. [The staff not covered by [Regulatory Guide 1.8] shall meet or exceed the minimum qualifications of [Regulations, Regulatory Guides, or ANSI Standards acceptable to NRC staff].

Insert B →

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5.5.4 Radioactive Effluent Controls Program (continued)

be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to ~~10 CFR 20, Appendix B, Table 2, Column 2;~~ Insert C

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

d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each unit to unrestricted areas, conforming to 10 CFR 50, Appendix I;

e. Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days;

f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2% of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I;

g. Limitations on the dose rate resulting from (from the site) radioactive material released in gaseous effluents to areas (at or) beyond the site boundary ~~conforming to the dose associated with 10 CFR 20, Appendix B, Table 2, Column 1;~~ Insert D

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

(continued)

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5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and

j. Limitations on the annual dose or dose commitment to any member of the public, beyond the site boundary, due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

Insert E

5.5.5 Component Cyclic or Transient Limit

This program provides controls to track the FSAR, Section [ ], cyclic and transient occurrences to ensure that components are maintained within the design limits.

[5.5.6 Pre-Stressed Concrete Containment Tendon Surveillance Program

This program provides controls for monitoring any tendon degradation in pre-stressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial operations. The Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with [Regulatory Guide 1.35, Revision 3, 1989].

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Tendon Surveillance Program inspection frequencies.]

5.5.7 Reactor Coolant Pump Flywheel Inspection Program

This program shall provide for the inspection of each reactor coolant pump flywheel per the recommendation of Regulatory position c.4.b of Regulatory Guide 1.14, Revision 1, August 1975.

(continued)

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5.6 Reporting Requirements (continued)

5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the pressurizer power operated relief valves or pressurizer safety valves, shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

[The individual specifications that address core operating limits must be referenced here.]

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

[Identify the Topical Report(s) by number, title, date, and NRC staff approval document, or identify the staff Safety Evaluation Report for a plant specific methodology by NRC letter and date.]

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

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

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heat up, cooldown, low temperature operation, criticality, and hydrostatic

(continued)

Insert F

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[5.7 High Radiation Area]

5.7.1 Pursuant to 10 CFR 20, paragraph 20.1601(c), in lieu of the requirements of 10 CFR 20.1601, each high radiation area, as defined in 10 CFR 20, in which the intensity of radiation is > 100 mrem/hr but < 1000 mrem/hr, shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP). Individuals qualified in radiation protection procedures (e.g., [Health Physics Technicians]) or personnel continuously escorted by such individuals may be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas with exposure rates ≤ 1000 mrem/hr, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas.

Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device that continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device that continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel are aware of them.

c. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the [Radiation Protection Manager] in the RWP.

5.7.2 In addition to the requirements of Specification 5.7.1, areas with radiation levels ≥ 1000 mrem/hr shall be provided with locked or continuously guarded doors to prevent unauthorized entry and the keys shall be maintained under the administrative control of the Shift Foreman on duty or health physics supervision. Doors shall remain locked except during periods of access by personnel

(continued)

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[5.7 High Radiation Area]

5.7.2 (continued)

under an approved RWP that shall specify the dose rate levels in the immediate work areas and the maximum allowable stay times for individuals in those areas. In lieu of the stay time specification of the RWP, direct or remote (such as closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area.

5.7.3 For individual high radiation areas with radiation levels of > 1000 mrem/hr, accessible to personnel, that are located within large areas such as reactor containment, where no enclosure exists for purposes of locking, or that cannot be continuously guarded, and where no enclosure can be reasonably constructed around the individual area, that individual area shall be barricaded and conspicuously posted, and a flashing light shall be activated as a warning device.

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5.2 Organization

5.2.2 Unit Staff (continued)

shall be assigned for each control room from which a reactor is operating in MODES 1, 2, 3, or 4.

[Two unit sites with both units shutdown or defueled require a total of three non-licensed operators for the two units.]

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

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

c → d. A [Health Physics Technician] shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

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

Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating personnel work an [8 or 12] hour day, nominal 40 hour week while the unit is operating. However, in the event that unforeseen problems require substantial amounts of overtime to be used, or during extended periods of shutdown for refueling, major maintenance, or major plant modification, on a temporary basis the following guidelines shall be followed:

1. An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time;

(continued)

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5.2.2 Unit Staff (continued)

2. An individual should not be permitted to work more than 16 hours in any 24 hour period, nor more than 24 hours in any 48 hour period, nor more than 72 hours in any 7 day period, all excluding shift turnover time;

3. A break of at least 8 hours should be allowed between work periods, including shift turnover time;

4. Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.

Insert A

Any deviation from the above guidelines shall be authorized in advance by the [Plant Superintendent] or his designee, in accordance with approved administrative procedures, or by higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation.

Insert G

Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the [Plant Superintendent] or his designee to ensure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized.

OR

The amount of overtime worked by unit staff members performing safety related functions shall be limited and controlled in accordance with the NRC Policy Statement on working hours (Generic Letter 82-12).

e → f. The [Operations Manager or Assistant Operations Manager] shall hold an SRO license.

f → g. The Shift Technical Advisor (STA) shall provide advisory technical support to the Shift Supervisor (SS) in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. In addition, the STA shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift.

An individual

This individual

unit operations shift crew

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5.0 ADMINISTRATIVE CONTROLS

5.3 Unit Staff Qualifications

Reviewer's Note: Minimum qualifications for members of the unit staff shall be specified by use of an overall qualification statement referencing an ANSI Standard acceptable to the NRC staff or by specifying individual position qualifications. Generally, the first method is preferable; however, the second method is adaptable to those unit staffs requiring special qualification statements because of unique organizational structures.

5.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of [Regulatory Guide 1.8, Revision 2, 1987, or more recent revisions, or ANSI Standard acceptable to the NRC staff]. C The staff not covered by Regulatory Guide 1.8 shall meet or exceed the minimum qualifications of Regulations, Regulatory Guides, or ANSI Standards acceptable to NRC staff].

Insert B

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5.5.4 Radioactive Effluent Controls Program (continued)

achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to ~~10 CFR 20, Appendix B, Table 2, Column 2;~~ Insert C

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

d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each unit to unrestricted areas, conforming to 10 CFR 50, Appendix I;

e. Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days;

f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2% of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I;

g. Limitations on the dose rate resulting from the site from radioactive material released ~~in gaseous effluents~~ to areas at or beyond the site boundary ~~conforming to the dose associated with 10 CFR 20, Appendix B, Table 2, Column 1;~~ Insert D

(continued)

5.5 Programs and Manuals

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5.5.4 Radioactive Effluent Controls Program (continued)

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and

j. Limitations on the annual dose or dose commitment to any member of the public, beyond the site boundary, due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

Insert E

5.5.5 Component Cyclic or Transient Limit

This program provides controls to track the FSAR Section [ ] cyclic and transient occurrences to ensure that components are maintained within the design limits.

[5.5.6 Pre-Stressed Concrete Containment Tendon Surveillance Program

This program provides controls for monitoring any tendon degradation in pre-stressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial operations. The Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with [Regulatory Guide 1.35, Revision 3, 1989].

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Tendon Surveillance Program inspection frequencies.]

5.5.7 Reactor Coolant Pump Flywheel Inspection Program

This program shall provide for the inspection of each reactor coolant pump flywheel per the recommendations of regulatory position c.4.b of Regulatory Guide 1.14, Revision 1, August 1975.

(continued)

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5.6 Reporting Requirements

5.6.2 Annual Radiological Environmental Operating Report (continued)

(ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements [in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979]. [The report shall identify the TLD results that represent collocated dosimeters in relation to the NRC TLD program and the exposure period associated with each result.] In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.6.3 Radioactive Effluent Release Report

----------NOTE----------
A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

The Radioactive Effluent Release Report covering the operation of the unit shall be submitted in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience[, including documentation of all challenges to the pressurizer

(continued)

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[5.7 High Radiation Area]

5.7.2 (continued)

under an approved RWP that shall specify the dose rate levels in the immediate work areas and the maximum allowable stay times for individuals in those areas. In lieu of the stay time specification of the RWP, direct or remote (such as closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area.

5.7.3 For individual high radiation areas with radiation levels of > 1000 mrem/hr, accessible to personnel, that are located within large areas such as reactor containment, where no enclosure exists for purposes of locking, or that cannot be continuously guarded, and where no enclosure can be reasonably constructed around the individual area, that individual area shall be barricaded and conspicuously posted, and a flashing light shall be activated as a warning device.

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5.6 Reporting Requirements

5.6.4 Monthly Operating Reports (continued)

power operated relief valves or pressurizer safety valves,] shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

[The individual specifications that address core operating limits must be referenced here.]

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

[Identify the Topical Report(s) by number, title, date, and NRC staff approval document, or identify the staff Safety Evaluation Report for a plant specific methodology by NRC letter and date.]

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

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

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heatup, cooldown, low temperature operation, critically, and hydrostatic

(continued)

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5.0 ADMINISTRATIVE CONTROLS

Insert F

[5.7 High Radiation Area]

5.7.1 Pursuant to 10 CFR 20, paragraph 20.1601(c), in lieu of the requirements of 10 CFR 20.1601, each high radiation area, as defined in 10 CFR 20, in which the intensity of radiation is > 100 mrem/hr but < 1000 mrem/hr, shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP). Individuals qualified in radiation protection procedures (e.g., [Health Physics Technicians]) or personnel continuously escorted by such individuals may be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas with exposure rates ≤ 1000 mrem/hr, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas.

Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device that continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device that continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel are aware of them.

c. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the [Radiation Protection Manager] in the RWP.

5.7.2 In addition to the requirements of Specification 5.7.1, areas with radiation levels ≥ 1000 mrem/hr shall be provided with locked or continuously guarded doors to prevent unauthorized entry and the keys shall be maintained under the administrative control of the Shift Foreman on duty or health physics supervision. Doors shall remain locked except during periods of access by personnel

(continued)

5.2.2 Unit Staff (continued)

shall be assigned for each control room from which a reactor is operating in MODES 1, 2, or 3.

[Two unit sites with both units shutdown or defueled require a total of three non-licensed operators for the two units.]

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

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

c → d. A [Health Physics Technician] shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

d → [e. Administrative procedures shall be developed (personnel) and implemented to limit the working hours of unit staff who perform safety related functions (e.g., licensed SROs (Senior Reactor Operators (SROs)), licensed ROs (Reactor Operators (ROs)), health physicists, auxiliary operators, and key maintenance personnel).

Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating personnel work an [8 or 12] hour day, nominal 40 hour week while the unit is operating. However, in the event that unforeseen problems require substantial amounts of overtime to be used, or during extended periods of shutdown for refueling, major maintenance, or major plant modification, on a temporary basis the following guidelines shall be followed:

1. An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time;]

(continued)

5.2.2 Unit Staff (continued)

2. An individual should not be permitted to work more than 16 hours in any 24 hour period, nor more than 24 hours in any 48 hour period, nor more than 72 hours in any 7 day period, all excluding shift turnover time;

3. A break of at least 8 hours should be allowed between work periods, including shift turnover time;

4. Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.

Insert A

Any deviation from the above guidelines shall be authorized in advance by the [Plant Superintendent] or his designee, in accordance with approved administrative procedures, or by higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation.

Insert G

Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the [Plant Superintendent] or his designee to ensure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized.

OR

The amount of overtime worked by unit staff members performing safety related functions shall be limited and controlled in accordance with the NRC Policy Statement on working hours (Generic Letter 82-12).

e f. The [Operations Manager or Assistant Operations Manager] shall hold an SRO license.

f g. The Shift Technical Advisor (STA) shall provide advisory technical support to the Shift Supervisor (SS) in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. In addition, the STA shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift.

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5.0 ADMINISTRATIVE CONTROLS

5.3 Unit Staff Qualifications

Reviewer's Note: Minimum qualifications for members of the unit staff shall be specified by use of an overall qualification statement referencing an ANSI Standard acceptable to the NRC staff or by specifying individual position qualifications. Generally, the first method is preferable; however, the second method is adaptable to those unit staffs requiring special qualification statements because of unique organizational structures.

5.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of [Regulatory Guide 1.8, Revision 2, 1987, or more recent revisions, or ANSI Standard acceptable to the NRC staff]. The staff not covered by [Regulatory Guide 1.8] shall meet or exceed the minimum qualifications of [Regulations, Regulatory Guides, or ANSI Standards acceptable to NRC staff].

Insert B

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5.5.4 Radioactive Effluent Controls Program (continued)

achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to ~~10 CFR 20, Appendix B, Table 2, Column 2~~; Insert C

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

d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each unit to unrestricted areas, conforming to 10 CFR 50, Appendix I;

e. Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days;

f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2% of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I;

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the site boundary ~~conforming to the dose associated with 10 CFR 20, Appendix B, Table 2, Column 1~~; Insert D

from the site

at or

(continued)

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5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

j. Limitations on the annual dose or dose commitment to any member of the public, beyond the site boundary, due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190; and

k. Limitations on venting and purging of the Mark II containment through the Standby Gas Treatment System to maintain releases as low as reasonably achievable (in BWR/4s with Mark II containments).

Insert E

5.5.5 Component Cyclic or Transient Limit

This program provides controls to track the FSAR Section [ ], cyclic and transient occurrences to ensure that components are maintained within the design limits.

[5.5.6 Pre-Stressed Concrete Containment Tendon Surveillance Program

This program provides controls for monitoring any tendon degradation in pre-stressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial operations. The Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with [Regulatory Guide 1.35, Revision 3, 1989].

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Tendon Surveillance Program inspection frequencies.]

(continued)

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5.6.2 Annual Radiological Environmental Operating Report (continued)

(ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements [in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979]. [The report shall identify the TLD results that represent collocated dosimeters in relation to the NRC TLD program and the exposure period associated with each result.] In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.6.3 Radioactive Effluent Release Report

----------NOTE----------
A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

The Radioactive Effluent Release Report covering the operation of the unit shall be submitted in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the safety/relief

(continued)

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5.6 Reporting Requirements

5.6.4 Monthly Operating Reports (continued)

valves,] shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

[The individual specifications that address core operating limits must be referenced here.]

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

[Identify the Topical Report(s) by number, title, date, and NRC staff approval document, or identify the staff Safety Evaluation Report for a plant specific methodology by NRC letter and date.]

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

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

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heatup, cooldown, low temperature operation, critically, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

(continued)

Insert f

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[5.7 High Radiation Area]

5.7.1 Pursuant to 10 CFR 20, paragraph 20.1601(c), in lieu of the requirements of 10 CFR 20.1601, each high radiation area, as defined in 10 CFR 20, in which the intensity of radiation is > 100 mrem/hr but < 1000 mrem/hr, shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP). Individuals qualified in radiation protection procedures (e.g., [Health Physics Technicians]) or personnel continuously escorted by such individuals may be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas with exposure rates ≤ 1000 mrem/hr, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas.

Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device that continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device that continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel are aware of them.

c. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the [Radiation Protection Manager] in the RWP.

5.7.2 In addition to the requirements of Specification 5.7.1, areas with radiation levels ≥ 1000 mrem/hr shall be provided with locked or continuously guarded doors to prevent unauthorized entry and the keys shall be maintained under the administrative control of the Shift Foreman on duty or health physics supervision. Doors shall remain locked except during periods of access by personnel under an approved RWP that shall specify the dose rate levels in the immediate work

(continued)

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[5.7 High Radiation Area]

5.7.2 (continued)

areas and the maximum allowable stay times for individuals in those areas. In lieu of the stay time specification of the RWP, direct or remote (such as closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area.

5.7.3 For individual high radiation areas with radiation levels of > 1000 mrem/hr, accessible to personnel, that are located within large areas such as reactor containment, where no enclosure exists for purposes of locking, or that cannot be continuously guarded, and where no enclosure can be reasonably constructed around the individual area, that individual area shall be barricaded and conspicuously posted, and a flashing light shall be activated as a warning device.

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5.2 Organization

5.2.2 Unit Staff (continued)

shall be assigned for each control room from which a reactor is operating in MODES 1, 2, or 3.

[Two unit sites with both units shutdown or defueled require a total of three non-licensed operators for the two units.]

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

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

c → ~~d~~. A [Health Physics Technician] shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

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

Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating personnel work an [8 or 12] hour day, nominal 40 hour week, while the unit is operating. However, in the event that unforeseen problems require substantial amounts of overtime to be used, or during extended periods of shutdown for refueling, major maintenance, or major plant modification, on a temporary basis the following guidelines shall be followed:

(continued)

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5.2 Organization

5.2.2 Unit Staff (continued)

1. An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time;

2. An individual should not be permitted to work more than 16 hours in any 24 hour period, nor more than 24 hours in any 48 hour period, nor more than 72 hours in any 7 day period, all excluding shift turnover time;

3. A break of at least 8 hours should be allowed between work periods, including shift turnover time;

4. Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.

Insert A

Any deviation from the above guidelines shall be authorized in advance by the [Plant Superintendent] or his designee, in accordance with approved administrative procedures, or by higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation.

Insert G

Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the [Plant Superintendent] or his designee to ensure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized.

OR

The amount of overtime worked by unit staff members performing safety related functions shall be limited and controlled in accordance with the NRC Policy Statement on working hours (Generic Letter 82-12).

e → f. The [Operations Manager or Assistant Operations Manager] shall hold an SRO license.

f → g. The Shift Technical Advisor (STA) shall provide advisory technical support to the Shift Supervisor (SS) in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. In addition, the STA shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift.

Unit operations shift crew

An individual

This individual

5.0 ADMINISTRATIVE CONTROLS

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5.3 Unit Staff Qualifications

Reviewer's Note: Minimum qualifications for members of the unit staff shall be specified by use of an overall qualification statement referencing an ANSI Standard acceptable to the NRC staff or by specifying individual position qualifications. Generally, the first method is preferable; however, the second method is adaptable to those unit staffs requiring special qualification statements because of unique organizational structures.

5.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of [Regulatory Guide 1.8, Revision 2, 1987, or more recent revisions, or ANSI Standard acceptable to the NRC staff]. [The staff not covered by [Regulatory Guide 1.8] shall meet or exceed the minimum qualifications of [Regulations, Regulatory Guides, or ANSI Standards acceptable to NRC staff].

Insert B

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5.5.4 Radioactive Effluent Controls Program (continued)

the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to ~~10 CFR 20, Appendix B, Table 2, Column 2;~~ Insert C

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

d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each unit to unrestricted areas, conforming to 10 CFR 50, Appendix I;

e. Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days;

f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2% of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I;

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the site boundary ~~conforming to the dose associated with 10 CFR 20, Appendix B, Table 2, Column 1;~~ Insert D

(from the site) (at or)

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each

(continued)

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5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and

j. Limitations on the annual dose or dose commitment to any member of the public, beyond the site boundary, due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

Insert E

5.5.5 Component Cyclic or Transient Limit

This program provides controls to track the FSAR, Section [ ], cyclic and transient occurrences to ensure that components are maintained within the design limits.

5.5.6 Pre-Stressed Concrete Containment Tendon Surveillance Program

This program provides controls for monitoring any tendon degradation in pre-stressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial operations. The Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with [Regulatory Guide 1.35, Revision 3, 1989].

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Tendon Surveillance Program inspection frequencies.

5.5.7 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components including applicable supports. The program shall include the following:

(continued)

5.6 Reporting Requirements

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5.6.2 Annual Radiological Environmental Operating Report (continued)

(ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements [in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979]. [The report shall identify the TLD results that represent collocated dosimeters in relation to the NRC TLD program and the exposure period associated with each result.] In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.6.3 Radioactive Effluent Release Report

---NOTE---
A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

The Radioactive Effluent Release Report covering the operation of the unit shall be submitted in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience[, including documentation of all challenges to the safety/relief

(continued)

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5.6 Reporting Requirements

5.6.4 Monthly Operating Reports (continued)

~~valves,~~ shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

[The individual specifications that address core operating limits must be referenced here.]

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

[Identify the Topical Report(s) by number, title, date, and NRC staff approval document, or identify the staff Safety Evaluation Report for a plant specific methodology by NRC letter and date.]

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

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

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heatup, cooldown, low temperature operation, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

(continued)

Insert f

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5.0 ADMINISTRATIVE CONTROLS

[5.7 High Radiation Area]

5.7.1 Pursuant to 10 CFR 20, paragraph 20.1601(c), in lieu of the requirements of 10 CFR 20.1601, each high radiation area, as defined in 10 CFR 20, in which the intensity of radiation is > 100 mrem/hr but < 1000 mrem/hr, shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP). Individuals qualified in radiation protection procedures (e.g., [Health Physics Technicians]) or personnel continuously escorted by such individuals may be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas with exposure rates ≤ 1000 mrem/hr, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas.

Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device that continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device that continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel are aware of them.

c. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the [Radiation Protection Manager] in the RWP.

5.7.2 In addition to the requirements of Specification 5.7.1, areas with radiation levels ≥ 1000 mrem/hr shall be provided with locked or continuously guarded doors to prevent unauthorized entry and the keys shall be maintained under the administrative control of the Shift Foreman on duty or health physics supervision. Doors shall remain locked except during periods of access by personnel

(continued)

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[5.7 High Radiation Area]

5.7.2 (continued)

under an approved RWP that shall specify the dose rate levels in the immediate work areas and the maximum allowable stay times for individuals in those areas. In lieu of the stay time specification of the RWP, direct or remote (such as closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area.

5.7.3 For individual high radiation areas with radiation levels of > 1000 mrem/hr, accessible to personnel, that are located within large areas such as reactor containment, where no enclosure exists for purposes of locking, or that cannot be continuously guarded, and where no enclosure can be reasonably constructed around the individual area, that individual area shall be barricaded and conspicuously posted, and a flashing light shall be activated as a warning device.

Proposed Section 5.7 from the April 9, 1997 C. Grimes letter, marked to show differences from TSTF-258

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5.0
ADMINISTRATIVE CONTROLS

5.7
High Radiation Area

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

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

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

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

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

otherwise

d. Each individual or group entering such an area shall possess:

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

Continued

5.7 High Radiation Area

5.7.1 (continued)

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

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

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

   (i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

   (ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with and control every individual in the area.

individuals in the area who are covered by such surveillance.

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them.

5.7.2 <u>High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from Any Surface Penetrated by the Radiation, but Less than 500 rads/hour at 1 Meter from the Radiation Source or from Any Surface Penetrated by the Radiation:</u>

a. Each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked

Continued

These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

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5.7 High Radiation Area

5.7.2 (continued) door or gate ~~that prevents unauthorized~~ or continuously guarded entry, and, in addition:

STEP MODEL

1. All such door and gate keys shall be maintained under the administrative control of the shift supervisor, radiation protection manager, or his or her designee.

2. Doors and gates shall remain locked except during ~~periods of personnel or equipment entry or exit.~~ access by personnel or equipment under an approved RWP.

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

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

d. Each individual or group entering such an area shall possess:

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

2. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area with the means to communicate with ~~and control every individual in the area~~ individuals in the area who are covered by such surveillance, or

STEP MODEL

Continued

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5.7 High Radiation Area

5.7.2 (continued)

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

   (i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

   (ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with ~~and control every individual in the area~~, or ~~individuals in the area who are covered by such surveillance~~

<STET MODEL>

4. In those cases where options (2) and (3), above, are impractical or determined to be inconsistent with the "As Low As is Reasonably Achievable" principle, a radiation monitoring device that continuously displays radiation dose rates in the area.

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas ~~shall~~ be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them.

<NEW Rev 1 wording>

f. Such individual areas that are within a larger area ~~that is controlled as a high radiation area,~~ where no enclosure exists for the purpose of locking and where no enclosure can reasonable be constructed around the individual area need not be controlled by a locked door or gate, but shall be barricaded, ~~and conspicuous~~, conspicuously posted, and a clearly visible flashing light shall be activated at the area as a warning device. nor continuously guarded

These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

Technical Specification Task Force
Improved Standard Technical Specifications Change Traveler

SFDP Clarifications

NUREGs Affected: ☑ 1430 ☑ 1431 ☑ 1432 ☑ 1433 ☑ 1434

Classification: 1) Correct Specifications Recommended for CLIIP?: (Unassigned)

Correction or Improvement: (Unassigned)

Industry Contact: Steve Wideman, (620) 364-4037, stwidem@wcnoc.com

Add to LCO 3.0.6 Bases clarification of "appropriate LCO for loss of function" and clarify in the requirements for the SFDP that consideration does not have to be made for a loss of power in determining loss of function.
The NUREGs were developed such that the Actions for a single support system inoperability would be addressed by that support system's Actions - without cascading to the supported system; even if both trains of the support system were inoperable resulting from a loss of function. This intent is clarified in the LCO 3.0.6 Bases. Without this clarification, supported systems with a single support system (such as both Containment Spray and ECCS trains supported by the Refueling Water Tank) would be declared inoperable when the support system is inoperable under the provisions of LCO 3.0.6 even though the support system Actions were designed to provide the appropriate response.

Also the NUREGs were developed with the appropriate "loss of function" (i.e., cross-train) check for electrical power inoperabilities contained within the LCO 3.8.1 Actions, without reliance on the SFDP. The NUREG Bases for LCO 3.8.1, Required Actions A.2 and B.2 (last paragraph in each) were added during development to attempt to clarify this issue. However, the actual requirements for the SFDP in Chapter 5.0 are sufficiently ambiguous to have resulted in misinterpretation. Therefore, clarification is added to the requirements for the SFDP, consistent with the intent of the NUREGs.

Revision History

OG Revision 0 Revision Status: Closed

Revision Proposed by: WOG MiniGroup

Revision Description:
Original Issue

Owners Group Review Information

Date Originated by OG: 14-Jan-97

Owners Group Comments:
(No Comments)

Owners Group Resolution: Approved Date: 19-Mar-97

TSTF Review Information

TSTF Received Date: 27-Mar-97 Date Distributed for Review: 06-Jan-98

OG Review Completed: ☑ BWOG ☑ WOG ☑ CEOG ☑ BWROG

TSTF Comments:

Originally distributed on 4/8/97
CEOG Comments from 4/24/97: Applicable, accepts.

2/5/98 - make changes to insert. Applicable to all OGs.

TSTF Resolution: Approved Date: 05-Feb-98

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OG Revision 0 Revision Status: Closed

NRC Review Information

NRC Received Date: 29-May-98

NRC Comments: Date of NRC Letter: 13-Jan-99

7/16/98 - The proposed additions are not necessary; what "misinterpretations" have resulted from the current wording of LCO 3.0.6 and the SFDP? The SFDP should take into consideration a complete loss of power. The proposed additions to the SFDP do not appear to be incorrect, though they also do not appear to be necessary. Likewise the first paragraph of the LCO 3.0.6 insert appears correct. However, the second paragraph of the LCO 3.0.6 insert is incorrect; it is not true for all circumstances. The following two sentences are not always true: "Where a loss of function is solely due to a single TS support system . . . The appropriate LCO is the LCO for the support system. The ACTIONS for a support systems LCO adequately addresses the inoperabilites of that system. . ." For a loss of function, the above two sentences contradict LCO 3.0.6.
9/24/98 - NRC agrees to reconsider rejection.
11/12/98 - NRC still reviewing. B. Tjader will contact B. Ford on 11/19/98 in afternoon to provide status or approval.
05/23/99 - Superceded to incorporate clarification of loss of function.
12/16/98 - There was discussion of both of the proposed changes of TSTF-273 and the NRC agreed conceptually with both changes. However, the NRC Technical Specifications Branch would like the proposed change regarding single LCO systems not triggering the SFDP to be addressed in the specification addressing the SFDP and in the Bases. The TSTF agreed to provide wording to the SFDP program to incorporate the agreed to concepts based on the initial wording of the SFDP and as agreed to by the TSB and TSTF during the discussion. TSTF will provide a revision to TSTF 273 to the NRC by 2/1/99 which addresses the agreements reached during this meeting.
1/13/99 - Recommendation is revised to modify TSTF-273 to change the SFDP A/C TS to reflect the original "Rev 0" meaning, and to make the Bases consistent with the revised TS.

Final Resolution: Superceded by Revision Final Resolution Date: 16-Dec-98

TSTF Revision 1 Revision Status: Closed

Revision Proposed by: TSTF

Revision Description:
Added statement to SFDP to clarify loss of function caused by the inoperability of a single TS support system (from TSTF-Weber review).

TSTF Review Information

TSTF Received Date: 28-May-99 Date Distributed for Review: 15-Jun-99

OG Review Completed: ☑ BWOG ☑ WOG ☑ CEOG ☑ BWROG

TSTF Comments:

(No Comments)

TSTF Resolution: Approved Date: 15-Jun-99

NRC Review Information

NRC Received Date: 23-Jun-99

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TSTF Revision 1 Revision Status: Closed

NRC Comments:

Received rewording suggestion for Insert 1 from NRC.

Final Resolution: Superceded by Revision Final Resolution Date: 17-Jul-99

TSTF Revision 2 Revision Status: Active

Revision Proposed by: NRC

Revision Description:
Made wording changes to Insert 1 per NRC suggestion.

TSTF Review Information

TSTF Received Date: 17-Jul-99 Date Distributed for Review: 17-Jul-99

OG Review Completed: ☑ BWOG ☑ WOG ☑ CEOG ☑ BWROG

TSTF Comments:
(No Comments)

TSTF Resolution: Approved Date: 17-Jul-99

NRC Review Information

NRC Received Date: 20-Jul-99

Date of NRC Letter: 16-Aug-99

Final Resolution: NRC Approves Final Resolution Date: 16-Aug-99

Affected Technical Specifications

LCO 3.0.6 Bases	LCO Applicability	
5.5.15	Safety Function Determination Program	NUREG(s)- 1430 1431 1432 Only
5.5.12	Safety Function Determination Program	NUREG(s)- 1433 1434 Only

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Insert 1

This loss of safety function does not require the assumption of additional single failures or loss of offsite power. Since operation is being restricted in accordance with the ACTIONS of the support system, any resulting temporary loss of redundancy or single failure protection is taken into account. Similarly, the ACTIONS for inoperable offsite circuit(s) and inoperable diesel generator(s) provide the necessary restriction for cross train inoperabilities. This explicit cross train verification for inoperable AC electrical power sources also acknowledges that supported system(s) are not declared inoperable solely as a result of inoperability of a normal or emergency electrical power source (refer to the definition of OPERABILITY).

When a loss of safety function is determined to exist, and the SFDP requires entry into the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists, consideration must be given to the specific type of function affected. Where a loss of function is solely due to a single Technical Specification support system (e.g., loss of automatic start due to inoperable instrumentation, or loss of pump suction source due to low tank level) the appropriate LCO is the LCO for the support system. The ACTIONS for a support system LCO adequately addresses the inoperabilities of that system without reliance on entering its supported system LCO. When the loss of function is the result of multiple support systems, the appropriate LCO is the LCO for the supported system.

Insert 2

When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

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INSERT 1

BASES

LCO 3.0.6 (continued)

Required Actions of the LCO in which the loss of safety function exists are required to be entered.

LCO 3.0.7

There are certain special tests and operations required to be performed at various times over the life of the unit. These special tests and operations are necessary to demonstrate select unit performance characteristics, to perform special maintenance activities, and to perform special evolutions. Test Exception LCOs [3.1.9, 3.1.10, 3.1.11, and 3.4.19] allow specified Technical Specification (TS) requirements to be changed to permit performances of these special tests and operations, which otherwise could not be performed if required to comply with the requirements of these TS. Unless otherwise specified, all the other TS requirements remain unchanged. This will ensure all appropriate requirements of the MODE or other specified condition not directly associated with or required to be changed to perform the special test or operation will remain in effect.

The Applicability of a Test Exception LCO represents a condition not necessarily in compliance with the normal requirements of the TS. Compliance with Test Exception LCOs is optional. A special operation may be performed either under the provisions of the appropriate Test Exception LCO or under the other applicable TS requirements. If it is desired to perform the special operation under the provisions of the Test Exception LCO, the requirements of the Test Exception LCO shall be followed.

