ENCLOSURE 2

VOLUME 16

ST. LUCIE PLANT
UNIT 1 AND UNIT 2

IMPROVED TECHNICAL SPECIFICATIONS CONVERSION

ITS CHAPTER 5.0
ADMINISTRATIVE CONTROLS

Revision 0

LIST OF ATTACHMENTS

1. ITS Section 5.1, Responsibility
2. ITS Section 5.2, Organization
3. ITS Section 5.3, Unit Staff Qualifications
4. ITS Section 5.4, Procedures
5. ITS Section 5.5, Programs and Manuals
6. ITS Section 5.6, Reporting Requirements
7. ITS Section 5.7, High Radiation Area
8. Relocated/Deleted Current Technical Specifications (CTS)

ATTACHMENT 1

ITS Section 5.1, Responsibility

Current Technical Specifications (CTS) Markup
and Discussion of Changes (DOCs)

ITS

A01

The plant manager or his designee shall approve, prior to implementation, each proposed test, experiment or modification to systems or equipment that affect nuclear safety.

ITS 5.1

M01

6.0 ADMINISTRATIVE CONTROLS

5.2 **6.1 RESPONSIBILITY**

5.1.1 6.1.1 The plant manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

the

5.1.2 6.1.2 The Shift Supervisor shall be responsible for the control room command function. During any absence of the Shift Supervisor from the control room while ~~either~~ unit is in MODE 1, 2, 3, or 4, an individual with an active Senior Reactor Operator (SRO) license shall be designated to assume the control room command function. During any absence of the Shift Supervisor from the control room while ~~both units are~~ in MODE 5 or 6, an individual with an active SRO license or Reactor Operator license shall be designated to assume the control room command function.

L01

the unit is

6.2 ORGANIZATION

See ITS 5.2

ONSITE AND OFFSITE ORGANIZATION

6.2.1 An onsite and an offsite organization shall be established for unit operation and corporate management. This onsite and offsite organizations shall include the positions for activities affecting the safety of the nuclear power plant.

a. Lines of authority, responsibility and communication shall be established and defined from the highest management levels through intermediate levels to and including all operating organization positions. Those relationships shall be documented and updated, as appropriate, in the form of organizational charts. These organizational charts will be documented in the Quality Assurance Topical Report and updated in accordance with 10 CFR 50.54(a)(3). The plant-specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications shall be documented in the UFSAR or the Quality Assurance Topical Report.

b. A specified corporate officer shall be responsible for overall plant nuclear safety. This individual shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support in the plant so that continued nuclear safety is assured.

c. The plant manager shall be responsible for overall safe operation and shall have control over those onsite resources necessary for safe operation and maintenance of the plant.

d. Although the individuals who train the operating staff and those who carry out the quality assurance functions may report to the appropriate manager onsite, they shall have sufficient organizational freedom to be independent from operating pressures.

e. Although health physics individuals may report to any appropriate manager onsite, for matters relating to radiological health and safety of employees and the public, the radiation protection manager shall have direct access to that onsite individual having responsibility for overall unit management. Health physics personnel shall have the authority to cease any work activity when worker safety is jeopardized or in the event of unnecessary personnel radiation exposures.

ITS

ITS 5.1

A01

The plant manager or his designee shall approve, prior to implementation, each proposed test, experiment or modification to systems or equipment that affect nuclear safety.

M01

6.0 ADMINISTRATIVE CONTROLS

5.2 **6.1 RESPONSIBILITY**

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L01

the unit is

6.2 ORGANIZATION

See ITS 5.2

ONSITE AND OFFSITE ORGANIZATION

6.2.1 An onsite and an offsite organization shall be established for unit operation and corporate management. This onsite and offsite organization shall include the positions for activities affecting the safety of the nuclear power plant.

a. Lines of authority, responsibility and communication shall be established and defined from the highest management levels through intermediate levels to and including all operating organization positions. Those relationships shall be documented and updated, as appropriate, in the form of organizational charts. These organizational charts will be documented in the Quality Assurance Topical Report and updated in accordance with 10 CFR 50.54(a)(3). The plant-specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications shall be documented in the UFSAR or the Quality Assurance Topical Report.

b. A specified corporate officer shall be responsible for overall plant nuclear safety. This individual shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support in the plant so that continued nuclear safety is assured.

c. The plant manager shall be responsible for overall unit safe operation and shall have control over those onsite resources necessary for safe operation and maintenance of the plant.

d. Although the individuals who train the operating staff and those who carry out the quality assurance functions may report to the appropriate manager onsite, they shall have sufficient organizational freedom to be independent from operating pressures.

e. Although health physics individuals may report to any appropriate manager onsite, for matters relating to radiological health and safety of employees and the public, the radiation protection manager shall have direct access to that onsite individual having responsibility for overall unit management. Health physics personnel shall have the authority to cease any work activity when worker safety is jeopardized or in the event of unnecessary personnel radiation exposures.

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG - 1432, Rev. 5.0, "Standard Technical Specifications - Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

M01 CTS 6.1.1 provides requirements related to responsibility of the plant manager but does not contain any information concerning the plant manager's (or designee's) role in plant activities that affect nuclear safety. ITS 5.1.1 requires the plant manager or his designee to approve, prior to implementation, each proposed test, experiment, or modification to systems or equipment that affect nuclear safety. This changes the CTS by requiring the plant manager or his designee to approve certain activities to systems and equipment, prior to implementation, that affect nuclear safety.

The purpose of CTS 6.1.1 is to provide specific responsibilities for the plant manager. The proposed change adds a requirement for the plant manager (or his designee) to approve, prior to implementation, proposed tests, experiments, and modifications to systems or equipment that affect nuclear safety. The addition of this responsibility will not alter any of the existing responsibilities. The requirement is being added to ensure the plant manager is cognizant of those activities to plant systems or equipment that can potentially affect nuclear safety. This change is designated as more restrictive because it adds an additional responsibility for the plant manager in the administrative controls chapter of the ITS.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

L01 *(Category 1 – Relaxation of LCO Requirements)* CTS 6.1.2 requires an individual with an active Senior Reactor Operator (SRO) license to provide control room command function during any absence of the Shift Supervisor from the control room while either unit is in MODE 1, 2, 3, or 4. An individual with an active Reactor Operator (RO) license may also assume the control room command function during any absence of the Shift Supervisor from the control room while both units are in MODE 5 or 6. ITS 5.1.2 provides similar requirements except an individual with an active RO license may assume the control room command function during the absence of the Shift Supervisor while the unit associated with the control room is in MODE 5 or 6 irrespective of the operational status of the other unit. This changes the CTS by relaxing the control room command function requirement from requiring an SRO to assume these responsibilities while either unit is in MODE 1, 2, 3, or 4 to requiring the SRO to assume the control room command function only when the associated unit is in MODE 1, 2, 3, or 4 and allowing the RO to assume these responsibilities when the unit associated with the control room is in MODE 5 or 6.

The purpose of CTS 6.1.2 is to ensure compliance with the 10 CFR 50.54(m)(iii) requirement that, when a unit is in an operational mode other than cold shutdown or refueling, a person holding a senior operator license for the unit is present at the controls at all times and that when the unit is fueled, including cold shutdown and refueling operational modes, either a senior licensed operator or a licensed operator is present at the controls at all times. Each PSL unit is controlled from a separate control room. The CTS requirement is written for a two unit site with a common control room to ensure an SRO maintains the control room command function when either unit is in MODES 1, 2, 3, and 4. However, with separate and independent control rooms, the assignment of the control room command function is based on the operational status of the associated unit and is irrespective of the operational status of other unit.

This change is acceptable because the ITS continues to assure that a senior licensed operator (i.e., Shift Supervisor or SRO) maintains control room command function in the control room when the associated unit is in MODE 1, 2, 3, or 4 and either a senior licensed operator or a licensed operator (i.e., RO) maintains control room command function in the control room when the associated unit is shutdown and still fueled (i.e., in MODES 5 and 6). Therefore, ITS 5.1.2 will continue to satisfy the requirements of 10 CFR 50.54(m)(iii). Additionally, this change does not alter the minimum emergency response organizational requirements of the PSL emergency plan and therefore, does not impact PSL compliance with 10 CFR 50.45, Emergency plan, or 10 CFR 50, Appendix E, Emergency Planning and Preparedness for Production and Utilization Facilities. This change is designated as less restrictive because less stringent requirements are being applied in the ITS than were applied in the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

5.0 ADMINISTRATIVE CONTROLS

6.1 ## 5.1 Responsibility

~~REVIEWER'S NOTES~~

~~1. Titles for members of the unit staff shall be specified by use of an overall statement referencing an ANSI Standard acceptable to the NRC staff from which the titles were obtained, or an alternative title may be designated for this position. Generally, the first method is preferable; however, the second method is adaptable to those unit staffs requiring special titles because of unique organizational structures.~~ 2

~~2. The ANSI Standard shall be the same ANSI Standard referenced in Section 5.3, Unit Staff Qualifications. If alternative titles are used, all requirements of these Technical Specifications apply to the position with the alternative title as apply with the specified title. Unit staff titles shall be specified in the Final Safety Analysis Report or Quality Assurance Plan. Unit staff titles shall be maintained and revised using those procedures approved for modifying/revising the Final Safety Analysis Report or Quality Assurance Plan.~~

6.1.1 5.1.1 The plant manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

DOC M01 The plant manager or his designee shall approve, prior to implementation, each proposed test, experiment or modification to systems or equipment that affect nuclear safety.

shift supervisor 3

6.1.2
DOC L01 5.1.2 The ~~[~~Shift Supervisor ~~(SS)]~~ shall be responsible for the control room command function. During any absence of the ~~[SS]~~ from the control room while the unit is in MODE 1, 2, 3, or 4, an individual with an active Senior Reactor Operator (SRO) license shall be designated to assume the control room command function. During any absence of the ~~[SS]~~ from the control room while the unit is in MODE 5 or 6, an individual with an active SRO license or Reactor Operator license shall be designated to assume the control room command function.

5.0 ADMINISTRATIVE CONTROLS

6.1 5.1 Responsibility

~~-----REVIEWER'S NOTES-----~~

~~1. Titles for members of the unit staff shall be specified by use of an overall statement referencing an ANSI Standard acceptable to the NRC staff from which the titles were obtained, or an alternative title may be designated for this position. Generally, the first method is preferable; however, the second method is adaptable to those unit staffs requiring special titles because of unique organizational structures.~~ 2

~~2. The ANSI Standard shall be the same ANSI Standard referenced in Section 5.3, Unit Staff Qualifications. If alternative titles are used, all requirements of these Technical Specifications apply to the position with the alternative title as apply with the specified title. Unit staff titles shall be specified in the Final Safety Analysis Report or Quality Assurance Plan. Unit staff titles shall be maintained and revised using those procedures approved for modifying/revising the Final Safety Analysis Report or Quality Assurance Plan.~~

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shift supervisor 3

6.1.2 DOC L01 5.1.2 The ~~[~~Shift Supervisor ~~(SS)]~~ shall be responsible for the control room command function. During any absence of the ~~[SS]~~ from the control room while the unit is in MODE 1, 2, 3, or 4, an individual with an active Senior Reactor Operator (SRO) license shall be designated to assume the control room command function. During any absence of the ~~[SS]~~ from the control room while the unit is in MODE 5 or 6, an individual with an active SRO license or Reactor Operator license shall be designated to assume the control room command function.

JUSTIFICATION FOR DEVIATIONS
ITS 5.1, RESPONSIBILITY

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal. Titles used in ITS Section 5.1 align with the titles used in ANSI Standard ANSI/ANS 3.1-1978, “Standard for Selection and Training of Personnel for Nuclear Power Plants,” and alternate titles are not used. ANSI/ANS 3.1-1978 is the same ANS Standard referenced in ITS Section 5.3 (CTS 6.3).

3. The ISTS contains bracketed information and/or values that are generic to Combustion Engineering vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 5.1, RESPONSIBILITY

There are no specific No Significant Hazards Considerations for this Section.

ATTACHMENT 2

ITS Section 5.2, Organization

Current Technical Specifications (CTS) Markup
and Discussion of Changes (DOCs)

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The plant manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Shift Supervisor shall be responsible for the control room command function. During any absence of the Shift Supervisor from the control room while either unit is in MODE 1, 2, 3, or 4, an individual with an active Senior Reactor Operator (SRO) license shall be designated to assume the control room command function. During any absence of the Shift Supervisor from the control room while both units are in MODE 5 or 6, an individual with an active SRO license or Reactor Operator license shall be designated to assume the control room command function.

See ITS 5.1

5.2

6.2 ORGANIZATION

ONSITE AND OFFSITE ORGANIZATION

5.2.1 6.2.1 ~~An o~~Onsite and ~~an~~ offsite organizations shall be established for unit operation and corporate management. The~~is~~ onsite and offsite organizations shall include the positions for activities affecting the safety of the nuclear power plant.

, respectively

5.2.1.a a. Lines of authority, responsibility and communication shall be established and defined from the highest management levels through intermediate levels to and including all operating organization positions. Those relationships shall be documented and updated, as appropriate, in ~~the form of~~ organizational charts. ~~These organizational charts will be documented in the Quality Assurance Topical Report and updated in accordance with 10 CFR 50.54(a)(3).~~ The plant-specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications shall be documented in the UFSAR or the Quality Assurance Topical Report.

, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation

A06

LA01

A02

5.2.1.c b. A specified corporate officer shall be responsible for overall plant nuclear safety. This individual shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support in the plant so that continued nuclear safety is assured.

5.2.1.b c. The plant manager shall be responsible for overall safe operation and shall have control over those onsite resources necessary for safe operation and maintenance of the plant.

; however

5.2.1.d d. ~~Although the i~~Individuals who train the operating staff and those who carry out the quality assurance functions may report to the appropriate manager onsite, they shall have sufficient organizational freedom to be independent from operating pressures.

5.2.1.d e. ~~Although h~~Health physics individuals may report to any appropriate manager onsite, ~~for matters relating to radiological health and safety of employees and the public, the radiation protection manager shall have direct access to that onsite individual having responsibility for overall unit management. Health physics personnel shall have the authority to cease any work activity when worker safety is jeopardized or in the event of unnecessary personnel radiation exposures.~~

; however, these individuals shall have sufficient freedom to ensure their operational independence.

LA02

6.0 ADMINISTRATIVE CONTROLS

5.2 **6.2 ORGANIZATION** (continued)

UNIT STAFF

5.2.2 6.2.2 The unit staff organization shall ~~meet the requirements of 10 CFR 50.54(m) and~~ include the following: A02

5.2.2.a a. A non-licensed operator shall be assigned to each reactor containing fuel and an additional non-licensed operator shall be assigned for each control room from which a reactor is operating in MODES 1, 2, 3, or 4. A ~~minimum~~ total of three non-licensed operators is required for both units when both units are in MODE~~S~~ 5 or 6. A01

5.2.2.b b. Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i) and 6.2.2.a and 6.2.2.e 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.

5.2.2.c c. A ~~health physics~~ radiation protection technician shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position. A03

5.2.2.d d. The operations manager or assistant operations manager shall hold an SRO license.

5.2.2.e e. An individual ~~(Shift Technical Advisor (STA))~~ shall provide advisory technical support to the unit operations shift crew in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. This individual shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift. ~~The STA position shall be manned in MODES 1, 2, 3, or 4~~. ~~by use of either a dedicated STA, a Shift Supervisor who meets the qualifications for the STA as required by Technical Specification 6.3.1, or an individual assigned to the unit with a Senior Reactor Operator's license who meets the qualifications for the STA as required by Technical Specification 6.3.1.~~ ~~If the STA position is filled by an STA qualified Shift Supervisor or dedicated STA, then the~~ individual may ~~fill the same position on Unit 2~~. A04 A04 A05 A04

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6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The plant manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Shift Supervisor shall be responsible for the control room command function. During any absence of the Shift Supervisor from the control room while either unit is in MODE 1, 2, 3, or 4, an individual with an active Senior Reactor Operator (SRO) license shall be designated to assume the control room command function. During any absence of the Shift Supervisor from the control room while both units are in MODE 5 or 6, an individual with an active SRO license or Reactor Operator license shall be designated to assume the control room command function.

See ITS 5.1

5.2

6.2 ORGANIZATION

ONSITE AND OFFSITE ORGANIZATION

5.2.1 6.2.1 ~~An o~~Onsite and ~~an~~ offsite organizations shall be established for unit operation and corporate management~~.~~, respectively. The~~is~~e onsite and offsite organizations shall include the positions for activities affecting the safety of the nuclear power plant.

5.2.1.a a. Lines of authority, responsibility and communication shall be established and defined from the highest management levels through intermediate levels to and including all operating organization positions. Those relationships shall be documented and updated, as appropriate, in ~~the form of~~ organizational charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. ~~These organizational charts will be documented in the Quality Assurance Topical Report~~ ~~and updated in accordance with 10 CFR 50.54(a)(3)~~. A06 LA01 A02

The plant-specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications shall be documented in the UFSAR or the Quality Assurance Topical Report.

5.2.1.c b. A specified corporate officer shall be responsible for overall plant nuclear safety. This individual shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support in the plant so that continued nuclear safety is assured.