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5.5 Programs and Manuals (continued)

5.5.15 Safety Function Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate limitations and remedial or compensatory actions may be identified to be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following:

a. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;

b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

d. Other appropriate limitations and remedial or compensatory actions.

, and assuming no concurrent loss of offsite power or loss of onsite diesel generator(s),

A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to the system(s) supported by the inoperable support system is also inoperable; or

b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to the support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

INSERT 2

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BASES

LCO 3.0.6 (continued)

system are OPERABLE, thereby ensuring safety function is retained. If this evaluation determines that a loss of safety function exists, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

INSERT 1

LCO 3.0.7

There are certain special tests and operations required to be performed at various times over the life of the unit. These special tests and operations are necessary to demonstrate select unit performance characteristics, to perform special maintenance activities, and to perform special evolutions. Test Exception LCOs [3.1.9, 3.1.10, 3.1.11, and 3.4.19] allow specified Technical Specification (TS) requirements to be changed to permit performances of these special tests and operations, which otherwise could not be performed if required to comply with the requirements of these TS. Unless otherwise specified, all the other TS requirements remain unchanged. This will ensure all appropriate requirements of the MODE or other specified condition not directly associated with or required to be changed to perform the special test or operation will remain in effect.

The Applicability of a Test Exception LCO represents a condition not necessarily in compliance with the normal requirements of the TS. Compliance with Test Exception LCOs is optional. A special operation may be performed either under the provisions of the appropriate Test Exception LCO or under the other applicable TS requirements. If it is desired to perform the special operation under the provisions of the Test Exception LCO, the requirements of the Test Exception LCO shall be followed.

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5.5 Programs and Manuals (continued)

5.5.15 Safety Function Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate actions may be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following:

a. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;

b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

d. Other appropriate limitations and remedial or compensatory actions.

, and assuming no concurrent loss of offsite power or loss of onsite diesel generator(s),

A loss of safety function exists when, assuming no concurrent single failure~~s~~, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to the system(s) supported by the inoperable support system is also inoperable; or

b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to the support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

INSERT 2

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BASES

LCO 3.0.6 (continued)

INSERT 1

retained. If this evaluation determines that a loss of safety function exists, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

LCO 3.0.7

Special tests and operations are required at various times over the unit's life to demonstrate performance characteristics, to perform maintenance activities, and to perform special evaluations. Because TS normally preclude these tests and operations, special test exceptions (STEs) allow specified requirements to be changed or suspended under controlled conditions. STEs are included in applicable sections of the Specifications. Unless otherwise specified, all other TS requirements remain unchanged and in effect as applicable. This will ensure that all appropriate requirements of the MODE or other specified condition not directly associated with or required to be changed or suspended to perform the special test or operation will remain in effect.

The Applicability of an STE LCO represents a condition not necessarily in compliance with the normal requirements of the TS. Compliance with STE LCOs is optional.

A special test may be performed under either the provisions of the appropriate STE LCO or the other applicable TS requirements. If it is desired to perform the special test under the provisions of the STE LCO, the requirements of the STE LCO shall be followed. This includes the SRs specified in the STE LCO.

Some of the STE LCOs require that one or more of the LCOs for normal operation be met (i.e., meeting the STE LCO requires meeting the specified normal LCOs). The Applicability, ACTIONS, and SRs of the specified normal LCOs, however, are not required to be met in order to meet the STE LCO when it is in effect. This means that, upon failure to meet a specified normal LCO, the associated ACTIONS of the STE LCO apply, in lieu of the ACTIONS of the normal LCO. Exceptions to the above do exist. There are instances when the Applicability of the specified normal LCO must be met, where its ACTIONS must be taken, where certain of its Surveillances must be performed, or where all of

(continued)

5.5 Programs and Manuals

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5.5.14 Technical Specifications (TS) Bases Control Program (continued)

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR.

d. Proposed changes that meet the criteria of Specification 5.5.14b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

5.5.15 Safety Functions Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate limitations and remedial or compensatory actions may be identified to be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following:

a. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;

b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

d. Other appropriate limitations and remedial or compensatory actions.

, and assuming no concurrent loss of offsite power or loss of onsite diesel generator(s),

A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to system(s) supported by the inoperable support system is also inoperable; or

(continued)

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5.5.15 Safety Functions Determination Program (continued)

b. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

INSERT 2

BASES INSERT 1 TSTF-273, Rev 2

LCO 3.0.6 (continued) the LCO in which the loss of safety function exists are required to be entered.

LCO 3.0.7 There are certain special tests and operations required to be performed at various times over the life of the unit. These special tests and operations are necessary to demonstrate select unit performance characteristics, to perform special maintenance activities, and to perform special evolutions. Special Operations LCOs in Section 3.10 allow specified TS requirements to be changed to permit performances of these special tests and operations, which otherwise could not be performed if required to comply with the requirements of these TS. Unless otherwise specified, all the other TS requirements remain unchanged. This will ensure all appropriate requirements of the MODE or other specified condition not directly associated with or required to be changed to perform the special test or operation will remain in effect.

The Applicability of a Special Operations LCO represents a condition not necessarily in compliance with the normal requirements of the TS. Compliance with Special Operations LCOs is optional. A special operation may be performed either under the provisions of the appropriate Special Operations LCO or under the other applicable TS requirements. If it is desired to perform the special operation under the provisions of the Special Operations LCO, the requirements of the Special Operations LCO shall be followed. When a Special Operations LCO requires another LCO to be met, only the requirements of the LCO statement are required to be met regardless of that LCO's Applicability (i.e., should the requirements of this other LCO not be met, the ACTIONS of the Special Operations LCO apply, not the ACTIONS of the other LCO). However, there are instances where the Special Operations LCO ACTIONS may direct the other LCOs' ACTIONS be met. The Surveillances of the other LCO are not required to be met, unless specified in the Special Operations LCO. If conditions exist such that the Applicability of any other LCO is met, all the other LCO's requirements (ACTIONS and SRs) are required to be met concurrent with the requirements of the Special Operations LCO.

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5.5.11 Technical Specifications (TS) Bases Control Program (continued)

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR.

d. Proposed changes that meet the criteria of Specification 5.5.11b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

5.5.12 Safety Function Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate limitations and remedial or compensatory actions may be identified to be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following:

a. Provisions for cross division checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;

b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

d. Other appropriate limitations and remedial or compensatory actions.

, and assuming no concurrent loss of offsite power or loss of onsite diesel generator(s),

A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to system(s) supported by the inoperable support system is also inoperable; or

(continued)

5.5 Programs and Manuals

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5.5.12 Safety Function Determination Program (SFDP) (continued)

b. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

INSERT 2

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BASES

LCO 3.0.6 (continued)

Cross division checks to identify a loss of safety function for those support systems that support safety systems are required. The cross division check verifies that the supported systems of the redundant OPERABLE support system are OPERABLE, thereby ensuring safety function is retained. If this evaluation determines that a loss of safety function exists, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

INSERT 1

LCO 3.0.7

There are certain special tests and operations required to be performed at various times over the life of the unit. These special tests and operations are necessary to demonstrate select unit performance characteristics, to perform special maintenance activities, and to perform special evolutions. Special Operations LCOs in Section 3.10 allow specified TS requirements to be changed to permit performances of these special tests and operations, which otherwise could not be performed if required to comply with the requirements of these TS. Unless otherwise specified, all the other TS requirements remain unchanged. This will ensure all appropriate requirements of the MODE or other specified condition not directly associated with or required to be changed to perform the special test or operation will remain in effect.

The Applicability of a Special Operations LCO represents a condition not necessarily in compliance with the normal requirements of the TS. Compliance with Special Operations LCOs is optional. A special operation may be performed either under the provisions of the appropriate Special Operations LCO or under the other applicable TS requirements. If it is desired to perform the special operation under the provisions of the Special Operations LCO, the requirements of the Special Operations LCO shall be followed. When a Special Operations LCO requires another LCO to be met, only the requirements of the LCO statement are required to be met regardless of that LCO's Applicability (i.e., should the requirements of this other LCO not be met, the ACTIONS of the Special Operations LCO apply, not the ACTIONS of the other LCO). However, there are instances where the Special Operations LCO ACTIONS may direct the other LCOs' ACTIONS be met. The Surveillances of

(continued)

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5.5 Programs and Manuals

5.5.11 Technical Specifications (TS) Bases Control Program (continued)

prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

5.5.12 Safety Function Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate limitations and remedial or compensatory actions may be identified to be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following:

a. Provisions for cross division checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;

b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

d. Other appropriate limitations and remedial or compensatory actions.

, and assuming no concurrent loss of offsite power or loss of onsite diesel generator(s),

A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to system(s) supported by the inoperable support system is also inoperable; or

b. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or

(continued)

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5.5.12 Safety Function Determination Program (SFDP) (continued)

c. A required system redundant to support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

INSERT 2

Technical Specification Task Force
Improved Standard Technical Specifications Change Traveler

Remove "applicable supports" from Inservice Testing Program

NUREGs Affected: ☑ 1430 ☑ 1431 ☑ 1432 ☑ 1433 ☑ 1434

Classification: 3) Improve Specifications Recommended for CLIIP?: (Unassigned)

Correction or Improvement: (Unassigned)

Industry Contact: Steve Wideman, (620) 364-4037, stwidem@wcnoc.com

The reference to the "applicable supports" was deleted from the description of the "Inservice Testing Program" contained in Section 5.5.
The Inservice Testing Program (IST) provides controls for testing Code Class 1, 2 and 3 components. The discussion of the IST Program in Section 5.5 of the STS was revised by the NRC to include the "applicable supports" in February 1992 due to concerns related to the relocation of the Snubber LCO from the ITS NUREGs. However, this is inappropriate; supports are addressed under the Inservice Inspection Program not the IST Program. Thus, the reference to the applicable supports in the IST Program description in Section 5.5 was deleted. Additionally, in the last six years, sixteen plants have implemented ITS with no known issues related to testing of snubbers or supports. ASME has developed OM-5 and other guidance to provide appropriate testing requirements for supports and snubbers.

Revision History

OG Revision 0 Revision Status: Active

Revision Proposed by: WOG

Revision Description:
Original Issue

Owners Group Review Information

Date Originated by OG: 20-Nov-97

Owners Group Comments:
(No Comments)

Owners Group Resolution: Approved Date: 20-Nov-97

TSTF Review Information

TSTF Received Date: 20-Nov-97 Date Distributed for Review: 06-Jan-98

OG Review Completed: ☑ BWOG ☑ WOG ☑ CEOG ☑ BWROG

TSTF Comments:

Revise justification: plants have to apply this ASME program to whatever Code Class 1,2 and 3 components are applicable, the supports are under the ISI not the IST. Applicable to all OGs

TSTF Resolution: Approved Date: 05-Feb-98

NRC Review Information

NRC Received Date: 29-May-98

Date of NRC Letter: 16-Jul-98

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OG Revision 0 Revision Status: Active

Final Resolution: NRC Approves Final Resolution Date: 16-Jul-98

Affected Technical Specifications

5.5.8	Inservice Testing Program	NUREG(s)- 1430 1431 1432 Only
5.5.7	Inservice Testing Program	NUREG(s)- 1433 1434 Only

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5.5 Programs and Manuals (continued)

5.5.8 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components ~~including applicable supports~~. The program shall include the following:

a. Testing frequencies specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as follows:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice testing activities	Required Frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies for performing inservice testing activities;

c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

d. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any TS.

5.5.9 Steam Generator (SG) Tube Surveillance Program

Reviewer's Note: The Licensees current licensing basis steam generator tube surveillance requirements shall be relocated from the LCO and included here. An appropriate administrative controls program format should be used.

(continued)

TSTF-279

5.5 Programs and Manuals (continued)

5.5.8 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components, ~~including applicable supports.~~ The program shall include the following:

a. Testing frequencies specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as follows:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice testing activities	Required Frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies for performing inservice testing activities;

c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

d. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any TS.

5.5.9 Steam Generator (SG) Tube Surveillance Program

[Reviewer's Note: The Licensee's current licensing basis steam generator tube surveillance requirements shall be relocated from the LCO and included here. An appropriate administrative controls program format should be used.]

(continued)

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5.5 Programs and Manuals (continued)

5.5.8 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components including applicable supports. The program shall include the following:

a. Testing frequencies specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as follows:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice testing activities	Required Frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies for performing inservice testing activities;

c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

d. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any TS.

5.5.9 Steam Generator (SG) Tube Surveillance Program

[Reviewer's Note: The Licensees current licensing basis steam generator tube surveillance requirements shall be relocated from the LCO and included here. An appropriate administrative controls program format should be used.]

(continued)

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5.5 Programs and Manuals (continued)

5.5.7 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components ~~including applicable supports~~. The program shall include the following:

a. Testing frequencies specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda are as follows:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice testing activities	Required Frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies for performing inservice testing activities;

c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

d. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any TS.

5.5.8 Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems at the frequencies specified in [Regulatory Guide], and in accordance with [Regulatory Guide 1.52, Revision 2, ASME N510-1989, and AG-1].

(continued)

TSTF-279

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and

j. Limitations on the annual dose or dose commitment to any member of the public due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

5.5.5 Component Cyclic or Transient Limit

This program provides controls to track the FSAR, Section [ ], cyclic and transient occurrences to ensure that components are maintained within the design limits.

5.5.6 Pre-Stressed Concrete Containment Tendon Surveillance Program

This program provides controls for monitoring any tendon degradation in pre-stressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial operations. The Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with [Regulatory Guide 1.35, Revision 3, 1989].

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Tendon Surveillance Program inspection frequencies.

5.5.7 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components ~~including applicable supports~~. The program shall include the following:

(continued)

Technical Specification Task Force
Improved Standard Technical Specifications Change Traveler

Add "Met vs. Perform" to Specification 1.4, Frequency

NUREGs Affected: ☑ 1430 ☑ 1431 ☑ 1432 ☑ 1433 ☑ 1434

Classification: 2) Consistency/Standardization Recommended for CLIIP?: (Unassigned)

Correction or Improvement: (Unassigned)

Industry Contact: Steve Wideman, (620) 364-4037, stwidem@wcnoc.com

Insert into Specification 1.4 a discussion paragraph and new example to facilitate the use and application of SR Notes that utilize "met" and "perform". Revise SRs as necessary to appropriately use "met" and "perform" exceptions. NUREG-1433 and 1434 contain a discussion in Specification 1.4 regarding the use of "met" and "performed" in SR Notes. Similarly, the Writer's Guide provides a distinction between these phrases. NUREG-1430, -1431, and -1432 do not contain this detail; however, various locations throughout these NUREGs provide Notes with "met" and "performed" distinctions. Inserting this material will provide for better use, application, and understanding of these Notes. Furthermore, this change will establish consistency between the NUREGs. With this clarification, several exceptions that are unclear or have incorrect usage of "met" and "perform" are also corrected.

Examples of Surveillance Notes are added. The examples parallel the existing example 1.4-3 of Notes that allow for the SR to "Not required to be performed . . .". The examples will alleviate misunderstanding and provide explicit direction for these types of SR Notes.

Revision History

OG Revision 0 Revision Status: Closed

Revision Proposed by: WOG

Revision Description:
Original Issue

Owners Group Review Information

Date Originated by OG: 19-Aug-97

Owners Group Comments:
(No Comments)

Owners Group Resolution: Approved Date: 19-Aug-97

TSTF Review Information

TSTF Received Date: 20-Nov-97 Date Distributed for Review: 06-Jan-98

OG Review Completed: ☑ BWOG ☑ WOG ☐ CEOG ☐ BWROG

TSTF Comments:

2/5/98 - Applicable to all PWRs. Excel to markup the CEOG and BWOG NUREGs and provide to the CEOG and BWOG chairmen prior to sending to NRC.

7/98 - Comments received from CEOG and BWOG chairmen. Traveler to be revised.

Redistributed to TSTF on 5/27/98
- 7/10/98 - Change "performed" to "met" and change/add appropriate Bases change on BWOG p 3.4-26
- Withdraw WOG SR 3.3.1.4 on Pg 3.3-11 from TSTF-284.

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OG Revision 0 Revision Status: Closed

- Change SR 3.3.8.3 and SR 3.3.8.4 to "Only"

TSTF Resolution: Approved Date: 10-Jul-98

NRC Review Information

NRC Received Date: 25-Sep-98

NRC Comments:

11/12/98 - The TSTF wants to pursue the change and come back to the NRC with a broader scope change of Met versus Performed. TSTF to inform the NRC of status of revisiting of Met vs.Performed in January, 1999.

Final Resolution: Superceded by Revision Final Resolution Date: 03-Feb-99

TSTF Revision 1 Revision Status: Closed

Revision Proposed by: WOG

Revision Description:
Deleted changes to NUREG-1431 (Westinghouse), SR 3.7.5.3 and 3.7.5.4.

TSTF Review Information

TSTF Received Date: 03-Feb-99 Date Distributed for Review:

OG Review Completed: ☐ BWOG ☐ WOG ☐ CEOG ☐ BWROG

TSTF Comments:

On hold.

05/23/99 This change is superceded to incorporate TSTF-270 R1.

TSTF Resolution: Superceeded Date:

TSTF Revision 2 Revision Status: Closed

Revision Proposed by: TSTF

Revision Description:
This change incorporates proposed TSTF-270 R1 and TSTF-288, consistent with proper use of "met" and "perform" notes. Also, examples of Surveillance Notes that allow for the SR to "Only required to be performed . . .", "Only required to be met . . .", and "not required to be met . . ." are added.

TSTF Review Information

TSTF Received Date: 28-May-99 Date Distributed for Review: 15-Jun-99

OG Review Completed: ☑ BWOG ☑ WOG ☑ CEOG ☑ BWROG

TSTF Comments:

(No Comments)

TSTF Resolution: Approved Date: 15-Jun-99

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TSTF Revision 2 Revision Status: Closed

NRC Review Information

NRC Received Date: 23-Jun-99

NRC Comments: Date of NRC Letter: 16-Dec-99

12/14/99 - Denny B to revise and send to TSTF for review.
1/10/00 - TSTF agrees to make changes.

Final Resolution: Superceded by Revision

TSTF Revision 3 Revision Status: Active

Revision Proposed by: NRC

Revision Description:
The "Description" section of Section 1.4 is revised to incorporate NRC comments and to make the section consistent between all 5 ISTS NUREGs. Notes are added to SR 3.9.3.2 (WOG and CEOG) and 3.9.4.2 (WOG) to properly address met vs. performed issues. The changes to these two SRs were orginally proposed in TSTF-92. The NRC requested that the change be incorporated into TSTF-284.

TSTF Review Information

TSTF Received Date: 10-Jan-00 Date Distributed for Review: 10-Jan-00

OG Review Completed: ☑ BWOG ☑ WOG ☑ CEOG ☑ BWROG

TSTF Comments:

(No Comments)

TSTF Resolution: Approved Date: 10-Jan-00

NRC Review Information

NRC Received Date: 11-Jan-00

Date of NRC Letter: 16-Feb-00

Final Resolution: NRC Approves Final Resolution Date: 16-Feb-00

Affected Technical Specifications

1.4	Frequency	
SR 3.9.3.2	Containment Penetrations	NUREG(s)- 1430 1432 Only
SR 3.9.4.2 Bases	Containment Penetrations	NUREG(s)- 1430 1432 Only
SR 3.1.3.2	MTC	NUREG(s)- 1430 Only
SR 3.1.3.2 Bases	MTC	NUREG(s)- 1430 Only

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SR 3.4.12.6	LTOP System	NUREG(s)- 1430 Only
SR 3.4.12.6 Bases	LTOP System	NUREG(s)- 1430 Only
SR 3.7.5.3	EFW System	NUREG(s)- 1430 Only
SR 3.7.5.3 Bases	EFW System	NUREG(s)- 1430 Only
SR 3.7.5.4	EFW System	NUREG(s)- 1430 Only
SR 3.7.5.4 Bases	EFW System	NUREG(s)- 1430 Only
SR 3.1.11.1	SDM Test Exceptions	NUREG(s)- 1431 Only
SR 3.1.11.2	SDM Test Exceptions	NUREG(s)- 1431 Only
SR 3.1.11.2 Bases	SDM Test Exceptions	NUREG(s)- 1431 Only
Action 3.4.11 Bases	Pressurizer PORVs	NUREG(s)- 1431 Only
Ref. 3.4.11 Bases	Pressurizer PORVs	NUREG(s)- 1431 Only
SR 3.4.11.1	Pressurizer PORVs	NUREG(s)- 1431 Only
SR 3.4.11.1 Bases	Pressurizer PORVs	NUREG(s)- 1431 Only
SR 3.4.11.2	Pressurizer PORVs	NUREG(s)- 1431 Only
SR 3.4.11.2 Bases	Pressurizer PORVs	NUREG(s)- 1431 Only
SR 3.9.4.2	Containment Penetrations	NUREG(s)- 1431 Only
SR 3.9.4.2 Bases	Containment Penetrations	NUREG(s)- 1431 Only
SR 3.1.4.1	MTC (Analog)	NUREG(s)- 1432 Only
SR 3.1.4.1	MTC (Digital)	NUREG(s)- 1432 Only
SR 3.1.4.1 Bases	MTC (Analog)	NUREG(s)- 1432 Only
SR 3.1.4.1 Bases	MTC (Digital)	NUREG(s)- 1432 Only
SR 3.1.4.2	MTC (Analog)	NUREG(s)- 1432 Only
SR 3.1.4.2	MTC (Digital)	NUREG(s)- 1432 Only
SR 3.1.4.2 Bases	MTC (Analog)	NUREG(s)- 1432 Only
SR 3.1.4.2 Bases	MTC (Digital)	NUREG(s)- 1432 Only
SR 3.1.7.1	Regulating CEA Insertion Limits (Analog)	NUREG(s)- 1432 Only
SR 3.1.7.1	Regulating CEA Insertion Limits (Digital)	NUREG(s)- 1432 Only

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SR 3.1.7.1 Bases	Regulating CEA Insertion Limits (Analog)	NUREG(s)- 1432 Only
SR 3.1.7.1 Bases	Regulating CEA Insertion Limits (Digital)	NUREG(s)- 1432 Only
SR 3.2.1.1	LHR (Analog)	NUREG(s)- 1432 Only
SR 3.2.1.1	LHR (Digital)	NUREG(s)- 1432 Only
SR 3.2.1.1 Bases	LHR (Analog)	NUREG(s)- 1432 Only
SR 3.2.1.1 Bases	LHR (Digital)	NUREG(s)- 1432 Only
SR 3.2.1.2	LHR (Analog)	NUREG(s)- 1432 Only
SR 3.2.1.2 Bases	LHR (Analog)	NUREG(s)- 1432 Only
SR 3.2.1.3	LHR (Analog)	NUREG(s)- 1432 Only
SR 3.2.3.1	Tq (Digital)	NUREG(s)- 1432 Only
SR 3.2.4.1	DNBR (Digital)	NUREG(s)- 1432 Only
SR 3.2.4.1 Bases	DNBR (Digital)	NUREG(s)- 1432 Only
SR 3.2.4.2	DNBR (Digital)	NUREG(s)- 1432 Only
SR 3.3.8.3	CPIS (Digital)	NUREG(s)- 1432 Only
SR 3.3.8.3 Bases	CPIS (Digital)	NUREG(s)- 1432 Only
SR 3.3.8.4	CPIS (Digital)	NUREG(s)- 1432 Only
SR 3.3.8.4 Bases	CPIS (Digital)	NUREG(s)- 1432 Only
Ref. 3.4.11 Bases	Pressurizer PORVs	NUREG(s)- 1432 Only
SR 3.4.11.1	Pressurizer PORVs	NUREG(s)- 1432 Only
SR 3.4.11.1 Bases	Pressurizer PORVs	NUREG(s)- 1432 Only
SR 3.4.11.2	Pressurizer PORVs	NUREG(s)- 1432 Only
SR 3.4.11.2 Bases	Pressurizer PORVs	NUREG(s)- 1432 Only
SR 3.7.5.3	AFW System	NUREG(s)- 1432 Only
SR 3.7.5.3	AFW System	NUREG(s)- 1432 Only
SR 3.7.5.4	AFW System	NUREG(s)- 1432 Only
SR 3.7.5.4 Bases	AFW System	NUREG(s)- 1432 Only

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INSERT 1 (PWR)

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

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

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

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

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

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

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

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

INSERT 1A (BWR/4 and BWR/6)

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

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

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

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

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

INSERT 2 (FYI: The source of this insert is NUREG 1433)

EXAMPLE 1.4-4

SURVEILLANCE REQUIREMENTS

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

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

EXAMPLES (continued)

EXAMPLE 1.4-[5]

SURVEILLANCE REQUIREMENTS

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

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

As the Note modifies the required performance of the Surveillance, the Note is construed to be part of the "specified Frequency." Should the 7 day interval be exceeded while operation is not in MODE 1, this Note allows entry into and operation in MODES 2 and 3 to perform the Surveillance. The Surveillance is still considered to be performed within the "specified Frequency" if completed prior to entering MODE 1. Therefore, if the Surveillance were not performed within the 7 day (plus the extension allowed by SR 3.0.2) interval, but operation was not in MODE 1, it would not constitute a failure of the SR or failure to meet the LCO. Also, no violation of SR 3.0.4 occurs when changing MODES, even with the 7 day Frequency not met, provided operation does not result in entry into MODE 1.

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

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INSERT 4

EXAMPLE 1.4-[6]

SURVEILLANCE REQUIREMENTS

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

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

WOG AND CE ONLY

Insert A

... Opening the block valve in this condition increases the risk of an unisolable leak from the RCS since the PORV is already inoperable.

¶ Note 2 modifies this SR to allow entry into and operation in MODE 3 prior to performing the SR. This allows the test to be performed in MODE 3 under operating temperature and pressure conditions, prior to entering MODE 1 or 2. [In accordance with Reference 4, administrative controls require this test be performed in MODE 3 or 4 to adequately simulate operating temperature and pressure effects on PORV operation.]

Insert B

The Note modifies this SR to allow entry into and operation in MODE 3 prior to performing the SR. This allows the test to be performed in MODE 3 under operating temperature and pressure conditions, prior to entering MODE 1 or 2. [In accordance with Reference 4, administrative controls require this test be performed in MODE 3 or 4 to adequately simulate operating temperature and pressure effects on PORV operation.]

Insert C

[4. Generic Letter 90-06, "Resolution of Generic Issue 70, 'Power-Operated Relief Valve and Block Valve Reliability,' and Generic Issue 94, 'Additional Low-Temperature Overpressure for Light-Water Reactors,' Pursuant to 10 CFR 50.54(f)," June 25, 1990.]

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1.0 USE AND APPLICATION

1.4 Frequency

PURPOSE

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

DESCRIPTION

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

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

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

INSERT 1

EXAMPLES

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

(continued)

EXAMPLES (continued)

EXAMPLE 1.4-3

SURVEILLANCE REQUIREMENTS

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

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

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

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

INSERT 2
INSERT 3
INSERT 4

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SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.3.2	----------NOTES---------- ~~1. This SR is not required to be performed prior to entry into MODE 1 or 2.~~ ~~2.~~ If the MTC is more negative than the COLR limit when extrapolated to the end of cycle, SR 3.1.3.2 may be repeated. Shutdown must occur prior to exceeding the minimum allowable boron concentration at which MTC is projected to exceed the lower limit. ---------- Verify extrapolated MTC is within the lower limit specified in the COLR.	Each fuel cycle within 7 EFPDs after reaching an equilibrium boron concentration equivalent to 300 ppm

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SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.12.4	Verify pressurizer level is ≤ [22] inches.	30 minutes during RCS heatup and cooldown AND 12 hours
SR 3.4.12.5	Verify PORV block valve is open.	12 hours
SR 3.4.12.6	----------NOTE---------- Only required to be met when complying with LCO 3.4.12.b. ---------- Verify RCS vent ≥ [0.75] square inch is open.	12 hours for unlocked open vent valve(s) AND 31 days for locked open vent valve(s)
SR 3.4.12.7	Perform CHANNEL FUNCTIONAL TEST for PORV.	Within [12] hours after decreasing RCS temperature to ≤ [283]°F AND 31 days thereafter

(continued)

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.5.1	Verify each EFW manual, power operated, and automatic valve in each water flow path and in both steam supply flow paths to the steam turbine driven pumps, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.7.5.2	----------NOTE---------- Not required to be performed for the turbine driven EFW pumps, until [24] hours after reaching [800] psig in the steam generators. ---------- Verify the developed head of each EFW pump at the flow test point is greater than or equal to the required developed head.	[31] days on a STAGGERED TEST BASIS
SR 3.7.5.3	----------NOTES---------- 1. Not required to be performed until [24] hours after reaching [800] psig in the steam generators. 2. Not ~~applicable~~ required to be met in MODE 4. ---------- Verify each EFW 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

(continued)

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SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.5.4	----------NOTES---------- 1. Not required to be performed until [24] hours after reaching [800] psig in the steam generators. 2. Not applicable in MODE 4. required to be met ---------- Verify each EFW pump starts automatically on an actual or simulated actuation signal.	[18] months
SR 3.7.5.5	Verify proper alignment of the required EFW flow paths by verifying [valve alignment/flow] from the condensate storage tank to each steam generator.	Prior to entering MODE 2 whenever plant has been in MODE 5 or 6 for > 30 days
[SR 3.7.5.6	Perform a CHANNEL FUNCTIONAL TEST for the EFW pump suction pressure interlocks.	31 days]
[SR 3.7.5.7	Perform a CHANNEL CALIBRATION for the EFW pump suction pressure interlocks.	[18] months]

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.3.1	Verify each required containment penetration is in the required status.	7 days
SR 3.9.3.2	Verify each required containment purge and exhaust valve actuates to the isolation position on an actual or simulated actuation signal.	[18] months

NOTE

Not required to be met for containment purge and exhaust valves in penetrations closed to comply with LCO C.1.

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.3.2 (continued)

check on the most negative (least positive) MTC value. The measurement is performed at any THERMAL POWER equivalent to an RCS boron concentration of 300 ppm (for steady state operation at RTP with all CONTROL RODS fully withdrawn) so that the projected EOC MTC may be evaluated before the reactor actually reaches the EOC condition. MTC values are extrapolated and compensated to permit direct comparison to the specified MTC limits.

The SR is modified by ~~two~~ Note~~s~~. ~~Note 1 indicates performance of SR 3.1.3.2 is not required prior to entering MODE 1 or 2. Although this Surveillance is applicable in MODES 1 and 2, the reactor must be critical before the Surveillance can be completed. Therefore, entry into the applicable MODE, prior to accomplishing the Surveillance, is necessary.~~

The Note ~~Note 2~~ indicates that SR 3.1.3.2 may be repeated, and shutdown must occur, prior to exceeding the minimum allowable boron concentration at which MTC is projected to exceed the lower limit. The minimum allowable boron concentration is obtained from the EOC MTC versus boron concentration slope with appropriate conservatisms. Thus, the projected EOC MTC is evaluated before the lower limit is actually reached.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 11.
2. FSAR, Chapter [14].
3. FSAR, Section [ ].
4. FSAR, Section [ ].

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.12.4 (continued)

variations. This Frequency may be discontinued when the ends of these conditions are satisfied, as defined in plant procedures. Thereafter, the Surveillance is required at 12 hour intervals.

These Frequencies are shown by operating practice sufficient to regularly assess indications of potential degradation and verify operation within the safety analysis.

SR 3.4.12.5

Verification that the PORV block valve is open ensures a flow path to the PORV. This is required at 12 hour intervals.

The interval has been shown by operating practice sufficient to regularly assess conditions for potential degradation and verify operation is within the safety analysis.

SR 3.4.12.6

When stipulated by LCO 3.4.12.b, the RCS vent of at least [0.75] square inches must be verified open for relief protection. For a vent valve not locked open, the Frequency is every 12 hours. For a valve locked open, the required Frequency is every 31 days.

Again, the Frequency intervals consider operating practice to determine adequacy to regularly assess conditions for potential degradation and verify operation within the safety analysis.

A Note modifies the SR by requiring the Surveillance to only be met when complying with LCO 3.4.12.b.

SR 3.4.12.7

A CHANNEL FUNCTIONAL TEST is required within [12] hours after decreasing RCS temperature to ≤ [283]°F and every 31 days thereafter to ensure the setpoint is proper for

(continued)

BASES

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SURVEILLANCE REQUIREMENTS

SR 3.7.5.3 (continued)

unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The [18] month Frequency is also acceptable based on operating experience and design reliability of the equipment. This SR is modified by a Note that states the SR is not required in MODE 4. In MODE 4, the required AFW train is already aligned and operating. This SR is modified by [a] [two] Note[s]. [Note 1 indicates that the SR be deferred until suitable test conditions are established. This deferral is required because there is insufficient steam pressure to perform the test.] [The] Note [2] states that the SR is not required in MODE 4. [In MODE 4, the required pump is already 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.]

to be met

SR 3.7.5.4

This SR verifies that the turbine driven EFW pumps start in the event of any accident or transient that generates an SFRCS signal by demonstrating that each turbine driven EFW pump starts automatically on an actual or simulated actuation signal. These pumps are not required in MODE 4. 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. This SR is modified by [a] [two] Note[s]. [Note 1 indicates that the SR be deferred until suitable test conditions are established. This deferral is required because there is insufficient steam pressure to perform the test.] [The] Note [2] states that the SR is not required in MODE 4. [In MODE 4, the required pump is already 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.]

to be met

[Reviewer's Note: Some plants may not routinely use the AFW for heat removal in MODE 4. The second justification is provided for plants that use a startup feedwater pump rather than AFW for startup and shutdown.]

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.9.3.1 (continued)

radioactivity within the containment will not result in a release of fission product radioactivity to the environment.

SR 3.9.3.2

This Surveillance demonstrates that each containment purge and exhaust valve actuates to its isolation position on manual initiation or on an actual or simulated high radiation signal. The 18 month Frequency maintains consistency with other similar ESFAS instrumentation and valve testing requirements. In LCO 3.3.15, "RB Purge Isolation High Radiation," the isolation instrumentation requires a CHANNEL CHECK every 12 hours and a CHANNEL FUNCTIONAL TEST every 92 days to ensure the channel OPERABILITY during refueling operations. Every 18 months a CHANNEL CALIBRATION is performed. The system actuation response time is demonstrated every 18 months, during refueling, on a STAGGERED TEST BASIS. SR 3.6.3.5 demonstrates that the isolation time of each valve is in accordance with the Inservice Testing Program requirements. These Surveillances performed during MODE 6 will ensure that the valves are capable of closing after a postulated fuel handling accident to limit a release of fission product radioactivity from the containment.

REFERENCES

1. GPU Nuclear Safety Evaluation SE-0002000-001, Rev. 0, May 20, 1988.

2. FSAR, Section [ ].

3. NUREG-0800, Section 15.7.4, Rev. 1, July 1981.

The SR is modified by a Note stating that this Surveillance is not required to be met for valves in isolate penetrations. The LCO provides the option to close penetrations in lieu of requiring automatic actuation capability.

1.0 USE AND APPLICATION

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1.4 Frequency

PURPOSE

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

DESCRIPTION

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

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

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

INSERT 1

EXAMPLES

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

(continued)

EXAMPLES (continued)

EXAMPLE 1.4-3

SURVEILLANCE REQUIREMENTS

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

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

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

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

INSERT 2
INSERT 3
INSERT 4

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.11.1	to be met ----------NOTE---------- Only required for control rods not fully inserted. ---------- Determine the position of each control rod.	2 hours
SR 3.1.11.2	to be met ----------NOTE---------- Only required for control rods not fully inserted. ---------- Trip each control rod from ≥ the 50% withdrawn position, and verify full control rod insertion.	Within 24 hours prior to reducing SDM outside limits

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ACTIONS

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.11.1	----------NOTE---------- (S) Not required to be ~~met~~ performed with block valve closed in accordance with the Required Action (S) of ~~Condition B or E~~ this LCO. ----------	2. Only required to be performed in MODES 1 and 2.
	Perform a complete cycle of each block valve.	92 days
SR 3.4.11.2	Perform a complete cycle of each PORV.	[18] months

1.

---------- NOTE ----------
Only required to be performed in MODES 1 and 2.

(continued)

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SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
[SR 3.4.12.4	Verify RHR suction valve is open for each required RHR suction relief valve.	12 hours]
SR 3.4.12.5	met ----------NOTE---------- Only required to be ~~performed~~ when complying with LCO 3.4.12.b. ---------- Verify RCS vent ≥ [2.07] square inches open.	12 hours for unlocked open vent valve(s) AND 31 days for locked open vent valve(s)
SR 3.4.12.6	Verify PORV block valve is open for each required PORV.	72 hours
[SR 3.4.12.7	Verify associated RHR suction isolation valve is locked open with operator power removed for each required RHR suction relief valve.	31 days]
SR 3.4.12.8	performed ----------NOTE---------- Not required to be met until 12 hours after decreasing RCS cold leg temperature to ≤ [275]°F. ---------- Perform a COT on each required PORV, excluding actuation.	31 days

(continued)

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.4.1	Verify each required containment penetration is in the required status.	7 days
SR 3.9.4.2	Verify each required containment purge and exhaust valve actuates to the isolation position on an actual or simulated actuation signal.	[18] months

NOTE
Not required to be met for containment purge and exhaust valve(s) in penetrations closed to comply with LCO C.1.

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BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.1.11.1

In order to establish an acceptable SDM during the measurement of control rod worths, it is necessary to know the position of each control rod. A test Frequency of 2 hours is reasonable, based on normal control rod motion during control rod worth measurements.

SR 3.1.11.1 has been modified by a Note establishing that the position of only those control rods not fully inserted must be determined. It is assumed that the position and worth of fully inserted control rods is known.

SR 3.1.11.2

One of the assumptions made in granting an STE for SDM, is that all control rods not fully inserted will fully insert when tripped. This Surveillance is performed to verify that fact.

The Frequency of 24 hours prior to reducing the plant SDM below the normal requirements is acceptable, based on the assumption that the control rods will remain OPERABLE and trippable for 24 hours and during the performance of the test.

to be met

SR 3.1.11.2 has been modified by a Note establishing that this Surveillance is only required for control rods not fully inserted. During the performance of control rod worth measurements, certain control rods remain fully inserted. Since these rods are not relied on to trip, there is no need to demonstrate that they will fully insert when tripped.

REFERENCES

1. 10 CFR 50, Appendix B, Section XI.
2. 10 CFR 50.59.
3. Regulatory Guide 1.68, Revision 2, August 1978.
4. ANSI/ANS-19.6.1-1985, December 13, 1985.
5. FSAR, Chapter [14].

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BASES (continued)

APPLICABILITY

In MODES 1, 2, and 3, the PORV and its block valve are required to be OPERABLE to limit the potential for a small break LOCA through the flow path. The most likely cause for a PORV small break LOCA is a result of a pressure increase transient that causes the PORV to open. Imbalances in the energy output of the core and heat removal by the secondary system can cause the RCS pressure to increase to the PORV opening setpoint. The most rapid increases will occur at the higher operating power and pressure conditions of MODES 1 and 2. The PORVs are also required to be OPERABLE in MODES 1, 2, and 3 to minimize challenges to the pressurizer safety valves.

Pressure increases are less prominent in MODE 3 because the core input energy is reduced, but the RCS pressure is high. Therefore, the LCO is applicable in MODES 1, 2, and 3. The LCO is not applicable in MODE 4 when both pressure and core energy are decreased and the pressure surges become much less significant. The PORV setpoint is reduced for LTOP in MODES 4, 5, and 6 with the reactor vessel head in place. LCO 3.4.12 addresses the PORV requirements in these MODES.

ACTIONS

Note 1 has been added to clarify that all pressurizer PORVs are treated as separate entities, each with separate Completion Times (i.e., the Completion Time is on a component basis). The exception for LCO 3.0.4, Note 2, permits entry into MODES 1, 2, and 3 to perform cycling of the PORVs or block valves to verify their OPERABLE status, in the event that testing ~~is~~ was not satisfactorily performed in lower MODES.

A.1

With the PORVs inoperable and capable of being manually cycled, either the PORVs must be restored or the flow path isolated within 1 hour. The block valves should be closed but power must be maintained to the associated block valves, since removal of power would render the block valve inoperable. Although a PORV may be designated inoperable, it may be able to be manually opened and closed, and therefore, able to perform its function. PORV inoperability may be due to seat leakage, instrumentation problems, automatic control problems, or other causes that do not prevent manual use and do not create a possibility for a

(continued)

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BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.11.1

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, Section XI (Ref. 3). 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).

This SR is modified by two Notes

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 Actions of this LCO.

Performed

Insert A

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 B

SR 3.4.11.3

Operating the solenoid air control valves and check valves on the air accumulators ensures the PORV control system actuates properly when called upon. The Frequency of [18] months is based on a typical refueling cycle and the Frequency of the other Surveillances used to demonstrate PORV OPERABILITY.

SR 3.4.11.4

This Surveillance is not required for plants with permanent 1E power supplies to the valves.

(continued)

BASES

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SURVEILLANCE REQUIREMENTS

SR 3.4.11.4 (continued)

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

REFERENCES

1. Regulatory Guide 1.32, February 1977.

2. FSAR, Section [15.2].

3. ASME, Boiler and Pressure Vessel Code, Section XI.

INSERT C

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.12.4 (continued)

The RHR suction valve is 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 valve remains open.

The ASME Code, Section XI (Ref. 8), test per Inservice Testing Program verifies OPERABILITY by proving proper relief valve mechanical motion and by measuring and, if required, adjusting the lift setpoint.

SR 3.4.12.5

The RCS vent of ≥ [2.07] square inches is proven OPERABLE by verifying its open condition either:

a. Once every 12 hours for a valve that cannot be locked.

b. Once every 31 days for a valve that is locked, sealed, or secured in position. A removed pressurizer safety valve fits this category.

The passive vent arrangement must only be open to be OPERABLE. This Surveillance is required to be performed if the vent is being used to satisfy the pressure relief requirements of the LCO 3.4.12b.

SR 3.4.12.6

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 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.

(continued)

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.12.6 (continued)

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 3.4.12.7

Each required RHR suction relief valve shall be demonstrated OPERABLE by verifying its RHR suction valve and RHR suction isolation valve are open and by testing it in accordance with the Inservice Testing Program. (Refer to SR 3.4.12.4 for the RHR suction valve Surveillance and for a description of the requirements of the Inservice Testing Program.) This Surveillance is only performed if the RHR suction relief valve is being used to satisfy this LCO.

Every 31 days the RHR suction isolation valve is verified locked open, with power to the valve operator removed, to ensure that accidental closure will not occur. The "locked open" valve must be locally verified in its open position with the manual actuator 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.

SR 3.4.12.8

Performance of a COT is required within 12 hours after decreasing RCS temperature to ≤ [275]°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 PTLR allowed maximum limits in the PTLR. PORV actuation could depressurize the RCS and is not required.

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

Performed

A Note has been added indicating that this SR is required to be ~~met~~ 12 hours after decreasing RCS cold leg temperature to ≤ [275]°F. The COT cannot be performed until in the LTOP MODES when the PORV lift setpoint can be reduced to the LTOP

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.4.1 (continued)

demonstrate that each valve operator has motive power, which will ensure that each valve is capable of being closed by an OPERABLE automatic containment purge and exhaust isolation signal.

The Surveillance is performed every 7 days during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment. The Surveillance interval is selected to be commensurate with the normal duration of time to complete fuel handling operations. A surveillance before the start of refueling operations will provide two or three surveillance verifications during the applicable period for this LCO. As such, this Surveillance ensures that a postulated fuel handling accident that releases fission product radioactivity within the containment will not result in a release of fission product radioactivity to the environment.

SR 3.9.4.2

This Surveillance demonstrates that each containment purge and exhaust valve actuates to its isolation position on manual initiation or on an actual or simulated high radiation signal. The 18 month Frequency maintains consistency with other similar ESFAS instrumentation and valve testing requirements. In LCO 3.3.6, the Containment Purge and Exhaust Isolation instrumentation requires a CHANNEL CHECK every 12 hours and a COT every 92 days to ensure the channel OPERABILITY during refueling operations. Every 18 months a CHANNEL CALIBRATION is performed. The system actuation response time is demonstrated every 18 months, during refueling, on a STAGGERED TEST BASIS. SR 3.6.3.5 demonstrates that the isolation time of each valve is in accordance with the Inservice Testing Program requirements. These Surveillances performed during MODE 6 will ensure that the valves are capable of closing after a postulated fuel handling accident to limit a release of fission product radioactivity from the containment.

The SR is modified by a Note stating that this Surveillance is not required to be met for valves in isolated penetrations. The LCO provides the option to close penetrations in lieu of requiring automatic actuation capability.

(continued)

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1.0 USE AND APPLICATION

1.4 Frequency

PURPOSE

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

DESCRIPTION

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

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

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

INSERT 1

EXAMPLES

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

EXAMPLES (continued)

EXAMPLE 1.4-3

SURVEILLANCE REQUIREMENTS

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

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

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

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

INSERT 2
INSERT 3
INSERT 4

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3.1 REACTIVITY CONTROL SYSTEMS

3.1.4 Moderator Temperature Coefficient (MTC) (Analog)

LCO 3.1.4 The MTC shall be maintained within the limits specified in the COLR. The maximum positive limit shall be that specified in Figure 3.1.4-1.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. MTC not within limits.	A.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.4.1	~~----------NOTE----------~~ ~~This Surveillance is not required to be performed prior to entry into MODE 2.~~ ~~----------~~ Verify MTC is within the upper limits specified in the COLR.	Prior to entering MODE 1 after each fuel loading

(continued)

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SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.4.2	----------NOTE---------- ~~1. This Surveillance is not required to be performed prior to entry into MODE 1 or 2.~~ ~~2.~~ If the MTC is more negative than the COLR limit when extrapolated to the end of cycle, SR 3.1.4.2 may be repeated. Shutdown must occur prior to exceeding the minimum allowable boron concentration at which MTC is projected to exceed the lower limit. ---------- Verify MTC is within the lower limit specified in the COLR.	Each fuel cycle within 7 effective full power days (EFPD) of reaching 40 EFPD core burnup <u>AND</u> Each fuel cycle within 7 EFPD of reaching $\frac{2}{3}$ of expected core burnup

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3.1 REACTIVITY CONTROL SYSTEMS

3.1.4 Moderator Temperature Coefficient (MTC) (Digital)

LCO 3.1.4 The MTC shall be maintained within the limits specified in the COLR, and a maximum positive limit as specified below:

a. [0.5 E-4 Δk/k/°F] when THERMAL POWER is ≤ 70% RTP; and

b. [0.0 Δk/k/°F] when THERMAL POWER is > 70% RTP.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. MTC not within limits.	A.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.4.1	~~NOTE~~ ~~This Surveillance is not required to be performed prior to entry into MODE 2.~~ Verify MTC within the upper limit specified in the COLR.	Prior to entering MODE 1 after each fuel loading

(continued)

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SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.4.2	----------NOTES---------- ~~1. This Surveillance is not required to be performed prior to entry into MODE 1 or 2.~~ ~~2.~~ If the MTC is more negative than the COLR limit when extrapolated to the end of cycle, SR 3.1.4.2 may be repeated. Shutdown must occur prior to exceeding the minimum allowable boron concentration at which MTC is projected to exceed the lower limit. ---------- Verify MTC is within the lower limit specified in the COLR.	Each fuel cycle within 7 effective full power days (EFPD) of reaching 40 EFPD core burnup <u>AND</u> Each fuel cycle within 7 EFPD of reaching $\frac{2}{3}$ of expected core burnup

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ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. (continued)		Once per 4 hours thereafter
E. Required Action and associated Completion Time not met.	E.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.7.1	----------NOTE---------- ~~This Surveillance is~~ not required to be performed ~~prior to~~ until 12 hours after entry into MODE 2. ---------- Verify each regulating CEA group position is within its insertion limits.	12 hours
SR 3.1.7.2	Verify the accumulated times during which the regulating CEA groups are inserted beyond the steady state insertion limits but within the transient insertion limits.	24 hours
SR 3.1.7.3	Verify PDIL alarm circuit is OPERABLE.	31 days

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.7.1	----------NOTE---------- ~~This Surveillance is~~ not required to be performed ~~prior to~~ entry into MODE 2. ---------- until 12 hours after Verify each regulating CEA group position is within its insertion limits.	12 hours
SR 3.1.7.2	Verify the accumulated times during which the regulating CEA groups are inserted beyond the steady state insertion limits but within the transient insertion limits.	24 hours
SR 3.1.7.3	Verify PDIL alarm circuit is OPERABLE.	31 days

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SURVEILLANCE REQUIREMENTS

----------NOTE----------
Either the Excore Detector Monitoring System or the Incore Detector Monitoring System shall be used to determine LHR.

SURVEILLANCE		FREQUENCY
SR 3.2.1.1	required to be met ----------NOTE---------- Only applicable when the Excore Detector Monitoring System is being used to determine LHR. ---------- Verify ASI alarm setpoints are within the limits specified in Figure 3.2.2-2 (ASI Operating Limits) in the COLR.	31 days
SR 3.2.1.2	required to be met ----------NOTES---------- 1. Only applicable when the Incore Detector Monitoring System is being used to determine LHR. 2. Not required to be performed below 20% RTP. ---------- Verify incore detector local power density alarms satisfy the requirements of the core power distribution map, which shall be updated at least once per 31 days of accumulated operation in MODE 1.	31 days

(continued)

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SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.2.1.3	required to be met ----------NOTES---------- 1. Only applicable when the Incore Detector Monitoring System is being used to determine LHR. 2. Not required to be performed below 20% RTP. ---------- Verify incore detector local power density alarm setpoints are less than or equal to the limits specified in the COLR.	31 days

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.1.1	required to be met ----------NOTE---------- Only ~~applicable~~ when COLSS is out of service. With COLSS in service, LHR is continuously monitored. ---------- Verify LHR, as indicated on each OPERABLE local power density channel, is ≤ [13.9 kW/ft].	2 hours
SR 3.2.1.2	Verify the COLSS margin alarm actuates at a THERMAL POWER equal to or less than the core power operating limit based on LHR.	31 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.3.1	----------NOTES---------- Only applicable when COLSS is out of service. With COLSS in service, this parameter is continuously monitored. ---------- Calculate T_q and verify it is within the limit.	12 hours
SR 3.2.3.2	Verify COLSS azimuthal tilt alarm is actuated at a T_q value less than the T_q value used in the CPCs.	31 days
SR 3.2.3.3	Independently confirm the validity of the COLSS calculated T_q by use of the incore detectors.	31 EFPD

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. DNBR outside the region of acceptable operation when COLSS is out of service.	B.1 Initiate SR 3.2.4.1. AND	15 minutes
	B.2 Restore DNBR to within limit.	4 hours
C. Required Action and associated Completion Time not met.	C.1 Reduce THERMAL POWER to ≤ 20% RTP.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.4.1	----------NOTE---------- Only required to be met with COLSS not in service and DNBR not within specified limits using any CPC channel. ---------- Verify no adverse trend in DNBR.	15 minutes

(continued)

TSTF-284 R.3

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.2.4.2	required to be met ----------NOTE---------- Only applicable when COLSS is out of service. With COLSS in service, this parameter is continuously monitored. ---------- Verify DNBR, as indicated on all OPERABLE DNBR channels, is within the limit of Figure 3.2.4-1 or 3.2.4-2 of the COLR, as applicable.	2 hours
SR 3.2.4.3	Verify COLSS margin alarm actuates at a THERMAL POWER level equal to or less than the core power operating limit based on DNBR.	31 days

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SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.8.3	Only required to be met ----------NOTE---------- ~~This SR is applicable~~ in MODES 1, 2, 3, and 4 only. ---------- Perform a CHANNEL FUNCTIONAL TEST on each required containment radiation monitor channel. Verify setpoint [Allowable Value] is in accordance with the following: Containment Gaseous Monitor: ≤ [2X background] Containment Particulate Monitor: ≤ [2X background] Containment Area Gamma Monitor: ≤ [325 mR/hr]	92 days
SR 3.3.8.4	Only required to be met ----------NOTE---------- ~~This SR is only applicable~~ during CORE ALTERATIONS or during movement of irradiated fuel assemblies within containment. ---------- Perform a CHANNEL FUNCTIONAL TEST on required containment radiation monitor channel. Verify setpoint [Allowable Value] is in accordance with the following: Containment Gaseous Monitor: ≤ [2X background] Containment Particulate Monitor: ≤ [2X background] Containment Iodine Monitor: ≤ [2X background] Containment Area Gamma Monitor: ≤ [2X background]	92 days

(continued)

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ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. (continued)	F.2 Restore at least one block valve to OPERABLE status.	2 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 (5)---------- 1. Not required to be performed with block valve closed in accordance with the Required Actions of this LCO. ---------- Perform a complete cycle of each block valve.	2. Only required to be performed in MODES 1 and 2. [92 days]
SR 3.4.11.2	Perform a complete cycle of each PORV.	[18] months
[SR 3.4.11.3	Perform a complete cycle of each solenoid air control valve and check valve on the air accumulators in PORV control systems.	[18] months]

(continued)

---- Note ----
Only required to be performed in MODES 1 and 2.

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.5.1	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
SR 3.7.5.2	----------NOTE---------- Not required to be performed for the turbine driven AFW pump until [24] hours after reaching [800] psig in the steam generators. ---------- Verify the developed head of each AFW pump at the flow test point is greater than or equal to the required developed head.	[31] days on a STAGGERED TEST BASIS
SR 3.7.5.3	----------NOTES---------- 1. Not required to be performed for the turbine driven AFW pump until [24] hours after reaching [800] psig in the steam generators. 2. 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

required to be met

(continued)

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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 reaching [800] psig in the steam generators. 2. Not applicable in MODE 4 when steam generator is relied upon for heat removal. ---------- required to be met Verify each AFW pump starts automatically on an actual or simulated actuation signal when in MODE 1, 2, or 3.	[18] months
SR 3.7.5.5	Verify the proper alignment of the required AFW flow paths by verifying flow from the condensate storage tank to each steam generator.	Prior to entering MODE 2 whenever unit has been in MODE 5 or 6 for > 30 days

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.3.1	Verify each required containment penetration is in the required status.	7 days
SR 3.9.3.2	Verify each required containment purge and exhaust valve actuates to the isolation position on an actual or simulated actuation signal.	[18] months

NOTE

Not required to be met for containment purge and exhaust valve(s) in penetrations closed to comply with LCO C.1.

BASES

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APPLICABILITY (continued)

temperature assumed in the safety analysis, is accepted as valid once the BOC and MOC measurements are used for normalization.

ACTIONS

A.1

MTC is a function of the fuel and fuel cycle designs, and cannot be controlled directly once the designs have been implemented in the core. If MTC exceeds its limits, the reactor must be placed in MODE 3. This eliminates the potential for violation of the accident analysis bounds. The associated Completion Time of 6 hours is reasonable, considering the probability of an accident occurring during the time period that would require an MTC value within the LCO limits, and the time for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.4.1 and SR 3.1.4.2

The SRs for measurement of the MTC at the beginning and middle of each fuel cycle provide for confirmation of the limiting MTC values. The MTC changes smoothly from most positive (least negative) to most negative value during fuel cycle operation, as the RCS boron concentration is reduced to compensate for fuel depletion. The requirement for measurement prior to operation > 5% RTP satisfies the confirmatory check on the most positive (least negative) MTC value. The requirement for measurement, within 7 days after reaching 40 effective full power days and ⅔ core burnup, satisfies the confirmatory check of the most negative MTC value. The measurement is performed at any THERMAL POWER, so that the projected EOC MTC may be evaluated before the reactor actually reaches the EOC condition. MTC values may be extrapolated and compensated to permit direct comparison to the specified MTC limits.

~~SR 3.1.4.2 is modified by a Note that indicates performance is not required prior to entering MODE 1 or 2. Although this Surveillance is applicable in MODES 1 and 2, the reactor must be critical before the Surveillance can be~~

(continued)

BASES

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SURVEILLANCE REQUIREMENTS

SR 3.1.4.1 and SR 3.1.4.2 (continued)

~~completed. Therefore, entry into the applicable MODE prior to accomplishing the Surveillance is necessary.~~

SR 3.1.4.2 is modified by a ~~second~~ Note, which indicates that if the extrapolated MTC is more negative than the EOC COLR limit, the Surveillance may be repeated, and that shutdown must occur prior to exceeding the minimum allowable boron concentration at which MTC is projected to exceed the lower limit. An engineering evaluation is performed if the extrapolated value of MTC exceeds the Specification limits.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 11.
2. FSAR, Section [ ].
3. FSAR, Section [ ].
4. FSAR, Section [ ].

BASES

TSTF-284 REV 3

APPLICABILITY (continued)

temperature assumed in the safety analysis, is accepted as valid once the BOC and MOC measurements are used for normalization.

ACTIONS

A.1

MTC is a function of the fuel and fuel cycle designs, and cannot be controlled directly once the designs have been implemented in the core. If MTC exceeds its limits, the reactor must be placed in MODE 3. This eliminates the potential for violation of the accident analysis bounds. The associated Completion Time of 6 hours is reasonable, considering the probability of an accident occurring during the time period that would require an MTC value within the LCO limits, and the time for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.4.1 and SR 3.1.4.2

The SRs for measurement of the MTC at the beginning and middle of each fuel cycle provide for confirmation of the limiting MTC values. The MTC changes smoothly from most positive (least negative) to most negative value during fuel cycle operation, as the RCS boron concentration is reduced to compensate for fuel depletion. The requirement for measurement prior to operation > 5% RTP satisfies the confirmatory check on the most positive (least negative) MTC value. The requirement for measurement, within 7 days after reaching 40 effective full power days and a ⅔ core burnup, satisfies the confirmatory check of the most negative MTC value. The measurement is performed at any THERMAL POWER so that the projected EOC MTC may be evaluated before the reactor actually reaches the EOC condition. MTC values may be extrapolated and compensated to permit direct comparison to the specified MTC limits.

SR 3.1.4.2 is modified by a Note that indicates performance is not required prior to entering MODE 1 or 2. Although this Surveillance is applicable in MODES 1 and 2, the reactor must be critical before the Surveillance can be

(continued)

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.4.1 and SR 3.1.4.2 (continued)

completed. Therefore, entry into the applicable MODE prior to accomplishing the Surveillance is necessary.

SR 3.1.4.2 is modified by a ~~second~~ Note, which indicates that if extrapolated MTC is more negative than the EOC COLR limit, the Surveillance may be repeated, and that shutdown must occur prior to exceeding the minimum allowable boron concentration at which MTC is projected to exceed the lower limit. An engineering evaluation is performed if the extrapolated value of MTC exceeds the Specification limits.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 11.
2. FSAR, Section [ ].
3. FSAR, Section [ ].
4. FSAR, Section [ ].

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BASES

ACTIONS

C.1 (continued)

regulating CEAs to withdraw to the acceptable region. It is reasonable to continue operation for 2 hours after it is discovered that the 5 day or 14 day EFPD limit has been exceeded. This Completion Time is based on limiting the potential xenon redistribution, the low probability of an accident, and the steps required to complete the action.

D.1

When the PDIL alarm circuit is inoperable, performing SR 3.1.7.1 within 1 hour and once per 4 hours thereafter ensures improper CEA alignments are identified before unacceptable flux distributions occur.

E.1

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

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1

With the PDIL alarm circuit OPERABLE, verification of each regulating CEA group position every 12 hours is sufficient to detect CEA positions that may approach the acceptable limits, and to provide the operator with time to undertake the Required Action(s) should the sequence or insertion limits be found to be exceeded. The 12 hour Frequency also takes into account the indication provided by the PDIL alarm circuit and other information about CEA group positions available to the operator in the control room.

SR 3.1.7.1 is modified by a Note indicating that entry is allowed into MODE 2 for 12 hours without having performed the SR. This is necessary, since the unit must be in the applicable MODES in order to perform Surveillances that demonstrate the LCO limits are met.

(continued)

BASES

TSTF-284 REV 3

ACTIONS

F.1 (continued)

MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1

With the PDIL alarm circuit OPERABLE, verification of each regulating CEA group position every 12 hours is sufficient to detect CEA positions that may approach the acceptable limits, and provides the operator with time to undertake the Required Action(s) should the sequence or insertion limits be found to be exceeded. The 12 hour Frequency also takes into account the indication provided by the PDIL alarm circuit and other information about CEA group positions available to the operator in ~~the control room~~.

SR 3.1.7.1 is modified by a Note indicating that entry is allowed into MODE 2 for 12 hours without having performed the SR. This is necessary, since the unit must be in the applicable MODES in order to perform Surveillances that demonstrate the LCO limits are met.

SR 3.1.7.2

Verification of the accumulated time of CEA group insertion between the long term steady state insertion limits and the transient insertion limits ensures the cumulative time limits are not exceeded. The 24 hour Frequency ensures the operator identifies a time limit that is being approached before it is reached.

SR 3.1.7.3

Demonstrating the PDIL alarm circuit OPERABLE verifies that the PDIL alarm circuit is functional. The 31 day Frequency takes into account other Surveillances being performed at shorter Frequencies that identify improper CEA alignments.

(continued)

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BASES

ACTIONS

A.1 (continued)

specified limits. One hour to restore the LHR to within its specified limits is reasonable and ensures that the core does not continue to operate in this Condition. The 1 hour Completion Time also allows the operator sufficient time for evaluating core conditions and for initiating proper corrective actions.

B.1

If the LHR cannot be returned to within its specified limits, THERMAL POWER must be reduced. The change to MODE 2 ensures that the core is operating within its thermal limits and places the core in a conservative condition. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 2 from full power MODE 1 conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

A Note was added to the SRs to require LHR to be determined by either the Excore Detector Monitoring System or the Incore Detector Monitoring System.

SR 3.2.1.1

Performance of this SR verifies that the Excore Detector Monitoring System can accurately monitor the LHR. Therefore, this SR is only applicable when the Excore Detector Monitoring System is being used to determine the LHR. The 31 day Frequency is appropriate for this SR because it is consistent with the requirements of SR 3.3.1.3 for calibration of the excore detectors using the incore detectors.

required to be met

The SR is modified by a Note that states that the SR is only applicable when the Excore Detection Monitoring System is being used to determine LHR. The reason for the Note is that the excore detectors input neutron flux information into the ASI calculation.

(continued)

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.2.1.2 and SR 3.2.1.3

Continuous monitoring of the LHR is provided by the Incore Detector Monitoring System and the Excore Detector Monitoring System. Either of these two core power distribution monitoring systems provides adequate monitoring of the core power distribution and is capable of verifying that the LHR does not exceed its specified limits.

Performance of these SRs verifies that the Incore Detector Monitoring System can accurately monitor LHR. Therefore, they are only applicable when the Incore Detector Monitoring System is being used to determine the LHR.

A 31 day Frequency is consistent with the historical testing frequency of the reactor monitoring system. The SRs are modified by two Notes. Note 1 allows the SRs to be performed only when the Incore Detector Monitoring System is being used to determine LHR. Note 2 states that the SRs are not required to be performed when THERMAL POWER is < 20% RTP. The accuracy of the neutron flux information from the incore detectors is not reliable at THERMAL POWER < 20% RTP.

met

REFERENCES

1. FSAR, Chapter [15].

2. FSAR, Chapter [6].

3. 10 CFR 50, Appendix A.

4. 10 CFR 50.46.