5.2.1.b c. The plant manager shall be responsible for overall unit safe operation and shall have control over those onsite resources necessary for safe operation and maintenance of the plant.

5.2.1.d d. ~~Although the i~~Individuals who train the operating staff and those who carry out the quality assurance functions may report to the appropriate manager onsite, ; however they shall have sufficient organizational freedom to be independent from operating pressures.

5.2.1.d e. ~~Although h~~Health physics individuals may report to any appropriate manager onsite, ; however, these individuals shall have sufficient freedom to ensure their operational independence. ~~for matters relating to radiological health and safety of employees and the public, the radiation protection manager shall have direct access to that onsite individual having responsibility for overall unit management. Health physics personnel shall have the authority to cease any work activity when worker safety is jeopardized or in the event of unnecessary personnel radiation exposures.~~ LA02

6.0 ADMINISTRATIVE CONTROLS

5.2 **6.2 ORGANIZATION** (Continued)

UNIT STAFF

A02

5.2.2 6.2.2 The unit staff organization shall ~~meet the requirements of 10 CFR 50.54(m) and~~ include the following:

5.2.2.a a. A non-licensed operator shall be assigned to each reactor containing fuel and an additional non-licensed operator shall be assigned for each control room from which a reactor is operating in MODES 1, 2, 3, or 4. A ~~minimum~~ total of three non-licensed operators is required for both units when both units are in MODE~~S~~ 5 or 6. A01

5.2.2.b b. Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i) and 6.2.2.a and 6.2.2.e 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. A03

5.2.2.c c. A ~~health physics~~ radiation protection technician shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

5.2.2.d d. The operations manager or assistant operations manager shall hold an SRO license.

5.2.2.e e. An individual ~~(Shift Technical Advisor (STA))~~ shall provide advisory technical support to the unit operations shift crew in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. This individual shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift. ~~The STA position shall be manned in MODES 1, 2, 3, or 4~~ ~~by use of either a dedicated STA, a Shift Supervisor who meets the qualifications for the STA as required by Technical Specification 6.3.1, or an individual assigned to the unit with a Senior Reactor Operator's license who meets the qualifications for the STA as required by Technical Specification 6.3.1~~. ~~If the STA position is filled by an STA qualified Shift Supervisor or dedicated STA, then the individual may fill the same position on Unit 1~~. A04 A04 A05 A04

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ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications - Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 CTS 6.2.1 requires, in part, to update the organizational charts documented in the Quality Assurance Topical Report (QATR) in accordance with 10 CFR 50.54(a)(3). CTS 6.2.2 requires, in part, for the unit staff organization to meet the requirements of 10 CFR 50.54(m). The ITS does not include these statements. This changes the CTS by removing the duplicate requirement to comply with the requirements of 10 CFR 50.54. Each PSL facility operating license requires the licensee to be subject to the conditions specified in 10 CFR Section 50.54. Therefore, it is unnecessary to state organizational charts in the QATR be updated in accordance with 10 CFR 50.54(a)(3) and the unit staff meet the requirements of 10 CFR 50.54(m) because it is duplicative. This change is designated as administrative because it does not result in technical changes to the CTS.

A03 CTS 6.2.2.c requires, in part, that a health physics technician be on site when fuel is in the reactor. ITS 5.2.2.c requires a radiation protection technician to be on site when fuel is in the reactor. This changes the CTS by modifying the generic title of health physics technician with the more appropriate generic title of radiation protection technician. The purpose of CTS 6.2.2.c is to ensure there is an individual responsible for radiation protection on site when fuel is in the reactor. This change is consistent with the generic position in the ISTS and the current plant-specific title for these technicians. The generic title also aligns with the position title provided in ANSI/ANS 3.1-1978, “American National Standard for Selection and Training of Nuclear Power Plant Personnel.” This change is designated as administrative because it does not result in technical changes to the CTS.

A04 CTS 6.2.2.e requires an individual (Shift Technical Advisor (STA)) to provide advisory technical support to the unit operations shift crew, specifically be manned in MODES 1, 2, 3, or 4, and if the STA position is filled by an STA qualified Shift Supervisor or dedicated STA, then the individual may fill the same position on both units. ITS 5.2.2.e requires an individual to provide advisory technical support to the unit operations shift crew but is revised to eliminate reference to the STA position, when the position must be manned, and the details of who may fill STA role for both units. This changes the CTS by eliminating the inference that the STA role is a separate shift crew position instead of a function.

The purpose of the CTS requirement is to ensure engineering expertise is available on shift to provide advisory technical support. NRC Generic Letter

86-04 “Policy Statement on Engineering Expertise on Shift,” promulgated the policy statement regarding engineering expertise on shift and established the requirements for eliminating the separate STA position. The NRC policy statement offers the licensees two options for meeting the NUREG-0737, “Clarification of TMI Action Plan Requirements,” Enclosure 3, Item I.A.1.1 requirement regarding engineering expertise on shift and meeting licensed operator staffing requirements pursuant to 10 CFR 50.54(m)(2). Option 1 provides for elimination of the separate STA position by allowing licensees to combine one of the required SRO positions with the STA position into a dual-role (SRO/STA) function. Option 2 allows a licensee to continue use of an NRC-approved STA program while meeting licensed operator staffing requirements. Per the policy statement, the Commission encourages licensees to move toward the dual-role (SRO/STA) function, with the eventual goal of the shift supervisor serving in the dual role. Additionally, Item I.A.1.1 of NUREG-0737, Enclosure 3, allows the individual providing advisory technical support function to serve more than one unit at a multiunit site if qualified to perform the advisor function for the various units.

ITS 5.2.2.e eliminates the title of "Shift Technical Advisor (STA)," because the on shift advisory technical support function may be fulfilled by one or more of the other on-shift individuals. This change is necessary so that it does not imply that the STA and the shift supervisor must be different individuals. As a result of eliminating the STA “position,” it is unnecessary to state when the STA “position” must be manned. In addition, Item I.A.1.1 of NUREG-0737, Enclosure 3, allows the individual providing advisory technical support to serve more than one unit at a multiunit site if qualified to perform the advisor function for the various units and there is no regulatory requirement that conflicts with this statement. Therefore, it is unnecessary to restate this as an allowance in the Technical Specifications and does not prevent PSL from utilizing this allowance. This change is designated as administrative because an individual with engineering expertise (i.e., dedicated individual or dual role individual) will continue to be required on shift and this same individual may provide support to both units as allowed by NUREG-0737. Therefore, the change does not result in technical changes to the CTS.

A05 CTS 6.2.2.e, in part, includes a description of who specifically may provide advisory technical support to the unit operations shift crew: either a dedicated STA, a Shift Supervisor who meets the qualifications for the STA as required by Technical Specification 6.3.1, or an individual assigned to the unit with a Senior Reactor Operator's license who meets the qualifications for the STA as required by Technical Specification 6.3.1. ITS 5.2.2.e requires an individual to provide advisory technical support to the unit operations shift crew but is revised to eliminate the details of who is qualified to assume the duties. This changes the CTS by eliminating details regarding the individual assigned to provide technical advisory support to the unit operations shift crew.

The purpose of the CTS requirement is to ensure engineering expertise is available on shift to provide advisory technical support. NRC Generic Letter 86-04, promulgated the policy statement regarding engineering expertise on shift and established the requirements for eliminating the separate STA position. The NRC policy statement allowed for the elimination of the separate STA position by allowing licensees to combine one of the required SRO positions with the STA position into a dual-role (SRO/STA) function. The policy statement indicates that either a dedicated STA, a Shift Supervisor who meets the qualifications or an individual assigned to the unit with a Senior Reactor Operator's license who meets the qualifications may serve the dual-role function.

It is an ITS convention to not include cross-references to comply with other Specification requirements. Compliance with other Specifications is understood. Compliance with CTS 6.3.1 (ITS 5.3.1) is required and referencing this Specification is unnecessary. Refer to ITS 5.3, "Unit Staff Qualifications," for changes to CTS 6.3.1 requirements. This change is designated as administrative because an individual with engineering expertise and appropriate qualifications will continue to be required on shift. Therefore, the change does not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 (*Type 4 – Removal of LCO, SR, or other TS requirement to the TRM, UFSAR, ODCM, QAP, CLRT Program, IST Program, ISI Program, or Surveillance Frequency Control Program)* CTS 6.2.1.a requires, in part, to document the organizational charts in the QATR. ITS 5.2.1.a does not include this requirement. The requirement that organizational charts be documented in the QATR is moved to the Updated Final Safety Analysis Report (UFSAR). This changes the CTS by moving a requirement to the UFSAR. The removal of this requirement from the Technical Specifications is acceptable because this type of information is administrative and not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS retains the requirement to document and update, in organizational charts, the relationship of the management levels through intermediate levels to and including operating organization positions, as appropriate. Also, this change is acceptable because the removed information will be adequately controlled in the UFSAR. The UFSAR is controlled under 10 CFR 50.59 which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because requirements for meeting Technical Specifications are being relocated from the Technical Specifications to the UFSAR.

LA02 (*Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements*) CTS 6.2.1.e states that, although health physics individuals may report to any appropriate manager onsite, for matters relating to radiological health and safety of employees and the public, the radiation protection manager shall have direct access to that onsite individual having responsibility for overall unit management. Health physics personnel shall have the authority to cease any work activity when worker safety is jeopardized or in the event of unnecessary personnel radiation exposures. ITS 5.2.1.d provides a similar requirement but does not provide procedural detail related to organizational freedom for health physics individuals (e.g., radiation manager shall have direct access to the individual having responsibility for overall unit management and health physics individuals' authority to cease any work activity when worker safety is jeopardized). ITS 5.2.1.d states, in part, "The individuals who… carry out health physics… functions… shall have sufficient organizational freedom to ensure their independence from operating pressures." The procedural detail of what activities constitute operational independence for health physics individuals, including the radiation protection manager's direct access to overall unit management individual, is moved to the UFSAR. This changes the CTS by moving procedural detail related to health physics individuals' responsibilities to the UFSAR and replacing the information with a general statement. The removal of these details from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS retains the requirement for health physics individuals to have sufficient freedom to ensure their operational independence. Also, this change is acceptable because the removed information will be adequately controlled in the UFSAR. The UFSAR is controlled under 10 CFR 50.59 which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

<u>LESS RESTRICTIVE CHANGES</u>

None

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

5.0 ADMINISTRATIVE CONTROLS

6.2 5.2 Organization

6.2.1 5.2.1 Onsite and Offsite Organizations

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

6.2.1.a a. Lines of authority, responsibility, and communication shall be defined and established throughout highest management levels, intermediate levels, and all operating organization positions. These relationships shall be documented and updated, as appropriate, in organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements including the plant-specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications shall be documented in the ~~[FSAR/QA Plan]~~ Updated Final Safety Analysis Report or the Quality Assurance Topical Report, 2

6.2.1.c b. The plant manager shall be responsible for overall safe operation of the plant and shall have control over those onsite activities necessary for safe operation and maintenance of the plant,

6.2.1.b c. A specified corporate officer shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety, and 4

6.2.1.d
6.2.1.e d. The individuals who train the operating staff, carry out health physics functions, or perform quality assurance functions may report to the appropriate onsite manager; however, these individuals shall have sufficient organizational freedom to ensure their independence from operating pressures.

6.2.2 5.2.2 Unit Staff

The unit staff organization shall include the following:

6.2.2.a a. A non-licensed operator shall be assigned to each reactor containing fuel and an additional non-licensed operator shall be assigned for each control room from which a reactor is operating in MODES 1, 2, 3, or 4. A total of three non-licensed operators are required for both units when both units are in MODE 5 or 6.

~~-------------------------------------REVIEWER'S NOTE-------------------------------------~~
~~Two unit sites with both units shutdown or defueled require a total of three non-licensed operators for the two units.~~
~~--~~ 3

5.2 Organization

6.2.2 5.2.2 Unit Staff (continued)

6.2.2.b b. 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 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.

6.2.2.c c. A radiation protection technician shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

6.2.2.d d. The operations manager or assistant operations manager shall hold an SRO license.

6.2.2.e e. An individual shall provide advisory technical support to the unit operations shift crew in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. This individual shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift.

5.0 ADMINISTRATIVE CONTROLS

6.2 5.2 Organization

6.2.1 5.2.1 Onsite and Offsite Organizations

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

6.2.1.a a. Lines of authority, responsibility, and communication shall be defined and established throughout highest management levels, intermediate levels, and all operating organization positions. These relationships shall be documented and updated, as appropriate, in organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements including the plant-specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications shall be documented in the ~~[FSAR/QA Plan]~~, 2

Updated Final Safety Analysis Report or the Quality Assurance Topical Report

6.2.1.c b. The plant manager shall be responsible for overall safe operation of the plant and shall have control over those onsite activities necessary for safe operation and maintenance of the plant,

6.2.1.b c. A specified corporate officer shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety, and 4

functions

6.2.1.d
6.2.1.e d. The individuals who train the operating staff, carry out health physics, or perform quality assurance functions may report to the appropriate onsite manager; however, these individuals shall have sufficient organizational freedom to ensure their independence from operating pressures.

6.2.2 5.2.2 Unit Staff

The unit staff organization shall include the following:

6.2.2.a a. A non-licensed operator shall be assigned to each reactor containing fuel and an additional non-licensed operator shall be assigned for each control room from which a reactor is operating in MODES 1, 2, 3, or 4.

~~-----REVIEWER'S NOTE-----~~
~~Two unit sites with both units shutdown or defueled require a total of three non-licensed operators for the two units.~~ 3
~~-----~~

A total of three non-licensed operators are required for both units when both units are in MODE 5 or 6.

5.2 Organization

6.2.2 5.2.2 Unit Staff (continued)

6.2.2.b b. 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 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.

6.2.2.c c. A radiation protection technician shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

6.2.2.d d. The operations manager or assistant operations manager shall hold an SRO license.

6.2.2.e e. An individual shall provide advisory technical support to the unit operations shift crew in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. This individual shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift.

JUSTIFICATION FOR DEVIATIONS
ITS 5.2, ORGANIZATION

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to Combustion Engineering vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

3. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal. Since the PSL is a two unit site, ISTS 5.2.2.a includes an additional statement requiring a total of three non-licensed operators for both units when both units are in MODE 5 or 6 (i.e., shutdown) consistent with the CTS requirement. This requirement does not extend to when both units are defueled because the ISTS nor 10 CFR 50.54(m) require licensed operators assigned to the reactor or control room when the reactor is defueled. Therefore, it unnecessary to require non-licensed operators when both units are defueled.

4. Grammatical correction made to the ITS to clarify ISTS 5.5.2.d includes individuals required to carry out health physics “functions.”

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 5.2, ORGANIZATION

There are no specific No Significant Hazards Considerations for this Section.

ATTACHMENT 3

ITS Section 5.3, Unit Staff Qualifications

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

6.0 ADMINISTRATIVE CONTROLS

5.3

6.3 UNIT STAFF QUALIFICATIONS

5.3.1 6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI / ANS-3.1-1978 for comparable positions, except for~~:~~ the radiation protection manager.

~~(1)~~ ~~t~~The radiation protection manager ~~who~~ shall meet or exceed the qualifications of Regulatory Guide 1.8, Revision 1, September 1975~~,~~.

~~(2) the Shift Technical Advisor who shall have specific training in plant design and plant operating characteristics, including transients and accidents, and any of the following educational requirements:~~

L01

- ~~Bachelor's degree in engineering from an accredited institution; or~~
- ~~Professional Engineer's (PE) license obtained by successful completion of the PE examination; or~~
- ~~Bachelor's degree in engineering technology from an accredited institution, including course work in the physical, mathematical, or engineering sciences, or~~
- ~~Bachelor's degree in physical science from an accredited institution, including course work in the physical, mathematical, or engineering sciences.~~

LA01

(3) the Multi-Discipline Supervisors who shall meet or exceed the following requirements:

a. Education: Minimum of a high school diploma or equivalent.

b. Experience: Minimum of four years of related technical experience, which shall include three years power plant experience of which one year is at a nuclear power plant.

c. Training: Complete the Multi-Discipline Supervisor training program.

5.3.2 6.3.2 For the purpose of 10 CFR 55.4, a licensed senior reactor operator and a licensed reactor operator are those individuals who, in addition to meeting the requirements of ~~6.3.1~~ Specification 5.3.1, perform the functions described in 10 CFR 50.54(m).

~~6.4 DELETED~~

~~6.5 DELETED~~

~~6.0 ADMINISTRATIVE CONTROLS~~

~~DELETED~~

~~6.0 ADMINISTRATIVE CONTROLS~~

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~~6.6 DELETED~~

~~6.7 DELETED~~

6.0 ADMINISTRATIVE CONTROLS

5.3 6.3 UNIT STAFF QUALIFICATIONS

5.3.1 6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI / ANS-3.1-1978 for comparable positions, except for~~:~~ the radiation protection manager.

~~(1) t~~The radiation protection manager who shall meet or exceed the qualifications of Regulatory Guide 1.8, Revision 1, September 1975~~;~~.