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BASES

ACTIONS

C.1 (continued)

inoperability, core power must be reduced. Reduction of core power to < 20% RTP ensures that the core is operating within its thermal limits and places the core in a conservative condition based on the trip setpoints generated by the CPCs, which assume a minimum core power of 20% RTP. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach 20% RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1

With the COLSS out of service, the operator must monitor the LHR with each OPERABLE local power density channel. A 2 hour Frequency is sufficient to allow the operator to identify trends that would result in an approach to the LHR limits.

required to be met

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

SR 3.2.1.2

Verification that the COLSS margin alarm actuates at a THERMAL POWER level equal to or less than the core power operating limit based on the LHR in units of kilowatts per foot ensures the operator is alerted when conditions approach the LHR operating limit.

The 31 day Frequency for performance of this SR is consistent with the historical testing frequency of reactor protection and monitoring systems. The Surveillance Frequency for testing protection systems was extended to 92 days by CEN 327. Monitoring systems were not addressed in CEN 327; therefore, this Frequency remains at 31 days.

(continued)

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BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1

With the COLSS out of service, the operator must monitor the DNBR as indicated on any of the OPERABLE DNBR channels of the CPCs to verify that the DNBR is within the specified limits, shown in either Figure 3.2.4-1 or 3.2.4-2 of the COLR, as applicable. A 2 hour Frequency is adequate to allow the operator to identify trends in conditions that would result in an approach to the DNBR limit.

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

required to be met

SR 3.2.4.2

Verification that the COLSS margin alarm actuates at a power level equal to or less than the core power operating limit, as calculated by the COLSS, based on the DNBR, ensures that the operator is alerted when operating conditions approach the DNBR operating limit. The 31 day Frequency for performance of this SR is consistent with the historical testing frequency of reactor protection and monitoring systems. The Surveillance Frequency for testing protection systems was extended to 92 days by CEN 327. Monitoring systems were not addressed in CEN 327; therefore, this Frequency remains at 31 days.

REFERENCES

1. FSAR, Chapter [15].
2. FSAR, Chapter [6].
3. CE-1 Correlation for DNBR.
4. 10 CFR 50, Appendix A, GDC 10.
5. 10 CFR 50.46.

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.3.8.3 (continued)

(Ref. 4). The Frequency of 92 days is based on plant operating experience with regard to channel OPERABILITY and drift, which demonstrates that failure of more than one channel of a given Function in any 92 day Frequency is a rare event.

A Note to the SR indicates this Surveillance is ~~applicable~~ required to be met in MODES 1, 2, 3, and 4 only.

SR 3.3.8.4

A CHANNEL FUNCTIONAL TEST is performed on the required containment radiation monitoring channel to ensure the entire channel will perform its intended function. Setpoints must be found within the Allowable Values specified in SR 3.3.8.4 and left consistent with the assumptions of the plant specific setpoint methodology (Ref. 4). The Frequency of 92 days is based on plant operating experience with regard to channel OPERABILITY and drift, which demonstrates that failure of more than one channel of a given Function in any 92 day interval is a rare event.

A Note to the SR indicates that this test is only ~~applicable~~ required to be met during CORE ALTERATIONS or during movement of irradiated fuel assemblies within containment.

SR 3.3.8.5

Proper operation of the individual initiation relays is verified by actuating these relays during the CHANNEL FUNCTIONAL TEST of the Actuation Logic every [18] months. This will actuate the Function, operating all associated equipment. Proper operation of the equipment actuated by each train is thus verified. The Frequency of [18] months is based on plant operating experience with regard to channel OPERABILITY and drift, which demonstrates that failure of more than one channel of a given Function during any [18] month interval is a rare event. A Note to the SR indicates that this Surveillance includes verification of operation for each initiation relay.

(continued)

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BASES

APPLICABILITY (continued)

MODES 4, 5, and 6 with the reactor vessel head in place. LCO 3.4.12 addresses the PORV requirements in these MODES.

ACTIONS

A.1

The ACTIONS are modified by two Notes. Note 1 clarifies that all pressurizer PORVs are treated as separate entities, each with separate Completion Times (i.e., the Completion Time is on a component basis). Note 2 is an exception to LCO 3.0.4. The exception for LCO 3.0.4 permits entry into MODES 1, 2, and 3 to perform cycling of the PORV or block valve to verify their OPERABLE status, in the event that testing was not satisfactorily performed in lower MODES. Testing is typically not performed in lower MODES.

With the PORV inoperable and capable of being manually cycled, either the PORV must be restored or the flow path isolated within 1 hour. The block valve should be closed but power must be maintained to the associated block valve, since removal of power would render the block valve inoperable. Although the PORV may be designated inoperable, it may be able to be manually opened and closed and in this manner can be used to perform its function. PORV inoperability may be due to seat leakage, instrumentation problems, automatic control problems, or other causes that do not prevent manual use and do not create a possibility for a small break LOCA. For these reasons, the block valve may be closed but the Action requires power be maintained to the valve. This Condition is only intended to permit operation of the plant for a limited period of time not to exceed the next refueling outage (MODE 6) so that maintenance can be performed on the PORVs to eliminate the problem condition. The PORVs should normally be available for automatic mitigation of overpressure events and should be returned to OPERABLE status prior to entering startup (MODE 2).

Quick access to the PORV for pressure control can be made when power remains on the closed block valve. The Completion Time of 1 hour is based on plant operating experience that minor problems can be corrected or closure can be accomplished in this time period.

(continued)

BASES

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ACTIONS

F.1 and F.2 (continued)

control is reasonable based on the small potential for challenges to the system during this time and provides the operator time to correct the situation.

G.1 and G.2

If the Required Actions and associated Completion Times of Condition E or F are not met, then the plant must be brought to a MODE in which the LCO does not apply. The plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging safety systems. Similarly, the Completion Time of 12 hours to reach MODE 4 is reasonable considering that a plant can cool down within that time frame on one safety system train. In MODES 4 and 5, maintaining PORV OPERABILITY may be required. See LCO 3.4.12.

SURVEILLANCE REQUIREMENTS

SR 3.4.11.1

Block valve cycling verifies that it can be closed if necessary. The basis for the Frequency of [92 days] is ASME XI (Ref. 3). ~~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 surveillance interval 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 Action fulfills the SR).~~

This SR is modified by two Notes.

The Note 1 modifies this SR by stating that this SR is not required to be performed with the block valve closed in accordance with the Required Actions of this LCO.

Insert A

(continued)

BASES

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.11.2

SR 3.4.11.2 requires complete cycling of each PORV. PORV cycling demonstrates its function. The Frequency of [18] months is based on a typical refueling cycle and industry accepted practice.

Insert B

SR 3.4.11.3

Operating the solenoid air control valves and check valves on the air accumulators ensures the PORV control system actuates properly when called upon. The Frequency of [18] months is based on a typical refueling cycle and the Frequency of the other surveillances used to demonstrate PORV OPERABILITY.

SR 3.4.11.4

This Surveillance is not required for plants with permanent 1E power supplies to the valves. The test demonstrates that emergency power can be provided and is performed by transferring power from the normal supply to the emergency supply and cycling the valves. The Frequency of [18] months is based on a typical refueling cycle and industry accepted practice.

REFERENCES

1. NUREG-0737, Paragraph III, G.I, November 1980.

2. Inspection and Enforcement (IE) Bulletin 79-05B, April 21, 1979.

3. ASME, Boiler and Pressure Vessel Code, Section XI.

Insert C

SURVEILLANCE REQUIREMENTS

SR 3.7.5.2 (continued)

TEST BASIS results in testing each pump once every 3 months, as required by Reference 2.

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 an insufficient steam pressure to perform the test.

SR 3.7.5.3

This SR ensures that AFW can be delivered to the appropriate steam generator, in the event of any accident or transient that generates an EFAS signal, 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 the design reliability and operating experience of the equipment.

This SR is modified by a Note indicating that the SR should be deferred until suitable test conditions have been established. This deferral is required because there is an insufficient steam pressure to perform the test.

Also, this SR is modified by a Note that states the SR is not required in MODE 4. In MODE 4, the required AFW train is already aligned and operating.

to be met

SR 3.7.5.4

This SR ensures that the AFW pumps will start in the event of any accident or transient that generates an EFAS signal by demonstrating that each AFW pump starts automatically on an actual or simulated actuation signal. The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the

(continued)

TSTF-284/B3

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.5.4 (continued)

potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 18 month Frequency is acceptable, based on the design reliability and operating experience of the equipment.

This SR is modified by [a] [two] Note[s]. [Note 1 indicates that the SR be deferred until suitable test conditions are established. This deferral is required because there is insufficient steam pressure to perform the test.] [The] Note [2] states that the SR is not required in MODE 4. [In MODE 4, the required pump is already 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.]

to be met

[Reviewer's Note: Some plants may not routinely use the AFW for heat removal in MODE 4. The second justification is provided for plants that use a startup feedwater pump rather than AFW for startup and shutdown.]

SR 3.7.5.5

This SR ensures that the AFW System is properly aligned by verifying the flow path to each steam generator prior to entering MODE 2 operation, after 30 days in MODE 5 or 6. OPERABILITY of AFW flow paths must be verified before sufficient core heat is generated that would require the operation of the AFW System during a subsequent shutdown. The Frequency is reasonable, based on engineering judgment, and other administrative controls to ensure that flow paths remain OPERABLE. To further ensure AFW System alignment, the OPERABILITY of the flow paths is verified following extended outages to determine that no misalignment of valves has occurred. This SR ensures that the flow path from the CST to the steam generators is properly aligned by requiring a verification of minimum flow capacity of 750 gpm at 1270 psi. (This SR is not required by those units that use AFW for normal startup and shutdown.)

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.9.3.1 (continued)

OPERABLE automatic containment purge and exhaust isolation signal.

The Surveillance is performed every 7 days during CORE ALTERATIONS or movement of irradiated fuel assemblies within the containment. The Surveillance interval is selected to be commensurate with the normal duration of time to complete fuel handling operations. A surveillance before the start of refueling operations will provide two or three surveillance verifications during the applicable period for this LCO. As such, this Surveillance ensures that a postulated fuel handling accident that releases fission product radioactivity within the containment will not result in a release of fission product radioactivity to the environment.

SR 3.9.3.2

This Surveillance demonstrates that each containment purge and exhaust valve actuates to its isolation position on manual initiation or on an actual or simulated high radiation signal. The 18 month Frequency maintains consistency with other similar ESFAS instrumentation and valve testing requirements. In LCO 3.3.4 [(Digital) or 3.3.3 (Analog)], "Miscellaneous Actuations," the Containment Purge Isolation Signal System requires a CHANNEL CHECK every 7 days and a CHANNEL FUNCTIONAL TEST every 31 days to ensure the channel OPERABILITY during refueling operations. Every 18 months a CHANNEL CALIBRATION is performed. The system actuation response time is demonstrated every 18 months, during refueling, on a STAGGERED TEST BASIS. SR 3.6.3.5 demonstrates that the isolation time of each valve is in accordance with the Inservice Testing Program requirements. These surveillances performed during MODE 6 will ensure that the valves are capable of closing after a postulated fuel handling accident to limit a release of fission product radioactivity from the containment.

The SR is modified by a Note stating that this Surveillance is not required to be met for valves in isolated penetrations. The LCO provides the option to close penetrations in lieu of requiring automatic actuation capability.

(continued)

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1.0 USE AND APPLICATION

1.4 Frequency

PURPOSE

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

DESCRIPTION

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

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

Sometimes special situations dictate when the requirements of a Surveillance are to be met. They are "otherwise stated" conditions allowed by SR 3.0.1. They may be stated as clarifying Notes in the Surveillance, as part of the Surveillance, or both. ~~Example 1.4-4 discusses these special situations.~~

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

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

(continued)

TSTF-284, Rev 3

1.4 Frequency

Insert 1A

DESCRIPTION (continued)

criteria. SR 3.0.4 restrictions would not apply if both the following conditions are satisfied:

a. The Surveillance is not required to be performed; and

b. The Surveillance is not required to be met or, even if required to be met, is not known to be failed.

EXAMPLES

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

EXAMPLE 1.4-1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Perform CHANNEL CHECK.	12 hours

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

(continued)

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EXAMPLES

EXAMPLE 1.4-3 (continued)

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

EXAMPLE 1.4-4

SURVEILLANCE REQUIREMENTS

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

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

INSERT 3

INSERT 4

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1.0 USE AND APPLICATION

1.4 Frequency

PURPOSE

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

DESCRIPTION

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

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

Sometimes special situations dictate when the requirements of a Surveillance are to be met. They are "otherwise stated" conditions allowed by SR 3.0.1. They may be stated as clarifying Notes in the Surveillance, as part of the Surveillance, or both. ~~Example 1.4-4 discusses these special situations.~~

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

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

(continued)

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1.4 Frequency

Insert 1A

DESCRIPTION (continued)

criteria. SR 3.0.4 restrictions would not apply if both the following conditions are satisfied:

a. The Surveillance is not required to be performed; and

b. The Surveillance is not required to be met or, even if required to be met, is not known to be failed.

EXAMPLES

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

EXAMPLE 1.4-1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Perform CHANNEL CHECK.	12 hours

Example 1.4-1 contains the type of SR most often encountered in the Technical Specifications (TS). The Frequency specifies an interval (12 hours) during which the associated Surveillance must be performed at least one time. Performance of the Surveillance initiates the subsequent interval. Although the Frequency is stated as 12 hours, an extension of the time interval to 1.25 times the interval specified in the Frequency is allowed by SR 3.0.2 for operational flexibility. The measurement of this interval continues at all times, even when the SR is not required to be met per SR 3.0.1 (such as when the equipment is inoperable, a variable is outside specified limits, or the unit is outside the Applicability of the LCO). If the interval specified by SR 3.0.2 is exceeded while the unit is in a MODE or other specified condition in the Applicability of the LCO, and the performance of the Surveillance is not otherwise modified (refer to Examples 1.4-3 and 1.4-4), then SR 3.0.3 becomes applicable.

(continued)

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EXAMPLES

EXAMPLE 1.4-3 (continued)

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

EXAMPLE 1.4-4

SURVEILLANCE REQUIREMENTS

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

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

INSERT 3

INSERT 4

Technical Specification Task Force
Improved Standard Technical Specifications Change Traveler

Administrative Controls Program 5.5.2.b Test Interval and Exception

NUREGs Affected: ☑ 1430 ☑ 1431 ☑ 1432 ☑ 1433 ☑ 1434

Classification: 2) Consistency/Standardization Recommended for CLIIP?: (Unassigned)

Correction or Improvement: (Unassigned)

Industry Contact: Tom Silko, (802) 258-4146, tsilko@entergy.com

Program 5.5.2, "Primary Coolant Sources Outside Containment," is revised to clarify the intent of refueling cycle intervals with respect to the system integrated leak test requirements (i.e., [18] month intervals) and to add the following sentence, "The provisions of SR 3.0.2 are applicable."
ISTS 5.5.2.b provides integrated leak test requirements for each system at refueling cycle intervals or less. ISTS 5.5.5.2.b is revised to require integrated leak test requirements for each system at [18] month intervals or less. ISTS 5.5.2.b is essentially a Surveillance Requirement. Since normal "refueling cycle intervals" are 18 months, presenting the requirement in this manner achieves consistency with similar requirements in the ISTS. The ISTS Surveillance Requirements specify "[18] months" and not refueling cycle intervals for Surveillance performed at refueling intervals. This change also allows approved changes to ISTS 5.5.2.b associated with implementation of 24 month refueling cycles to be explicitly documented. As a result of explicitly stating the interval for the test, it will no longer be possible to account for shutdowns or power reductions that may occur during the cycle in order to satisfy the interval requirements for the tests required by ISTS 5.5.2 b i.e., a refueling cycle may be longer than [18] months ,in order to achieve the required fuel burnup. but the testing of ISTS 5.5.2.b would be required to be performed once per [18] months. For consistency with normal Surveillance Requirements in the ISTS LCO Sections that allow a 25% extension of the Frequency in accordance with ISTS SR 3 0.2, ISTS 5.5.2.b is considered a Surveillance Requirement. ISTS 5.5.2 is revised to allow the provisions of ISTS SR 3 0 2 to be applicable to ISTS 5.5.2 b. The applicability of ISTS SR 3.0.2 must be explicitly stated in ISTS 5.5.2 since ISTS SR 3.0.2 only applies to the ISTS LCO Sections (i.e., ISTS LCO Sections 3.1 through 3.9 or 3.10).

Revision History

OG Revision 0 Revision Status: Active

Revision Proposed by: Brunswick

Revision Description:
Original Issue

Owners Group Review Information

Date Originated by OG: 03-Nov-97

Owners Group Comments:
(No Comments)

Owners Group Resolution: Approved Date: 03-Nov-97

TSTF Review Information

TSTF Received Date: 03-Nov-97 Date Distributed for Review: 28-May-98

OG Review Completed: ☑ BWOG ☑ WOG ☑ CEOG ☑ BWROG

TSTF Comments:
Applicable to BWR/4 and BWR/6.

TSTF Resolution: Approved Date: 10-Jul-98

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OG Revision 0 Revision Status: Active

NRC Review Information

NRC Received Date: 13-Nov-98

Date of NRC Letter: 31-Oct-00

Final Resolution: NRC Approves

Final Resolution Date: 31-Oct-00

Affected Technical Specifications

5.5.2 Primary Coolant Sources Outside of Containment

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INSERT 1

least once per [18] months.

The provisions of SR 3.0.2 are applicable.

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5.5 Programs and Manuals

5.5.1 Offsite Dose Calculation Manual (ODCM) (continued)

page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

5.5.2 Primary Coolant Sources Outside Containment

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The systems include [Low Pressure Injection, Reactor Building Spray, Makeup and Purification, and Hydrogen Recombiner]. The program shall include the following:

a. Preventive maintenance and periodic visual inspection requirements; and

b. Integrated leak test requirements for each system at refueling cycle intervals or less.

Insert 1

5.5.3 Post Accident Sampling

This program provides controls that ensure the capability to obtain and analyze reactor coolant, radioactive gases, and particulates in plant gaseous effluents and containment atmosphere samples under accident conditions. The program shall include the following:

a. Training of personnel;

b. Procedures for sampling and analysis; and

c. Provisions for maintenance of sampling and analysis equipment.

5.5.4 Radioactive Effluent Controls Program

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to

(continued)

TSTF-299

5.5 Programs and Manuals

5.5.1 Offsite Dose Calculation Manual (ODCM) (continued)

page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

5.5.2 Primary Coolant Sources Outside Containment

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The systems include [Recirculation Spray, Safety Injection, Chemical and Volume Control, gas stripper, and Hydrogen Recombiner]. The program shall include the following:

a. Preventive maintenance and periodic visual inspection requirements; and

Insert 1

b. Integrated leak test requirements for each system at ~~refueling cycle intervals or less.~~

5.5.3 Post Accident Sampling

This program provides controls that ensure the capability to obtain and analyze reactor coolant, radioactive gases, and particulates in plant gaseous effluents and containment atmosphere samples under accident conditions. The program shall include the following:

a. Training of personnel;

b. Procedures for sampling and analysis; and

c. Provisions for maintenance of sampling and analysis equipment.

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(continued)

TSTF-299

5.5 Programs and Manuals

5.5.1 Offsite Dose Calculation Manual (ODCM) (continued)

the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

5.5.2 Primary Coolant Sources Outside Containment

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The systems include [Recirculation Spray, Safety Injection, Chemical and Volume Control, gas stripper, and Hydrogen Recombiner]. The program shall include the following:

a. Preventive maintenance and periodic visual inspection requirements; and

Insert 1

b. Integrated leak test requirements for each system at refueling cycle intervals or less.

5.5.3 Post Accident Sampling

This program provides controls that ensure the capability to obtain and analyze reactor coolant, radioactive gases, and particulates in plant gaseous effluents and containment atmosphere samples under accident conditions. The program shall include the following:

a. Training of personnel;

b. Procedures for sampling and analysis; and

c. Provisions for maintenance of sampling and analysis equipment.

5.5.4 Radioactive Effluent Controls Program

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably

(continued)

TSTF-299

5.5 Programs and Manuals

5.5.1 Offsite Dose Calculation Manual (ODCM) (continued)

that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

5.5.2 Primary Coolant Sources Outside Containment

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The systems include [the Low Pressure Core Spray, High Pressure Coolant Injection, Residual Heat Removal, Reactor Core Isolation Cooling, hydrogen recombiner, process sampling, and Standby Gas Treatment]. The program shall include the following:

a. Preventive maintenance and periodic visual inspection requirements; and

Insert 1

b. Integrated leak test requirements for each system at refueling cycle intervals or less.

5.5.3 Post Accident Sampling

This program provides controls that ensure the capability to obtain and analyze reactor coolant, radioactive gases, and particulates in plant gaseous effluents and containment atmosphere samples under accident conditions. The program shall include the following:

a. Training of personnel;

b. Procedures for sampling and analysis; and

c. Provisions for maintenance of sampling and analysis equipment.

5.5.4 Radioactive Effluent Controls Program

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably

(continued)

TSTF-299

5.5 Programs and Manuals

5.5.1 Offsite Dose Calculation Manual (ODCM) (continued)

Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

5.5.2 Primary Coolant Sources Outside Containment

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The systems include [the Low Pressure Core Spray, High Pressure Core Spray, Residual Heat Removal, Reactor Core Isolation Cooling, hydrogen recombiner, process sampling, and Standby Gas Treatment]. The program shall include the following:

a. Preventive maintenance and periodic visual inspection requirements; and

Insert 1

b. Integrated leak test requirements for each system at ~~refueling cycle intervals or less.~~

5.5.3 Post Accident Sampling

This program provides controls that ensure the capability to obtain and analyze reactor coolant, radioactive gases, and particulates in plant gaseous effluents and containment atmosphere samples under accident conditions. The program shall include the following:

a. Training of personnel;

b. Procedures for sampling and analysis; and

c. Provisions for maintenance of sampling and analysis equipment.

5.5.4 Radioactive Effluent Controls Program

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of

(continued)

Technical Specification Task Force
Improved Standard Technical Specifications Change Traveler

Determination of Cumulative and Projected Dose Contributions in RECP

NUREGs Affected: ☑ 1430 ☑ 1431 ☑ 1432 ☑ 1433 ☑ 1434

Classification: 3) Improve Specifications Recommended for CLIIP?: (Unassigned)

Correction or Improvement: (Unassigned)

Industry Contact: Steve Wideman, (620) 364-4037, stwidem@wcnoc.com

Revise 5.5.4.e wording to describe the actual intent of the dose projections. New proposed words are the same as pre-GL 89-01 implementation.
The NRC Staff's draft STS for 4-loop Westinghouse plants (8/14/87 letter to Texas Utilities) included Radioactive Effluent Technical Specifications.

SR 4.11.1.2 for DOSE states, "Cumulative dose contributions from liquid effluents for the current calendar quarter and the current calendar year shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days."

SR 4.11.1.3.1 for LIQUID RADWASTE TREATMENT SYSTEM states, "Doses due to liquid releases from each unit to UNRESTRICTED AREAS shall be projected at least once per 31 days in accordance with the methodology and parameters in the ODCM when Liquid Radwaste Treatment Systems are not being fully utilized."

Generic Letter 89-01 appears to have combined these two Surveillance Requirements for cumulative and projected doses. In combining these requirements in Generic Letter 89-01, the new program element can be interpreted to require determining projected dose contribution for the current calendar quarter and current calendar year every 31 days. Therefore, the proposed change clarifies the wording in 5.5.4.e to not require dose projections for a calendar quarter and a calendar year very 31 days.

Revision History

OG Revision 0 Revision Status: Closed

Revision Proposed by: Diablo Canyon

Revision Description:
Original Issue

Owners Group Review Information

Date Originated by OG: 10-Oct-96

Owners Group Comments:
(No Comments)

Owners Group Resolution: Approved Date: 10-Oct-96

TSTF Review Information

TSTF Received Date: 11-Oct-96 Date Distributed for Review: 29-Oct-96

OG Review Completed: ☑ BWOG ☑ WOG ☑ CEOG ☑ BWROG

TSTF Comments:

CEOG/BWOG/BWROG - rewrite justification and bring back to TSTF.

TSTF Resolution: Rejected Date: 19-Dec-96

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OG Revision 0 Revision Status: Closed

OG Revision 1 Revision Status: Closed

Revision Proposed by: WOG

Revision Description:
Revised justification to address TSTF comments on Revision 0.

Owners Group Review Information

Date Originated by OG: 20-Nov-97

Owners Group Comments:
(No Comments)

Owners Group Resolution: Approved Date: 20-Nov-97

TSTF Review Information

TSTF Received Date: 20-Nov-97 Date Distributed for Review: 06-Jan-98

OG Review Completed: ☑ BWOG ☑ WOG ☑ CEOG ☑ BWROG

TSTF Comments:
(No Comments)

TSTF Resolution: Approved Date: 25-Jun-98

NRC Review Information

NRC Received Date: 13-Nov-98

NRC Comments: Date of NRC Letter: 16-Mar-00

3/16/2000 - NRC provided comments:
The change is unacceptable because it does not specify a time period for the determination of performing a cumulative dose calculation. An acceptable requirement is: "Determination of cumulative dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days. Determination of projected dose contribution from radioactive effluents in accordance with the methodology in the ODCM at least every 31 days."

Final Resolution: Superceded by Revision Final Resolution Date: 16-Mar-00

TSTF Revision 1 Revision Status: Active

Revision Proposed by: NRC

Revision Description:
Revised Insert A to clarify that the determination of cumulative dose contributions from radioactive effluents for the current calendar quarter and current calendar year at least every 31 days.

TSTF Review Information

TSTF Received Date: 13-Jun-00 Date Distributed for Review: 13-Jun-00

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TSTF Revision 1 Revision Status: Active

OG Review Completed: ☑ BWOG ☑ WOG ☑ CEOG ☑ BWROG

TSTF Comments:
(No Comments)

TSTF Resolution: Approved Date: 13-Jun-00

NRC Review Information

NRC Received Date: 20-Jun-00

Final Resolution: NRC Approves Final Resolution Date: 06-Jul-00

Affected Technical Specifications

5.5.4.e Radioactive Effluent Controls Program

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Insert A

Determination of cumulative dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days. Determination of projected dose contributions from radioactive effluents in accordance with the methodology in the ODCM at least every 31 days.

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5.5.4 Radioactive Effluent Controls Program (continued)

be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to 10 CFR 20, Appendix B, Table 2, Column 2;

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

d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each unit to unrestricted areas, conforming to 10 CFR 50, Appendix I;

Insert A

e. ~~Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days;~~

f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2% of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I;

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the site boundary conforming to the dose associated with 10 CFR 20, Appendix B, Table 2, Column 1;

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

(continued)

TSTF-308, Rev. 1

5.5.4 Radioactive Effluent Controls Program (continued)

be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to 10 CFR 20, Appendix B, Table 2, Column 2;

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

d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each unit to unrestricted areas, conforming to 10 CFR 50, Appendix I;

Insert A

e. ~~Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days;~~

f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2% of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I;

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the site boundary conforming to the dose associated with 10 CFR 20, Appendix B, Table 2, Column 1;

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

(continued)

TSTF-308, Rev.1

5.5.4 Radioactive Effluent Controls Program (continued)

achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to 10 CFR 20, Appendix B, Table 2, Column 2;

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

d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each unit to unrestricted areas, conforming to 10 CFR 50, Appendix I;

Insert A

e. ~~Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days;~~

f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2% of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I;

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the site boundary conforming to the dose associated with 10 CFR 20, Appendix B, Table 2, Column 1;

(continued)

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5.5.4 Radioactive Effluent Controls Program (continued)

achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to 10 CFR 20, Appendix B, Table 2, Column 2;

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

d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each unit to unrestricted areas, conforming to 10 CFR 50, Appendix I;

Insert A

e. ~~Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days;~~

f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2% of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I;

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the site boundary conforming to the dose associated with 10 CFR 20, Appendix B, Table 2, Column 1;

(continued)

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5.5.4 Radioactive Effluent Controls Program (continued)

the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to 10 CFR 20, Appendix B, Table 2, Column 2;

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

d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each unit to unrestricted areas, conforming to 10 CFR 50, Appendix I;

e. Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days;

Insert A

f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2% of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I;

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the site boundary conforming to the dose associated with 10 CFR 20, Appendix B, Table 2, Column 1;

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each

(continued)

Technical Specification Task Force
Improved Standard Technical Specifications Change Traveler

Containment Structural Integrity

NUREGs Affected: ☑ 1430 ☑ 1431 ☑ 1432 ☑ 1433 ☑ 1434

Classification: 1) Technical Change

Recommended for CLIIP?: Yes

Correction or Improvement: Correction

NRC Fee Status: Not Exempt

Benefit: Allows Less Stringent Testing

Industry Contact: Paul Infanger, (352) 563-4796, paul.infanger@pgnmail.com

See attached justification.

Revision History

OG Revision 0 Revision Status: Closed

Revision Proposed by: Oconee

Revision Description:
Original Issue

Owners Group Review Information

Date Originated by OG: 02-Apr-98

Owners Group Comments:
ONS-28

Owners Group Resolution: Approved Date: 02-Apr-98

TSTF Review Information

TSTF Received Date: 01-May-98 Date Distributed for Review: 12-Oct-98

OG Review Completed: ☑ BWOG ☑ WOG ☑ CEOG ☑ BWROG

TSTF Comments:

4/28/99 - Revise to bracket the Surveillance and the Program with a Reviewer's Note stating that the SR and the Program may be deleted by plants that have adopted ASME Section XI, Subsections IWE and IWC. Expand the Containment Bases to discuss structural integrity and reference 10 CFR 50.55a.

TSTF Resolution: Superceeded Date: 20-Nov-98

OG Revision 1 Revision Status: Closed

Revision Proposed by: TSTF

Revision Description:
Complete replacement of Revision 0. Revised to bracket the Surveillance and the Program with a Reviewer's Note stating that the SR and the Program may be deleted by plants that have adopted ASME Section XI, Subsections IWE and IWC. Expand the Containment Bases to discuss structural integrity and reference 10 CFR 50.55a.

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OG Revision 1 Revision Status: Closed

TSTF Review Information

TSTF Received Date: 02-Jun-99 Date Distributed for Review: 08-Jun-99

OG Review Completed: ☑ BWOG ☑ WOG ☑ CEOG ☑ BWROG

TSTF Comments:

Change "IWC" to "IWL" in justification and insert.

TSTF Resolution: Approved Date: 07-Jul-99

NRC Review Information

NRC Received Date: 20-Jul-99

NRC Comments:

2/11/00 - NRC provided comments. TSTF to respond.

1/13/00 - Noel spoke to NRC. NRC agrees that the requirements are in the Code, but also wants them in Tech Spec. Further discussion is required.

5/11/2001 - BWOG chairman to discuss with NRC to identify concerns by 5/31/2001.

Final Resolution: Reviewer Recommends Changes

TSTF Revision 1 Revision Status: Active

Revision Proposed by: Wolf Creek

Revision Description:
Complete replacement of Revision 1. Revises ISTS consistent with many approved plant-specific amendments. Modeled closely on the Wolf Creek approved amendment.

TSTF Review Information

TSTF Received Date: 05-May-05 Date Distributed for Review: 05-May-05

OG Review Completed: ☑ BWOG ☑ WOG ☑ CEOG ☑ BWROG

TSTF Comments:

(No Comments)

TSTF Resolution: Approved Date: 20-Jun-05

NRC Review Information

NRC Received Date: 29-Jun-05

NRC Comments: Date of NRC Letter: 06-Dec-05

The NRC agreed to approve this Traveler as an administrative change to Revision 3.1 of the ISTS NUREGs. This is documented in a letter dated 12/6/05. The first plant to submit an amendment request based on this Traveler will be considered the "lead plant" submittal and a generic Safety Evaluation will be written for the Traveler.

Final Resolution: NRC Approves Final Resolution Date: 06-Dec-05

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Technical Specification Task Force
Improved Standard Technical Specifications Change Traveler

Containment Structural Integrity

NUREGs Affected: ☑ 1430 ☑ 1431 ☑ 1432 ☑ 1433 ☑ 1434

Classification: 1) Technical Change

Recommended for CLIIP?: Yes

Correction or Improvement: Correction

NRC Fee Status: Exemption Requested

Benefit: Allows Less Stringent Testing

Industry Contact: Paul Infanger, (352) 563-4796, paul.infanger@pgnmail.com

See attached justification.

Revision History

OG Revision 0 Revision Status: Closed

Revision Proposed by: Oconee

Revision Description:
Original Issue

Owners Group Review Information

Date Originated by OG: 02-Apr-98

Owners Group Comments:
ONS-28

Owners Group Resolution: Approved Date: 02-Apr-98

TSTF Review Information

TSTF Received Date: 01-May-98 Date Distributed for Review: 12-Oct-98

OG Review Completed: ☑ BWOG ☑ WOG ☑ CEOG ☑ BWROG

TSTF Comments:

4/28/99 - Revise to bracket the Surveillance and the Program with a Reviewer's Note stating that the SR and the Program may be deleted by plants that have adopted ASME Section XI, Subsections IWE and IWC. Expand the Containment Bases to discuss structural integrity and reference 10 CFR 50.55a.

TSTF Resolution: Superceeded Date: 20-Nov-98

OG Revision 1 Revision Status: Closed

Revision Proposed by: TSTF

Revision Description:
Complete replacement of Revision 0. Revised to bracket the Surveillance and the Program with a Reviewer's Note stating that the SR and the Program may be deleted by plants that have adopted ASME Section XI, Subsections IWE and IWC. Expand the Containment Bases to discuss structural integrity and reference 10 CFR 50.55a.

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OG Revision 1 Revision Status: Closed

TSTF Review Information

TSTF Received Date: 02-Jun-99 Date Distributed for Review: 08-Jun-99

OG Review Completed: ☑ BWOG ☑ WOG ☑ CEOG ☑ BWROG

TSTF Comments:

Change "IWC" to "IWL" in justification and insert.

TSTF Resolution: Approved Date: 07-Jul-99

NRC Review Information

NRC Received Date: 20-Jul-99

NRC Comments:

2/11/00 - NRC provided comments. TSTF to respond.

1/13/00 - Noel spoke to NRC. NRC agrees that the requirements are in the Code, but also wants them in Tech Spec. Further discussion is required.

5/11/2001 - BWOG chairman to discuss with NRC to identify concerns by 5/31/2001.

Final Resolution: Reviewer Recommends Changes

TSTF Revision 1 Revision Status: Active

Revision Proposed by: Wolf Creek

Revision Description:
Complete replacement of Revision 1. Revises ISTS consistent with many approved plant-specific amendments. Modeled closely on the Wolf Creek approved amendment.

TSTF Review Information

TSTF Received Date: 05-May-05 Date Distributed for Review: 05-May-05

OG Review Completed: ☑ BWOG ☑ WOG ☑ CEOG ☑ BWROG

TSTF Comments:

(No Comments)

TSTF Resolution: Approved Date: 20-Jun-05

Affected Technical Specifications

5.5.6	Pre-Stressed Concrete Containment Tendon Surveillance Program	
Ref. 3.6.1 Bases	Containment	NUREG(s)- 1430 1431 1432 Only
SR 3.6.1.1 Bases	Containment	NUREG(s)- 1430 1431 1432 Only
SR 3.6.1.2 Bases	Containment	NUREG(s)- 1430 1431 1432 Only

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5.5.16	Containment Leakage Rate Testing Program	NUREG(s)- 1430 1431 1432 Only
SR 3.6.1.1.1 Bases	Primary Containment	NUREG(s)- 1433 1434 Only
5.5.13	Primary Containment Leakage Rate Testing Program	NUREG(s)- 1433 1434 Only
Ref. 3.6.1.1 Bases	Primary Containment	NUREG(s)- 1434 Only
SR 3.6.1.1.2 Bases	Primary Containment	NUREG(s)- 1434 Only

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1.0 Description

The proposed Traveler revises Improved Standard Technical Specification (ISTS) "Pre-Stressed Containment Tendon Surveillance Program," the NUREG-1430, NUREG-1431, and NUREG-1432 "Containment Leakage Rate Testing Program," and the NUREG-1433 and NUREG-1434 "Primary Containment Leakage Rate Testing Program," for consistency with the requirements of 10 CFR 50.55a(g)(4) for components classified as Code Class CC. This regulation requires licensees to update their containment inservice inspection requirements in accordance with Subsections IWE and IWL of Section XI, Division I of the ASME Boiler and Pressure Vessel Code as limited by 10 CFR 50.55a(b)(2)(vi) and modified by 10 CFR 50.55a(b)(2)(viii) and 10 CFR50.55a(b)(2)(ix).

The pressurized water reactor (PWR) ISTS NUREGs (NUREG-1430, NUREG-1431, and NUREG-1432) use the term "containment." The boiling water reactor (BWR) ISTS NUREGs (NUREG-1433 and NUREG-1434) use the term "primary containment." For simplicity, the term "containment" is used throughout this document (except for quoted titles) to refer to both PWR and BWR containment structures.

As a result of this Traveler, licensees will be required to perform one less visual inspection of the containment during the ten year interval. However, the requirements for inspection in Subsection IWE and IWL of Section XI are more rigorous than those currently required to be performed.

The proposed changes have been approved by the NRC for plant-specific Technical Specifications several times. See References 3, 4, 5, 6, 7, and 8. This proposed Traveler is nearly identical to the approved amendment for Wolf Creek (Reference 8). The NRC issued changes to 10 CFR 50.55a require that all plants revise their Technical Specifications. Approval of this Traveler and issuance of a Consolidated Line Item Improvement Program (CLIIP) Notice of Availability will save the NRC and the licensees resources in processing this required change to licensee’s technical specifications while improving quality and consistency.

2.0 Proposed Change

The proposed change will revise:

Technical Specification 5.5.6, "Pre-Stressed Concrete Containment Tendon Surveillance Program," in all ISTS NUREGs

These specifications are revised to indicate that the Containment Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with Section XI, Subsection IWL of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50.55a, except where an alternative, exemption, or relief has been authorized by the NRC. Additionally, the provisions of Surveillance Requirement (SR) 3.0.2 are deleted from these specifications.

Technical Specification 5.5.16, "Containment Leakage Rate Testing Program" in the PWR ISTS NUREGs and Technical Specification 5.5.13, "Primary Containment Leakage Rate Testing Program" in the BWR ISTS NUREGs.

These specifications are revised to add the following exceptions to Regulatory Guide 1.163, "Performance- Based Containment Leak-Testing Program,"

"1. The visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by ASME Section XI Code, Subsection IWL, except where relief has been authorized by the NRC.

2. The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by ASME Section XI Code, Subsection IWE, except where relief has been authorized by the NRC."

The TS Bases for SR 3.6.1.1 and SR 3.6.1.2 in the PWR ISTS NUREGs and the BWR ISTS NUREGs TS Bases for SR 3.6.1.1.1 and the NUREG-1434 TS Bases for SR 3.6.1.1.2 are revised for consistency with the requirements of the ASME Code Section XI, Subsection IWL, and applicable addenda as required by 10 CFR 50.55a.

The SR 3.6.1.1 Bases in NUREG-1431 contain a paragraph describing the basis for the Surveillance Frequency. The PWR ISTS NUREG Bases for SR 3.6.1.1, and the BWR ISTS NUREG Bases for SR 3.6.1.1.1 do not contain a description of the basis for the Surveillance Frequency. Such a description is required by the ISTS Writer's Guide. The NUREG-1431 Bases paragraph is added to the other NUREG ISTS Bases for consistency.

3.0 Background

On January 7, 1994, the Nuclear Regulatory Commission (NRC) published a proposed amendment to the regulations to incorporate by reference the 1992 Edition with the 1992 Addenda of Subsections IWE and IWL of Section XI, Division I of the ASME Boiler and Pressure Vessel Code (the Code). The final rule, Subpart 50.55a(g)(6)(ii)(B) of Title 10 of the Code of Federal Regulations (10 CFR), became effective on September 9, 1996, and requires licensees to implement Subsections IWE and IWL, with specified modifications and limitations, by September 9, 2001.

The containment consists of a prestressed, reinforced concrete, cylindrical structure with a hemispherical dome. The post-tensioning System used for the shell and dome of the containment employs tendons. Each tendon consists of high strength steel wires and anchoring components. The prestressing load is transferred, by cold formed button heads on the ends of the individual wires through stressing washers, to steel bearing plates

embedded in the structure. The unbonded tendons are installed in tendon ducts and tensioned in a predetermined sequence.

4.0 Technical Analysis

Technical Specification 5.5.6, "Pre-Stressed Concrete Containment Tendon Surveillance Program," states in part, "The Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with [Regulatory Guide 1.35, Revision 3, 1990]." As identified above, 10 CFR 50.55a(g)(4) requires licensees to update their containment inservice inspection requirements in accordance with Subsections IWE and IWL of Section XI, Division I of the ASME Boiler and Pressure Vessel Code as limited by 10 CFR 50.55a(b)(2)(vi) and modified by 10 CFR 50.55a(b)(2)(viii) and 10 CFR 50.55a(b)(2)(ix). The requirements in 10 CFR 50.55a(g)(4) and ASME Code Section XI, Subsection IWL, do not reference Regulatory Guide 1.35, Revision 3. As such, the ISTS are inconsistent with the requirements of 10 CFR 50.55a.

10 CFR 50.55a(g)(5)(ii) states, in part: "If a revised inservice inspection program for a facility conflicts with the technical specification for the facility, the licensee shall apply to the Commission for Amendment of the technical specifications to conform the technical specification to the revised program." Based on the requirements in 10 CFR 50.55a, licensees will be required to update their plant-specific technical specifications. Licensee's containment inservice inspection programs are required to be in accordance with ASME Code Section XI, Subsection IWL, as modified by 10 CFR 50.55a(b)(2)(viii), except where an exemption or relief has been authorized by the NRC.

Additionally, since the tendon inspection frequencies will be in accordance with ASME Section XI, Subsection IWL, the provisions of SR 3.0.2 are no longer applicable and are deleted from Technical Specification 5.5.6. As discussed in the Technical Specification Bases for SR 3.0.2, the requirements of regulations take precedence over the Technical Specifications. As such, 10 CFR 50.55a requires the implementation of ASME Section XI, Subsection IWL and specifies the requirements for extending inspection frequencies.

The Technical Specification requirements for the [Primary] Containment Leakage Rate Testing Program specify that the program shall be in accordance with the guidelines contained in Regulatory Guide 1.163. Regulatory Position C.3 of the regulatory guide states that "Section 9.2.1, 'Pretest Inspection and Test Methodology,' of NEI 94-01 provides guidance for the visual examination of accessible interior and exterior surfaces of the containment system for structural problems. These examinations should be conducted prior to initiating a Type A test, and during two other refueling outages before the next Type A test if the interval for the Type A test has been extended to 10 years, in order to allow for early uncovering of evidence of structural deterioration." There are no specific requirements in NEI 94-01 for the visual examination except that it is to be a general visual examination of accessible interior and exterior surfaces of the primary containment components.

In addition to the requirements of Regulatory Guide 1.163 and NEI 94-01, the concrete surfaces of the containment must be visually examined in accordance with the ASME Section XI Code, Subsection IWL, and the liner plate inside containment must be visually examined in accordance with Subsection IWE. The frequency of visual examination of the concrete surfaces per Subsection IWL is once every five years, and the frequency of visual examination of the liner plate per Subsection IWE is, in general, three visual examinations over a 10-year period. The visual examinations performed pursuant to Subsection IWL may be performed at any time during power operation or during shutdown, and the visual examinations performed pursuant to Subsection IWE are performed during refueling outages since this in the only time that the liner plate is fully accessible.

The visual examinations performed pursuant to Subsections IWL and IWE are more rigorous than those performed pursuant to Regulatory Guide 1.163 and NEI 94-01. For example, Subarticle IWE-2320 requires the general visual examination to be the responsibility of an individual who is knowledgeable in the requirements for design, inservice inspection, and testing of Class MC and metallic liners of Class CC components. Subsection IWE, Subarticle-2330 requires the examination to be performed either directly or remotely, by an examiner with visual acuity sufficient to detect evidence of degradation.

Similarly, Subarticle IWL-2320 states that:

> "The Responsible Engineer shall be a Registered Professional Engineer experienced in evaluating the inservice condition of structural concrete. The Responsible Engineer shall have knowledge of the design and Construction Codes and other criteria use in design and construction of concrete containments in nuclear power plants.
>
> The Responsible Engineer shall be responsible for the following:
> (a) development of plans and procedures for examination of concrete surfaces;
> (b) approval, instruction, and training of concrete examination personnel
> (c) evaluation of examination results;
> (d) preparation or review of Repair/Replacement Plans and procedures;
> (e) review of procedures for pressure tests following repair/replacement procedures;
> (f) submittal of report to the Owner documenting results of examinations and repairs."

Based on the above, the Responsible Engineer will ensure that a comprehensive visual examination of the concrete is performed in accordance with Code requirements except where relief has been granted by the NRC. Furthermore, with respect to examinations performed pursuant to both Subsections IWL and IWE, visual examinations of both the concrete surfaces and the liner plate must be reviewed by an Inspector employed by a State or municipality of the United States or an Inspector regularly employed by an insurance company authorized to write boiler and pressure vessel insurance, in

accordance with IWA-2110 and IWA-2120. The combination of the Code requirements for the rigor of the visual examinations plus the third party review will more than offset the fact that one fewer visual examination of the concrete will be performed during a 10-year interval. The fact that the concrete visual examination pursuant to Subsection IWL may be performed during power operation as opposed to during a refueling outage will have no effect on the quality of the examination and will provide flexibility in scheduling of the visual examinations.

5.0 Regulatory Analysis

5.1 No Significant Hazards Consideration

The TSTF has evaluated whether or not a significant hazards consideration is involved with the proposed generic change by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

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

Response: No

The proposed change revises the TS administrative controls programs for consistency with the requirements of 10 CFR 50, paragraph 55a(g)(4) for components classified as Code Class CC.

The revised requirements do not affect the function of the containment post-tensioning system components. The post-tensioning systems are passive components whose failure modes could not act as accident initiators or precursors.

The proposed change affects the frequency of visual examinations that will be performed for the concrete surfaces of the containment for the purpose of the [Primary] Containment Leakage Rate Testing Program. In addition, the proposed change allows those examinations to be performed during power operation as opposed to during a refueling outage. The frequency of visual examinations of the concrete surfaces of the containment and the mode of operation during which those examinations are performed has no relationship to or adverse impact on the probability of any of the initiating events assumed in the accident analyses. The proposed change would allow visual examinations that are performed pursuant to NRC approved ASME Section XI Code requirements (except where relief has been granted by the NRC) to meet the intent of visual examinations required by Regulatory Guide 1.163, without requiring additional visual examinations pursuant to the Regulatory Guide. The intent of early detection of deterioration will continue to be met by the more rigorous requirements of the Code required visual examinations. As such, the safety function of the containment as a fission product barrier is maintained.

The proposed change does not impact any accident initiators or analyzed events or assumed mitigation of accident or transient events. It does not involve the addition or removal of any equipment, or any design changes to the facility.

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 accident previously evaluated?

Response: No

The proposed change revises the TS administrative controls programs for consistency with the requirements of 10 CFR 50, paragraph 55a(g)(4) for components classified as Code Class CC.

The function of the containment post-tensioning system components are not altered by this change. The change affects the frequency of visual examinations that will be performed for the concrete surfaces containments. In addition, the proposed change allows those examinations to be performed during power operation as opposed to during a refueling outage. The proposed change does not involve a modification to the physical configuration of the plant (i.e., no new equipment will be installed) or change in the methods governing normal plant operation. The proposed change will not impose any new or different requirements or introduce a new accident initiator, accident precursor, or malfunction mechanism. Additionally, there is no change in the types or increases in the amounts of any effluent that may be released off-site and there is no increase in individual or cumulative occupational exposure.

Therefore, the proposed change does 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 a margin of safety?

Response: No

The proposed change revises the Improved Standard Technical Specification Administrative Controls program requirements for consistency with the requirements of 10 CFR 50, paragraph 55a(g)(4) for components classified as Code Class CC.

The function of the containment post-tensioning system components are not altered by this change. The change affects the frequency of visual examinations that will be performed for the concrete surfaces containments. In addition, the proposed change allows those examinations to be performed during power operation as opposed to during a refueling outage. The safety function of the containment as a fission product barrier will be maintained.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

5.2 Applicable Regulatory Requirements/Criteria

The regulatory basis for PWR ISTS 3.6.1, "Containment," and BWR ISTS 3.6.1.1, "Primary Containment," is to ensure that the containment is capable of remaining leak-tight following a loss of coolant accident. This ensures that offsite radiation exposures are maintained within the limits of 10 CFR 100.

10 CFR 50, Appendix A, General Design Criterion 16, "Design," requires that reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as the postulated accident conditions require.

This Technical Specification change will not reduce the leak-tightness of the containment. Therefore, based on the considerations discussed above:

1) There is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner;

2) Such activities will be conducted in compliance with the Commission's regulations; and

3) Issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 Environmental Consideration

A review has determined that the proposed change would change a requirement with respect to installation or use of a facility component located within the restricted areas, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed change does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed change meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed change.

7.0 References

1. 10 CFR 50.55a

2. Regulatory Guide 1.163, "Performance-Based Containment Leak-Testing Program."

3. Letter dated January 18, 2000, to W. R. McCollum, Jr., Duke Energy Corporation, "Oconee Nuclear Station Units 1, 2, and 3 RE: Issuance of Amendments (TAC Nos. MA6568, MA6569, and MA6570)." Amendment Nos. 310

4. Letter dated June 6, 2001, to J. B. Beasley, Jr., Southern Nuclear Operating Company, Inc, "Vogtle Electric Generating Plant, Units 1 and 2 RE: Issuance of Amendments (TAC Nos. MB1097 and MB1098)." Amendment Nos. 122 and 100.

5. Letter dated January 30, 2001, to C. H. Cruse, Constellation Nuclear, "Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 RE: Containment Tendon Surveillance Program – Amendment (TAC Nos. MB0011 and MB0012)." Amendment Nos. 240 and 214.

6. Letter dated January 31, 2001, to T. F. Plunkett, Florida Power and Light Company, "Turkey Point Units 3 and 4 – Issuance of Amendments Regarding Changes to Containment Structural Integrity Technical Specifications (TAC Nos. MA9047 and MA9048)." Amendment Nos. 210 and 204.

7. Letter to R. R. Overbeck, Arizona Public Service Company, "Palo Verde Nuclear Generating Station, Units 1, 2, and 3 – Issuance of Amendment on Containment Tendon Surveillance Program and Containment Leakage Rate Testing Program (TAC Nos. MC1069, MC1070, and MC1071)." Amendment Nos. 151.

8. Letter dated March 17, 2004, to R. A. Muench, Wolf Creek Nuclear Operating Corporation, "Wolf Creek Generating Station – Issuance of Amendment Re: Containment Tendon Surveillance Program and Containment Leakage Rate Testing Program." Amendment No. 152.

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

j. Limitations on the annual dose or dose commitment to any member of the public, beyond the site boundary, due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluent Controls Program surveillance frequency.

5.5.5 Component Cyclic or Transient Limit

This program provides controls to track the FSAR, Section [ ], cyclic and transient occurrences to ensure that components are maintained within the design limits.

5.5.6 [Pre-Stressed Concrete Containment Tendon Surveillance Program

This program provides controls for monitoring any tendon degradation in pre-stressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial operations. The Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with ~~[Regulatory Guide 1.35, Revision 3, 1990]~~ Section XI, Subsection IWL of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50.55a, except where an alternative, exemption, or relief has been authorized by the NRC.

The provisions of ~~SR 3.0.2 and~~ SR 3.0.3 are applicable to the Tendon Surveillance Program inspection frequencies.]

5.5.7 Reactor Coolant Pump Flywheel Inspection Program

This program shall provide for the inspection of each reactor coolant pump flywheel per the recommendation of Regulatory position c.4.b of Regulatory Guide 1.14, Revision 1, August 1975.

5.5.8 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components. The program shall include the following:

a. Testing frequencies specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as follows:

5.5 Programs and Manuals

5.5.16 Containment Leakage Rate Testing Program (continued)

a) Overall air lock leakage rate is ≤ [0.05 L_a] when tested at ≥ P_a.

b) For each door, leakage rate is ≤ [0.01 L_a] when pressurized to [≥ 10 psig].

d. The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

e. Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J.

[OPTION B]

a. A program shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September, 1995 [,as modified by the following exceptions:

1. The visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWL, except where relief has been authorized by the NRC.

2. The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI code, Subsection IWE, except where relief has been authorized by the NRC.

[3. . . .]

b. The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a, is [45 psig]. The containment design pressure is [50 psig].

c. The maximum allowable containment leakage rate, L_a, at P_a, shall be [ ]% of containment air weight per day.

d. Leakage rate acceptance criteria are:

1. Containment leakage rate acceptance criterion is ≤ 1.0 L_a. During the first unit startup following testing in accordance with this program, the

5.5 Programs and Manuals

5.5.16 Containment Leakage Rate Testing Program (continued)

e. The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

f. Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J.

[OPTION A/B Combined]

a. A program shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J. [Type A][Type B and C] test requirements are in accordance with 10 CFR 50, Appendix J, Option A, as modified by approved exemptions. [Type B and C][Type A] test requirements are in accordance with 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. The 10 CFR 50, Appendix J, Option B test requirements shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September, 1995-[,as modified by the following exceptions:

1. The visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWL, except where relief has been authorized by the NRC.

2. The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI code, Subsection IWE, except where relief has been authorized by the NRC.

[3. . . .]

b. The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a, is [45 psig]. The containment design pressure is [50 psig].

c. The maximum allowable containment leakage rate, L_a, at P_a, shall be [ ]% of containment air weight per day.

d. Leakage rate acceptance criteria are:

1. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60\ L_a$ for the Type B and C

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1

Maintaining the containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements of the Containment Leakage Rate Testing Program. The containment concrete visual examinations may be performed during either power operation, e.g., performed concurrently with other containment inspection-related activities such as tendon testing, or during a maintenance or refueling outage. The visual examinations of the steel liner plate inside containment are performed during maintenance or refueling outages since this is the only time the liner plate is fully accessible.

Failure to meet air lock and purge valve with resilient seal leakage limits specified in LCO 3.6.2 and LCO 3.6.3 does not invalidate the acceptability of these overall leakage determinations unless their contribution to overall Type A, B, and C leakage causes that to exceed limits. As left leakage prior to the first startup after performing a required Containment Leakage Rate Testing Program leakage test is required to be < 0.6 L_a for combined Type B and C leakage, and [< 0.75 L_a for Option A] [≤ 0.75 L_a for Option B] for overall Type A leakage. At all other times between required leakage rate tests, the acceptance criteria is based on an overall Type A leakage limit of ≤ 1.0 L_a. At ≤ 1.0 L_a the offsite dose consequences are bounded by the assumptions of the safety analysis. SR Frequencies are as required by the Containment Leakage Rate Testing Program. These periodic testing requirements verify that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis.

SR Frequencies are as required by the Containment Leakage Rate Testing Program. These periodic testing requirements verify that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis.

----------REVIEWER'S NOTE----------
Regulatory Guide 1.163 and NEI 94-01 include acceptance criteria for as-left and as-found Type A leakage rates and combined Type B and C leakage rates, which may be reflected in the Bases.

[SR 3.6.1.2

For ungrouted, post tensioned tendons, this SR ensures that the structural integrity of the containment will be maintained in accordance with the provisions of the Containment Tendon Surveillance Program. Testing and Frequency are ~~consistent with the recommendations of Regulatory Guide 1.35~~ in accordance with the ASME code, Section XI, Subsection IWL (Ref. 4), and applicable addenda as required by 10 CFR 50.55a.]

REFERENCES

1. 10 CFR 50, Appendix J, Option [A][B].
2. FSAR, Sections [14.1 and 14.2].
3. FSAR, Section [5.6].
4. ~~Regulatory Guide 1.35, Revision [1]~~ ASME Code, Section XI, Subsection IWL.

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I, and

j. Limitations on the annual dose or dose commitment to any member of the public, beyond the site boundary, due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluent Controls Program surveillance frequency.

5.5.5 Component Cyclic or Transient Limit

This program provides controls to track the FSAR, Section [ ], cyclic and transient occurrences to ensure that components are maintained within the design limits.

5.5.6 [Pre-Stressed Concrete Containment Tendon Surveillance Program

This program provides controls for monitoring any tendon degradation in pre-stressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial operations. The Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with ~~[Regulatory Guide 1.35, Revision 3, 1990]~~ Section XI, Subsection IWL of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50.55a, except where an alternative, exemption, or relief has been authorized by the NRC.

The provisions of ~~SR 3.0.2 and~~ SR 3.0.3 are applicable to the Tendon Surveillance Program inspection frequencies.]

5.5.7 Reactor Coolant Pump Flywheel Inspection Program

This program shall provide for the inspection of each reactor coolant pump flywheel per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975.

In lieu of Position C.4.b(1) and C.4.b(2), a qualified in-place UT examination over the volume from the inner bore of the flywheel to the circle one-half of the outer radius or a surface examination (MT and/or PT) of exposed surfaces of the removed flywheels may be conducted at approximately 10 year intervals coinciding with the Inservice Inspection schedule as required by ASME Section XI.

5.5 Programs and Manuals

5.5.16 Containment Leakage Rate Testing Program (continued)

a) Overall air lock leakage rate is ≤[0.05 L_a] when tested at ≥ P_a.

b) For each door, leakage rate is ≤[0.01 L_a] when pressurized to [≥ 10 psig].

d. The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

e. Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J.

[OPTION B]

a. A program shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September, 1995~~[~~, as modified by the following exceptions:

1. The visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWL, except where relief has been authorized by the NRC.

2. The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI code, Subsection IWE, except where relief has been authorized by the NRC.

[3. . . .]

b. The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a, is [45 psig]. The containment design pressure is [50 psig].

c. The maximum allowable containment leakage rate, L_a, at P_a, shall be [ ]% of containment air weight per day.

d. Leakage rate acceptance criteria are:

1. Containment leakage rate acceptance criterion is 1.0 L_a. During the first unit startup following testing in accordance with this program, the

5.5 Programs and Manuals

5.5.16 Containment Leakage Rate Testing Program (continued)

e. The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

f. Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J.

[OPTION A/B Combined]

a. A program shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J. [Type A][Type B and C] test requirements are in accordance with 10 CFR 50, Appendix J, Option A, as modified by approved exemptions. [Type B and C][Type A] test requirements are in accordance with 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. The 10 CFR 50, Appendix J, Option B test requirements shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September, 1995~~[~~,as modified by the following exceptions:

1. The visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWL, except where relief has been authorized by the NRC.

2. The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI code, Subsection IWE, except where relief has been authorized by the NRC.

[3. . . .]

b. The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a, [45 psig]. The containment design pressure is [50 psig].

c. The maximum allowable containment leakage rate, L_a, at P_a, shall be [ ]% of containment air weight per day.

d. Leakage rate acceptance criteria are:

1. Containment leakage rate acceptance criterion is ≤1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are < 0.60 L_a for the Type B and C tests and [< 0.75 L_a for Option A Type A tests][≤ 0.75 L_a for Option B Type A tests].

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1

Maintaining the containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements of the Containment Leakage Rate Testing Program. The containment concrete visual examinations may be performed during either power operation, e.g., performed concurrently with other containment inspection-related activities such as tendon testing, or during a maintenance or refueling outage. The visual examinations of the steel liner plate inside containment are performed during maintenance or refueling outages since this is the only time the liner plate is fully accessible.

Failure to meet air lock [and purge valve with resilient seal] leakage limits specified in LCO 3.6.2 [and LCO 3.6.3] does not invalidate the acceptability of these overall leakage determinations unless their contribution to overall Type A, B, and C leakage causes that to exceed limits. As left leakage prior to the first startup after performing a required Containment Leakage Rate Testing Program leakage test is required to be < 0.6 L_a for combined Type B and C leakage, and [< 0.75 L_a for Option A] [0.75 L_a for Option B] for overall Type A leakage. At all other times between required leakage rate tests, the acceptance criteria is based on an overall Type A leakage limit of 1.0 L_a. At 1.0 L_a the offsite dose consequences are bounded by the assumptions of the safety analysis.

SR Frequencies are as required by the Containment Leakage Rate Testing Program. These periodic testing requirements verify that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis.

----------REVIEWER'S NOTE----------
Regulatory Guide 1.163 and NEI 94-01 include acceptance criteria for as-left and as-found Type A leakage rates and combined Type B and C leakage rates, which may be reflected in the Bases.

[SR 3.6.1.2

For ungrouted, post tensioned tendons, this SR ensures that the structural integrity of the containment will be maintained in accordance with the provisions of the Containment Tendon Surveillance Program. Testing and Frequency are ~~consistent with the recommendations of Regulatory Guide 1.35~~ in accordance with the ASME code, Section XI, Subsection IWL (Ref. 4), and applicable addenda as required by 10 CFR 50.55a.]

REFERENCES

1. 10 CFR 50, Appendix J, Option [A][B].

2. FSAR, Chapter [15].

3. FSAR, Section [6.2].

4. ~~Regulatory Guide 1.35, Revision [1]~~ ASME Code, Section XI, Subsection IWL.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1

Maintaining the containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements of the Containment Leakage Rate Testing Program. The containment concrete visual examinations may be performed during either power operation, e.g., performed concurrently with other containment inspection-related activities such as tendon testing, or during a maintenance or refueling outage. The visual examinations of the steel liner plate inside containment are performed during maintenance or refueling outages since this is the only time the liner plate is fully accessible.

Failure to meet air lock [, secondary containment bypass leakage path and purge valve with resilient seal] leakage limits specified in LCO 3.6.2 [and LCO 3.6.3] does not invalidate the acceptability of these overall leakage determinations unless their contribution to overall Type A, B, and C leakage causes that to exceed limits. As left leakage prior to the first startup after performing a required Containment Leakage Rate Testing Program leakage test is required to be < 0.6 L_a for combined Type B and C leakage, and [< 0.75 L_a for Option A][≤ 0.75 L_a for Option B] for overall Type A leakage. At all other times between required Containment Leakage Rate Testing Program leakage rate tests, the acceptance criteria is based on an overall Type A leakage limit of ≤ 1.0 L_a. At ≤ 1.0 L_a the offsite dose consequences are bounded by the assumptions of the safety analysis.

SR Frequencies are as required by the Containment Leakage Rate Testing Program. These periodic testing requirements verify that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis.

----------REVIEWER'S NOTE----------
Regulatory Guide 1.163 and NEI 94-01 include acceptance criteria for as-left and as-found Type A leakage rates and combined Type B and C leakage rates, which may be reflected in the Bases.

[SR 3.6.1.2

For ungrouted, post tensioned tendons, this SR ensures that the structural integrity of the containment will be maintained in accordance with the provisions of the Containment Tendon Surveillance Program. Testing and Frequency are ~~consistent with the recommendations of Regulatory Guide 1.35~~ in accordance with the ASME code, Section XI, Subsection IWL (Ref. 4), and applicable addenda as required by 10 CFR 50.55a.]

REFERENCES

1. 10 CFR 50, Appendix J, Option [A][B].