~~(2) the Shift Technical Advisor who shall have specific training in plant design and plant operating characteristics, including transients and accidents, and any of the following educational requirements:~~

L01

- ~~Bachelor's degree in engineering from an accredited institution; or~~
- ~~Professional Engineer's (PE) license obtained by successful completion of the PE examination; or~~
- ~~Bachelor's degree in engineering technology from an accredited institution, including course work in the physical, mathematical, or engineering sciences, or~~
- ~~Bachelor's degree in physical science from an accredited institution, including course work in the physical, mathematical, or engineering sciences.~~

(3) the Multi-Discipline Supervisors who shall meet or exceed the following requirements: LA01

a. Education: Minimum of a high school diploma or equivalent.

b. Experience: Minimum of four years of related technical experience, which shall include three years power plant experience of which one year is at a nuclear power plant.

c. Training: Complete the Multi-Discipline Supervisor training program.

5.3.2 6.3.2 For the purpose of 10 CFR 55.4, a licensed senior reactor operator and a licensed reactor operator are those individuals who, in addition to meeting the requirements of ~~6.3.1~~ Specification 5.3.1, perform the functions described in 10 CFR 50.54(m).

~~6.0 ADMINISTRATIVE CONTROLS~~

~~6.4 DELETED~~

~~6.5 DELETED~~

~~6.0 ADMINISTRATIVE CONTROLS~~

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ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG - 1432, Rev. 5.0, "Standard Technical Specifications - Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 (*Type 4 – Removal of LCO, SR, or other TS requirement to the TRM, UFSAR, ODCM, QAP, CLRT Program, IST Program, ISI Program, or Surveillance Frequency Control Program)* CTS 6.3.1(3) provides minimum education, experience, and training requirements for multi-discipline supervisors. ITS 5.3.1 does not include these requirements. The minimum education, experience, and training requirements for multi-discipline supervisors are moved to the Updated Final Safety Analysis Report (UFSAR). This changes the CTS by moving an administrative requirement to the UFSAR. The removal of this requirement from the Technical Specifications is acceptable because this type of information is administrative and not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS retains minimum qualification requirements for members of the facility staff for comparable positions as defined in ANSI/ANS-3.1-1978 or Regulatory Guide 1.8, as appropriate. Also, this change is acceptable because the removed information will be adequately controlled in the UFSAR. The UFSAR is controlled under 10 CFR 50.59 which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because administrative requirements in Technical Specifications are being relocated from the Technical Specifications to the UFSAR.

LESS RESTRICTIVE CHANGES

L01 (*Category 1 – Relaxation of LCO Requirements*) CTS 6.3.1(2) provides specific training requirements and minimum educational requirements for the Shift Technical Advisor. ITS 5.3.1 does not include these requirements. This changes the CTS by deleting explicit administrative training and qualification requirements for the shift technical advisor function.

The purpose of CTS 6.3.1(2) is to ensure the individual that provides advisory technical support to the unit operations shift crew (i.e., shift technical advisor) meets minimum qualifications and training pursuant to NUREG-0737, "Clarification of TMI Action Plan Requirements," Enclosure 3, Item I.A.1.1 This change is acceptable because the Florida Power & Light Company Quality Assurance Topical Report requires training for positions identified in 10CFR50.120, which includes the shift technical advisor position, be accomplished according to programs accredited by the National Nuclear Accrediting Board of the National Academy for Nuclear Training that implements a systematic approach to training. The systematic approach to training process develops and maintains training programs to ensure individuals are trained and meet minimum educational requirements necessary to perform their required tasks. Therefore, an individual who provides the shift technical advisor function will continue to be adequately trained and qualified to provide engineering expertise on shift. This change is designated as less restrictive because less stringent administrative LCO requirements are being applied in the ITS than were applied in the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

CTS

5.0 ADMINISTRATIVE CONTROLS

6.3 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.~~ 2
~~----------~~

6.3.1 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]~~. 3

ANSI/ANS-3.1-1978 for comparable positions, except for the radiation protection manager. The radiation protection manager shall meet or exceed the minimum qualifications of Regulatory Guide 1.8, Revision 1, September 1975.

6.3.2 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 Specification 5.3.1, perform the functions described in 10 CFR 50.54(m).

5.0 ADMINISTRATIVE CONTROLS

6.3 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.~~ 2

6.3.1 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]~~. 3

6.3.2 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 Specification 5.3.1, perform the functions described in 10 CFR 50.54(m).

ANSI/ANS-3.1-1978 for comparable positions, except for the radiation protection manager. The radiation protection manager shall meet or exceed the minimum qualifications of Regulatory Guide 1.8, Revision 1, September 1975.

JUSTIFICATION FOR DEVIATIONS
ITS 5.3, UNIT STAFF QUALIFICATIONS

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal. ANSI/ANS-3.1-1978 and Regulatory Guide 1.8, Revision 1, September 1975 are documents acceptable to the NRC staff.

3. The ISTS contains bracketed information and/or values that are generic to Combustion Engineering vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 5.3, UNIT STAFF QUALIFICATIONS

There are no specific No Significant Hazards Considerations for this Section.

ATTACHMENT 4

ITS Section 5.4, Procedures

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

6.0 ADMINISTRATIVE CONTROLS

5.4 **6.8 PROCEDURES ~~AND PROGRAMS~~**

5.4.1 6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

5.4.1.a
5.4.1.b
a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, Revision 2, February 1978~~,~~. ~~and those~~ required for implementing the requirements of NUREG 0737.

b. The emergency operating procedures
and NUREG-0737, Supplement 1, as stated in Generic Letter 82-33
A02

~~b. Refueling operations.~~

~~c. Surveillance and test activities of safety-related equipment.~~
A03

~~d. Not Used.~~

~~e. Not Used.~~

~~f. Not Used.~~

~~g. PROCESS CONTROL PROGRAM implementation.~~ LA01

~~h. OFFSITE DOSE CALCULATION MANUAL implementation.~~ A04

assurance
5.4.1.c i. Quality ~~Control Program~~ for effluent monitoring, ~~using the guidance in Regulatory Guide 1.21, Revision 1, June 1974~~.

assurance
5.4.1.c j. Quality ~~Control Program~~ for environmental monitoring ~~using the guidance in Regulatory Guide 4.1, Revision 1, April 1975.~~ LA02

~~**6.8.2 DELETED**~~

Insert proposed Specification 5.4.1.d
M01

ADMINISTRATIVE CONTROLS

~~6.6 DELETED~~

~~6.7 DELETED~~

5.4 **6.8 PROCEDURES** ~~AND PROGRAMS~~

5.4.1 6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

5.4.1.a
5.4.1.b
a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, Revision 2, February 1978~~,~~. ~~and those~~ b. The emergency operating procedures required for implementing the requirements of NUREG 0737 and NUREG-0737, Supplement 1, as stated in Generic Letter 82-33.

A02

~~b. Refueling operations.~~

~~c. Surveillance and test activities of safety-related equipment.~~

A03

~~d. Not Used.~~

~~e. Not Used.~~

6.0 ADMINISTRATIVE CONTROLS

5.4 **6.8 PROCEDURES ~~AND PROGRAMS~~** (Continued)

~~f. Not Used.~~

~~g. PROCESS CONTROL PROGRAM implementation.~~ LA01

~~h. OFFSITE DOSE CALCULATION MANUAL implementation.~~ A04

assurance

5.4.1.c i. Quality ~~Control Program~~ for effluent monitoring~~, using the guidance in Regulatory Guide 1.21, Revision 1, June 1974.~~

assurance

5.4.1.c j. Quality ~~Control Program~~ for environmental monitoring ~~using the guidance in Regulatory Guide 4.1, Revision 1, April 1975.~~ LA02

Insert proposed Specification 5.4.1.d M01

~~**6.8.2 DELETED**~~

~~**6.8.3 DELETED**~~

DISCUSSION OF CHANGES
ITS 5.4, PROCEDURES

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG - 1432, Rev. 5.0, "Standard Technical Specifications - Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 CTS 6.8.1.a requires written procedures to be established, implemented, and maintained, in part, for those required for implementing the requirements of NUREG-0737. ITS 5.4.1.b provides a similar requirement specifically stating, “The emergency operating procedures required to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1, as stated in Generic Letter 82-33.” Supplement 1 to NUREG-0737 enclosed with Generic Letter 82-33 is part of the implementation of the Three Mile Island (TMI) action items of NUREG-0737. The purpose of the generic letter was to provide additional clarification regarding, in part, upgrade of emergency operating procedures associated with TMI Action Item I.C.1. As a result, the change clarifies the existing requirement to comply with NUREG-0737 and NUREG-0737, Supplement 1, as stated in Generic Letter 82-33, for emergency operating procedures. This change is designated as an administrative change and is acceptable because it does not result in a technical change to the CTS.

A03 CTS 6.8.1.b and c require written procedures be established, implemented, and maintained covering refueling operations and surveillance and test activities of safety-related equipment, respectively. ITS 5.4.1 does not explicitly require these activities but rather requires written procedures shall be established, implemented, and maintained to the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978 consistent with CTS 6.8.1.a. This changes the CTS by removing the specific wording of CTS 6.8.1.b and CTS 6.8.1.c. This change is considered administrative because the recommendations of Regulatory Guide 1.33, Revision 2, Appendix A, February 1978 require procedures for refueling operations and surveillance tests for safety related activities. Therefore, specifically citing these requirements individually is unnecessary. This change is designated as administrative because it does not result in a technical change to the CTS.

A04 CTS 6.8.1.h requires written procedures be established, implemented, and maintained for OFFSITE DOSE CALCULATION MANUAL (ODCM) implementation. ITS 5.4.1 requires procedures for various activities but does not specifically list the ODCM. This changes the CTS by removing the specific requirement for written procedures to implement the ODCM. This change is acceptable because implementing procedures for the ODCM are required by ITS 5.4.1.d. ITS 5.4.1.d (as described in DOC M01) requires that written procedures be established, implemented, and maintained for programs and manuals listed in ITS Section 5.5. ITS Section 5.5 includes the ODCM. Therefore, it is not necessary to specifically identify the ODCM in ITS 5.4.1. This change is designated as administrative because it does not result in a technical change to the CTS.

MORE RESTRICTIVE CHANGES

M01 ITS 5.4.1.d requires written procedures be established, implemented, and maintained for programs specified in Specification 5.5. The CTS does not include this requirement for any program except the OFFSITE DOSE CALCULATION MANUAL. This changes the CTS by adopting a new requirement for procedures to address programs described in ITS Section 5.5. The purpose of ITS 5.4.1.d is to ensure that written procedures are established, implemented, and maintained covering programs specified in ITS Section 5.5. This change is acceptable because it requires written procedures, including proper procedure control, to address programs required by ITS Section 5.5. This change is designated as more restrictive because it imposes new administrative requirements for procedures within the Technical Specifications.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 (*Type 4 – Removal of LCO, SR, or other TS requirement to the TRM, UFSAR, ODCM, QAP, CLRT Program, IST Program, or ISI Program*) CTS 6.8.1.g requires that written procedures for the PROCESS CONTROL PROGRAM (PCP) be established, implemented, and maintained. The ITS does not include these requirements. This changes the CTS by moving the requirements to the Updated Safety Analysis Report (UFSAR). The removal of these details, which are related to meeting Specification requirements, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety.

The PCP implements the requirements of 10 CFR 20, 10 CFR 61, and 10 CFR 71 and written procedures are necessary to ensure compliance with regulations. Regulations provide an adequate level of control for the affected requirements, and thus, inclusion of this requirement in the Technical Specifications is not necessary. Also, this change is acceptable because these details will be adequately controlled in the UFSAR. Any changes to the UFSAR are made under 10 CFR 50.59 or 10 CFR 50.71(e), which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because details for meeting Technical Specification and regulatory requirements are being removed from the Technical Specifications.

LA02 (*Type 4 – Removal of LCO, SR, or other TS requirement to the TRM, UFSAR, ODCM, QAP, CLRT Program, IST Program, or ISI Program*) CTS 6.8.1.i and j require written procedures be established, implemented, and maintained covering the Quality Control Program for effluent and environmental monitoring, respectively, "using the guidance in Regulatory Guide 1.21, Revision 1, 1974, and Regulatory Guide 4.1, Revision 1, April 1975." ITS 5.4.1.c does not include the Regulatory Guide references. This changes the CTS by moving the references to the regulatory guides to the UFSAR. The removal of these details, which are related to meeting Technical Specification requirements, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement for written procedures covering quality assurance for effluent and environmental monitoring. Also, this change is acceptable because these details will be adequately controlled in the UFSAR. Any changes to the UFSAR are made under 10 CFR 50.59 or 10 CFR 50.71(e), which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because references for meeting Technical Specification requirements are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

None

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

CTS

5.0 ADMINISTRATIVE CONTROLS

6.8 ## 5.4 Procedures

6.8.1 5.4.1 Written procedures shall be established, implemented, and maintained covering the following activities:

6.8.1.a a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978,

6.8.1.a b. The emergency operating procedures required to implement the requirements of NUREG-0737 and ~~to~~ NUREG-0737, Supplement 1, as stated in ~~[~~Generic Letter 82-33~~]~~, 2 3

6.8.1.i 6.8.1.j c. Quality assurance for effluent and environmental monitoring, and

d. ~~Fire Protection Program implementation,~~ and 4

DOC M01 ~~e.~~ d. All programs specified in Specification 5.5.

~~[f. Modification of core protection calculator (CPC) addressable constants. These procedures shall include provisions to ensure that sufficient margin is maintained in CPC type I addressable constants to avoid excessive operator interaction with CPCs during reactor operation.~~ 3

~~Modifications to the CPC software (including changes of algorithms and fuel cycle specific data) shall be performed in accordance with the most recent version of "CPC Protection Algorithm Software Change Procedure," CEN-39(A)-P, which has been determined to be applicable to the facility. Additions or deletions to CPC addressable constants or changes to addressable constant software limit values shall not be implemented without prior NRC approval.]~~

CTS

5.0 ADMINISTRATIVE CONTROLS

6.8 5.4 Procedures

6.8.1 5.4.1 Written procedures shall be established, implemented, and maintained covering the following activities:

6.8.1.a a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978,

6.8.1.a b. The emergency operating procedures required to implement the requirements of NUREG-0737 and ~~to~~ NUREG-0737, Supplement 1, as stated in ~~[~~Generic Letter 82-33~~]~~, 2 3

6.8.1.i 6.8.1.j c. Quality assurance for effluent and environmental monitoring, ~~and~~

d. ~~Fire Protection Program implementation,~~ and 4

DOC M01 ~~e.~~ All programs specified in Specification 5.5.

~~[f. Modification of core protection calculator (CPC) addressable constants. These procedures shall include provisions to ensure that sufficient margin is maintained in CPC type I addressable constants to avoid excessive operator interaction with CPCs during reactor operation.~~ 3

~~Modifications to the CPC software (including changes of algorithms and fuel cycle specific data) shall be performed in accordance with the most recent version of "CPC Protection Algorithm Software Change Procedure," CEN-39(A)-P, which has been determined to be applicable to the facility. Additions or deletions to CPC addressable constants or changes to addressable constant software limit values shall not be implemented without prior NRC approval.]~~

JUSTIFICATION FOR DEVIATIONS
ITS 5.4, PROCEDURES

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. Grammatical error corrected.

3. The ISTS contains bracketed information and/or values that are generic to Combustion Engineering vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

4. ISTS 5.4.1.d is deleted consistent with St. Lucie (PSL) Unit 1 and Unit 2 current licensing basis. PSL Units 1 and 2 transitioned to a risk-informed, performance-based fire protection program in accordance with National Fire Protection Association Standard NFPA 805 pursuant to 10 CFR 50.48(c). This transition was approved March 31, 2016 as issued in License Amendments 231 and 181, for Unit 1 and Unit 2 respectively (ADAMS Accession No. ML15344A346). As stated in the NRC Safety Evaluation supporting the license amendments, the administrative requirement that procedures be established, implemented, and maintained for Fire Protection Program implementation is contained in regulations 10 CFR 50.48(a) and 10 CFR 50.48(c); and in NFPA 805 Chapter 3. The NRC staff concluded that maintaining a procedure requirement for Fire Protection Program (FPP) implementation in Technical Specifications is redundant to the NFPA 805 requirement to establish FPP procedures, and as such, is unnecessary. Therefore, ISTS 5.4.1.d is not included in the ITS. The list in ITS 5.4.1 has been relabeled, as applicable.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 5.4, PROCEDURES

There are no specific No Significant Hazards Considerations for this Section.

ATTACHMENT 5

ITS Section 5.5, Programs and Manuals

Current Technical Specifications (CTS) Markup
and Discussion of Changes (DOCs)

6.0 ADMINISTRATIVE CONTROLS

~~6.8.3 DELETED~~

5.5 ~~6.8.4~~ The following programs shall be established, implemented, and maintained.

5.5.2 a. Primary Coolant Sources Outside Containment

A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include ~~the Shutdown Cooling System~~, Low Pressure Safety Injection High Pressure Safety Injection ~~System~~, Containment Spray ~~System~~, and RCS Sampling. The program shall include the following: A02

5.5.2.a (i) Preventive maintenance and periodic visual inspection requirements, and

5.5.2.b (ii) Integrated leak test requirements for each system ~~at least once per 18 months~~ in accordance with the Surveillance Frequency Control Program. LA01

The provisions of ~~Specification 4.0.2~~ SR 3.0.2 are applicable.