2. FSAR, Chapter [15].

3. FSAR, Section [6.2].

4. ~~Regulatory Guide 1.35, Revision [1]~~ ASME Code, Section XI, Subsection IWL.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1

Maintaining the containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements of the Containment Leakage Rate Testing Program. The containment concrete visual examinations may be performed during either power operation, e.g., performed concurrently with other containment inspection-related activities such as tendon testing, or during a maintenance or refueling outage. The visual examinations of the steel liner plate inside containment are performed during maintenance or refueling outages since this is the only time the liner plate is fully accessible.

Failure to meet air lock [, secondary containment bypass leakage path, and purge valve with resilient seal] leakage limits specified in LCO 3.6.2 [and LCO 3.6.3] does not invalidate the acceptability of these overall leakage determinations unless their contribution to overall Type A, B, and C leakage causes that to exceed limits. As left leakage prior to the first startup after performing a required Containment Leakage Rate Testing Program leakage test is required to be < 0.6 L_a for combined Type B and C leakage, and [<0.75 L_a for Option A] [≤ 0.75 L_a for Option B] for overall Type A leakage. At all other times between required leakage rate tests, the acceptance criteria is based on an overall Type A leakage limit of ≤ 1.0 L_a. At ≤ 1.0 L_a the offsite dose consequences are bounded by the assumptions of the safety analysis.

SR Frequencies are as required by the Containment Leakage Rate Testing Program. These periodic testing requirements verify that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis.

----------REVIEWER'S NOTE----------
Regulatory Guide 1.163 and NEI 94-01 include acceptance criteria for as-left and as-found Type A leakage rates and combined Type B and C leakage rates, which may be reflected in the Bases.

[SR 3.6.1.2

For ungrouted, post tensioned tendons, this SR ensures that the structural integrity of the containment will be maintained in accordance with the provisions of the Containment Tendon Surveillance Program. Testing and Frequency are ~~consistent with the recommendations of Regulatory Guide 1.35~~ in accordance with the ASME code, Section XI, Subsection IWL (Ref. 4), and applicable addenda as required by 10 CFR 50.55a.]

REFERENCES

1. 10 CFR 50, Appendix J, Option [A][B].

2. FSAR, Chapter [15].

3. FSAR, Section [6.2].

4. ~~Regulatory Guide 1.35, Revision [1]~~ ASME Code, Section XI, Subsection IWL.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1

Maintaining the containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements of the Containment Leakage Rate Testing Program. The containment concrete visual examinations may be performed during either power operation, e.g., performed concurrently with other containment inspection-related activities such as tendon testing, or during a maintenance or refueling outage. The visual examinations of the steel liner plate inside containment are performed during maintenance or refueling outages since this is the only time the liner plate is fully accessible.

Failure to meet air lock [and purge valve with resilient seal] leakage limits specified in LCO 3.6.2 [and LCO 3.6.3] does not invalidate the acceptability of these overall leakage determinations unless their contribution to overall Type A, B, and C leakage causes that to exceed limits. As left leakage prior to the first startup after performing a required Containment Leakage Rate Testing Program leakage test is required to be < 0.6 L_a for combined Type B and C leakage, and [< 0.75 L_a for Option A] [≤ 0.75 L_a for Option B] for overall Type A leakage. At all other times between required leakage rate tests, the acceptance criteria is based on an overall Type A leakage limit of ≤ 1.0 L_a. At ≤ 1.0 L_a the offsite dose consequences are bounded by the assumptions of the safety analysis.

SR Frequencies are as required by the Containment Leakage Rate Testing Program. These periodic testing requirements verify that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis.

----------REVIEWER'S NOTE----------
Regulatory Guide 1.163 and NEI 94-01 include acceptance criteria for as-left and as-found Type A leakage rates and combined Type B and C leakage rates, which may be reflected in the Bases.

[SR 3.6.1.2

For ungrouted post tensioned tendons, this SR ensures that the structural integrity of the containment will be maintained in accordance with the provisions of the Containment Tendon Surveillance Program. Testing and Frequency are ~~consistent with the recommendations of Regulatory Guide 1.35~~ in accordance with the ASME code, Section XI, Subsection IWL (Ref. 4), and applicable addenda as required by 10 CFR 50.55a.]

REFERENCES

1. 10 CFR 50, Appendix J, Option [A][B].

2. FSAR, Chapter [15].

3. FSAR, Section [6.2].

4. ~~Regulatory Guide 1.35, Revision [1]~~ ASME Code, Section XI, Subsection IWL.

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I, and

j. Limitations on the annual dose or dose commitment to any member of the public, beyond the site boundary, due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluent Controls Program surveillance frequency.

5.5.5 Component Cyclic or Transient Limit

This program provides controls to track the FSAR, Section [ ], cyclic and transient occurrences to ensure that components are maintained within the design limits.

5.5.6 [Pre-Stressed Concrete Containment Tendon Surveillance Program

This program provides controls for monitoring any tendon degradation in pre-stressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial operations. The Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with ~~[Regulatory Guide 1.35, Revision 3, 1990]~~ Section XI, Subsection IWL of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50.55a, except where an alternative, exemption, or relief has been authorized by the NRC.

The provisions of ~~SR 3.0.2 and~~ SR 3.0.3 are applicable to the Tendon Surveillance Program inspection frequencies.]

5.5.7 Reactor Coolant Pump Flywheel Inspection Program

This program shall provide for the inspection of each reactor coolant pump flywheel per the recommendation of Regulatory position c.4.b of Regulatory Guide 1.14, Revision 1, August 1975.

5.5.8 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components. The program shall include the following:

a. Testing frequencies specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as follows:

5.5 Programs and Manuals

5.5.16 Containment Leakage Rate Testing Program (continued)

1. Containment leakage rate acceptance criterion is ≤ 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are < 0.60 L_a for the Type B and C tests and < 0.75 L_a for Type A tests.

2. Air lock testing acceptance criteria are:

   a) Overall air lock leakage rate is ≤ [0.05 L_a] when tested at ≥ P_a.

   b) For each door, leakage rate is ≤ [0.01 L_a] when pressurized to [≥10 psig].

d. The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

e. Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J.

[OPTION B]

a. A program shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September, 1995 [,as modified by the following exceptions:

   1. The visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWL, except where relief has been authorized by the NRC.

   2. The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI code, Subsection IWE, except where relief has been authorized by the NRC.

   [3. . . .]

b. The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a is [45 psig]. The containment design pressure is [50 psig].

5.5 Programs and Manuals

5.5.16 Containment Leakage Rate Testing Program (continued)

2. Air lock testing acceptance criteria are:

a) Overall air lock leakage rate is ≤ [0.05 L_a] when tested at ≥ P_a.

b) For each door, leakage rate is ≤ [0.01 L_a] when pressurized to [≥10 psig].

e. The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

f. Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J.

[OPTION A/B Combined]

a. A program shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J. [Type A][Type B and C] test requirements are in accordance with 10 CFR 50, Appendix J, Option A, as modified by approved exemptions. [Type B and C] [Type A] test requirements are in accordance with 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. The 10 CFR 50, Appendix J, Option B test requirements shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September, 1995 [,as modified by the following exceptions:

1. The visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWL, except where relief has been authorized by the NRC.

2. The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI code, Subsection IWE, except where relief has been authorized by the NRC.

[3. . . .]

b. The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a is [45 psig]. The containment design pressure is [50 psig].

c. The maximum allowable containment leakage rate, L_a, at P_a, shall be [ ]% of containment air weight per day.

BASES

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material into containment. 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, containment is not required to be OPERABLE in MODE 5 to prevent leakage of radioactive material from containment. The requirements for containment during MODE 6 are addressed in LCO 3.9.3, "Containment Penetrations."

ACTIONS

A.1

In the event containment is inoperable, containment must be restored to OPERABLE status within 1 hour. The 1 hour Completion Time provides a period of time to correct the problem commensurate with the importance of maintaining containment during MODES 1, 2, 3, and 4. This time period also ensures that the probability of an accident (requiring containment OPERABILITY) occurring during periods when containment is inoperable is minimal.

B.1 and B.2

If containment 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 3.6.1.1

Maintaining the containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements of the Containment Leakage Rate Testing Program.

The containment concrete visual examinations may be performed during either power operation, e.g., performed concurrently with other containment inspection-related activities such as tendon testing, or during a maintenance or refueling outage. The visual examinations of the steel liner plate inside containment are performed during maintenance or refueling outages since this is the only time the liner plate is fully accessible.

Failure to meet air lock and purge valve with resilient seal leakage limits specified in LCO 3.6.2 and LCO 3.6.3 does not invalidate the acceptability of these overall leakage determinations unless their contribution to overall Type A, B, and C leakage causes that to exceed limits. As left leakage

BASES

SURVEILLANCE REQUIREMENTS (continued)

Option B] for overall Type A leakage. At all other times between required leakage rate tests, the acceptance criteria is based on an overall Type A leakage limit of ≤ 1.0 L_a. At ≤ 1.0 L_a the offsite dose consequences are bounded by the assumptions of the safety analysis. SR Frequencies are as required by the Containment Leakage Rate Testing Program. These periodic testing requirements verify that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis.

SR Frequencies are as required by the Containment Leakage Rate Testing Program. These periodic testing requirements verify that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis.

----------REVIEWER'S NOTE----------
Regulatory Guide 1.163 and NEI 94-01 include acceptance criteria for as-left and as-found Type A leakage rates and combined Type B and C leakage rates, which may be reflected in the Bases.

[SR 3.6.1.2

For ungrouted, post tensioned tendons, this SR ensures that the structural integrity of the containment will be maintained in accordance with the provisions of the Containment Tendon Surveillance Program. Testing and Frequency are ~~consistent with the recommendations of Regulatory Guide 1.35~~ in accordance with the ASME code, Section XI, Subsection IWL (Ref. 4), and applicable addenda as required by 10 CFR 50.55a.]

REFERENCES

1. 10 CFR 50, Appendix J, Option [A][B].

2. FSAR, Section [ ].

3. FSAR, Section [ ].

4. ~~Regulatory Guide 1.35, Revision [1]~~ ASME Code, Section XI, Subsection IWL.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1

Maintaining containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements of the Containment Leakage Rate Testing Program. The containment concrete visual examinations may be performed during either power operation, e.g., performed concurrently with other containment inspection-related activities such as tendon testing, or during a maintenance or refueling outage. The visual examinations of the steel liner plate inside containment are performed during maintenance or refueling outages since this is the only time the liner plate is fully accessible.

Failure to meet air lock and purge valve with resilient seal specific leakage limits specified in LCO 3.6.2 and LCO 3.6.3 does not invalidate the acceptability of these overall leakage determinations unless their contribution to overall Type A, B, and C leakage causes that to exceed limits. As left leakage prior to the first startup after performing a required Containment Leakage Rate Testing Program leakage test is required to be < 0.6 L_a for combined Type B and C leakage, and [< 0.75 L_a for Option A] [≤ 0.75 L_a for Option B] for overall Type A leakage. At all other times between required leakage rate tests, the acceptance criteria is based on an overall Type A leakage limit of ≤ 1.0 L_a. At ≤ 1.0 L_a the offsite dose consequences are bounded by the assumptions of the safety analysis. SR Frequencies are as required by the Containment Leakage Rate Testing Program. These periodic testing requirements verify that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis.

SR Frequencies are as required by the Containment Leakage Rate Testing Program. These periodic testing requirements verify that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis.

----------REVIEWER'S NOTE----------
Regulatory Guide 1.163 and NEI 94-01 include acceptance criteria for as-left and as-found Type A leakage rates and combined Type B and C leakage rates, which may be reflected in the Bases.

[SR 3.6.1.2

For ungrouted, post tensioned tendons, this SR ensures that the structural integrity of the containment will be maintained in accordance with the provisions of the Containment Tendon Surveillance Program. Testing and Frequency are ~~consistent with the recommendations of Regulatory Guide 1.35~~ in accordance with the ASME code, Section XI,

	Subsection IWL (Ref. 4), and applicable addenda as required by 10 CFR 50.55a.]
REFERENCES	1. 10 CFR 50, Appendix J, Option [A][B]. 2. FSAR, Section [ ]. 3. FSAR, Section [ ]. 4. ~~Regulatory Guide 1.35, Revision [1]~~ ASME Code, Section XI, Subsection IWL.

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

j. Limitations on the annual dose or dose commitment to any member of the public, beyond the site boundary, due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190, and

k. Limitations on venting and purging of the Mark II containment through the Standby Gas Treatment System to maintain releases as low as reasonably achievable (in BWR/4s with Mark II containments).

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluent Controls Program surveillance frequency.

5.5.5 Component Cyclic or Transient Limit

This program provides controls to track the FSAR Section [ ], cyclic and transient occurrences to ensure that components are maintained within the design limits.

5.5.6 [Pre-Stressed Concrete Containment Tendon Surveillance Program

This program provides controls for monitoring any tendon degradation in pre-stressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial operations. The Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with ~~[Regulatory Guide 1.35, Revision 3, 1990]~~ Section XI, Subsection IWL of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50.55a, except where an alternative, exemption, or relief has been authorized by the NRC.

The provisions of ~~SR 3.0.2 and~~ SR 3.0.3 are applicable to the Tendon Surveillance Program inspection frequencies.]

5.5.7 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components. The program shall include the following:

5.5 Programs and Manuals

5.5.13 Primary Containment Leakage Rate Testing Program (continued)

b. The maximum allowable containment leakage rate, L_a, at P_a, shall be [ ]% of containment air weight per day.

c. Leakage rate acceptance criteria are:

1. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60\ L_a$ for the Type B and C tests and $< 0.75\ L_a$ for Type A tests.

2. Air lock testing acceptance criteria are:

a) Overall air lock leakage rate is $\leq$ [0.05 L_a] when tested at $\geq P_a$.

b) For each door, leakage rate is $\leq$ [0.01 L_a] when pressurized to [$\geq$ 10 psig].

d. The provisions of SR 3.0.3 are applicable to the Primary Containment Leakage Rate Testing Program.

e. Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J.

[OPTION B]

a. A program shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September, 1995[,as modified by the following exceptions:

1. The visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWL, except where relief has been authorized by the NRC.

2. The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI code, Subsection IWE, except where relief has been authorized by the NRC.

[3. . . .]

5.5 Programs and Manuals

5.5.13 Primary Containment Leakage Rate Testing Program (continued)

1. Containment leakage rate acceptance criterion is ≤ 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are < 0.60 L_a for the Type B and C tests and ≤ 0.75 L_a for Type A tests.

2. Air lock testing acceptance criteria are:

   a) Overall air lock leakage rate is ≤ [0.05 L_a] when tested at ≥ P_a.

   b) For each door, leakage rate is ≤ [0.01 L_a] when pressurized to [≥ 10 psig].

e. The provisions of SR 3.0.3 are applicable to the Primary Containment Leakage Rate Testing Program.

f. Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J.

[OPTION A/B Combined]

a. A program shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J. [Type A][Type B and C] test requirements are in accordance with 10 CFR 50, Appendix J, Option A, as modified by approved exemptions. [Type B and C][Type A] test requirements are in accordance with 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. The 10 CFR 50, Appendix J, Option B test requirements shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September, 1995~~[~~, as modified by the following exceptions:

   1. The visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWL, except where relief has been authorized by the NRC.

   2. The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI code, Subsection IWE, except where relief has been authorized by the NRC.

   [3. . . .]

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

j. Limitations on the annual dose or dose commitment to any member of the public, beyond the site boundary, due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluent Controls Program surveillance frequency.

5.5.5 Component Cyclic or Transient Limit

This program provides controls to track the FSAR, Section [ ], cyclic and transient occurrences to ensure that components are maintained within the design limits.

5.5.6 [Pre-Stressed Concrete Containment Tendon Surveillance Program

This program provides controls for monitoring any tendon degradation in pre-stressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial operations. The Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with ~~[Regulatory Guide 1.35, Revision 3, 1990]~~ Section XI, Subsection IWL of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50.55a, except where an alternative, exemption, or relief has been authorized by the NRC.

The provisions of ~~SR 3.0.2 and~~ SR 3.0.3 are applicable to the Tendon Surveillance Program inspection frequencies.]

5.5.7 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components. The program shall include the following:

a. Testing frequencies specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as follows:

5.5 Programs and Manuals

5.5.13 Primary Containment Leakage Rate Testing Program (continued)

a) Overall air lock leakage rate is ≤ [0.05 L_a] when tested at ≥ P_a.

b) For each door, leakage rate is ≤ [0.01 L_a] when pressurized to [≥ 10 psig].

d. The provisions of SR 3.0.3 are applicable to the Primary Containment Leakage Rate Testing Program.

e. Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J.

[OPTION B]

a. A program shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September, 1995[,as modified by the following exceptions:

1. The visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWL, except where relief has been authorized by the NRC.

2. The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI code, Subsection IWE, except where relief has been authorized by the NRC.

[3. . . .]

b. The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a, is [45 psig]. The containment design pressure is [50 psig].

c. The maximum allowable containment leakage rate, L_a, at P_a, shall be [ ]% of containment air weight per day.

d. Leakage rate acceptance criteria are:

1. Containment leakage rate acceptance criterion is ≤ 1.0 L_a. During the first unit startup following testing in accordance with this program, the

5.5 Programs and Manuals

5.5.13 Primary Containment Leakage Rate Testing Program (continued)

e. The provisions of SR 3.0.3 are applicable to the Primary Containment Leakage Rate Testing Program.

f. Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J.

[OPTION A/B Combined]

a. A program shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J. [Type A][Type B and C] test requirements are in accordance with 10 CFR 50, Appendix J, Option A, as modified by approved exemptions. [Type B and C][Type A] test requirements are in accordance with 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. The 10 CFR 50, Appendix J, Option B test requirements shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September, 1995-[,as modified by the following exceptions:

1. The visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWL, except where relief has been authorized by the NRC.

2. The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI code, Subsection IWE, except where relief has been authorized by the NRC.

[3. . . .]

b. The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a, is [45 psig]. The containment design pressure is [50 psig].

c. The maximum allowable containment leakage rate, L_a, at P_a, shall be [ ]% of containment air weight per day.

d. Leakage rate acceptance criteria are:

1. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60\ L_a$ for the Type B and C

BASES

ACTIONS

A.1

In the event that primary containment is inoperable, primary containment must be restored to OPERABLE status within 1 hour. The 1 hour Completion Time provides a period of time to correct the problem that is commensurate with the importance of maintaining primary containment OPERABILITY during MODES 1, 2, and 3. This time period also ensures that the probability of an accident (requiring primary containment OPERABILITY) occurring during periods where primary containment is inoperable is minimal.

B.1 and B.2

If primary containment cannot be restored 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 12 hours and to MODE 4 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 3.6.1.1.1

Maintaining the primary containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements of the Primary Containment Leakage Rate Testing Program. The primary containment concrete visual examinations may be performed during either power operation, e.g., performed concurrently with other primary containment inspection-related activities such as tendon testing, or during a maintenance or refueling outage. The visual examinations of the steel liner plate inside primary containment are performed during maintenance or refueling outages since this is the only time the liner plate is fully accessible.

Failure to meet air lock leakage testing (SR 3.6.1.2.1 and SR 3.6.1.2.4), [secondary containment bypass leakage (SR 3.6.1.3.9),] resilient seal primary containment purge valve leakage testing (SR 3.6.1.3.6), or main steam isolation valve leakage (SR 3.6.1.3.10) does not necessarily result in a failure of this SR. The impact of the failure to meet these SRs must be evaluated against the Type A, B, and C acceptance criteria of the Primary Containment Leakage Rate Testing Program. As left leakage prior to the first startup after performing a required Primary Containment Leakage Rate Testing Program leakage test is required to be $< 0.6\ L_a$ for combined Type B and C leakage, and [$< 0.75\ L_a$ for Option A] [$\leq 0.75\ L_a$ for Option B] for overall Type A leakage. At all other times between required leakage rate tests, the acceptance criteria is based on an overall

Type A leakage limit of ≤ 1.0 L_a. At ≤ 1.0 L_a the offsite dose consequences are bounded by the assumptions of the safety analysis. The Frequency is required by the Primary Containment Leakage Rate Testing Program.

SR Frequencies are as required by the Primary Containment Leakage Rate Testing Program. These periodic testing requirements verify that the primary containment leakage rate does not exceed the leakage rate assumed in the safety analysis.

BASES

SURVEILLANCE REQUIREMENTS (continued)

----------REVIEWER'S NOTE----------
Regulatory Guide 1.163 and NEI 94-01 include acceptance criteria for as-left and as-found Type A leakage rates and combined Type B and C leakage rates, which may be reflected in the Bases.

[SR 3.6.1.1.2

The structural integrity of the primary containment is ensured by the successful completion of the Primary Containment Tendon Surveillance Program and by associated visual inspections of the steel liner and penetrations for evidence of deterioration or breach of integrity. This ensures that the structural integrity of the primary containment will be maintained in accordance with the provisions of the Primary Containment Tendon Surveillance Program. Testing and Frequency are ~~consistent with the recommendations of Regulatory Guide 1.35~~ in accordance with the ASME code, Section XI, Subsection IWL (Ref. 5), and applicable addenda as required by 10 CFR 50.55a.]

REFERENCES

1. FSAR, Section [6.2].
2. FSAR, Section [15.6.5].
3. 10 CFR 50, Appendix J, Option [A][B].
4. FSAR, Section [ ].
5. ~~Regulatory Guide 1.35, Revision [1]~~ ASME Code, Section XI, Subsection IWL.

Technical Specification Task Force
Improved Standard Technical Specifications Change Traveler

Cancellation of NRC Environmental Monitoring Program with States

NUREGs Affected: ☑ 1430 ☑ 1431 ☑ 1432 ☑ 1433 ☑ 1434

Classification 2) Consistency/Standardization

Recommended for CLIIP?: (Unassigned)

Correction or Improvement: Unassigned

NRC Fee Status: Exempt

Benefit: Prevents Unnecessary Actions

Industry Contact: Dana Millar, (601) 368-5445, DMILLAR@entergy.com

In press release no. 98-08, dated January 13, 1998, the NRC announced that it ended its contract with 34 states to perform radiation monitoring around certain facilities at the end of 1997. In section 5.6.2 of the NUREG, there is a statement in the Annual Radiological Environmental Operating Report that the TLD results that represent collocated dosimeters in relation to the NRC TLD Program and the exposure period associated with each result be included in the report. The TLD Program that the NUREG references is the same program that the NRC cancelled at the end of 1997.

In press release no. 98-08, dated January 13, 1998, the NRC announced that it ended its contract with 34 states to perform radiation monitoring around certain facilities at the end of 1997. There is no longer a need to reference this program in the NUREG.

Revision History

OG Revision 0 Revision Status: Active

Revision Proposed by: Palo Verde

Revision Description:
Original Issue

Owners Group Review Information

Date Originated by OG: 17-Mar-99

Owners Group Comments
(No Comments)

Owners Group Resolution: Approved Date: 17-Mar-99

TSTF Review Information

TSTF Received Date: 16-Jun-99 Date Distributed for Review 17-Jun-99

OG Review Completed: ☑ BWOG ☑ WOG ☑ CEOG ☑ BWROG

TSTF Comments:
Applicable to all.

TSTF Resolution: Approved Date: 07-Jul-99

NRC Review Information

NRC Received Date: 20-Jul-99

NRC Comments: Date of NRC Letter: 01-Nov-99

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OG Revision 0 Revision Status: Active

NRC to provide status by 10/14/99.

Final Resolution: NRC Approves Final Resolution Date: 01-Nov-99

Affected Technical Specifications

5.6.2 Reporting Requirements

Change Description: Annual Radiological Environmental Operating Report

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TSTF-348

5.6 Reporting Requirements

5.6.2 Annual Radiological Environmental Operating Report (continued)

(ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements [in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979]. [The report shall identify the TLD results that represent collocated dosimeters in relation to the NRC TLD program and the exposure period associated with each result.] In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.6.3 Radioactive Effluent Release Report

----------NOTE----------
A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

The Radioactive Effluent Release Report covering the operation of the unit shall be submitted in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

(continued)

TSTF-348

5.6 Reporting Requirements

5.6.2 Annual Radiological Environmental Operating Report (continued)

(ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements [in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979]. [The report shall identify the TLD results that represent collocated dosimeters in relation to the NRC TLD program and the exposure period associated with each result.] In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

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(continued)

5.6 Reporting Requirements

TSTF-348

5.6.2 Annual Radiological Environmental Operating Report (continued)

(ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements [in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979]. ~~[The report shall identify the TLD results that represent collocated dosimeters in relation to the NRC TLD program and the exposure period associated with each result.]~~ In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

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5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience[, including documentation of all challenges to the pressurizer

(continued)

TSTF-348

5.6 Reporting Requirements

5.6.2 Annual Radiological Environmental Operating Report (continued)

(ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements [in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979]. [The report shall identify the TLD results that represent collocated dosimeters in relation to the NRC TLD program and the exposure period associated with each result.] In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

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5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience[, including documentation of all challenges to the safety/relief

(continued)

TSTF-348

5.6 Reporting Requirements

5.6.2 Annual Radiological Environmental Operating Report (continued)

(ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements [in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979]. ~~[The report shall identify the TLD results that represent collocated dosimeters in relation to the NRC TLD program and the exposure period associated with each result.]~~ In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.6.3 Radioactive Effluent Release Report

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5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience[, including documentation of all challenges to the safety/relief

(continued)

Technical Specification Task Force
Improved Standard Technical Specifications Change Traveler

Change to the VFTP in ITS Section 5.0 in accordance with GL 99-02

NUREGs Affected: ☑ 1430 ☑ 1431 ☑ 1432 ☑ 1433 ☑ 1434

Classification: 3) Improve Specifications Recommended for CLIIP?: (Unassigned)

Correction or Improvement: (Unassigned)

Industry Contact: Patricia Furio, (410) 495-4374, patricia.s.furio@ccnppi.com

A revision is needed to change the VFTP in ITS Section 5.0 to reflect the changes of Generic Letter 99-02, Laboratory Testing of Nuclear-Grade Activated Charcoal, issued on June 3, 1999. A primary change identified in GL 99-02 was the specification of ASTM D3803-1989 for charcoal filter laboratory testing. The NUREGs already identified this particular ASTM standard, however, GL also allows the use of a different safety factor than that currently in the NUREG. The allowed safety factor is identified in the existing Reviewer's Note of ITS 5.511 which is being replaced with the Reviewer's Note contained in GL 99-02 attachment 2. The change in safety factor results in a change to the Penetration values specified in TS 5.5.11.

The Bases of various Specifications are revised to eliminate statements that the filter testing is performed in accordance with Regulatory Guide 1.52. Under this change, filter testing is performed in accordance with the ATSM standard. When necessary, references are renumbered to accommodate this change.

In addition, References to Regulatory Guide 1.52 are revised to be consistent.

This is needed to reflect the changes identified in GL 99-02, Laboratory Testing of Nuclear-Grade Activated Charcoal, issued on June 3, 1999. In previous testing standards, it was erroneously assumed that the high temperature/high relative humidity provided the most severe test conditions. Later with more testing experience, it was determined that the most conservative testing of charcoal is at low temperature/high humidity. The NRC position in GL 99-02 is that testing to a standard other than ASTM D3803-1989 may result in an overestimation of the capability of the charcoal to adsorb radiation due to a lack of proper control of testing conditions (humidity and temperature). Since testing charcoal per ASTM D3803-1989 will yield more accurate results that better characterize the carbon's ability to adsorb radioactive gases, the use of a lower safety factor is allowed.

Revision History

OG Revision 0 Revision Status: Active

Revision Proposed by: Palo Verde

Revision Description:
Original Issue

Owners Group Review Information

Date Originated by OG: 11-Dec-99

Owners Group Comments:
(No Comments)

Owners Group Resolution: Approved Date: 11-Dec-99

TSTF Review Information

TSTF Received Date: 11-Jan-00 Date Distributed for Review: 11-Jan-00

OG Review Completed: ☑ BWOG ☑ WOG ☑ CEOG ☑ BWROG

TSTF Comments:

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OG Revision 0 Revision Status: Active

(No Comments)

TSTF Resolution: Approved Date: 18-Feb-00

NRC Review Information

NRC Received Date: 13-Mar-00

NRC Comments: Date of NRC Letter: 31-Oct-00

9/14/00 - Editorial changes made in WOG-ED-27 to add a Bases change to the ECCS PREACS specification and to delete the phrase "greater than or equal to" from the last sentence of 5.5.11.c in the WOG, BWR/4, and BWR/6 markups.

Final Resolution: NRC Approves Final Resolution Date: 13-Apr-00

Affected Technical Specifications

5.5.11	Ventilation Filter Testing Program	
SR 3.7.10.2 Bases	CREVS	NUREG(s)- 1430 Only
Ref. 3.7.12 Bases	EVS	NUREG(s)- 1430 Only
SR 3.7.12.2 Bases	EVS	NUREG(s)- 1430 Only
Ref. 3.7.13 Bases	FSPVS	NUREG(s)- 1430 Only
SR 3.7.13.2 Bases	FSPVS	NUREG(s)- 1430 Only
SR 3.6.11.2 Bases	ICS (Atmospheric and Subatmospheric)	NUREG(s)- 1431 Only
SR 3.6.13.2 Bases	SBACS (Dual and Ice Condenser)	NUREG(s)- 1431 Only
SR 3.7.10.2 Bases	CREFS	NUREG(s)- 1431 Only
SR 3.7.12.2 Bases	ECCS PREACS	NUREG(s)- 1431 Only
SR 3.7.13.2 Bases	FBACS	NUREG(s)- 1431 Only
SR 3.7.14.2 Bases	PREACS	NUREG(s)- 1431 Only
Ref. 3.6.10 Bases	ICS (Atmospheric and Dual)	NUREG(s)- 1432 Only
SR 3.6.10.2 Bases	ICS (Atmospherics and Dual)	NUREG(s)- 1432 Only
Ref. 3.6.11 Bases	Shield Building (Dual)	NUREG(s)- 1432 Only
Ref. 3.6.13 Bases	SBEACS (Dual)	NUREG(s)- 1432 Only
SR 3.6.13.2 Bases	SBEACS (Dual)	NUREG(s)- 1432 Only

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SR 3.7.11.2 Bases	CREACS	NUREG(s)- 1432 Only
SR 3.7.13.2 Bases	ECCS PREACS	NUREG(s)- 1432 Only
Ref. 3.7.14 Bases	FBACS	NUREG(s)- 1432 Only
SR 3.7.14.2 Bases	FBACS	NUREG(s)- 1432 Only
Ref. 3.7.15 Bases	PREACS	NUREG(s)- 1432 Only
Ref. 3.6.4.3 Bases	SGT System	NUREG(s)- 1433 Only
SR 3.6.4.3.s Bases	SGT System	NUREG(s)- 1433 Only
SR 3.7.4.2 Bases	[MCREC] System	NUREG(s)- 1433 Only
SR 3.6.4 Bases	SGT System	NUREG(s)- 1434 Only
SR 3.6.4.2 Bases	SGT System	NUREG(s)- 1434 Only
Ref. 3.7.3 Bases	[CRFA] System	NUREG(s)- 1434 Only
SR 3.7.3.2 Bases	[CRFA] System	NUREG(s)- 1434 Only

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INSERT 1

---------- REVIEWER'S NOTE ----------

The use of any standard other than ASTM D3803-1989 to test the charcoal sample may result in an overestimation of the capability of the charcoal to adsorb radioiodine. As a result, the ability of the charcoal filters to perform in a manner consistent with the licensing basis for the facility is indeterminate.

ASTM D3803-1989 is a more stringent testing standard because it does not differentiate between used and new charcoal, it has a longer equilibration period performed at a temperature of 30 °C [86 °F] and a relative humidity (RH) of 95% (or 70% RH with humidity control), and it has more stringent tolerances that improve repeatability of the test.

$$\text{Allowable Penetration} = \frac{[100\% - \text{Methyl Iodide Efficiency} * \text{ for Charcoal Credited In Licensee's Accident Analysis}]}{\text{Safety Factor}}$$

When ASTM D3803-1989 is used with 30 °C [86 °F] and 95% RH (or 70% RH with humidity control) is used, the staff will accept the following:

Safety factor $\geq$ 2 for systems with or without humidity control.

Humidity control can be provided by heaters or an NRC-approved analysis that demonstrates that the air entering the charcoal will be maintained less than or equal to 70 percent RH under worst-case design-basis conditions.

If the system has a face velocity greater than 110 percent of 0.203 m/s (40 ft/min), the face velocity should be specified.

* This value should be the efficiency that was incorporated in the licensee's accident analysis which was reviewed and approved by the staff in a safety evaluation.

INSERT 2

Face Velocity (fps)
See Reviewer's Note

TSTF-362

5.5 Programs and Manuals

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

b. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass < [0.5]% when tested in accordance with [Regulatory Guide 1.52, Revision 2, and ASME N510-1989] at the system flowrate specified below [± 10%].

ESF Ventilation System	Flowrate
[ ]	[ ]

c. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in [Regulatory Guide 1.52, Revision 2], shows the methyl iodide penetration less than the value specified below when tested in accordance with ~~[ASTM D3803-1989]~~ at a temperature of ~~[~~30°C~~]~~ and ~~greater than or equal to~~ the relative humidity specified below.

(86°F)

Insert 2

ESF Ventilation System	Penetration	RH
[ ]	[See Reviewer's Note]	[See Reviewer's Note]

~~Reviewer's Note: Allowable penetration = [100% - methyl iodide efficiency for charcoal credited in staff safety evaluation]/(safety factor).~~

~~Safety factor = [5] for systems with heaters.~~
~~= [7] for systems without heaters.~~

Insert 1

d. Demonstrate for each of the ESF systems that the pressure drop across the combined HEPA filters, the prefilters, and the charcoal adsorbers is less than the value specified below when tested in accordance with [Regulatory Guide 1.52,

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.10.2

This SR verifies that the required CREVS testing is performed in accordance with the [Ventilation Filter Testing Program (VFTP)]. The CREVS filter tests are in accordance with Regulatory Guide 1.52 (Ref. 3). The [VFTP] includes testing HEPA filter performance, charcoal absorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal. Specific test frequencies and additional information are discussed in detail in the [VFTP].

SR 3.7.10.3

This SR verifies that [each CREVS train starts] [or the control room isolates] and operates on an actual or simulated actuation signal. The Frequency of [18] months is consistent with that specified in Reference 3.

SR 3.7.10.4

This SR verifies the integrity of the control room enclosure and the assumed inleakage rates of the potentially contaminated air. The control room positive pressure, with respect to potentially contaminated adjacent areas, is periodically tested to verify that the CREVS is functioning properly. During the emergency mode of operation, the CREVS is designed to pressurize the control room ≥ [0.125] inches water gauge positive pressure, with respect to adjacent areas, to prevent unfiltered inleakage. The CREVS is designed to maintain this positive pressure with one train at a flow rate of ≤ [3300] cfm. This value includes [300] cfm of outside air. The Frequency of [18] months on a STAGGERED TEST BASIS is consistent with industry practice and other filtration SRs.

REFERENCES

1. FSAR, Section [9.4].

2. FSAR, Chapter [15].

3. Regulatory Guide 1.52.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.12.2

This SR verified that the required EVS testing is performed in accordance with the [Ventilation Filter Testing Program (VFTP)]. The EVS filter tests are in accordance with Regulatory Guide 1.52 (Ref. 5). The [VFTP] includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the [VFTP].

SR 3.7.12.3

This SR verifies that each EVS train starts and operates on an actual or simulated actuation signal. The [18] month Frequency is consistent with that specified in Reference 5.

SR 3.7.12.4

This SR verifies the integrity of the negative pressure boundary area. The ability of the EVS to maintain a negative pressure, with respect to potentially uncontaminated adjacent areas, is periodically tested to verify proper functioning of the EVS. During the [post accident] mode of operation, the EVS is designed to maintain a slight negative pressure in the negative pressure boundary area with respect to adjacent areas to prevent unfiltered LEAKAGE. The EVS is designed to maintain this negative pressure at a flow rate of [3000] cfm from the negative pressure boundary area. The Frequency of [18] months on a STAGGERED TEST BASIS is consistent with industry practice and other filtration SRs.

[SR 3.7.12.5

Operating the EVS filter bypass damper is necessary to ensure that the system functions properly. The OPERABILITY of the EVS filter bypass damper is verified if it can be closed. An [18] month Frequency is consistent with that specified in Reference 5.]

(continued)

TSTF 362

BASES (continued)

REFERENCES

1. FSAR, Section [6.2.3].
2. FSAR, Section [9.4.2].
3. FSAR, Section [15.4.7].
4. Regulatory Guide 1.25.
5. 10 CFR 100.11.
6. Regulatory Guide 1.52. , Rev [2]

TSTF-362

BASES (continued)

REFERENCES

1. FSAR, Section [6.2.3].
2. FSAR, Section [9.4.2].
3. FSAR, Section [15.4.6].
4. 10 CFR 100.11.
5. Regulatory Guide 1.52, Rev [2].

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.13.2 (continued)

Program (VFTP)]. The FSPVS filter tests are in accordance with Regulatory Guide 1.52 (Ref. 6). The [VFTP] includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the [VFTP].

SR 3.7.13.3

This SR verifies that each FSPVS train starts and operates on an actual or simulated actuation signal. The 18 month Frequency is consistent with that specified in Reference 6.

SR 3.7.13.4

This SR verifies the integrity of the fuel handling area. The ability of the fuel handling area to maintain a negative pressure, with respect to potentially uncontaminated adjacent areas, is periodically tested to verify proper function of the FSPVS. During the [post accident] mode of operation, the FSPVS is designed to maintain a slight negative pressure in the fuel handling area to prevent unfiltered LEAKAGE. The FSPVS is designed to maintain this negative pressure at a flow rate of ≤ [3000] cfm to the fuel handling area. The Frequency of [18] months on a STAGGERED TEST BASIS is consistent with industry practice.

SR 3.7.13.5

Operating the FSPVS filter bypass damper is necessary to ensure that the system functions properly. The OPERABILITY of the FSPVS filter bypass damper is verified if it can be opened. A Frequency of [18] months is specified in Reference 6.

(continued)

TSTF 362

5.5 Programs and Manuals

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

b. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass < [0.05]% when tested in accordance with [Regulatory Guide 1.52, Revision 2, and ASME N510-1989] at the system flowrate specified below [± 10%].

ESF Ventilation System	Flowrate
[ ]	[ ]

c. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in [Regulatory Guide 1.52, Revision 2], shows the methyl iodide penetration less than the value specified below when tested in accordance with ~~[ASTM D3803-1989]~~ at a temperature of ~~≤ [30°C]~~ (86°F) and ~~greater than or equal to~~ the relative humidity specified below.

Insert 2

ESF Ventilation System	Penetration	RH
[ ]	See Reviewer's Note	See Reviewer's Note

~~Reviewer's Note: Allowable penetration = [100% - methyl iodide efficiency for charcoal credited in staff safety evaluation]/(safety factor).~~

~~Safety factor = [5] for systems with heaters.~~
~~= [7] for systems without heaters.~~

Insert 1

d. Demonstrate for each of the ESF systems that the pressure drop across the combined HEPA filters, the prefilters, and the charcoal adsorbers is less than the value specified below when tested in accordance with [Regulatory Guide 1.52,

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.10.1 (continued)

testing each train once every month provides an adequate check of this system. Monthly heater operations dry out any moisture accumulated in the charcoal from humidity in the ambient air. [Systems with heaters must be operated for ≥ 10 continuous hours with the heaters energized. Systems without heaters need only be operated for ≥ 15 minutes to demonstrate the function of the system.] The 31 day Frequency is based on the reliability of the equipment and the two train redundancy availability.

SR 3.7.10.2

This SR verifies that the required CREFS testing is performed in accordance with the [Ventilation Filter Testing Program (VFTP)]. The CREFS filter tests are in accordance with Regulatory Guide 1.52 (Ref. 3). The [VFTP] includes testing the performance of the HEPA filter, charcoal adsorber efficiency, minimum flow rate, and the physical properties of the activated charcoal. Specific test Frequencies and additional information are discussed in detail in the [VFTP].

SR 3.7.10.3

This SR verifies that each CREFS train starts and operates on an actual or simulated actuation signal. The Frequency of [18] months is specified in Regulatory Guide 1.52 (Ref. 3).

SR 3.7.10.4

This SR verifies the integrity of the control room enclosure, and the assumed inleakage rates of the potentially contaminated air. The control room positive pressure, with respect to potentially contaminated adjacent areas, is periodically tested to verify proper functioning of the CREFS. During the emergency mode of operation, the CREFS is designed to pressurize the control room ≥ [0.125] inches water gauge positive pressure with respect to adjacent areas in order to prevent unfiltered inleakage. The CREFS is designed to maintain this positive pressure

(continued)

This page was inserted by WOG-ED-27. WOG-ED-27

BASES

ACTIONS

B.1 and B.2 (continued)

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 3.7.12.1

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not 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 charcoal from humidity in the ambient air. [Systems with heaters must be operated ≥ 10 continuous hours with the heaters energized. Systems without heaters need only be operated for ≥ 15 minutes to demonstrate the function of the system.] The 31 day Frequency is based on the known reliability of equipment and the two train redundancy available.

SR 3.7.12.2

This SR verifies that the required ECCS PREACS testing is performed in accordance with the [Ventilation Filter Testing Program (VFTP)]. ~~The ECCS PREACS filter tests are in accordance with Reference 4.~~ The [VFTP] includes testing HEPA filter performance, charcoal adsorbers efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test Frequencies and additional information are discussed in detail in the [VFTP].

SR 3.7.12.3

This SR verifies that each ECCS PREACS train starts and operates on an actual or simulated actuation signal. The [18] month Frequency is consistent with that specified in Reference 4.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.13.2 (continued)

Program (VFTP)]. The FBACS filter tests are in accordance with Regulatory Guide 1.52 (Ref. 6). The [VFTP] includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the [VFTP].

SR 3.7.13.3

This SR verifies that each FBACS train starts and operates on an actual or simulated actuation signal. The [18] month Frequency is consistent with Reference 6.

SR 3.7.13.4

This SR verifies the integrity of the fuel building enclosure. The ability of the fuel building to maintain negative pressure with respect to potentially uncontaminated adjacent areas is periodically tested to verify proper function of the FBACS. During the [post accident] mode of operation, the FBACS is designed to maintain a slight negative pressure in the fuel building, to prevent unfiltered LEAKAGE. The FBACS is designed to maintain a ≤ [-0.125] inches water gauge with respect to atmospheric pressure at a flow rate of [20,000] cfm to the fuel building. The Frequency of [18] months is consistent with the guidance provided in NUREG-0800, Section 6.5.1 (Ref. 7).

An [18] month Frequency (on a STAGGERED TEST BASIS) is consistent with Reference 6.

SR 3.7.13.5

Operating the FBACS filter bypass damper is necessary to ensure that the system functions properly. The OPERABILITY of the FBACS filter bypass damper is verified if it can be closed. An [18] month Frequency is consistent with Reference 6.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.14.1 (continued)

reliability of equipment and the two train redundancy available.

SR 3.7.14.2

This SR verifies that the required PREACS testing is performed in accordance with the [Ventilation Filter Testing Program (VFTP)]. The PREACS filter tests are in accordance with Regulatory Guide 1.52 (Ref. 5). The [VFTP] includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the [VFTP].

SR 3.7.14.3

This SR verifies that each PREACS starts and operates on an actual or simulated actuation signal. The [18] month Frequency is consistent with that specified in Reference 5.

SR 3.7.14.4

This SR verifies the integrity of the penetration room enclosure. The ability of the penetration room to maintain a negative pressure, with respect to potentially uncontaminated adjacent areas, is periodically tested to verify proper function of PREACS. During the [post accident] mode of operation, the PREACS is designed to maintain a ≤ [-0.125] inches water gauge relative to atmospheric pressure at a flow rate of [3000] cfm in the penetration room, with respect to adjacent areas, to prevent unfiltered LEAKAGE. The Frequency of [18] months is consistent with the guidance provided in NUREG-0800 (Ref. 6).

The minimum system flow rate maintains a slight negative pressure in the penetration room area, and provides sufficient air velocity to transport particulate

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.11.1

Operating each ICS train for ≥ 15 minutes 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. For systems with heaters, operation with the heaters on (automatic heater cycling to maintain temperature) 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 considering the known reliability of fan motors and controls, the two train redundancy available, and the iodine removal capability of the Containment Spray System independent of the ICS.

SR 3.6.11.2

This SR verifies that the required ICS filter testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). ~~The ICS filter tests are in accordance with Regulatory Guide 1.52 (Ref. 3).~~ The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the VFTP.

SR 3.6.11.3

The automatic startup test verifies that both trains of equipment start upon receipt of an actual or simulated test signal. 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 Frequency was developed considering that

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

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 3.6.13.1

Operating each SBACS train for ≥ 15 minutes 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. For systems with heaters, operation with the heaters on (automatic heater cycling to maintain temperature) 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.

SR 3.6.13.2

This SR verifies that the required SBACS filter testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). ~~The SBACS filter tests are in accordance with Regulatory Guide 1.52 (Ref. 4).~~ The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the VFTP.

(continued)

ACTIONS

B.1 and B.2 (continued)

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 3.7.12.1

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not 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 charcoal from humidity in the ambient air. [Systems with heaters must be operated ≥ 10 continuous hours with the heaters energized. Systems without heaters need only be operated for ≥ 15 minutes to demonstrate the function of the system.] The 31 day Frequency is based on the known reliability of equipment and the two train redundancy available.

SR 3.7.12.2

This SR verifies that the required ECCS PREACS testing is performed in accordance with the [Ventilation Filter Testing Program (VFTP)]. ~~The ECCS PREACS filter tests are in accordance with Reference 4.~~ The [VFTP] includes testing HEPA filter performance, charcoal adsorbers efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test Frequencies and additional information are discussed in detail in the [VFTP].

SR 3.7.12.3

This SR verifies that each ECCS PREACS train starts and operates on an actual or simulated actuation signal. The [18] month Frequency is consistent with that specified in Reference 4.

(continued)

5.5 Programs and Manuals

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

N510-1989, at the system flowrate specified as follows [± 10%]:

ESF Ventilation System	Flowrate
[ ]	[ ]

b. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass < [0.05]% when tested in accordance with [Regulatory Guide 1.52, Revision 2, and ASME N510-1989] at the system flowrate specified as follows [± 10%]:

ESF Ventilation System	Flowrate
[ ]	[ ]

c. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in [Regulatory Guide 1.52, Revision 2], shows the methyl iodide penetration less than the value specified below when tested in accordance with [ASTM D3803-1989] at a temperature of [30°C] (86°F) and greater than or equal to the relative humidity specified as follows:

ESF Ventilation System	Penetration	RH
[ ]	See Reviewer's Note	See Reviewer's Note

Insert 2

(continued)

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5.5 Programs and Manuals

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

Insert 1

~~Reviewer's Note: Allowable penetration = [100% - methyl iodide efficiency for charcoal credited in staff safety evaluation]/ (safety factor).~~

~~Safety factor = [5] for systems with heaters.~~
~~= [7] for systems without heaters.~~

d. For each of the ESF systems, demonstrate the pressure drop across the combined HEPA filters, the prefilters, and the charcoal adsorbers is less than the value specified below when tested in accordance with [Regulatory Guide 1.52, Revision 2, and ASME N510-1989] at the system flowrate specified as follows [± 10%]:

ESF Ventilation System	Delta P	Flowrate
[ ]	[ ]	[ ]

[e. Demonstrate that the heaters for each of the ESF systems dissipate the following specified value [± 10%] when tested in accordance with [ASME N510-1989]:]

ESF Ventilation System	Wattage
[ ]	[ ]

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.

5.5.12 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides control for potentially explosive gas mixtures contained in the [Waste Gas Holdup System], [the quantity of radioactivity contained in gas storage tanks or fed into the offgas treatment system, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks]. The

(continued)

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.11.1 (continued)

testing each train once every month provides an adequate check on this system.

Monthly heater operations dry out any moisture accumulated in the charcoal from humidity in the ambient air. [Systems with heaters must be operated for ≥ 10 continuous hours with the heaters energized. Systems without heaters need only be operated for ≥ 15 minutes to demonstrate the function of the system.] The 31 day Frequency is based on the known reliability of the equipment, and the two train redundancy available.

SR 3.7.11.2

This SR verifies that the required CREACS testing is performed in accordance with the [Ventilation Filter Testing Program (VFTP)]. ~~The CREACS filter tests are in accordance with Regulatory Guide 1.52 (Ref. 3).~~ The [VFTP] includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the [VFTP].

SR 3.7.11.3

This SR verifies each CREACS train starts and operates on an actual or simulated actuation signal. The Frequency of [18] months is consistent with that specified in Reference 3.

SR 3.7.11.4

This SR verifies the integrity of the control room enclosure and the assumed inleakage rates of potentially contaminated air. The control room positive pressure, with respect to potentially contaminated adjacent areas, is periodically tested to verify proper function of the CREACS. During the emergency radiation state of the emergency mode of

(continued)

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BASES

ACTIONS (continued)

B.1 and B.2

If the ECCS PREACS train cannot be restored to OPERABLE status within the associated Completion Time, the unit must be 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 3.7.13.1

Standby systems should be checked periodically to ensure that they function properly. Since 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 charcoal from humidity in the ambient air. [Systems with heaters must be operated for ≥ 10 continuous hours with the heaters energized. Systems without heaters need only be operated for ≥ 15 minutes to demonstrate the function of the system.] The 31 day Frequency is based on the known reliability of equipment, and the two train redundancy available.

SR 3.7.13.2

This SR verifies that the required ECCS PREACS testing is performed in accordance with the [Ventilation Filter Testing Program (VFTP)]. ~~The ECCS PREACS filter tests are in accordance with Regulatory Guide 1.52 (Ref. 4).~~ The [VFTP] includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the [VFTP].

(continued)

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BASES

ACTIONS

C.1 and C.2 (continued)

If the system is not placed in operation, this action requires suspension of fuel movement, which precludes a fuel handling accident. This does not preclude the movement of fuel to a safe position.

D.1

When two trains of the FBACS are inoperable during movement of irradiated fuel assemblies in the fuel building, action must be taken to place the unit in a condition in which the LCO does not apply. This LCO involves immediately suspending movement of irradiated fuel assemblies in the fuel building. This does not preclude the movement of fuel to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.7.14.1

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not severe, testing each train once every month provides an adequate check on this system. Monthly heater operation dries out any moisture accumulated in the charcoal from humidity in the ambient air. [Systems with heaters must be operated for ≥ 10 continuous hours with the heaters energized. Systems without heaters need only be operated for ≥ 15 minutes to demonstrate the function of the system.] The 31 day Frequency is based on the known reliability of the equipment and the two train redundancy available.

SR 3.7.14.2

This SR verifies the performance of FBACS filter testing in accordance with the [Ventilation Filter Testing Program (VFTP)]. The FBACS filter tests are in accordance with the Regulatory Guide 1.52 (Ref. 6). The [VFTP] includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific

(co

BASES (continued)

REFERENCES

1. FSAR, Section [6.5.1].
2. FSAR, Section [9.4.5].
3. FSAR, Section [15.7.4].
4. Regulatory Guide 1.25.
5. 10 CFR 100.
6. Regulatory Guide 1.52, Rev. [2].
7. NUREG-0800, Section 6.5.1, July 1981.

BASES

REFERENCES (continued)

3. FSAR, Section [15.6.5].

4. Regulatory Guide 1.52, Rev. [2].

5. 10 CFR 100.11.

6. NUREG-0800, Section 6.5.1.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.10.1

Operating each ICS train for ≥ 15 minutes 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. For systems with heaters, operation with the heaters on (automatic heater cycling to maintain temperature) 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 considering the known reliability of fan motors and controls, the two train redundancy available, and the iodine removal capability of the Containment Spray System independent of the ICS.

SR 3.6.10.2

This SR verifies that the required ICS filter testing is performed in accordance with the Vent Filter Testing Program (VFTP). The ICS filter tests are in accordance with Regulatory Guide 1.52 (Ref. 3). The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the VFTP.

SR 3.6.10.3

The automatic startup test verifies that both trains of equipment start upon receipt of an actual or simulated test signal. 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 Frequency was developed considering that

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.10.3 (continued)

the system equipment OPERABILITY is demonstrated on a 31 day Frequency by SR 3.6.10.1.

SR 3.6.10.4

The ICS filter bypass dampers are tested to verify OPERABILITY. The dampers are in the bypass position during normal operation and must reposition for accident operation to draw air through the filters. The [18] month Frequency is considered to be acceptable based on the damper reliability and design, the mild environmental conditions in the vicinity of the dampers, and the fact that operating experience has shown that the dampers usually pass the Surveillance when performed at the [18] month Frequency.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 41, GDC 42, and GDC 43.
2. FSAR, Section [ ].
3. Regulatory Guide 1.52, Revision [~~1~~ 2].
4. FSAR, Section [ ].

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.11.4 (continued)

testing of the ability of the SBEACS to "pull down" the required negative pressure every [18] months.

REFERENCES

1. Regulatory Guide 1.52, Revision [~~1~~ 2].

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BASES

ACTIONS

B.1 and B.2 (continued)

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

Operating each SBEACS train for ≥ 15 minutes 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. For systems with heaters, operation with the heaters on (automatic heater cycling to maintain temperature) 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 considering the known reliability of fan motors and controls, the two train redundancy available, and the iodine removal capability of the Containment Spray System.

SR 3.6.13.2

This SR verifies that the required SBEACS filter testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The Emergency Ventilation System filter tests are in accordance with Regulatory Guide 1.52 (Ref. 4). The VFTP includes testing of HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the VFTP.

SR 3.6.13.3

The automatic startup ensures that each SBEACS 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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.13.3 (continued)

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 SBEACS equipment OPERABILITY is demonstrated at a 31 day Frequency by SR 3.6.13.1.

[SR 3.6.13.4

The filter bypass dampers are tested to verify OPERABILITY. The dampers are in the bypass position during normal operation and must reposition for accident operation to draw air through the filters. The [18] month Frequency is considered to be acceptable based on the damper reliability and design, the mild environmental conditions in the vicinity of the dampers, and the fact that operating experience has shown that the dampers usually pass the Surveillance when performed at the [18] month Frequency.]

SR 3.6.13.5

The SBEACS train flow rate is verified ≥ [ ] cfm to ensure that the flow rate is adequate to "pull down" the shield building pressure as required. This test also will verify the proper functioning of the fans, dampers, filters, absorbers, etc., when this SR is performed in conjunction with SR 3.6.11.4.

The [18] month on a STAGGERED TEST BASIS Frequency is consistent with the Regulatory Guide 1.52 (Ref. 4) guidance.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 41.

2. FSAR, Section [ ].

3. FSAR, Section [ ].

4. Regulatory Guide 1.52, Revision [~~1~~ 2].

5.5 Programs and Manuals

5.5.8 Ventilation Filter Testing Program (VFTP) (continued)

a. Demonstrate for each of the ESF systems that an inplace test of the HEPA filters shows a penetration and system bypass < [0.05]% when tested in accordance with [Regulatory Guide 1.52, Revision 2, and ASME N510-1989] at the system flowrate specified below [+ 10%].

ESF Ventilation System	Flowrate
[ ]	[ ]

b. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass < [0.05]% when tested in accordance with [Regulatory Guide 1.52, Revision 2, and ASME N510-1989] at the system flowrate specified below [± 10%].

ESF Ventilation System	Flowrate
[ ]	[ ]

c. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in [Regulatory Guide 1.52, Revision 2], shows the methyl iodide penetration less than the value specified below when tested in accordance with [ASTM D3803-1989] at a temperature of (30°C) and greater than or equal to the relative humidity specified below.

(86°F)

Insert 2

ESF Ventilation System	Penetration	RH
[ ]	See Reviewer's Note	See Reviewer's Note

(continued)

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5.5 Programs and Manuals

5.5.8 Ventilation Filter Testing Program (VFTP) (continued)

[Reviewer's Note: Allowable penetration = [100% - methyl iodide efficiency for charcoal credited in staff safety evaluation]/ (safety factor).

Safety factor = [5] for systems with heaters.
= [7] for systems without heaters.]

Insert 1

d. Demonstrate for each of the ESF systems that the pressure drop across the combined HEPA filters, the prefilters, and the charcoal adsorbers is less than the value specified below when tested in accordance with [Regulatory Guide 1.52, Revision 2, and ASME N510-1989] at the system flowrate specified as follows [± 10%]:

ESF Ventilation System	Delta P	Flowrate
[ ]	[ ]	[ ]

[e. Demonstrate that the heaters for each of the ESF system dissipate the value specified below [± 10%] when tested in accordance with [ASME N510-1989]:

ESF Ventilation System	Wattage
[ ]	[ ]

]

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.

5.5.9 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the [Waste Gas Holdup System], [the quantity of radioactivity contained in gas storage tanks or fed into the offgas treatment system, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks]. The

(continued)

BASES

ACTIONS

E.1, E.2, and E.3 (continued)

draindown and subsequent potential for fission product release. Actions must continue until OPDRVs are suspended.

Required Action D.1 has been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.6.4.3.1

Operating each SGT subsystem for ≥ [10] continuous hours ensures that [both] subsystems 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 (automatic heater cycling to maintain temperature)] for ≥ [10] continuous hours every 31 days eliminates moisture on the adsorbers and HEPA filters. The 31 day Frequency was developed in consideration of the known reliability of fan motors and controls and the redundancy available in the system.

SR 3.6.4.3.2

This SR verifies that the required SGT filter testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). ~~The SGT System filter tests are in accordance with Regulatory Guide 1.52 (Ref. 3).~~ The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the VFTP.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.4.3.3

This SR verifies that each SGT subsystem starts on receipt of an actual or simulated initiation signal. While this Surveillance can be performed with the reactor at power, operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency. The LOGIC SYSTEM FUNCTIONAL TEST in SR 3.3.6.2.6 overlaps this SR to provide complete testing of the safety function. Therefore, the Frequency was found to be acceptable from a reliability standpoint.

SR 3.6.4.3.4

This SR verifies that the filter cooler bypass damper can be opened and the fan started. This ensures that the ventilation mode of SGT System operation is available. While this Surveillance can be performed with the reactor at power, operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency, which is based on the refueling cycle. Therefore, the Frequency was found to be acceptable from a reliability standpoint.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 41.

2. FSAR, Section [6.2.3].

3. Regulatory Guide 1.52, Rev. [2].

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.4.1 (continued)

moisture that has accumulated in the charcoal as a result of humidity in the ambient air. [Systems with heaters must be operated for ≥ 10 continuous hours with the heaters energized. Systems without heaters need only be operated for ≥ 15 minutes to demonstrate the function of the system.] Furthermore, the 31 day Frequency is based on the known reliability of the equipment and the two subsystem redundancy available.

SR 3.7.4.2

This SR verifies that the required [MCREC] testing is performed in accordance with the [Ventilation Filter Testing Program (VFTP)]. The [MCREC] filter tests are in accordance with Regulatory Guide 1.52 (Ref. 5). The [VFTP] includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the [VFTP].

SR 3.7.4.3

This SR verifies that on an actual or simulated initiation signal, each [MCREC] subsystem starts and operates. The LOGIC SYSTEM FUNCTIONAL TEST in SR 3.3.7.1.5 overlaps this SR to provide complete testing of the safety function. The [18] month Frequency is specified in Reference 5.

[SR 3.7.4.4

This SR verifies the integrity of the control room enclosure and the assumed inleakage rates of potentially contaminated air. The control room positive pressure, with respect to potentially contaminated adjacent areas (the turbine building), is periodically tested to verify proper function of the [MCREC] System. During the emergency mode of operation, the [MCREC] System is designed to slightly pressurize the control room ≥ [0.1] inches water gauge positive pressure with respect to the turbine building to]

(continued)

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5.5.8 Ventilation Filter Testing Program (VFTP) (continued)

accordance with [Regulatory Guide 1.52, Revision 2, and ASME N510-1989] at the system flowrate specified below [± 10%]:

ESF Ventilation System	Flowrate
[ ]	[ ]

b. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass < [0.05]% when tested in accordance with [Regulatory Guide 1.52, Revision 2, and ASME N510-1989] at the system flowrate specified below [± 10%]:

ESF Ventilation System	Flowrate
[ ]	[ ]

c. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in [Regulatory Guide 1.52, Revision 2], shows the methyl iodide penetration less than the value specified below when tested in accordance with ~~[~~ASTM D3803-1989~~]~~ at a temperature of ~~[~~30°C~~]~~ (86°F) and ~~greater than or equal to~~ the relative humidity specified below:

Insert 2

ESF Ventilation System	Penetration	RH
[ ]	See Reviewer's Note	See Reviewer's Note

~~Reviewer's Note: Allowable penetration = [100% - methyl iodide efficiency for charcoal credited in staff safety evaluation]/(safety factor).~~

~~Safety factor = [5] for systems with heaters.~~
~~= [7] for systems without heaters.~~

Insert 1

(continued)

BASES

ACTIONS

E.1, E.2, and E.3 (continued)

suspended. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, actions must be immediately initiated to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Action must continue until OPDRVs are suspended.

Required Action E.1 has been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies would not be sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.6.4.3.1

Operating each SGT subsystem for ≥ [10] continuous hours ensures that both subsystems 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 (automatic heater cycling to maintain temperature)] for ≥ [10] continuous hours every 31 days eliminates moisture on the adsorbers and HEPA filters. The 31 day Frequency was developed in consideration of the known reliability of fan motors and controls and the redundancy available in the system.

SR 3.6.4.3.2

This SR verifies that the required SGT filter testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). ~~The SGT System filter tests are in accordance with Regulatory Guide 1.52 (Ref. 4).~~ The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specified test frequencies

(continued)

BASES

SURVEILLANCE

SR 3.6.4.3.2 (continued)

and additional information are discussed in detail in the VFTP.

SR 3.6.4.3.3

This SR requires verification that each SGT subsystem starts upon receipt of an actual or simulated initiation signal. The LOGIC SYSTEM FUNCTIONAL TEST in SR 3.3.6.2.5 overlaps this SR to provide complete testing of the safety function. While this Surveillance can be performed with the reactor at power, operating experience has shown these components usually pass the Surveillance 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.

[SR 3.6.4.3.4

This SR requires verification that the SGT filter cooler bypass damper can be opened and the fan started. This ensures that the ventilation mode of SGT System operation is available. While this Surveillance can be performed with the reactor at power, operating experience has shown these components usually pass the Surveillance 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.]

REFERENCES

1. 10 CFR 50, Appendix A, GDC 41.
2. FSAR, Section [6.2.3].
3. FSAR, Section [15.6.5].
4. Regulatory Guide 1.52, Rev. [2].

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1 (continued)

each subsystem once every month provides an adequate check on this system. Monthly heater operation dries out any moisture accumulated in the charcoal from humidity in the ambient air. [Systems with heaters must be operated for ≥ 10 continuous hours with the heaters energized. Systems without heaters need only be operated for ≥ 15 minutes to demonstrate the function of the system.] Furthermore, the 31 day Frequency is based on the known reliability of the equipment and the two subsystem redundancy available.

SR 3.7.3.2

This SR verifies that the required CRFA testing is performed in accordance with the [Ventilation Filter Testing Program (VFTP)]. The CRFA filter tests are in accordance with Regulatory Guide 1.52 (Ref. 5). The [VFTP] includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the [VFTP].