~~b. In-Plant Radioiodine Monitoring~~

~~A program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:~~ LA02

~~(i) Training of personnel,~~

~~(ii) Procedures for monitoring, and~~

~~(iii) Provisions for maintenance of sampling and analysis equipment.~~

6.0 ADMINISTRATIVE CONTROLS

5.5.7 c. Secondary Water Chemistry

A program for monitoring of secondary water chemistry to inhibit steam generator tube degradation. This program shall include:

5.5.7.a (i) Identification of a sampling schedule for the critical variables and control points of these variables,

5.5.7.b (ii) Identification of the procedures used to measure the values of the critical variables,

5.5.7.c (iii) Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in-leakage, 5.5.7.d (iv) Procedures for the recording and management of data,

5.5.7.e (v) Procedures defining corrective actions for all off-control point chemistry conditions, and

5.5.7.f (vi) A procedure identifying ~~(a)~~ the authority responsible for the interpretation of the data, and ~~(b)~~ the sequence and timing of administrative events required to initiate corrective action.

~~d. Backup Method for Determining Subcooling Margin~~

~~A program which will ensure the capability to accurately monitor the Reactor Coolant System subcooling margin. This program shall include the following:~~

~~(i) Training of personnel, and~~

~~(ii) Procedures for monitoring.~~

LA03

~~e. DELETED~~

5.5.3 f. Radioactive Effluent Controls Program

A program shall be provided conforming with 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 ~~(1)~~ shall be contained in the ODCM, ~~(2)~~ shall be implemented by ~~operating~~ procedures, and ~~(3)~~ shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

functional capability

5.5.3.a 1) Limitations on the ~~operability~~ of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM,

5.5.3.b 2) Limitations on the concentrations of radioactive material released in liquid effluents to UNRESTRICTED AREAS conforming to ten times the concentration values in 10 CFR 20.1001 - 20.240~~1~~, Appendix B, Table 2, Column 2.

2

A03

5.5.3.c 3) 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,

5.5.3.d 4) Limitations on the annual and quarterly doses or dose commitment on a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released from each unit to UNRESTRICTED AREAS conforming to Appendix I to 10 CFR Part 50,

5.5.3.e 5) 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.

5.5.3.f 6) Limitations on the ~~operability~~ functional capability and use of the liquid and gaseous effluent treatment systems to ensure that the appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a 31-day period would exceed 2 percent of the guidelines for the annual dose or dose commitment conforming to Appendix I to 10 CFR Part 50,

5.5.3.g 7) Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas at or beyond the SITE BOUNDARY shall be limited to the following:

a) For noble gases: Less than or equal to 500 mrem/yr to the total body and less than or equal to 3000 mrem/yr to the skin, and

b) For Iodine-131, for Iodine-133, for tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: Less than or equal to 1500 mrem/yr to any organ;

5.5.3.h 8) 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 Appendix I to 10 CFR Part 50,

5.5.3.i 9) Limitations on the annual and quarterly doses to a MEMBER OF THE PUBLIC from Iodine-131, Iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50,

5.5.3.j 10) 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 Part 190.

The provisions of ~~Specifications 4.0.2~~ SR 3.0.2 and ~~4.0.3~~ SR 3.0.3 are applicable to the Radioactive Effluent Controls Program surveillance frequency.

~~g. Radiological Environmental Monitoring Program~~

LA04

~~A program shall be provided to monitor the radiation and radio-nuclides in the environs of the plant. The program shall provide (1) representative measurements of radioactivity in the highest potential exposure pathways, and (2) verification of the accuracy of the effluent monitoring program and modeling of the environmental exposure pathways. The program shall (1) be contained in the ODCM,~~

~~(2) conform to the guidance of Appendix I to 10 CFR Part 50, and~~
~~(3) include the following:~~

~~1) Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ODCM.~~

~~2) A Land Use Census to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the monitoring program are made if required by the results of this census, and~~

~~3) Participation in a Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.~~

LA04

5.5.13 h. Containment Leakage Rate Testing Program

5.5.13.a A program to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50 Appendix J, Option B, as modified by approved exemptions. This program is in accordance with the guidelines contained in NEI 94-01, Revision 2-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J~~,~~." ~~except that the next Type A test performed after the December 8, 2005 Type A test shall be performed no later than December 8, 2020~~. A04

5.5.13.b The peak calculated containment internal pressure for the design basis loss of coolant accident P_a, is 42.8 psig. The containment design pressure is 44 psig.

5.5.13.c The maximum allowed containment leakage rate, L_a, at P_a, shall be 0.50% of containment air weight per day.

5.5.13.d Leakage rate acceptance criteria are:

5.5.13.d.1 a. 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, $\leq$ 0.75 L_a for Type A tests, and $\leq$ 0.096 L_a for secondary containment bypass leakage paths.

See ITS 3.6.3

5.5.13.d.2 b. Air lock testing acceptance criteria are:

5.5.13.d.2.a) 1) Overall air lock leakage rate is $\leq$ 0.05 L_a when tested at $\geq$ P_a.

5.5.13.d.2.b) 2) For ~~the~~ each personnel air lock door ~~seal~~, leakage rate is < 0.01 L_a when pressurized to $\geq$ 1.0 P_a.

5.5.13.d.2.c) 3) For ~~the~~ each emergency air lock door ~~seal~~, leakage rate is < 0.01 L_a when pressurized to $\geq$ 10 psig.

~~The provisions of T.S. 4.0.2 do not apply to test frequencies in the Containment Leak Rate Testing Program.~~ A05

5.5.13.e The provisions of ~~T.S. 4.0.3~~ SR 3.0.3 are applicable to the Containment Leak Rate Testing Program.

~~i. Deleted~~ Insert proposed 5.5.13.f A05

5.5.11 j. Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

5.5.11.a 1. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

5.5.11.b 2. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

5.5.11.b.1 a. a change in the TS incorporated in the license; or

5.5.11.b.2 b. a change to the ~~updated~~ UFSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

5.5.11.c 3. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the UFSAR.

5.5.11.d 4. Proposed changes that meet the criteria of ~~Specification 6.8.4.j.2.a or 6.8.4.j.2.b~~ 5.5.11.b, above, shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

ADMINISTRATIVE CONTROLS (continued)

5.5.8 k. 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 1.52, Revision 3.~~ Insert 1

A06

5.5.8.a 1. Demonstrate for each of the ESF systems that an inplace test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass ~~less than the value specified below~~ ≤ 0.05% when tested in accordance with ASME N510-1989 at the system flowrate specified below. ± 10%

ESF Ventilation System	~~Penetration~~	Flowrate
Control Room Emergency Ventilation System (CREVS)	~~≤ 0.05%~~	2000 ~~± 200 cfm~~
Shield Building Ventilation System SBVS	~~≤ 0.05%~~	6000 ~~± 600 cfm~~
Emergency Core Cooling System (ECCS) Area Ventilation System	~~≤ 0.05%~~	30,000 ~~± 3000 cfm~~

5.5.8.b 2. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass ~~less than the value specified below~~ ≤ 0.05% when tested in accordance with ASME N510-1989 at the system flowrate specified below. ± 10%

ESF Ventilation System	~~Penetration~~	Flowrate
CREVS ~~Control Room Emergency Ventilation~~	≤ 0.05%	2000 ~~± 200 cfm~~
SBVS ~~Shield Building Ventilation System~~	~~≤ 0.05%~~	6000 ~~± 600 cfm~~
ECCS Area Ventilation System	~~≤ 0.05%~~	30,000 ~~± 3000 cfm~~

5.5.8.c 3. 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 3, shows the methyl iodide penetration ~~less than the value specified below~~ when tested in accordance with ASTM D3803-1989 at a ≤ 2.5% temperature of 30°C (86°F) and ~~the~~ a relative humidity ~~specified below.~~ ≤ 70%

ESF Ventilation System	~~Penetration~~	~~RH~~
CREVS ~~Control Room Emergency Ventilation~~	≤ 2.5%	~~70%~~
SBVS ~~Shield Building Ventilation System~~	~~≤ 2.5%~~	~~70%~~
ECCS Area Ventilation System	~~≤ 2.5%~~	~~70%~~

5.5.8.d 4. Demonstrate for each of the ESF systems that the pressure drop across the combined HEPA filters and charcoal adsorbers is less than the value specified below when tested at the system flowrate specified below. ± 10%

ESF Ventilation System	Delta P	Flowrate
CREVS ~~Control Room Emergency Ventilation~~	< 4.15" W.G.	2000 ~~± 200 cfm~~
SBVS ~~Shield Building Ventilation System~~	≤ 6.15" W.G.	6000 ~~± 600 cfm~~
ECCS Area Ventilation System	< 4.15" W.G.	30,000 ~~± 3000 cfm~~

LA01

5.5.8.f 5. At a frequency in accordance with the Surveillance Frequency Control Program ~~At least once per 18 months~~, demonstrate that the heaters for each of the ESF systems dissipate the value specified below when tested in accordance with ASME N510-1989.

ESF Ventilation System	Wattage
SBVS ~~Shield Building Ventilation System~~	
Main Heaters	30 ± 3 kW
Auxiliary Heaters	1.5 ± 0.25 kW

ITS

Insert 1

5.5.8 and the Fuel Pool Area Ventilation System in accordance with Regulatory Guide 1.52, Revision 3, ASME N510-1989, ASTM D3803-1989 and, ANSI N510-1975, as described herein.

ADMINISTRATIVE CONTROLS (continued)

The provisions of SR ~~4.0.2~~ 3.0.2 and SR ~~4.0.3~~ 3.0.3 are applicable to the VFTP test frequencies. M01

5.5.6 l. Steam Generator (SG) Program ← Insert proposed 5.5.5, "RCP Flywheel Inspection Program"

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following provisions:

5.5.6.a a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met.

5.5.6.b b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational leakage.

1. Structural integrity performance criterion: All in-service SG tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cooldown and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

2. Accident induced leakage performance criterion: The primary-to-secondary accident induced leakage rate for any design basis accident, other than SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 0.5 gpm total through all SGs and 0.25 gpm through any one SG.

3. The operational leakage performance criterion is specified in LCO ~~3.4.6.2.c~~ 3.4.13, "Reactor Coolant System Operational Leakage."

5.5.6.c c. Provisions for SG tube plugging ~~repair~~ criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged. A07

5.5.6 l. Steam Generator (SG) Program (continued)

5.5.6.d d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube plugging ~~repair~~ criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tube may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations. A07

5.5.6.d.1 1. Inspect 100% of the tubes in each SG during the first refueling outage following SG replacement.

5.5.6.d.2 2. Inspect 100% of the tubes at ~~sequential periods of 144, 108, 72, and, thereafter, 60~~ 96 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. ~~In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outages nearest the end of the period. No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected.~~ L01

5.5.6.d.3 3. If crack indications are found in any SG tube, then the next inspection for each affected and potentially affected SG for the degradation mechanism that caused the crack indication shall ~~not exceed 24 effective full power months or one~~ be at the next refueling outage ~~(whichever is less)~~. If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

5.5.6.e e. Provisions for monitoring operational primary-to-secondary leakage.

M02

Insert proposed 5.5.14, "Battery Monitoring and Maintenance Program"

5.5.15 m. Control Room Envelope Habitability Program

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

The program shall include the following elements:

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

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

5.5.15 m. Control Room Envelope Habitability Program (continued)

c. Requirements for (i) determining the unfiltered air inleakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

d. Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by one train of the CREVS, operating at the flow rate required by the VFTP, at a Frequency in accordance with the Surveillance Frequency Control Program ~~of 36 months on a STAGGERED TEST BASIS~~. The results shall be trended and used as part of the 36 month assessment of the CRE boundary. LA01

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

f. The provisions of SR 3.0.2 ~~4.0.2~~ are applicable to the Frequencies for assessing CRE habitability, determining CRE unfiltered inleakage, and measuring CRE pressure and assessing the CRE boundary as required by paragraphs c and d, respectively.

n. Diesel Fuel Oil Testing Program

5.5.10

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:

5.5.10.a (i) 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 or a water and sediment content within limits;

5.5.10.b (ii) Other properties for ASTM 2D fuel oil are within limits within 31 days following ~~sampling and~~ addition to storage tanks; and

5.5.10.c (iii) Total particulate concentration of the fuel oil is ≤ 10 mg/l when tested in accordance with the Surveillance Frequency Control Program ~~every 31 days~~. LA01

The provisions of SR 3.0.2 ~~4.0.2~~ and SR 3.0.3 ~~4.0.3~~ are applicable to the Diesel Fuel Oil Testing Program test frequencies.

o. Surveillance Frequency Control Program

5.5.16

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

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

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

SR 3.0.2 SR 3.0.3

c. The provisions of ~~Surveillance Requirements 4.0.2~~ and ~~4.0.3~~ are applicable to the frequencies established in the Surveillance Frequency Control Program.

See ITS 3.0

p. Snubber Testing Program

This program conforms to the examination, testing and service life monitoring for dynamic restraints (snubbers) in accordance with 10 CFR 50.55a inservice inspection (ISI) requirements for supports. The program shall be in accordance with the following:

1. This program shall meet 10 CFR 50.55a(g) ISI requirements for supports.

2. The program shall meet the requirements for ISI of supports set forth in subsequent editions of the Code of Record and addenda of the American Society of Mechanical Engineers (ASME) Boiler and Pressure (BPV) Code and the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code) that are incorporated by reference in 10 CFR 50.55a(b) subject to the conditions listed in 10 CFR 50.55a(b) and subject to Commission approval.

3. The program shall, as required by 10 CFR 50.55a(b)(3)(v), meet Subsection ISTA, "General Requirements" and Subsection ISTD, "Preservice and Inservice Examination and Testing of Dynamic Restraints (Snubbers) in Light-Water Reactor Nuclear Power Plants".

4. The 120-month program updates shall be made in accordance with 10 CFR 50.55a(g)(4), 10 CFR 50.55a(g)(3)(v) and 10 CFR 50.55a(b) (including 10 CFR 50.55a(b)(3)(v)) subject to the conditions listed therein.

5.5.4 q. Component Cyclic or Transient Limit Program

U

The program provides controls to track the FSAR, Section 5.2, cyclic and transient occurrences to ensure that components are maintained within the design limits.

5.5.17 r. Risk Informed Completion Time Program

This program provides controls to calculate a Risk Informed Completion Time (RICT) and must be implemented in accordance with NEI 06-09-A, "~~Risk-Informed Technical Specifications Initiative 4b:~~ Risk-Managed Technical Specifications (RMTS) Guidelines~~,~~." Revision 0~~-A, November 2006.~~ The program shall include the following:

a. The RICT may not exceed 30 days;

b. A RICT may only be utilized in MODES 1 and 2;

5.5.17 c. When a RICT is being used, any plant configuration change within the scope of the Configuration Risk Management Program must be considered for the effect on the RICT.

1. For planned changes, the revised RICT must be determined prior to implementation of the change in configuration.

2. For emergent conditions, the revised RICT must be determined within the time limits of the Required Action Completion Time (i.e., not the RICT) or 12 hours after the plant configuration change, whichever is less.

3. Revising the RICT is not required If the plant configuration change would lower plant risk and would result in a longer RICT.

d. Use of a RICT is not permitted for entry into a configuration which represents a loss of a specified safety function or inoperability of all required trains of a system required to be OPERABLE.

e. If the extent of condition evaluation for inoperable structures, systems, or components (SSCs) is not complete prior to exceeding the Completion Time, the RICT shall account for the increased possibility of common cause failure (CCF) by either:

1. Numerically accounting for the increased possibility of CCF in the RICT calculation, or

2. Risk Management Actions (RMAs) not already credited in the RICT calculation shall be implemented that support redundant or diverse SSCs that perform the function(s) of the inoperable SSCs, and, if practicable, reduce the frequency of initiating events that challenge the function(s) performed by the inoperable SSCs.

s. Safety Function Determination Program (SFDP)

5.5.12

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 taken as a result of the support system inoperability and corresponding exception to entering supported system Conditions and Required Actions ~~ACTIONS~~. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following: A08 A09

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 ~~allowed outage time~~ is not inappropriately extended as a result of multiple support system inoperabilites, and A09

d. Other appropriate limitations and remedial or compensatory actions.

5.5.12 A loss of safety function exists when, assuming no concurrent single failure, no concurrent loss of offsite power, or no concurrent loss of onsite diesel generator(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.

Conditions and Required Actions

A09

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 ~~ACTIONS~~ of the LCO in which the loss of safety function exists are required to be entered. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate ~~ACTIONS~~ to enter are those of the support system.

Conditions and Required Actions

A09

6.9 REPORTING REQUIREMENTS

See ITS 5.6

ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the NRC.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following:

(1) receipt of an operating license,

(2) amendment of the license involving a planned increase in power level,

(3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and

(4) modifications that may have significantly altered the nuclear, thermal or hydraulic performance of the plant.

6.13 PROCESS CONTROL PROGRAM (PCP)

Changes to the PCP:

1. Shall be documented and this documentation shall contain:

   a) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and

   b. A determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.

2. Shall become effective after the approval of the plant manager.

See CTS 6.0

5.5.1 ## 6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

Changes to the ODCM:

and records of reviews performed shall be retained.

M03

5.5.1.a 1. Shall be documented ~~and~~ this documentation shall contain:

5.5.1.a.1 a) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and

5.5.1.a.2 b. A determination that the change ~~will~~ maintain the level of radioactive effluent control required by 10 CFR 20.1302, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50 and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

5.5.1.b 2. Shall become effective after the approval of the plant manager.

5.5.1.c 3. Shall be submitted to the Commission in the form of a complete, legible copy of the entire ODCM as part of or concurrent with the ~~Annual~~ Radioactive Effluent Release Report for the period of the report in which any change to the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (e.g., month/year) the change was implemented.

IDENTIFED LEAKAGE

1.15 IDENTIFIED LEAKAGE shall be:

a. Leakage (except CONTROLLED LEAKAGE) into closed systems, such as pump seal or valve packing leaks that are captured, and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE, or

c. Reactor Coolant System leakage through a steam generator to the secondary system (Primary-to-secondary leakage).

INSERVICE TESTING PROGRAM

1.16 The INSERVICE TESTING PROGRAM is the licensee program that fulfills the requirements of 10 CFR 50.55a(f).

MEMBER(S) OF THE PUBLIC

1.17 MEMBER OF THE PUBLIC means an individual in a controlled or unrestricted area. However, an individual is not a member of the public during any period in which the individual receives an occupational dose.

OFFSITE DOSE CALCULATION MANUAL (ODCM)

5.5.1.a
5.5.1.b ~~1.18~~ THE OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain ~~(1)~~ the Radioactive Effluent Controls and Radiological Environmental Monitoring ~~Programs required by Section 6.8.4~~ and ~~(2)~~ descriptions of the information that should be included in the Annual Radiological Environmental Operating and ~~Annual~~ Radioactive Effluent Release Reports required by Specifications ~~6.9.1.7~~ and ~~6.9.1.8~~.

activities 5.6.1 5.6.2

CONTAINMENT SYSTEMS

3/4.6.6 SECONDARY CONTAINMENT

SHIELD BUILDING VENTILATION SYSTEM

See ITS 3.6.9

LIMITING CONDITION FOR OPERATION

3.6.6.1 Two independent shield building ventilation systems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one shield building ventilation system inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

NOTE
Action not applicable when second shield building ventilation system intentionally made inoperable.

b. With two shield building ventilation systems inoperable, within 1 hour verify at least one train of containment spray is OPERABLE, and restore at least one shield building ventilation system to OPERABLE status within 24 hours; otherwise, be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.6.1 Each shield building ventilation system shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by initiating, from the control room, flow through the HEPA filter and charcoal adsorber train and verifying that the train operates for at least 15 continuous minutes with the heaters on.

b. By performing required shield building ventilation system filter testing in accordance with the Ventilation Filter Testing Program.

5.5.8
5.5.8.e

c. In accordance with the Surveillance Frequency Control Program by:

1. Verifying that the air flow distribution is uniform within 20% across HEPA filters and charcoal adsorbers when tested in accordance with ASME N510-1989.

See ITS 3.6.9

2. Verifying that the filtration system starts automatically on a Containment Isolation Signal (CIS).

3. Verifying that the filter cooling makeup air and cross connection valves can be manually opened.

See ITS 3.6.7

4. Verifying that each system produces a negative pressure of $\geq$ 2.0 inches W.G. in the annulus within 2 minutes after a Containment Isolation Signal (CIS).

See ITS 3.7.12

PLANT SYSTEMS

3/4.7.8 ECCS AREA VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.8.1 Two independent ECCS area exhaust air filter trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one ECCS area exhaust air filter train inoperable, restore the inoperable train to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

SURVEILLANCE REQUIREMENTS

4.7.8.1 Each ECCS area exhaust air filter train shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by initiating, from the control room, flow through the HEPA filter and charcoal adsorber train and verifying that the train operates for at least 15 minutes.

b. By performing required ECCS area ventilation system filter testing in accordance with the Ventilation Filter Testing Program.

5.5.8 c. In accordance with the Surveillance Frequency Control Program:

5.5.8.e 1. Verifying that the air flow distribution is uniform within 20% across HEPA filters and charcoal adsorbers when tested in accordance with ASME N510-1989.

See ITS 3.7.12 2. Verifying that the filter train starts on a Safety Injection Actuation Signal.

TABLE 4.8-2

BATTERY SURVEILLANCE REQUIREMENT

	CATEGORY A[(1)]	CATEGORY B[(2)]	
Parameter	Limits for each designated pilot cell	Limits for each connected cell	Allowable[(3)] value for each connected cell
Electrolyte Level	> Minimum level indication mark, and ≤ 1/4" above maximum level indication mark	> Minimum level indication mark, and ≤ 1/4" above maximum level indication mark	Above top of plates and not overflowing
5.5.14. b.1 and b.2 Float Voltage	≥ 2.13 volts	≥ 2.13 volts	> 2.07 volts
Specific Gravity[(a)]	≥ 1.195[(b)]	≥ 1.190 Average of all connected cells > 1.200	Not more than .020 below the average of all connected cells Average of all connected cells ≥ 1.190[(b)]

(a) Corrected for electrolyte temperature and level.

(b) Or battery charging current is less than 2 amps when on charge.

5.5.14 ~~(c) Corrected for average electrolyte temperature.~~

LA05

(1) For any Category A parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that within 24 hours all the Category B measurements are taken and found to be within their allowable values, and provided all Category A and B parameter(s) are restored to within limits within the next 6 days.

(2) For any Category B parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that the Category B parameters are within their allowable values and provided the Category B parameter(s) are restored to within the limits within 7 days.

(3) With any Category B parameter not within its allowable value, declare the battery inoperable.

See ITS 3.8.6

REFUELING OPERATIONS

FUEL POOL VENTILATION SYSTEM – FUEL STORAGE

See ITS 3.7.17

LIMITING CONDITION FOR OPERATION

3.9.12 At least one fuel pool ventilation system shall be OPERABLE.

APPLICABILITY: Whenever recently irradiated fuel is in the spent fuel pool.

ACTION:

a. With no fuel pool ventilation system OPERABLE, suspend all operations involving movement of recently irradiated fuel within the spent fuel pool or crane operation with loads over the recently irradiated spent fuel until at least one fuel pool ventilation system is restored to OPERABLE status.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.12 The above required fuel pool ventilation system shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by initiating flow through the HEPA filter and charcoal adsorber train and verifying that the train operates for at least 15 minutes.

5.5.8
Insert 2

b. In accordance with the Surveillance Frequency Control Program or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system by:

REFUELING OPERATIONS

SURVEILLANCE REQUIREMENTS (Continued)

5.5.8.h 1. Verifying that the charcoal adsorbers remove ≥ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 10,350 cfm ± 10%.

5.5.8.g 2. Verifying that the HEPA filter banks remove ≥ 99% of the ~~DOP~~ dioctyl phthalate when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 10,350 cfm ± 10%.

5.5.8.i 3. Verifying that a laboratory analysis of a carbon sample from either at least one test canister or at least two carbon samples removed from one of the charcoal adsorbers demonstrates a removal efficiency of ≥ 85% for radioactive methyl iodide when the sample is tested in accordance with ASTM D3803-1989 (30°C, 95% RH). The carbon samples not obtained from test canisters shall be prepared by either:

5.5.8.i.1 a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

5.5.8.i.2 b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

~~4. Verifying a system flow rate of 10,350 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.~~ A10

REFUELING OPERATIONS

SURVEILLANCE REQUIREMENTS (Continued)

5.5.8 c. In accordance with the Surveillance Frequency Control Program by:

5.5.8.j 1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is < 4.15 inches Water Gauge while operating the ventilation system at a flow rate of 10,350 cfm ± 10%.

5.5.8.k 2. Verifying that the air flow distribution is uniform within 20% across HEPA filters and charcoal adsorbers when tested in accordance with ANSI N510-1975.

3. Verifying that the ventilation system maintains the spent fuel storage pool area at a negative pressure of ≥ 1/8 inches Water Gauge relative to the outside atmosphere during system operation. See ITS 3.7.17

5.5.8 d. After each complete or partial replacement of a HEPA filter

5.5.8.g bank by verifying that the HEPA filter banks remove ≥ 99% of the dioctyl phthalate ~~DOP~~ when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 10,350 cfm ± 10%.

5.5.8 e. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove

5.5.8.h ≥ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 10,350 cfm ± 10%.

~~Pages 3/4 11-2 through 3/4 11-13 (Amendment No. 123) have been deleted from the Technical Specifications. The next page is 3/4 11-14.~~

~~RADIOACTIVE EFFLUENTS~~

and Storage Tank Radioactivity Monitoring Program

EXPLOSIVE GAS ~~MIXTURE~~

Add proposed ITS 5.5.9 generic program statement

A11

~~LIMITING CONDITION FOR OPERATION~~

5.5.9.a 3.11.2.5 The concentration of oxygen in the waste gas decay tanks shall be limited ~~to less than or equal to 2% by volume whenever the hydrogen concentration exceeds 4% by volume.~~

~~**APPLICABILITY:** At all times.~~

LA06

~~**ACTION:**~~

~~a. With the concentration of oxygen in the waste gas decay tank greater than 2% by volume but less than or equal to 4% by volume, reduce the oxygen concentration to the above limits within 48 hours.~~

~~b. With the concentration of oxygen in the waste gas decay tank greater than 4% by volume and the hydrogen concentration greater than 2% by volume, immediately suspend all additions of waste gases to the system and immediately commence reduction of the concentration of oxygen to less than or equal to 2% by volume.~~

~~c. The provisions of Specification 3.0.3 are not applicable.~~

SURVEILLANCE REQUIREMENTS

5.5.9 a and b 4.11.2.5.1 The concentration of oxygen in the waste gas decay tank shall be determined to be within the above limits by ~~continuously*~~ monitoring the waste gases in the ~~on service~~ waste gas decay tank.

5.5.9 a and b 4.11.2.5.2 ~~With the oxygen concentration in the on service waste gas decay tank greater than 2% by volume as determined by Specification 4.11.2.5.1,~~ the concentration of hydrogen in the waste gas decay tank shall be determined to be within the above limits ~~by gas partitioner sample at least once per 24 hours.~~

LA06

~~* When continuous monitoring capability is inoperable, waste gases shall be monitored in accordance with the actions specified for the Waste Gas Decay Tanks Explosive Gas Monitoring System in Chapter 13 of the Updated Final Safety Analysis Report.~~

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

A11

~~RADIOACTIVE EFFLUENTS~~

Explosive Gas and Storage Tank Radioactivity Monitoring Program

~~GAS STORAGE TANKS~~

Add proposed ITS 5.5.9 generic program statement A11

~~LIMITING CONDITION FOR OPERATION~~

5.5.9.a 3.11.2.6 The quantity of radioactivity contained in each gas storage tank ~~shall be limited to less than or equal to 165,000 curies noble gases (considered as Xe-133).~~ LA06

~~**APPLICABILITY:** At all times.~~

~~**ACTION:**~~

~~a. With the quantity of radioactive material in any gas storage tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank.~~

~~b. The provisions of Specification 3.0.3 are not applicable.~~

SURVEILLANCE REQUIREMENTS

5.5.9 a and b 4.11.2.6 The quantity of radioactive material contained in each gas storage tank shall be determined to be within the above limit ~~in accordance with the Surveillance Frequency Control Program when radioactive materials are being added to the tank when reactor coolant system activity exceeds 518.9 μCi/gram DOSE EQUIVALENT XE-133.~~ LA06

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies. A11

6.0 ADMINISTRATIVE CONTROLS

PROCEDURES AND PROGRAMS (Continued)

5.5 ~~6.8.4~~ The following programs shall be established, implemented, and maintained.

5.5.2 a. Primary Coolant Sources Outside Containment

A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include ~~the Shutdown Cooling System~~ Low Pressure Safety Injection, High Pressure Safety Injection ~~System~~, Containment Spray System, and RCS Sampling. The program shall include the following: A02

5.5.2.a (i) Preventive maintenance and periodic visual inspection requirements, and

5.5.2.b (ii) Integrated leak test requirements for each system ~~at least once per 18 months~~ in accordance with the Surveillance Frequency Control Program. LA01

The provisions of ~~Specification 4.0.2~~ SR 3.0.2 are applicable.

~~b. In-Plant Radioiodine Monitoring~~

~~A program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:~~ LA02

~~(i) Training of personnel,~~

~~(ii) Procedures for monitoring, and~~

~~(iii) Provisions for maintenance of sampling and analysis equipment.~~

PROCEDURES AND PROGRAMS (continued)

5.5.7 c. Secondary Water Chemistry

A program for monitoring of secondary water chemistry to inhibit steam generator tube degradation. This program shall include:

5.5.7.a (i) Identification of a sampling schedule for the critical variables and control points of these variables,

5.5.7.b (ii) Identification of the procedures used to measure the values of the critical variables,

5.5.7.c (iii) Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in-leakage,

5.5.7.d (iv) Procedures for the recording and management of data,

5.5.7.e (v) Procedures defining corrective actions for all off-control point chemistry conditions, and

5.5.7.f (vi) A procedure identifying ~~(a)~~ the authority responsible for the interpretation of the data, and ~~(b)~~ the sequence and timing of administrative events required to initiate corrective action.

~~d. Backup Method for Determining Subcooling Margin~~

~~A program which will ensure the capability to accurately monitor the Reactor Coolant System subcooling margin. This program shall include the following:~~ LA03

~~(i) Training of personnel, and~~

~~(ii) Procedures for monitoring.~~

~~e. DELETED~~

5.5.3 f. Radioactive Effluent Controls Program

A program shall be provided conforming with 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 ~~(1)~~ shall be contained in the ODCM, ~~(2)~~ shall be implemented by operating procedures, and ~~(3)~~ shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

5.5.3.a 1) Limitations on the ~~operability~~ functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM,

5.5.3.b 2) Limitations on the concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS conforming to ten times the concentration values in 10 CFR 20.1001 – 20.2401 2, Appendix B, Table 2, Column 2. A03

5.5.3.c 3) 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,

5.5.3.d 4) Limitations on the annual and quarterly doses or dose commitment on a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released from each unit to UNRESTRICTED AREAS conforming to Appendix I to 10 CFR Part 50,

5.5.3.e 5) 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.

5.5.3.f 6) Limitations on the ~~operability~~ functional capability and use of the liquid and gaseous effluent treatment systems to ensure that the appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a 31-day period would exceed 2 percent of the guidelines for the annual dose or dose commitment conforming to Appendix I to 10 CFR Part 50,

5.5.3.g 7) Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas at or beyond the SITE BOUNDARY shall be limited to the following:

a) For noble gases: Less than or equal to 500 mrem/yr to the total body and less than or equal to 3000 mrem/yr to the skin, and

b) For Iodine-131, for Iodine-133, for tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: Less than or equal to 1500 mrem/yr to any organ;

5.5.3.h 8) 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 Appendix I to 10 CFR Part 50,

5.5.3.i 9) Limitations on the annual and quarterly doses to a MEMBER OF THE PUBLIC from Iodine-131, Iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50,

5.5.3.j

10) 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 Part 190.

SR 3.0.2 SR 3.0.3

The provisions of ~~Specifications 4.0.2~~ and ~~4.0.3~~ are applicable to the Radioactive Effluent Controls Program surveillance frequency.

LA04

~~g. Radiological Environmental Monitoring Program~~

~~A program shall be provided to monitor the radiation and radionuclides in the environs of the plant. The program shall provide (1) representative measurements of radioactivity in the highest potential exposure pathways, and (2) verification of the accuracy of the effluent monitoring program and modeling of the environmental exposure pathways. The program shall (1) be contained in the ODCM, (2) conform to the guidance of Appendix I to 10 CFR Part 50, and (3) include the following:~~

~~1) Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ODCM.~~

~~2) A Land Use Census to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the monitoring program are made if required by the results of this census, and~~

~~3) Participation in a Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.~~

5.5.13

h. Containment Leakage Rate Testing Program

5.5.13.a

A program to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50 Appendix J, Option B, as modified by approved exemptions. This program is in accordance with the guidelines contained in NEI 94-01, Revision 2-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," except that the next Type A test performed after the December 18, 2007 Type A test shall be performed no later than December 18, 2022.

5.5.13.b

The peak calculated containment internal pressure for the design basis loss of coolant accident P_a, is 43.48 psig. The containment design pressure is 44 psig.

5.5.13.c

The maximum allowed containment leakage rate, L_a, at P_a, shall be 0.50% of containment air weight per day.

5.5.13.d Leakage rate acceptance criteria are:

5.5.13.d.1 a. 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, $\leq 0.75\ L_a$ for Type A tests, and $\leq 0.096\ L_a$ for secondary containment bypass leakage paths.