SR 3.7.3.3

This SR verifies that each [CRFA] subsystem starts and operates on an actual or simulated initiation signal. The LOGIC SYSTEM FUNCTIONAL TEST in SR 3.3.7.1.5 overlaps this SR to provide complete testing of the safety function. The [18] month Frequency is specified in Reference 5.

SR 3.7.3.4

This SR verifies the integrity of the control room enclosure and the assumed inleakage rates of potentially contaminated air. The control room positive pressure, with respect to potentially contaminated adjacent areas, is periodically tested to verify proper function of the [CRFA] System. During the emergency mode of operation, the [CRFA] System is designed to slightly pressurize the control room to [0.1] inches water gauge positive pressure with respect to adjacent areas to prevent unfiltered inleakage. The [CRFA]

(continued)

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.3.4 (continued)

System is designed to maintain this positive pressure at a flow rate of [500] cfm to the control room in the isolation mode. The Frequency of [18] months on a STAGGERED TEST BASIS is consistent with industry practice and other filtration system SRs.

REFERENCES

1. FSAR, Section [6.5.1].
2. FSAR, Section [9.4.1].
3. FSAR, Chapter [6].
4. FSAR, Chapter [15].
5. Regulatory Guide 1.52, Rev~~ision~~ [2], March ~~1978~~.

Technical Specification Task Force
Improved Standard Technical Specifications Change Traveler

Revise Topical Report references in ITS 5.6.5, COLR

NUREGs Affected: ☑ 1430 ☑ 1431 ☑ 1432 ☑ 1433 ☑ 1434

Classification: 3) Improve Specifications Recommended for CLIIP?: (Unassigned)

Correction or Improvement: (Unassigned)

Industry Contact: Tom Silko, (802) 258-4146, tsilko@entergy.com

The requirement in ITS 5.6.5, Core Operating Limits Report (COLR), to identify the Topical Report(s) by number, title, date, and NRC staff approval document is revised to allow the Topical Reports to be identified by number and title. A requirement is added to specify the complete citation in the COLR for each Topical Report, including the report number, title, revision, date, and any supplements.
In a letter from Mr. Stuart A. Richards (NRC) to Mr. James F. Mallay (Siemens Power Corporation) dated December 15, 1999, entitled "Acceptance for Siemens References to Approved Topical Reports in Technical Specifications," the NRC stated that it is acceptable for the references to Topical Reports in ITS Section 5.6.5, COLR, to give the Topical Report title and number as long as the complete citation is given in the COLR.

As stated in the letter, this method of referencing topical reports would allow licensees to use current topical reports to support limits in the COLR without having to submit an amendment to the facility operating license every time the topical report is revised. The COLR would provide specific information identifying the particular approved topical reports used to determine the core limits for the particular cycle in the COLR report. This would eliminate unnecessary expenditure of NRC and licensee resources, and would ease the burden of TS submittal and approval needed to license reload fuel.

Revision History

OG Revision 0 Revision Status: Active

Revision Proposed by: BWROG

Revision Description:
Original Issue

Owners Group Review Information

Date Originated by OG: 10-Feb-00

Owners Group Comments:
(No Comments)

Owners Group Resolution: Approved Date: 22-Feb-00

TSTF Review Information

TSTF Received Date: 10-Feb-00 Date Distributed for Review: 10-Feb-00

OG Review Completed: ☑ BWOG ☑ WOG ☑ CEOG ☑ BWROG

TSTF Comments:
(No Comments)

TSTF Resolution: Approved Date: 22-Feb-00

NRC Review Information

NRC Received Date: 13-Mar-00

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OG Revision 0 Revision Status: Active

Final Resolution: NRC Approves Final Resolution Date: 13-Apr-00

Affected Technical Specifications

5.6.5 Core Operating Limits Report (COLR)

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INSERT 1

The COLR will contain the complete identification for each of the TS referenced topical reports used to prepare the COLR (i.e., report number, title, revision, date, and any supplements).

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5.6 Reporting Requirements (continued)

5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience[, including documentation of all challenges to the pressurizer power operated relief valves or pressurizer safety valves,] shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

[The individual specifications that address core operating limits must be referenced here.]

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

[Identify the Topical Report(s) by number, and title, ~~date, and NRC staff approval document,~~ or identify the staff Safety Evaluation Report for a plant specific methodology by NRC letter and date.]

Insert 1

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

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

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heat up, cooldown, low temperature operation, criticality, and hydrostatic

(continued)

TSTF-363

5.6 Reporting Requirements (continued)

5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience[, including documentation of all challenges to the pressurizer power operated relief valves or pressurizer safety valves,] shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

[The individual specifications that address core operating limits must be referenced here.]

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

[Identify the Topical Report(s) by number, title, ~~date, and~~ NRC ~~staff approval document,~~ or identify the staff Safety Evaluation Report for a plant specific methodology by NRC letter and date.]

Insert 1

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

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

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heat up, cooldown, low temperature operation, criticality, and hydrostatic

(continued)

TSTF-363

5.6 Reporting Requirements

5.6.4 Monthly Operating Reports (continued)

power operated relief valves or pressurizer safety valves,] shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

[The individual specifications that address core operating limits must be referenced here.]

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

[Identify the Topical Report(s) by number and title, ~~date, and NRC staff approval document,~~ or identify the staff Safety Evaluation Report for a plant specific methodology by NRC letter and date.]

Insert 1

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

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

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heatup, cooldown, low temperature operation, critically, and hydrostatic

(continued)

TSTF-363

5.6 Reporting Requirements

5.6.4 Monthly Operating Reports (continued)

valves,] shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

[The individual specifications that address core operating limits must be referenced here.]

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

Insert 1

[Identify the Topical Report(s) by number and title, ~~date, and NRC staff approval document,~~ or identify the staff Safety Evaluation Report for a plant specific methodology by NRC letter and date.]

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

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

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heatup, cooldown, low temperature operation, critically, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

(continued)

TSTF-363

5.6 Reporting Requirements

5.6.4 Monthly Operating Reports (continued)

valves,] shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

[The individual specifications that address core operating limits must be referenced here.]

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

Insert 1

[Identify the Topical Report(s) by number and title, ~~(date, and NRC staff approval document)~~ or identify the staff Safety Evaluation Report for a plant specific methodology by NRC letter and date.]

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

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

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heatup, cooldown, low temperature operation, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

(continued)

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

TSTF-263

December 15, 1999

Mr. James F. Mallay
Director, Regulatory Affairs
Siemens Power Corporation
2101 Horn Rapids Road
Richland, Washington 99352-0130

SUBJECT: ACCEPTANCE FOR SIEMENS REFERENCES TO APPROVED TOPICAL REPORTS IN TECHNICAL SPECIFICATIONS (TAC NO. MA6492)

Dear Mr. Mallay:

By letter dated September 1, 1999, Siemens Power Corporation requested approval of a method of referencing topical reports in the technical specifications (TS) of operating power plants, that make use of a Core Operating Limits Report (COLR). The method would allow the references to approved topical reports in the TS to be cited using the report number and title, and a note that the COLR provides the complete citation for the reports used. The citation in the COLR would include the complete identification for each of the TS references to topical reports used to prepare the COLR (i.e., report number, title, revision, date and any supplements).

For example, the reference in the TS would be of the form, EMF-84-093(P)(A), "Steam Line Break Methodology for PWRs," approved version as specified in the COLR. The corresponding reference in the COLR would be, EMF-84-093(P)(A), Revision 1, "Steam Line Break Methodology for PWRs," Siemens Power Corporation, February 1999.

The NRC staff has reviewed your request and agrees that this would be an acceptable approach. As stated in your letter, this method of referencing topical reports would allow licensees to use current topical reports to support limits in the COLR without having to submit an amendment to the facility operating license every time the topical report is revised. The COLR would provide specific information identifying the particular approved topical reports used to determine the core limits for the particular cycle in the COLR report. This would eliminate unnecessary expenditure of NRC and licensee resources, and would ease the burden of TS submittal and approval needed to license reload fuel. As you know, only NRC-approved methodology may be used and that since the COLR is a FSAR-related document, changes to the COLR require prior licensee review for unreviewed safety questions under 10 CFR 50.59.

This completes the staff effort on TAC No. MA6492. If you have any questions, please call Mr. N. Kalyanam, Project Manager, at (301) 415-1480.

Sincerely,

Stuart A. Richards, Director
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Project No. 702

Technical Specification Task Force
Improved Standard Technical Specifications Change Traveler

Revision to TS Bases Control Program to Incorporate Changes to 10 CFR 50.59

NUREGs Affected: ☑ 1430 ☑ 1431 ☑ 1432 ☑ 1433 ☑ 1434

Classification: 2) Consistency/Standardization Recommended for CLIIP?: (Unassigned)

Correction or Improvement: (Unassigned)

Industry Contact: Steve Wideman, (620) 364-4037, stwidem@wcnoc.com

Technical Specification 5.5.14, Technical Specifications (TS) Bases Control Program, requires a program for processing changes to the Bases of the Technical Specifications. TS 5.5.14b. states: Licensees may make changes to the Bases without prior NRC approval provided the changes do not involve either of the following: 1. a change in the TS incorporated in the license; or 2. a change to the updated FSAR or Bases that involves an unreviewed safety question as defined in 10 CFR 50.59. TS 5.5.14b.2. is revised to state: "a change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59." based on the changes to 10 CFR 50.59 published in the Federal Register (Volume 64, Number 191) dated October 4, 1999.

BACKGROUND

10 CFR 50.59 establishes the conditions under which licensees may make changes to the facility or procedures and conduct test or experiments without prior NRC approval.

In 1999, the NRC revised it regulation (Federal Register - Volume 64, Number 191 dated October 4, 1999) controlling changes, tests and experiments performed by nuclear plant licensees. The changes were prompted by the need to resolve differences in interpretation of the rule's requirements by the industry and the NRC that came clear focus in 1996. The rule changes had two principal objectives, both aimed at restoring much needed regulatory stability to this extensively used regulation:

- Establish clear definitions to promote common understanding of the rule's requirements

- Clarify the criteria for determining when changes, tests, and experiments require prior NRC approval.

The changes approved by the Commission in 1999 made 10 CFR 50.59 more focused and efficient by:

- Providing greater flexibility to licensees, primarily by allowing changes that have minimal safety impact to be made without prior NRC approval

- Clarifying the threshold for "screening out" changes that do not require full evaluation under 10 CFR 50.59, primarily by adoption of key definitions.

Proposed changes, tests, and experiments that satisfy the definitions and one or more of the criteria in the rule must be reviewed and approved by the NRC before implementation.

NEED FOR CHANGE

As indicated above, the Bases Control Program required by TS 5.5.14 allows licensees to make changes to the Bases without NRC approval provided the change does not involve a change to the updated FSAR or Bases that involves an unreviewed safety question as defined in 10 CFR 50.59. With the revisions to 10 CFR 50.59, the definition of unreviewed safety question was eliminated. Therefore, the TS should be revised consistent with the revision to 10 CFR 50.59.

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PROPOSED CHANGE

The proposed change revises TS 5.5.14b.2. to state: "a change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59."

JUSTIFICATION

The NRC amended its regulations concerning the authority for licensees of production or utilization facilities, such as nuclear reactors, and independent spent fuel storage facilities, and for certificate holders for spent fuel storage casks, to make changes to the facility or procedures, or to conduct tests or experiments, without prior NRC approval. The final rule clarifies the specific types of changes, tests, and experiments conducted at a licensed facility or by a certificate holder that require evaluation, and revises the criteria that licensees and certificate holders must use to determine when NRC approval is needed before such changes, tests, or experiments can be implemented. The final rule also adds definitions for terms that have been subject to differing interpretations , and reorganizes the rule language for clarity.

10 CFR 50.59 was revised to state, in part:

(c)(1) A licensee may make changes in the facility as described in the final safety analysis report (as updated), make changes in the procedures as described in the final safety analysis report (as updated), and conduct tests or experiments not described in the final safety analysis report (as updated) without obtaining a license amendment pursuant to 10 CFR 50.90 only if:

(i) A change to the technical specifications incorporated in the license is not required, and
(ii) The change, test, or experiment does not meet any of the criteria in paragraph (c)(2) of this section.

(2) A licensee shall obtain a license amendment pursuant to 10 CFR 50.90 prior to implementing a proposed change, test, or experiment if the change, test, or experiment would:

(i) Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the final safety analysis report (as updated);
(ii) Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component (SSC) important to safety previously evaluated in the final safety analysis report (as updated);
(iii) Result in more than a minimal increase in the consequences of an accident previously evaluated in the final safety analysis report (as updated);
(iv) Result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the final safety analysis report (as updated);
(v) Create a possibility for an accident of a different type than any previously evaluated in the final safety analysis report (as updated);
(vi) Create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the final safety analysis report (as updated);
(vii) Result in a design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered: or
(viii) Results in a departure from a method of evaluation described in the FSAR (as updated) used in establishing the design bases or in the safety analyses.

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Determination of No Significant Hazards Considerations

A change is proposed to the Improved Technical Specifications, NUREGs 1430 - 1434, TS 5.5.14, Technical Specification (TS) Bases Control Program, to provide consistentency the changes to 10 CFR 50.59 published in the Federal Register (Volume 64, Number 191) dated October 4, 1999.

In accordance with the criteria set forth in 10 CFR 50.92, the Industry has evaluated these proposed Improved Technical Specification changes and determined they do not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

The proposed change deletes the reference to unreviewed safety question as defined in 10 CFR 50.59. Deletion of the definition of unreviewed safety question was approved by the NRC with the revisions to 10 CFR 50.59. Consequently, the probability of an accident previously evaluated is not significantly increased. Changes to the TS Bases are still evaluated in accordance with 10 CFR 50.59. As a result, the consequences of any accident previously evaluated are not significantly affected. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously analyzed?

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in the margin of safety?

The proposed change will not reduce a margin of safety because it has no direct effect on any safety analyses assumptions. Changes to the TS Bases that result in meeting the criteria in paragraph (c)(2) will still require NRC approval pursuant to 10 CFR 50.59. This change is administrative in nature based on the amending of 10 CFR 50.59. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Revision History

OG Revision 0 Revision Status: Active

Revision Proposed by: Wolf Creek

Revision Description:
Original Issue

Owners Group Review Information

Date Originated by OG: 08-Mar-00

Owners Group Comments:
(No Comments)

Owners Group Resolution: Approved Date: 08-Mar-00

TSTF Review Information

TSTF Received Date: 08-Mar-00 Date Distributed for Review: 08-Mar-00

OG Review Completed: ☑ BWOG ☑ WOG ☑ CEOG ☑ BWROG

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OG Revision 0 Revision Status: Active

TSTF Comments:

(No Comments)

TSTF Resolution: Approved Date: 08-Mar-00

NRC Review Information

NRC Received Date: 13-Mar-00

Final Resolution: NRC Approves Final Resolution Date: 16-Jun-00

Affected Technical Specifications

5.5.14	Technical Specifications (TS) Bases Control Program	NUREG(s)- 1430 1431 1432 Only
5.5.11	Technical Specifications (TS) Bases Control Program	NUREG(s)- 1433 1434 Only

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INSERT

A change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

TSTF-364

5.5 Programs and Manuals

5.5.13 Diesel Fuel Oil Testing Program (continued)

2. a flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and

3. a clear and bright appearance with proper color;

b. Other properties for ASTM 2D fuel oil are within limits within 30 days following sampling and addition to storage tanks; and

c. Total particulate concentration of the fuel oil is ≤ 10 mg/l when tested every 31 days in accordance with ASTM D-2276, Method A-2 or A-3.

5.5.14 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not involve either of the following:

1. A change in the TS incorporated in the license; or

Insert

2. A change to the updated FSAR or Bases that involves an unreviewed safety question as defined in 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR.

d. Proposed changes that meet the criteria of 5.5.14b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

WOG-ED-24 changed "do not involve" to "do not require"

(continued)

TSTF-364

5.5 Programs and Manuals

5.5.13 Diesel Fuel Oil Testing Program (continued)

2. a flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and

3. a clear and bright appearance with proper color;

b. Other properties for ASTM 2D fuel oil are within limits within 31 days following sampling and addition to storage tanks; and

c. Total particulate concentration of the fuel oil is ≤ 10 mg/l when tested every 31 days in accordance with ASTM D-2276, Method A-2 or A-3.

5.5.14 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not involve either of the following:

1. a change in the TS incorporated in the license; or

Insert

2. ~~a change to the updated FSAR or Bases that involves an unreviewed safety question as defined in 10 CFR 50.59.~~

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR.

d. Proposed changes that meet the criteria of Specification 5.5.14b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

WOG-ED-24 changed "do not involve" to "do not require"

(continued)

TSTF-364

5.5 Programs and Manuals (continued)

5.5.13 Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:

   1. An API gravity or an absolute specific gravity within limits,

   2. A flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and

   3. A clear and bright appearance with proper color;

b. Other properties for ASTM 2D fuel oil are within limits within 31 days following sampling and addition to storage tanks; and

c. Total particulate concentration of the fuel oil is ≤ 10 mg/l when tested every 31 days in accordance with ASTM D-2276, Method A-2 or A-3.

5.5.14 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not involve either of the following:

   A change in the TS incorporated in the license; or

   Insert → A change to the updated FSAR or Bases that involves an unreviewed safety question as defined in 10 CFR 50.59.

WOG-ED-24 changed "do not involve" to "do not require"

(continued)

TSTF-364

5.5 Programs and Manuals (continued)

5.5.10 Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:

   1. an API gravity or an absolute specific gravity within limits,

   2. a flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and

   3. a clear and bright appearance with proper color;

b. Other properties for ASTM 2D fuel oil are within limits within 31 days following sampling and addition to storage tanks; and

c. Total particulate concentration of the fuel oil is $\leq$ 10 mg/l when tested every 31 days in accordance with ASTM D-2276, Method A-2 or A-3.

5.5.11 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not involve either of the following:

   1. a change in the TS incorporated in the license; or

   2. a change to the updated FSAR or Bases that involves an unreviewed safety question as defined in 10 CFR 50.59.

Insert

WOG-ED-24 changed "do not involve" to "do not require"

(continued)

TSTF 364

5.5 Programs and Manuals

5.5.10 Diesel Fuel Oil Testing Program (continued)

a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:

1. an API gravity or an absolute specific gravity within limits,

2. a flash point and kinematic viscosity within limits for ASTM 2D fuel oil,

3. a clear and bright appearance with proper color;

b. Other properties for ASTM 2D fuel oil are within limits within 31 days following sampling and addition to storage tanks; and

c. Total particulate concentration of the fuel oil is ≤ 10 mg/l when tested every 31 days in accordance with ASTM D-2276, Method A-2 or A-3.

5.5.11 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not involve either of the following:

1. a change in the TS incorporated in the license; or

Insert

2. a change ~~to~~ the updated ~~FSAR~~ or Bases that ~~involves an~~ unreviewed safety question as defined in ~~10~~ CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR.

d. Proposed changes that meet the criteria of 5.5.11b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without

WOG-ED-24 changed "do not involve" to "do not require"

(continued)

Technical Specification Task Force
Improved Standard Technical Specifications Change Traveler

Revise PTLR Definition and References in ISTS 5.6.6, RCS PTLR

NUREGs Affected: ☑ 1430 ☑ 1431 ☐ 1432 ☑ 1433 ☑ 1434

Classification: 1) Technical Change Recommended for CLIIP?: No

Correction or Improvement: (Unassigned)

Industry Contact: Steve Wideman, (620) 364-4037, stwidem@wcnoc.com

The definition of PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) is revised to delete the reference to the Specifications containing the limits specified in the PTLR.

The requirement in ITS 5.6.6, "Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)," to identify the NRC staff approval document by date is revised to allow the Topical Report(s) to be identified by number and title. A requirement is added to the Reviewers Note to specify the complete citation in the PTLR for each Topical Report, include the report number, title, revision, date, and any supplements.

BACKGROUND

NRC Generic Letter 96-06, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," dated January 31, 1996, allows licensees to relocate the pressure temperature (P/T) limit curves from their plant technical specifications (TS) to a pressure temperature limits report (PTLR) or a similar document. The Low Temperature Overpressure Protection (LTOP) System limits were also allowed to be relocated to the same document. The methodology used to determine the P/T and LTOP System limit parameters must comply with the specific requirements of Appendices G and H to Part 50 of Title 10 of the Code of Federal Regulations (10 CFR), be documented in an NRC approved topical report or in a plant-specific submittal, and be incorporated by reference into the TS. Subsequent changes in the methodology must be approved by a license amendment.

Similar changes to NUREG-1432, Rev. 2, are addressed in TSTF-408.

PROPOSED CHANGE

1. The definition of PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) is revised to delete the reference to the Specifications containing the limits specified in the PTLR.

2. The requirement in ITS 5.6.6, "Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)," to identify the NRC staff approval document by date is revised to allow the Topical Report(s) to be identified by number and title. A requirement is added to the Reviewers Note to specify the complete citation in the PTLR for each Topical Report, include the report number, title, revision, date, and any supplements.

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JUSTIFICATION

1. The definition of PTLR identifies the specifications in which the pressure and temperature limits are addressed. Specification 5.6.6a. requires that the individual specifications that address RCS pressure and temperature limits be referenced. The proposed changes to the definition eliminate duplication between the definition of PTLR and Section 5.6.6.

2. The revision to ITS 5.6.6 to allow the Topical Reports to be identified by number and title would allow licensees to use current Topical Reports to support limits in the PTLR without having to submit an amendment to facility operating license every time the Topical Report is revised. The PTLR would provide specific information identifying the particular approved Topical Reports used to determine the P/T limits or LTOP System limits. This still provides the assurance that only the approved versions of the referenced Topical Reports will be used for the determination of the P/T limits or LTOP System limits since the complete citation will be provided in the PTLR. This proposed change is consistent with TSTF-363, "Revise Topical Report references in ITS 5.6.5, COLR," which was approved by the NRC on April 13, 2000.

The requirement to operate within the limits in the PTLR is specified in and controlled by the technical specifications. Only the figures, values, and parameters associated with the P/T limits and LTOP setpoints are relocated to the PTLR. The methodology for their development must be reviewed and approved by the NRC. The proposed changes do not change the requirements associated with the review and approval of the methodology or the requirement to operate within the limits specified in the PTLR.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION

In accordance with the criteria set forth in 10 CFR 50.92, the proposed changes to NUREG-1431 have been evaluated and determined they do not represent a significant hazards consideration. The following is provided in support of this conclusion:

Standard I - Involves a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated

The proposed changes to reference only the Topical Report Number and title do not alter the use of the analytical methods used to determine the P/T limits or LTOP setpoints that have been reviewed and approved by the NRC. This method of referencing Topical Reports would allow the use of current Topical Reports to support limits in the PTLR without having to submit an amendment to the operating license. Implementation of revisions to Topical Reports would still be reviewed in accordance with 10 CFR 50.59 and where required receive NRC review and approval.

The proposed changes do not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, or configuration of the facility or the manner in which the plant is operated and maintained. The proposed changes do not alter or prevent the ability of structures, systems, and components (SSCs) from performing their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits. The proposed changes do not affect the source term, containment isolation, or radiological release assumptions used in evaluating the radiological consequences of an accident previously evaluated. Further, the proposed changes do not increase the types or amounts of radioactive effluent that may be released offsite, nor significantly increase individual or cumulative occupational/public radiation exposures. The proposed changes are consistent with safety analysis assumptions and resultant consequences.

Therefore, it is concluded that this change does not increase the probability of occurrence of an accident previously evaluated.

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Standard II - Create the Possibility of a New or Different Kind of Accident from any Previously Evaluated

The proposed changes to reference only the Topical Report Number and title do not alter the use of the analytical methods used to determine the P/T limits or LTOP setpoints that have been reviewed and approved by the NRC. This method of referencing Topical Reports would allow the use of current Topical Reports to support limits in the PTLR without having to submit an amendment to the operating license. Implementation of revisions to Topical Reports would still be reviewed in accordance with 10 CFR 50.59 and where required receive NRC review and approval.

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 or eliminate any existing 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 change does not create the possibility of a new or different kind of accident from any previously evaluated.

Standard III - Involve a Significant Reduction in the Margin of Safety

The proposed changes to reference only the Topical Report Number and title do not alter the use of the analytical methods used to determine the P/T limits or LTOP setpoints that have been reviewed and approved by the NRC. This method of referencing Topical Reports would allow the use of current Topical Reports to support limits in the PTLR without having to submit an amendment to the operating license. Implementation of revisions to Topical Reports would still be reviewed in accordance with 10 CFR 50.59 and where required receive NRC review and approval. The proposed changes do not alter the manner in which safety limits, limiting safety system settings or limiting conditions for operation are determined. The setpoints at which protective actions are initiated are not altered by the proposed changes. Sufficient equipment remains available to actuate upon demand for the purpose of mitigating an analyzed event.

Therefore, it is concluded that this change does not involve a significant reduction in the margin of safety.

Revision History

OG Revision 0 Revision Status: Active

Revision Proposed by: Wolf Creek

Revision Description:
Original Issue

Owners Group Review Information

Date Originated by OG: 02-May-01

Owners Group Comments:
(No Comments)

Owners Group Resolution: Approved Date: 02-May-01

TSTF Review Information

TSTF Received Date: 02-May-01 Date Distributed for Review: 02-May-01

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OG Revision 0 Revision Status: Active

OG Review Completed: ☑ BWOG ☑ WOG ☑ CEOG ☑ BWROG

TSTF Comments:

(No Comments)

TSTF Resolution: Approved Date: 02-May-01

NRC Review Information

NRC Received Date: 19-Sep-01

NRC Comments: Date of NRC Letter: 21-Mar-02

Safety Evaluation included in NRC 3/21/2002 letter.

Final Resolution: NRC Approves Final Resolution Date: 21-Mar-02

Affected Technical Specifications

1.0	Definitions - PTLR
5.6.6	Reactor Coolant System PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

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TSTF-419

INSERT 1

[Identify the Topical Report(s) by number and title or identify the NRC Safety Evaluation for a plant specific methodology by NRC letter and date. The PTLR will contain the complete identification for each of the TS referenced Topical Reports used to prepare the PTLR (i.e., report number, title, revision, date, and any supplements).]

1.1 Definitions

MASTER RELAY TEST	A MASTER RELAY TEST shall consist of energizing each required master relays in the channel required for channel OPERABILITY and verifying the OPERABILITY of each required master relay. The MASTER RELAY TEST shall include a continuity check of each associated required slave relay. The MASTER RELAY TEST may be performed by means of any series of sequential, overlapping, or total steps.
MODE	A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.
OPERABLE - OPERABILITY	A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
PHYSICS TESTS	PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are: a. Described in Chapter [14, Initial Test Program] of the FSAR, b. Authorized under the provisions of 10 CFR 50.59, or c. Otherwise approved by the Nuclear Regulatory Commission.
PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates and the LTOP arming temperature, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.6. ~~Plant operation within these operating~~

1.1 Definitions

PRESSURE AND TEMPERATURE LIMITS REPORT (continued)

~~limits is addressed in LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," and LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."~~

QUADRANT POWER TILT RATIO (QPTR)	QPTR shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of [2893] 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 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. However, with all RCCAs verified fully inserted by two independent means, it is not necessary to account for a stuck RCCA in the SDM calculation. With any RCCA not capable of being fully inserted, the reactivity worth of the RCCA must be accounted for in the determination of SDM, and b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the [nominal zero power design level].

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (continued)

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

[Identify the Topical Report(s) by number and title or identify the staff Safety Evaluation Report for a plant specific methodology by NRC letter and date. The COLR will contain the complete identification for each of the TS referenced topical reports used to prepare the COLR (i.e., report number, title, revision, date, and any supplements).]

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

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

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heat up, cooldown, low temperature operation, criticality, and hydrostatic testing, LTOP arming, and PORV lift settings as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

[The individual specifications that address RCS pressure and temperature limits must be referenced here.]

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

INSERT 1

~~[Identify the NRC staff approval document by date.]~~

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

1.1 Definitions

OPERABLE - OPERABILITY (continued)

perform its specified safety function(s) are also capable of performing their related support function(s).

PHYSICS TESTS

PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation.

These tests are:

a. Described in Chapter [14, Initial Test Program] of the FSAR,

b. Authorized under the provisions of 10 CFR 50.59, or

c. Otherwise approved by the Nuclear Regulatory Commission.

PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.6. ~~Plant operation within these operating limits is addressed in LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," and LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."~~

QUADRANT POWER TILT (QPT)

QPT shall be defined by the following equation and is expressed as a percentage of the Power in any Core Quadrant (P_{quad}) to the Average Power of all Quadrants (P_{avg}).

$$QPT = 100 [(P_{quad} / P_{avg}) - 1]$$

RATED THERMAL POWER (RTP)

RTP shall be a total reactor core heat transfer rate to the reactor coolant of [2544] MWt.

REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME

The RPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until electrical power is interrupted at the control rod drive trip breakers. The response time may be measured by means of any series of sequential,

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (continued)

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

[Identify the Topical Report(s) by number and title or identify the staff Safety Evaluation Report for a plant specific methodology by NRC letter and date. The COLR will contain the complete identification for each of the TS referenced topical reports used to prepare the COLR (i.e., report number, title, revision, date, and any supplements).]

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

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

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heat up, cooldown, low temperature operation, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

[The individual specifications that address RCS pressure and temperature limits must be referenced here.]

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

INSERT 1 ~~[Identify the NRC staff approval document by date.]~~

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

1.1 Definitions

MINIMUM CRITICAL POWER RATIO (MCPR)	The MCPR shall be the smallest critical power ratio (CPR) that exists in the core [for each class of fuel]. The CPR is that power in the assembly that is calculated by application of the appropriate correlation(s) to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.
MODE	A MODE shall correspond to any one inclusive combination of mode switch position, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.
OPERABLE - OPERABILITY	A system, subsystem, division, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, division, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
PHYSICS TESTS	PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are: a. Described in Chapter [14, Initial Test Program] of the FSAR, b. Authorized under the provisions of 10 CFR 50.59, or c. Otherwise approved by the Nuclear Regulatory Commission.
PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.6. ~~Plant operation within these operating limits is addressed in LCO 3.4.10, "RCS Pressure and Temperature (P/T) Limits."~~

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (continued)

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

[Identify the Topical Report(s) by number and title or identify the staff Safety Evaluation Report for a plant specific methodology by NRC letter and date. The COLR will contain the complete identification for each of the TS referenced topical reports used to prepare the COLR (i.e., report number, title, revision, date, and any supplements).]

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

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

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heat up, cooldown, low temperature operation, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

[The individual specifications that address RCS pressure and temperature limits must be referenced here.]

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

INSERT 1

~~[Identify the NRC staff approval document by date.]~~

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

1.1 Definitions

MINIMUM CRITICAL POWER RATIO (MCPR)	The MCPR shall be the smallest critical power ratio (CPR) that exists in the core [for each class of fuel]. The CPR is that power in the assembly that is calculated by application of the appropriate correlation(s) to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.
MODE	A MODE shall correspond to any one inclusive combination of mode switch position, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.
OPERABLE - OPERABILITY	A system, subsystem, division, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, division, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
PHYSICS TESTS	PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are: a. Described in Chapter [14, Initial Test Program] of the FSAR, b. Authorized under the provisions of 10 CFR 50.59, or c. Otherwise approved by the Nuclear Regulatory Commission.
PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.6. ~~Plant operation within these operating limits is addressed in LCO 3.4.10, "RCS Pressure and Temperature (P/T) Limits."~~

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (continued)

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

[Identify the Topical Report(s) by number and title or identify the staff Safety Evaluation Report for a plant specific methodology by NRC letter and date. The COLR will contain the complete identification for each of the TS referenced topical reports used to prepare the COLR (i.e., report number, title, revision, date, and and any supplements).]

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

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

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heatup, cooldown, low temperature operation, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

[The individual specifications that address RCS pressure and temperature limits must be referenced here.]

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

INSERT 1

~~[Identify the NRC staff approval document by date.]~~

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.