See ITS 3.6.3

5.5.13.d.2 b. Air lock testing acceptance criteria are:

5.5.13.d.2.a) 1) Overall air lock leakage is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

air lock

5.5.13.d.2.b) 2) For each door ~~seal~~, leakage rate is $< 0.01\ L_a$ when pressurized to $\geq P_a$.

~~The provisions of T.S. 4.0.2 do not apply to test frequencies in the Containment Leak Rate Testing Program.~~ A05

5.5.13.e The provisions for ~~T.S. 4.0.3~~ SR 3.0.3 are applicable to the Containment Leak Rate Testing Program.

~~i. Deleted~~ Insert proposed 5.5.13.f A05

5.5.11 j. Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

5.5.11.a 1. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

5.5.11.b 2. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

5.5.11.b.1 a. a change in the TS incorporated in the license; or

5.5.11.b.2 b. a change to the ~~updated~~ UFSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

5.5.11.c 3. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the UFSAR.

5.5.11.d 4. Proposed changes that meet the criteria of ~~Specification 6.8.4.j.2.a or 6.8.4.j.2.b,~~ 5.5.11.b above, shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

5.5.8 k. 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~~ in accordance with Regulatory Guide 1.52, Revision 3~~.~~, ASME N510-1989 and ASTM D3803-1989, as described herein. A06

5.5.8.a 1. Demonstrate for each of the ESF systems that an inplace test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass ~~less than the value specified below~~ ≤ 0.05% when tested in accordance with ASME N510-1989 at the system flowrate specified below ± 10%.

ESF Ventilation System	~~**Penetration**~~	**Flowrate**
Control Room Emergency Air Cleanup System (CREVS)	~~≤ 0.05%~~	2000 ~~± 200 cfm~~
Shield Building Ventilation System SBVS	~~≤ 0.05%~~	6000 ~~± 600 cfm~~
Emergency Core Cooling System (ECCS) ~~ECCS~~ Area Ventilation System	~~≤ 0.05%~~	30,000 ~~± 3000 cfm~~

5.5.8.b 2. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass ~~less than the value specified~~ ≤ 0.05% below when tested in accordance with ASME N510-1989 at the system flowrate specified below ± 10%.

ESF Ventilation System	**Penetration**	**Flowrate**
CREVS ~~Control Room Emergency Air Cleanup~~	~~≤ 0.05%~~	2000 ~~± 200 cfm~~
SBVS ~~Shield Building Ventilation System~~	~~≤ 0.05%~~	6000 ~~± 600 cfm~~
ECCS Area Ventilation System	~~≤ 0.05%~~	30,000 ~~± 3000 cfm~~

k. Ventilation Filter Testing Program (VFTP) (continued)

5.5.8.c 3. 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 3, 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 ~~the~~ a relative humidity ~~specified below~~ ≤ 95%.

ESF Ventilation System	Penetration	~~RH~~
CREVS ~~Control Room Emergency Air Cleanup~~	≤ 0.175%	~~95%~~
SBVS ~~Shield Building Ventilation System~~	≤ 2.5%	~~95%~~
ECCS Area Ventilation System	≤ 2.5%	~~95%~~

5.5.8.d 4. For the ~~Control Room Emergency Air Cleanup System~~ CREVS and the ECCS Area Ventilation System, demonstrate that the pressure drop across the combined HEPA filters and charcoal adsorbers is less than the value specified below when tested at the system flowrate specified below. For the ~~Shield Building Ventilation System~~ SBVS, demonstrate that the pressure drop across the combined demisters, electric heaters, HEPA filters and charcoal adsorbers is less than the value specified below when tested at the system flowrate specified below.

ESF Ventilation System	Delta P	Flowrate
CREVS ~~Control Room Emergency Air Cleanup~~	< 7.4" W.G.	2000 ~~± 200 cfm~~
SBVS ~~Shield Building Ventilation System~~	< 8.5" W.G.	6000 ~~± 600 cfm~~
ECCS Area Ventilation System	< 4.35" W.G.	30,000 ~~± 3000 cfm~~

LA01

5.5.8.f 5. ~~At least once per 18 months~~ At a frequency in accordance with the Surveillance Frequency Control Program, demonstrate that the heaters for each of the ESF systems dissipate the value specified below when tested in accordance with ASME N510-1989.

ESF Ventilation System	Wattage
SBVS ~~Shield Building Ventilation System~~	
Main Heaters	30 ± 3 kW
Auxiliary Heaters	1.5 ± 0.25 kW

The provisions of SR ~~4~~ 3.0.2 and SR ~~4~~ 3.0.3 are applicable to the VFTP test frequencies.

5.5.6 l. Steam Generator (SG) Program

1. A SG Program shall be established and implemented for the replacement SGs to ensure that SG tube integrity is maintained. In addition, the SG Program shall include the following provisions:

5.5.6.a a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met.

ADMINISTRATIVE CONTROLS (continued)

l. Steam Generator (SG) Program (continued)

5.5.6 1. (continued)

5.5.6.b b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational leakage.

1. Structural integrity performance criterion: All in-service SG tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cooldown and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

2. Accident induced leakage performance criterion: The primary-to-secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 0.5 gallons per minute total through all SGs and 0.25 gallons per minute through any one SG.

3. The operational leakage performance criterion is specified in LCO ~~3.4.6.2.c~~ 3.4.13, "Reactor Coolant System Operational Leakage."

ADMINISTRATIVE CONTROLS (continued)

5.5.6 l. Steam Generator (SG) Program (continued)

1. (continued)

plugging A07

5.5.6.c c. Provisions for SG tube ~~repair~~ criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.

5.5.6.d d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube plugging ~~repair~~ criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations. A07

5.5.6.d.1 1. Inspect 100% of the tubes in each SG during the first refueling outage following SG replacement.

5.5.6.d.2 2. Inspect 100% of the tubes at ~~sequential periods of 144, 108, 72, and, thereafter,~~ 96 ~~60~~ effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. ~~In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outages nearest the end of the period. No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected.~~ L01

5.5.6.d.3 3. If crack indications are found in any SG tube, then the next inspection for each affected and potentially affected SG for the degradation mechanism that caused the crack indication shall be at the next ~~not exceed 24 effective full power months or one~~ refueling outage ~~(whichever is less)~~. If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

5.5.6.e e. Provisions for monitoring operational primary-to-secondary leakage

Insert proposed 5.5.14, “Battery Monitoring and Maintenance Program” M02

~~PAGES 6-15g AND 6-15h HAVE BEEN DELETED.~~
~~THE NEXT PAGE IS 6-15i.~~

5.5.15 m. Control Room Envelope Habitability Program

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

The program shall include the following elements:

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

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

c. Requirements for (i) determining the unfiltered air inleakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

d. Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by one train of the CREACS, operating at the flow rate required by the VFTP, ~~at a Frequency of 36 months on a STAGGERED TEST BASIS~~ in accordance with the Surveillance Frequency Control Program. The results shall be trended and used as part of the 36 month assessment of the CRE boundary. LA01

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

f. The provisions of SR ~~4~~ 3.0.2 are applicable to the Frequencies for assessing CRE habitability, determining CRE unfiltered inleakage, and measuring CRE pressure and assessing the CRE boundary as required by paragraphs c and d, respectively.

5.5.10 n. 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:

(i) 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 or a water and sediment content within limits;

in accordance with the Surveillance Frequency Control Program

LA01

(ii) Other properties for ASTM 2D fuel oil are within limits within 31 days following ~~sampling and~~ addition to storage tanks; and

(iii) Total particulate concentration of the fuel oil is ≤ 10 mg/l when tested ~~every 31 days~~.

3 3

The provisions of SR 4.0.2 and SR 4.0.3 are applicable to the Diesel Fuel Oil Testing Program test frequencies.

Insert proposed requirements based on RCP flywheel material

L02

5.5.5 o. Reactor Coolant Pump Flywheel Inspection Program

This program shall provide for the inspection of each reactor coolant pump flywheel by either a 100% volumetric inspection of the upper flywheel over the volume from the inner bore of the flywheel to the circle of one-half the outer radius or a surface examination (magnetic particle testing and/or penetrant testing) of exposed surfaces defined by the volume of the disassembled flywheel at least once every 10 years.

See ITS 3.0

p. Snubber Testing Program

This program conforms to the examination, testing and service life monitoring for dynamic restraints (snubbers) in accordance with 10 CFR 50.55a inservice inspection (ISI) requirements for supports. The program shall be in accordance with the following:

1. This program shall meet 10 CFR 50.55a(g) ISI requirements for supports.

2. The program shall meet the requirements for ISI of supports set forth in subsequent editions of the Code of Record and addenda of the American Society of Mechanical Engineers (ASME) Boiler and Pressure (BPV) Code and the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code) that are incorporated by reference in 10 CFR 50.55a(b) subject to limitations and modifications listed in 10 CFR 50.55a(b) and subject to Commission approval.

3. The program shall, as required by 10 CFR 50.55a(b)(3)(v)(B), meet Subsection ISTA, “General Requirements” and Subsection ISTD, “Preservice and Inservice Examination and Testing of Dynamic Restraints (Snubbers) in Light-Water Reactor Nuclear Power Plants”.

4. The 120-month program updates shall be made in accordance with 10 CFR 50.55a(g)(4), 10 CFR 50.55a(g)(3)(v) and 10 CFR 50.55a(b) (including 10 CFR 50.55a(b)(3)(v)(B)) subject to the limitations and modifications listed therein.

5.5.16 q. Surveillance Frequency Control Program

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

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

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

SR 3.0.2 SR 3.0.3

c. The provisions of ~~Surveillance Requirements 4.0.2~~ and ~~4.0.3~~ are applicable to the frequencies established in the Surveillance Frequency Control Program.

5.5.4 r. Component Cyclic or Transient Limit Program

U

The Program provides controls to track the FSAR, Section 3.9, cyclic and transient occurrences to ensure that components are maintained within the design limits.

5.5.17 s. Risk Informed Completion Time Program

This program provides controls to calculate a Risk Informed Completion Time (RICT) and must be implemented in accordance with NEI 06-09-A, "~~Risk-Informed Technical Specifications Initiative 4b:~~ Risk-Managed Technical Specifications (RMTS) Guidelines," Revision 0~~-A, November 2006~~. The program shall include the following:

a. The RICT may not exceed 30 days;

b. A RICT may only be utilized in MODES 1 and 2;

c. When a RICT is being used, any plant configuration change within the scope of the Configuration Risk Management Program must be considered for the effect on the RICT.

1. For planned changes, the revised RICT must be determined prior to implementation of the change in configuration.

2. For emergent conditions, the revised RICT must be determined within the time limits of the Required Action Completion Time (i.e., not the RICT) or 12 hours after the plant configuration change, whichever is less.

3. Revising the RICT is not required if the plant configuration change would lower plant risk and would result in a longer RICT.

d. Use of a RICT is not permitted for entry into a configuration which represents a loss of a specified safety function or inoperability of all required trains of a system required to be OPERABLE.

ADMINISTRATIVE CONTROLS

5.5.17 e. If the extent of condition evaluation for inoperable structures, systems, or components (SSCs) is not complete prior to exceeding the Completion Time, the RICT shall account for the increased possibility of common cause failure (CCF) by either:

1. Numerically accounting for the increased possibility of CCF in the RICT calculation, or

2. Risk Management Actions (RMAs) not already credited in the RICT calculation shall be implemented that support redundant or diverse SSCs that perform the function(s) of the inoperable SSCs, and, if practicable, reduce the frequency of initiating events that challenge the function(s) performed by the inoperable SSCs.

5.5.12 t. Safety Function Determination Program (SFDP)

limitations and remedial or compensatory

A08

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 Conditions and Required Actions ~~ACTIONS~~. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following: A09

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 ~~allowed outage time~~ is not inappropriately extended as a result of multiple support system inoperabilites, and A09

d. Other appropriate limitations and remedial or compensatory actions.

A loss of safety function exists when, assuming no concurrent single failure, no concurrent loss of offsite power, or no concurrent loss of onsite diesel generator(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 ~~ACTIONS~~ of the LCO in which the loss of safety function exists are required to be entered. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions ~~ACTIONS~~ to enter are those of the support system. A09 A09

ADMINISTRATIVE CONTROLS

6.13 PROCESS CONTROL PROGRAM (PCP)

Changes to the PCP:

1. Shall be documented and this documentation shall contain:

   a) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and

   b. A determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.

2. Shall become effective after the approval of the plant manager.

See CTS 6.0

5.5.1 ### 6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

Changes to the ODCM:

M03

5.5.1.a 1. Shall be documented ~~and~~ and records of reviews performed shall be retained. this documentation shall contain:

5.5.1.a.1 a) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and

5.5.1.a.2 b) A determination that the change ~~will~~ maintain the level of radioactive effluent control required by 10 CFR 20.1302, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50 and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

5.5.1.b 2. Shall become effective after the approval of the plant manager.

5.5.1.c 3. Shall be submitted to the Commission in the form of a complete, legible copy of the entire ODCM as part of or concurrent with the ~~Annual~~ Radioactive Effluent Release Report for the period of the report in which any change to the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (e.g., month/year) the change was implemented.

INSERVICE TESTING PROGRAM

1.16 The INSERVICE TESTING PROGRAM is the licensee program that fulfills the requirements of 10 CFR 50.55a(f).

MEMBER(S) OF THE PUBLIC

1.17 MEMBER OF THE PUBLIC means an individual in a controlled or unrestricted area. However, an individual is not a member of the public during any period in which the individual receives an occupational dose.

OFFSITE DOSE CALCULATION MANUAL (ODCM)

5.5.1.a
5.5.1.b

~~1.18~~ THE OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain ~~(1)~~ the Radioactive Effluent Controls and Radiological Environmental Monitoring ~~Programs required by Section 6.8.4~~ activities and ~~(2)~~ descriptions of the information that should be included in the Annual Radiological Environmental Operating and ~~Annual~~ Radioactive Effluent Release Reports required by Specifications ~~6.9.1.7~~ 5.6.1 and ~~6.9.1.8~~ 5.6.2.

OPERABLE – OPERABILITY

1.19 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s), and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

OPERATIONAL MODE – MODE

1.20 An OPERATIONAL MODE (i.e., MODE) shall correspond to any one inclusive combination of core reactivity condition, power level and average reactor coolant temperature specified in Table 1.2.

PHYSICS TESTS

1.21 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and (1) described in Chapter 14.0 of the FSAR, (2) authorized under the provisions of 10 CFR 50.59, or (3) otherwise approved by the Commission.

See ITS 1.1

CONTAINMENT SYSTEMS

3/4.6.6 SECONDARY CONTAINMENT

SHIELD BUILDING VENTILATION SYSTEM (SBVS)

See ITS 3.6.9

LIMITING CONDITION FOR OPERATION

3.6.6.1 Two independent Shield Building Ventilation Systems shall be OPERABLE.

APPLICABILITY: At all times in MODES 1, 2, 3, and 4.
In addition, during movement of recently irradiated fuel assemblies or during crane operations with loads over recently irradiated fuel assemblies in the Spent Fuel Storage Pool in MODES 5 and 6.

ACTION:

a. With the SBVS inoperable solely due to loss of the SBVS capability to provide design basis filtered air evacuation from the Spent Fuel Pool area, only ACTION-c is required. If the SBVS is inoperable for any other reason, concurrently implement ACTION-b and ACTION-c.

b. (1) With one SBVS inoperable in MODE 1, 2, 3, or 4, restore the inoperable system to OPERABLE status within 7 days; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

NOTE
Action not applicable when second SBVS intentionally made inoperable.

(2) With both SBVSs inoperable, within 1 hour verify at least one train of containment spray is OPERABLE, and restore at least one SBVS to OPERABLE status within 24 hours; otherwise, be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

c. (1) With one SBVS inoperable in any MODE, restore the inoperable system to OPERABLE status within 7 days; otherwise, suspend movement of recently irradiated fuel assemblies within the Spent Fuel Storage Pool and crane operations with loads over recently irradiated fuel in the Spent Fuel Storage Pool.

(2) With both SBVS inoperable in any MODE, immediately suspend movement of recently irradiated fuel assemblies within the Spent Fuel Storage Pool and crane operations with loads over recently irradiated fuel in the Spent Fuel Storage Pool.

SURVEILLANCE REQUIREMENTS

4.6.6.1 Each Shield Building Ventilation System shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 15 continuous minutes with the heaters on.

5.5.8

In accordance with the frequencies of Regulatory Guide 1.52, Revision 3, except that the testing frequency of “24 months” is required at a frequency in

b. ~~In~~ accordance with the Surveillance Frequency Control Program ~~or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:~~ A12

~~1. Performing a visual examination of SBVS in accordance with ASME N510-1989.~~ LA07

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS (continued)

5.5.8.e 2. Performing airflow distribution to HEPA filters and charcoal adsorbers in accordance with ASME N510-1989. The distribution shall be ± 20% of the average flow per unit.

c. By performing required shield building ventilation system filter testing in accordance with the Ventilation Filter Testing Program.

See ITS 3.6.9

d. In accordance with the Surveillance Frequency Control Program by:

1. Verifying that the system starts on a Unit 2 containment isolation signal and on a fuel pool high radiation signal.

2. Verifying that the filter cooling makeup and cross connection valves can be manually opened.

3. Verifying that each system produces a negative pressure of greater than or equal to 2.0 inches WG in the annulus within 99 seconds after a start signal.

4. Verifying that each system achieves a negative pressure of greater than 0.125 inch WG in the fuel storage building after actuation of a fuel storage building high radiation test signal.

See ITS 3.6.7

See ITS 3.8.6

A01

ITS

TABLE 4 8-2

BATTERY SURVEILLANCE REQUIREMENT

	CATEGORY A[(1)]	CATEGORY B[(2)]	
Parameter	Limits for each designated pilot cell	Limits for each connected cell	Allowable[(3)] value for each connected cell
Electrolyte Level	> Minimum level indication mark, and ≤ 1/4" above maximum level indication mark	> Minimum level indication mark, and ≤ 1/4" above maximum level indication mark	Above top of plates, and not overflowing
5.5.14. b.1 and b.2 Float Voltage	≥ 2.13 volts	≥ 2.13 volts[(c)]	> 2.07 volts
		≥ 1.190	Not more than .020 below the average of all connected cells
Specific Gravity[(a)]	≥ 1.195[(b)]	Average of all connected cells > 1.200	Average of all connected cells ≥ 1.190[(b)]

(a) Corrected for electrolyte temperature and level.

(b) Or battery charging current is less than 2 amps when on charge.

5.5.14 ~~(c) Corrected for average electrolyte temperature.~~

LA05

(1) For any Category A parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that within 24 hours all the Category B measurements are taken and found to be within their allowable values, and provided all Category A and B parameter(s) are restored to within limits within the next 6 days.

(2) For any Category B parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that the Category B parameters are within their allowable values and provided the Category B parameter(s) are restored to within the limits within 7 days.

(3) With any Category B parameter not within its allowable value, declare the battery inoperable.

See ITS 3.8.6

~~Pages 3/4 11-2 through 3/4 11-13 (Amendment No. 61) have been deleted from the Technical Specifications. The next page is 3/4 11-14.~~

~~RADIOACTIVE EFFLUENTS~~

and Storage Tank Radioactivity Monitoring Program

EXPLOSIVE GAS ~~MIXTURE~~

Add proposed ITS 5.5.9 generic program statement

A11

~~LIMITING CONDITION FOR OPERATION~~

5.5.9.a 3.11.2.5 The concentration of oxygen in the waste gas decay tanks shall be limited ~~to less than or equal to 2% by volume whenever the hydrogen concentration exceeds 4% by volume.~~

~~APPLICABILITY: At all times.~~

LA06

~~ACTION:~~

~~a. With the concentration of oxygen in the waste gas decay tank greater than 2% by volume but less than or equal to 4% by volume, reduce the oxygen concentration to the above limits within 48 hours.~~

~~b. With the concentration of oxygen in the waste gas decay tank greater than 4% by volume and the hydrogen concentration greater than 2% by volume, immediately suspend all additions of waste gases to the system and immediately commence reduction of the concentration of oxygen to less than or equal to 2% by volume.~~

~~c. The provisions of Specification 3.0.3 are not applicable.~~

SURVEILLANCE REQUIREMENTS

5.5.9 a and b 4.11.2.5.1 The concentration of oxygen in the waste gas decay tank shall be determined to be within the above limits by ~~continuously*~~ monitoring the waste gases in the ~~on service~~ waste gas decay tank.

5.5.9 a and b 4.11.2.5.2 ~~With the oxygen concentration in the on service waste gas decay tank greater than 2% by volume as determined by Specification 4.11.2.5.1,~~ the concentration of hydrogen in the waste gas decay tank shall be determined to be within the above limits ~~by gas partitioner sample at least once per 24 hours.~~

LA06

~~* When continuous monitoring capability is inoperable, waste gases shall be monitored in accordance with the actions specified for the Waste Gas Decay Tanks Explosive Gas Monitoring System in Chapter 13 of the Updated Final Safety Analysis Report.~~

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

A11

~~RADIOACTIVE EFFLUENTS~~

Explosive Gas and Storage Tank Radioactivity Monitoring Program

~~GAS STORAGE TANKS~~

A11

Add proposed ITS 5.5.9 generic program statement

~~LIMITING CONDITION FOR OPERATION~~

5.5.9.a 3.11.2.6 The quantity of radioactivity contained in each gas storage tank ~~shall be limited to less than or equal to 165,000 curies noble gases (considered as Xe-133).~~ LA06

~~APPLICABILITY: At all times.~~

~~ACTION:~~

~~a. With the quantity of radioactive material in any gas storage tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank.~~

~~b. The provisions of Specification 3.0.3 are not applicable.~~

SURVEILLANCE REQUIREMENTS

5.5.9 a and b 4.11.2.6 The quantity of radioactive material contained in each gas storage tank shall be determined to be within the above limit ~~in accordance with the Surveillance Frequency Control Program when radioactive materials are being added to the tank when reactor coolant system activity exceeds 518.9 μCi/gram DOSE EQUIVALENT XE-133.~~ LA06

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies. A11

DISCUSSION OF CHANGES
ITS 5.5, PROGRAMS AND MANUALS

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG - 1432, Rev. 5.0, "Standard Technical Specifications - Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 CTS 6.8.4.a contains a program requiring leakage testing of primary coolant sources outside containment. Included in the listing of systems is the Shutdown Cooling System. ITS 5.5.2 requires the same program and has renamed the Shutdown Cooling System to the Low Pressure Safety Injection System. This changes the CTS by renaming the low pressure cooling/injection system.

The purpose of this administrative program is to ensure systems outside containment that have the potential to contain highly radioactive liquid following a design basis accident (DBA) are leakage tested periodically. The Shutdown Cooling System performs the function of normal shutdown decay heat removal and low pressure safety injection. The titles Low Pressure Safety Injection System and Shutdown Cooling System may be used interchangeable when discussing the system. Since the purpose of the program is to ensure leak tight systems following a DBA, it is more appropriately to use the safety injection function nomenclature. This change is consistent with the ISTS. The change is designated as administrative and is acceptable because it does not result in technical changes to the CTS.

A03 CTS 6.8.4.f 2) requires the Radioactive Effluent Controls Program to provide limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas and references Appendix B of 10 CFR 20.1001 – 20.2401. ITS contains the same requirement but references Appendix B of 10 CFR 20.1001 – 20.2402 consistent with the ISTS. This changes the CTS by changing the 10 CFR 20 reference from 2401 to 2402.

This change is considered acceptable because the 10 CFR 20 regulation encompasses 1001 – 2402. The change is designated as an administrative change to correct the regulation reference and is acceptable because it does not result in technical change to the CTS programmatic requirement.

A04 **Unit 1 only:** CTS 6.8.4.h contains an exception to Containment Leakage Rate Testing Program requirements required by 10 CFR 50.54(o) and 10 CFR 50 Appendix J in accordance with the guidelines contained in NEI-94-01. CTS 6.8.4.h states, in part, that the next Type A test performed after the December 8, 2005 Type A test shall be performed no later than December 8, 2020. ITS 5.5.13 does not include this Type A test exception. This changes the CTS by removing a 10 CFR 50, Appendix J exception that is no longer applicable.

The Type A test was performed in November 2019. Therefore, this exception to the Type A containment leakage rate testing frequency specified in the regulation and supporting documents is no longer required. The change is designated as administrative and is acceptable because it does not result in technical changes to the CTS.

A05 TS 6.8.4.h states, in part, that the provisions of T.S. 4.0.2 do not apply to test frequencies in the Containment Leak Rate Testing Program. ITS 5.5.13.f states, "Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J." This changes the CTS by exchanging the requirement that CTS 4.0.2 does not apply with the statement to clarify that Containment Leakage Rate Testing Program test frequencies required by a regulation cannot be extended by the use of any Technical Specification allowance, including CTS 4.0.2.

The change is designated as an administrative change to clarify that no generic testing frequency allowance shall be construed to allow modifying a testing Frequency required pursuant to a regulation and is acceptable because it does not result in technical change to the CTS programmatic requirement.

A06 Unit 1 CTS 4.6.6.1.c.1 and 4.7.8.1.c.1 and Unit 2 CTS 4.6.6.1.b.2 require a test to verify airflow distribution to HEPA filters and charcoal adsorbers in accordance with ASME N510-1989 for the Shield Building Ventilation System (SBVS) and the Unit 1 Emergency Core Cooling System (ECCS) Area Ventilation System. Additionally, Unit 1 CTS 4.9.12 provides filter testing requirements for the Fuel Pool Area Ventilation System, which are tested in accordance with ANSI N510-1975 and ASTM D3803-1989, as applicable. ITS includes the Surveillances associated with the ventilation filter testing in the Ventilation Filter Testing Program (ITS 5.5.8). As such, the general program statement has been modified to include, in addition to Regulatory Guide 1.52, the ASME, ANSI, and ASTM Standards applicable to the filter testing and a listing of which Surveillance Frequency requirements are associated with each filter test. This changes the CTS by moving additional ventilation filter testing Surveillances associated with the SBVS, Unit 1 ECCS Area Ventilation System, and Unit 1 Fuel Pool Area Ventilation System to the Ventilation Filter Testing Program.

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A07 CTS 6.8.4 l.c, states the provisions for steam generator (SG) tube repair criteria. ITS 5.5.6.c refers to the SG tube plugging criteria. This changes the CTS by referring to SG plugging criteria instead of "repair" criteria.

SG repair applied to the original SGs. However, the unit contains replacement SGs containing Alloy 690TT Tubing. As further stated in CTS 6.8.4.l.c, if SG tube degradation is discovered, the tubes shall be plugged. Therefore, there is no repair criteria, only plugging criteria, defined in CTS. This change is designated as administrative and is acceptable because it does not result in technical changes to the CTS.

A08 Unit 1 CTS 6.8.4.s and Unit 2 CTS 6.8.4.t state that the Safety Function Determination Program (SFDP) ensures loss of safety function is detected and appropriate actions taken, and other appropriate actions may be taken as a result of the support system inoperability. ITS 5.5.12 states that other appropriate limitations and remedial or compensatory actions may be taken as a result of the support system inoperability. The changes the CTS by adding additional clarification on what other appropriate action may be taken; limitation, remedial, or compensatory.

The addition of the description of the type of action that may be taken is acceptable because it is describing the intent of the CTS SFDP purpose. This change is designated as administrative because it does not result in technical changes to the CTS.

A09 Unit 1 CTS 6.8.4.s and Unit 2 CTS 6.8.4.t format regarding the SFDP refers to remedial actions with their associated prescribed period for completion as "ACTION" and "allowed outage time." ITS format refers to these same remedial action and prescribed completion periods as "Condition and Required Action" and "Completion Time." This changes the CTS by changing how the remedial actions and associated period for action completion is referred to consistent with the ITS format.

The change in format of these remedial action requirements is acceptable because it continues to portray the intent of the CTS. This change is designated as administrative because it does not result in technical changes to the CTS.

A10 **Unit 1 only:** CTS 4.9.12.b.4 requires verifying a system flow rate of 10,350 cfm ± 10% for the required fuel pool area ventilation train during system operation when tested in accordance with ANSI N510-1975. ITS 5.5.8 does not explicitly require this verification because it is duplicative of the flow rate requirement specified in CTS 4.9.12.b.1 and b.2. This changes the CTS by deleting a duplicative requirement to verify fuel pool area ventilation train flowrate while operating the ventilation system. This change is designated as administrative because it does not result in technical changes to the CTS.

A11 The explosive gas mixture requirements in CTS 3.11.2.5 and the gas storage tank requirements of CTS 3.11.2.6 have been placed in a program in the proposed Administrative Controls Chapter 5.0 (ITS 5.5.9). As such, a general program statement has been added. Also, a statement of applicability of ITS SR 3.0.2 and SR 3.0.3 is needed to clarify the allowances for Surveillance Frequency extensions do apply. This changes the CTS by moving the explosive gas mixture and gas storage tank requirements to a program in ITS Section 5.5 and specifically stating the applicability of ITS SR 3.0.2 and SR 3.0.3 in the program.

The addition of the program statement is acceptable because it is describing the intent of the CTS Specification. The addition of ITS SR 3.0.2 and SR 3.0.3 statement is a clarification needed to maintain provisions that are currently allowed in the LCO and SR sections of the CTS, therefore it is considered acceptable. This change is designated as administrative because it does not result in technical changes to the CTS.

A12 **Unit 2 only:** CTS 4.6.6.1.b requires SBVS filter testing, in part, after any structural maintenance on the HEPA filter or charcoal adsorber housings; or following painting, fire, or chemical release in any ventilation zone communicating with the system. ITS 5.5.8 does not include these specific testing frequency requirements for the SBVS filters. CTS 6.8.4.k currently requires filter testing in accordance with Regulatory Guide (RG) 1.52, Revision 3. RG 1.52 contains the similar testing frequency requirements as stated in CTS 4.6.6.1.b. ITS 5.5.8 retains the requirement that the SBVS filter testing be performed, as applicable, in accordance with Regulatory Guide (RG) 1.52, Revision 3. Therefore, CTS 4.6.6.1.b testing frequency requirements specified herein are duplicative to those required by CTS 6.8.4.k (ITS 5.5.8) and are unnecessary to include in the ITS. As a result, this change is designated as administrative because it does not result in technical changes to the CTS.

<u>MORE RESTRICTIVE CHANGES</u>

M01 **Unit 1 only:** CTS does not provide administrative controls for reactor RCP flywheel inspections. ITS 5.5.5, "Reactor Coolant Pump (RCP) Flywheel Inspection Program" is added to the ITS. The proposed program will provide for the inspection of each RCP flywheel per the recommendation of Regulatory position c.4.b of Regulatory Guide 1.14, Revision 1, August 1975 except for an RCP flywheel consisting of ASTM A-516-69 (i.e., SA-516) Grade 65 material. For an RCP flywheel composed of ASTM A-516-69 Grade 65 material, the inspection shall consist of either a 100% volumetric inspection of the upper flywheel over the volume from the inner bore of the flywheel to the circle of one-half the outer radius or a surface examination (magnetic particle testing and/or penetrant testing) of exposed surfaces defined by the volume of the disassembled flywheel at least once every 10 years. This changes the CTS by adding an additional administrative control program.

The purpose of the RCP flywheel program is to implement requirements to minimize the potential for failures of the flywheels of RCP motors in light-water-cooled power reactors. At PSL, RCP flywheels are interchanged between Unit 1 and Unit 2 during refurbishment and replacement. Between 2011 and 2017, the Unit 1 RCP flywheels, which consist of ASTM A-516-69, Grade 65 material, were exchanged with the Unit 2 RCP flywheels, which consist of ASTM A543 (three flywheels Class 1, Type B and one flywheel Grade 70) material. RCP flywheels with ASTM A543 material are inspected per Regulatory Guide 1.14, Revision 1, August 1975. As such, the added program will require the Unit 1 RCP flywheels to be inspected per the recommendation of Regulatory position c.4.b of Regulatory Guide 1.14 consistent with the ISTS. Additionally, the requirements specified for the PSL RCP flywheels composed of ASTM A-516-69 Grade 65 material are included in the ITS to support future RCP flywheel exchanges between Unit 1 and Unit 2. These requirements are consistent with the current requirements in Unit 2 CTS 6.8.4.o that were approved in Unit 2 Amendment No. 205, dated November 18, 2020 (ADAMS Accession No. ML20259A298). This change is designated as more restrictive because it imposes additional programmatic requirements in the Technical Specifications.

M02 CTS does not include a requirement for the Battery Monitoring and Maintenance Program. The ITS includes a requirement for this program. This changes the CTS by adding the ITS 5.5.14, "Battery Monitoring and Maintenance Program."

The Battery Monitoring and Maintenance Program is included to provide for battery restoration and maintenance per the guidance of IEEE-450. The specific wording associated with this program may be found in ITS 5.5.14. The Notice of Availability for TSTF-500, Revision 2, "DC Electrical Rewrite - Update to TSTF-360," (*76FR54510*) references the model application and safety evaluation for plant-specific adoption of TSTF–500, Revision 2 (NRC ADAMS Accession No. ML111751792). PSL has verified the applicable information specified in Section 2.2 of the TSTF-500 model application, including applicable UFSAR information. PSL will update the UFSAR, as necessary, to include any UFSAR information listed in Section 2.2 of the TSTF-500 model application that is not currently reflected in the PSL Unit 1 and Unit 2 UFSARs. This change is acceptable because it supports implementation of the requirements of the ITS. This change is designated as more restrictive because it imposes additional programmatic requirements in the Technical Specifications.

M03 CTS 6.14, in part, requires changes to the Offsite Dose Calculation Manual (ODCM) to be documented with a list of items contained in the documentation. ITS 5.5.1 also requires the same documentation and includes a requirement that records of reviews performed shall be retained. This change is necessary to incorporate the guidance related to ODCM changes specified in NRC Generic Letter GL 89-01, which states, "…the staff has concluded that records of licensee reviews performed for changes made to the ODCM… should be documented and retained for the duration of the unit operating license. This approach is in lieu of the current requirements that the reasons for changes to the ODCM… be addressed in the Semiannual Effluent Release Report." This change is designated as more restrictive because it imposes additional programmatic requirements in the Technical Specifications.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 5 – Removal of SR Frequency to the Surveillance Frequency Control Program)* CTS 6.8.4.a contains a program requiring leakage testing of primary coolant sources outside containment with a stated frequency of every 18 months. CTS 6.8.4.k.5 contains a program requiring testing of the Shield Building Ventilation System (SBVS) filter heaters testing every 18 months. CTS 6.8.4.m.d requires measurement, at designated locations, of the control room envelope (CRE) pressure relative to external areas adjacent to the CRE boundary during the pressurization mode of operation of the CREVS, operating at the flow rate required by the VFTP, at a frequency of 36 months on a Staggered Test Basis. CTS 6.8.4.n (iii) requires that total particulate concentration of the diesel generator fuel oil is ≤ 10 mg/l when tested every 31 days. ITS 5.5.2, 5.5.8, 5.5.15, and 5.5.10 contain similar requirements, respectively, but specify the periodic Frequency as "In accordance with the Surveillance Frequency Control Program." This changes the CTS by moving the specified periodic Frequency for these tests to the Surveillance Frequency Control Program.

The purpose of these Surveillances is to assure that the necessary quality of systems and components is maintained. The removal of these details related to Surveillance Requirement Frequencies from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The existing Frequency is removed from Technical Specifications and placed under licensee control pursuant to the methodology described in NEI 04-10. The surveillance test requirements remain in the Technical Specifications. The control of changes to the Surveillance Frequencies is in accordance with the Surveillance Frequency Control Program, which is retained in ITS Section 5.5. The Surveillance Frequency Control Program provides the necessary administrative controls to require that surveillances related to testing, calibration and inspection are conducted at a frequency to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met pursuant to the requirements of 10 CFR 50.36(c)(3). The proposed change to relocate periodic frequencies in the administrative controls section of Technical Specifications has been previously approved for Wolf Creek Generating Station Unit 1 in Amendment 227, dated April 8, 2021 (NRC ADAMS Accession No. ML21053A117), River Bend Station Unit 1 in Amendment 196, dated April 29, 2019 (NRC ADAMS Accession No. ML19066A008), and Grand Gulf Nuclear Station Unit 1 in Amendment 219, dated June 11, 2019 (NRC ADAMS Accession No. ML19094A799). PSL Unit 1 and Unit 2 adopted a Surveillance Frequency Control Program in Amendment Nos. 223 (Unit 1) and 173 (Unit 2) (ADAMS Accession No. ML15127A066) as contained in CTS 6.8.4.o (Unit 1) and 6.8.4.q (Unit 2) (ITS 5.5.16). This change is acceptable because the testing frequencies will be adequately controlled in accordance with the Surveillance Frequency Control Program requirements retained in ITS, which ensure changes are properly evaluated. This change is designated as a less restrictive removal of detail change, because the Surveillance Frequencies are being removed from the Technical Specifications.

LA02 *(Type 4 – Removal of LCO, SR, or other TS Requirement to the TRM, UFSAR, ODCM, NQAP, CLRT Program, IST Program, or ISI Program)* CTS 6.8.4.b, "In-Plant Radioiodine Monitoring," describes a program to ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. ITS 5.5 does not include this program. This changes the CTS by moving the requirements for the In-Plant Radioiodine Monitoring Program to the Technical Requirements Manual (TRM).

The removal of this requirement from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The CTS 6.8.4.b program is designed to minimize radiation exposure to plant personnel in vital areas of the plant after an accident and has no impact on nuclear safety or the health and safety of the public. This change is acceptable because the program requirements will be adequately controlled in the TRM. Changes to the TRM will be controlled by the provisions of 10 CFR 50.59, which ensure changes are properly evaluated. This change is designated as a less restrictive removal of requirement change because requirements are being removed from the Technical Specifications.

LA03 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 6.8.4 contains a Backup Method for Determining Subcooling Margin Program. This program ensures the capability to accurately monitor RCS subcooling margin by training and procedures. ITS 5.5 does not include this administrative program. This changes the CTS by moving administrative procedural detail to the updated Final Safety Analysis Report (UFSAR).

The removal of these administrative procedural details is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS retains the requirement to maintain the post accident monitoring instrument channels OPERABLE in MODES 1, 2, and 3 that are used to determine RCS subcooling margin and provide input into the RCS subcooling margin monitor (e.g. core exit thermocouples, RCS hot and cold leg temperature, and pressurizer pressure). Also, this change is acceptable because the removed information will be adequately controlled in the UFSAR. The UFSAR is controlled under 10 CFR 50.59 which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LA04 *(Type 4 – Removal of LCO, SR, or other TS Requirement to the TRM, UFSAR, ODCM, NQAP, CLRT Program, IST Program, or ISI Program)* CTS 6.8.4.g, "Radiological Environmental Monitoring Program," describes a program to monitor the radiation and radio-nuclides in the environs of the plant. ITS Section 5.5 does not require this program. This changes the CTS by moving the requirements for the Radiological Environmental Monitoring Program to the ODCM.

The purpose of these program requirements is to provide representative measurements of radioactivity in the highest potential exposure pathways, and verification of the accuracy of the effluent monitoring program and modeling of environmental exposure pathways. The removal of the requirement for this program from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. ITS 5.6.2, “Radiological Effluent Release Report,” continues to require an annual report of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. Also, this change is acceptable because these types of procedural details will be adequately controlled in the ODCM. Changes to the ODCM are controlled by the ODCM change control process specified in ITS 5.5.1, which ensures changes are properly evaluated. This change is designated as a less restrictive removal of requirement change because the requirements for a program are being removed from the Technical Specifications.

LA05 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS Table 4.8-2 footnote (c) states that the specified individual battery cell voltage limit for each connected cell is corrected for average electrolyte temperature. ITS 5.5.14 does not include the procedural detail regarding corrected cell voltage based on electrolyte temperature. This changes the CTS by moving procedural information from the Specification to the Battery Monitoring and Maintenance Program implementing document.

The removal of these details from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. ITS 5.5.14 retains the requirement specifying the individual battery cell voltage limit. Also, this change is acceptable because these types of procedural details will be adequately controlled by the requirements of a program required by ITS Section 5.5. Changes to the program requirements will be controlled by the provisions of 10 CFR 50.59, which ensure changes are properly evaluated. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LA06 *(Type 3 - Removing Procedural Detail for Meeting TS Requirements or Reporting Requirements)* CTS 3.11.2.5 includes the details for implementing the requirements for the explosive gas mixture. CTS 3.11.2.6 includes the details for implementing the requirements for the gas storage tanks. The details for implementing these requirements, including the specific limits for the explosive gas mixture and limit for quantity of radioactivity contained the gas storage tanks, are not included in the ITS. The ITS only includes a requirement to maintain a program for these requirements. This changes the CTS by moving these procedural details for implementing the requirements, including the specific limits, from the Technical Specifications to the TRM.

The removal of these details for the specific explosive gas limits and limits on quantity of radioactivity in the gas storage tanks, including the Applicability, Actions, and Surveillance Requirements, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. ITS 5.5.9 retains the requirement to include a program, which provides controls for potentially explosive gas mixtures contained in the waste gas decay tanks and the quantity of radioactivity contained in the gas storage tanks. Also, this change is acceptable because these types of procedural details will be adequately controlled in the TRM. Changes to the TRM will be controlled by the provisions of 10 CFR 50.59, which ensure changes are properly evaluated. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LA07 **Unit 2 only:** *(Type 3 – Removing Procedural Details for Meeting TS Requirements and Related Reporting Problems)* CTS 4.6.6.1.b.1 requires a visual examination of the SBVS in accordance with ASME 510-1989. The ITS does not include this requirement. This changes the CTS by moving this requirement to the UFSAR.

The purpose of CTS 4.6.6.1.b is to ensure that in accordance with the Surveillance Frequency Control Program or; after any structural maintenance on the HEPA filter or charcoal adsorber housings; or following painting, fire, or chemical release in any ventilation zone communicating with the system, a visual inspection is performed on the SBVS. This is a good housekeeping practice that should be part of any recovery of a system from maintenance activities and periodically to visually identify gross system anomalies that would indicate the system is degraded. However, this detail is not necessary to be included in the ITS to provide adequate protection of the public health and safety. ITS 3.6.9 and ITS 5.5.8 retains Surveillance requirements and filter testing requirements to ensure the SBVS is OPERABLE. Also, this change is acceptable because the removed information will be adequately controlled in the UFSAR. Changes to the UFSAR are controlled by 10 CFR 50.59 or 10 CFR 50.71(e), which ensures that any changes to the UFSAR are properly evaluated. This change is designated as a less restrictive removal of detail change because information relating to meeting a TS requirement is being removed from the Technical Specifications.

<u>LESS RESTRICTIVE CHANGES</u>

L01 *(Category 7 – Relaxation of Surveillance Frequency)* CTS 6.8.4.I.d.2 requires, in part, inspecting 100% of the steam generator (SG) tubes at sequential periods of 144, 108, 72, and thereafter 60 effective full power months (EFPMs) and, in addition, requires inspecting 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outages nearest the end of the period. No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected. ITS 5.5.6.d.2 requires inspection of 100% of the tubes in each steam generator at least every 96 EFPMs. CTS 6.8.4.I.d.3 requires additional inspection for each SG when crack indications are found in any tube at a frequency not to exceed 24 EFPMs or one refueling outage (whichever is less). ITS 5.5.6.d.3 requires the additional inspection on each affected and potentially affected SG at the next refueling outage. This changes the CTS by modifying the inspection frequency to a single requirement to inspect 100% of the SG tubes at a maximum frequency of 96 EFPMs and modifying the inspection frequency when crack indications are discovered to the next refueling outage.

The purpose of the inspection frequencies associated with the SG tubes is to ensure appropriate inspections are performed consistent with accepted NRC and industry practice as identified in Nuclear Energy Institute (NEI) 97-06, “Steam Generator Program Guidelines,” and its referenced Electric Power Research Institute SG examination guidelines, which establish the content of the Steam Generator Program. These guidelines minimize the potential of SG tube failures to support maintaining SG and reactor coolant pressure boundary structural integrity.

The proposed changes are consistent with the ISTS and Technical Specification Task Force (TSTF) traveler TSTF-577-A, Revision 1, "Revised Frequencies for Steam Generator Tube Inspections." PSL Units 1 and 2 SGs contain Alloy 690 thermal treated (TT) tubes. For Alloy 690TT tubing, TSTF-577, which is incorporated in Revision 5 of the ISTS, revised the frequencies related to 100% and 50% inspection of the tubes such that both the maximum time between inspections and the time to inspect 100 percent of the tubes be 96 EFPMs. TSTF-577 also revised the frequency when crack indications are found in any tube to eliminate 24 EFPMs and require the inspection at the next refueling outage. The nuclear industry's Steam Generator Task Force presented a technical basis supporting the 96 EFPM Alloy 690TT inspection interval during the February 13, 2019 (ADAMS Package Accession No. ML19044A416), and February 24, 2020 (ADAMS Package Accession No. ML20066E421), public meetings with the NRC staff.

As stated in the NRC Safety Evaluation (SE) accompanying TSTF-577-A, proposed changes in TSTF-577 are acceptable because they continue to ensure SG tube integrity and, therefore, protect the public health and safety. In particular, the structural integrity performance criterion and accident-induced leakage performance criterion will continue to be met with the proposed revised SG inspection intervals (maximum allowable time between SG inspections) and inspection periods (maximum allowable time between 100 percent of SG tubes inspections). That considered, the changes to the SG tube inspection frequencies are acceptable. For Unit 1, inspection of 100% of the tubes in each SG was completed in the Spring 2021 refueling outage. For Unit 2, inspection of 100% of the tubes in each SG was completed for the current inspection period prior to the Fall 2021 refueling outage. Therefore, the 96 EFPM inspection period will begin from these SG tube inspection outages.

TSTF traveler TSTF-577-A incorporated changes to the Standard Technical Specifications (STSs) under the consolidated line item improvement process (CLIIP). TSTF-577-A was approved for use by the NRC as documented in the accompanying SE dated April 14, 2021 (ADAMS Accession No. ML21098A188, ML21096A274). PSL has reviewed the NRC SE and concluded that the justification presented in TSTF-577-A and the SE prepared by the NRC staff are applicable to PSL and justify this change.

This change is designated as less restrictive because the maximum inspection frequencies for the 100% and 50% tube inspections was extended to beyond the current inspection periods.

L02 **Unit 2 only:** CTS 6.8.4.o requires administrative controls for reactor coolant pump (RCP) flywheel inspections. ITS 5.5.5 includes the same requirements and includes the CTS 6.8.4.o program requirements that were issued prior to Amendment No. 205 with a distinction on applicability of the two sets of requirements. The proposed program will provide for the inspection of each RCP flywheel per the recommendation of Regulatory position c.4.b of Regulatory Guide 1.14, Revision 1, August 1975 except for an RCP flywheel consisting of ASTM A-516-69 (i.e., SA-516) Grade 65 material. For an RCP flywheel composed of ASTM A-516-69 Grade 65 material, the inspection shall consist of either a 100% volumetric inspection of the upper flywheel over the volume from the inner bore of the flywheel to the circle of one-half the outer radius or a surface examination (magnetic particle testing and/or penetrant testing) of exposed surfaces defined by the volume of the disassembled flywheel at least once every 10 years. This changes the CTS by restoring program requirements that were issued prior to Amendment No. 205 in addition to the current requirements.

The purpose of the RCP flywheel inspection program is to implement requirements to minimize the potential for failures of the flywheels of RCP motors in light-water-cooled power reactors. At PSL, RCP flywheels are interchanged between Unit 1 and Unit 2 during refurbishment and replacement. Between 2011 and 2017, the Unit 2 RCP flywheels, which consist of ASTM A-543 (three flywheels Class 1, Type B and one flywheel Grade 70) material, were exchanged with the Unit 1 RCP flywheels, which consist of ASTM A-516-69, Grade 65 material. Unit 2 CTS 6.8.4.o was modified to relax the RCP flywheel inspection requirements similar to the conclusions specified in the NRC Safety Evaluation of Topical Report SIR-94-080, Revision 1. The relaxed inspection requirements were approved in Amendment No. 205, dated November 18, 2020 (ADAMS Accession No. ML20259A298). Previously, PSL Unit 2 RCP flywheels were inspected per NRC Regulatory Guide 1.14, "Reactor Coolant Pump Flywheel Integrity," Revision 1 and the requirements were previously contained in the CTS pre-Amendment 205 requirements. NRC Regulatory Guide 1.14 describes a method acceptable to the NRC staff of implementing RCP flywheel inspection requirements and include a recommended inspection schedule. Since the PSL RCP flywheels consisting of ASTM A-543 material were previously inspected per Regulatory Guide 1.14, Revision 1, August 1975 and the requirements were previously approved for use for Unit 2, the pre-Amendment 205 requirements are proposed to be restored in support of future RCP flywheel exchanges between Unit 1 and Unit 2. This change is designated as less restrictive because RCP flywheel inspection requirements, which have been previously approved for use at PSL, have been added.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

The following programs shall be established, implemented, and maintained.

Definition 1.18

5.5.1 Offsite Dose Calculation Manual (ODCM)

a. The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the radiological environmental monitoring program and

b. The ODCM shall also contain the radioactive effluent controls and radiological environmental monitoring activities, and descriptions of the information that should be included in the Annual Radiological Environmental Operating, and Radioactive Effluent Release Reports required by Specification ~~[~~5.6.1~~]~~ and Specification ~~[~~5.6.2~~]~~. 2

Licensee initiated changes to the ODCM:

6.14.1 DOC M03

a. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

1. Sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s) and

2. A determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.1302, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations,

6.14.2

b. Shall become effective after the approval of the plant manager, and

6.14.3

c. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

5.5 Programs and Manuals

6.8.4.a 5.5.2 Primary Coolant Sources Outside Containment

High Pressure Safety Injection, Containment

Safety

Reactor Coolant Sampling

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: 2

a. Preventive maintenance and periodic visual inspection requirements and

b. Integrated leak test requirements for each system at ~~least once per [18] months~~.

a Frequency in accordance with the Surveillance Frequency Control Program

The provisions of SR 3.0.2 are applicable. 3

~~[5.5.3 Post Accident Sampling~~ 2

~~-----REVIEWER'S NOTE-----~~
~~This program may be eliminated based on the implementation of Topical Report CE NPSD-1157, Rev. 1, "Technical Justification for the Elimination of the Post-Accident Sampling System from the Plant Design and Licensing Basis for CEOG Utilities," and the associated NRC Safety Evaluation dated May 16, 2000, and implementation of the following commitments:~~

~~1. [Licensee] has developed contingency plans for obtaining and analyzing highly radioactive samples of reactor coolant, containment sump, and containment atmosphere. The contingency plans will be contained in emergency plan implementing procedures and implemented with the implementation of the License amendment. Establishment of contingency plans is considered a regulatory commitment.~~

~~2. The capability for classifying fuel damage events at the Alert level threshold has been established for [PLANT] at radioactivity levels of 300 mCi/cc dose equivalent iodine. This capability may utilize the normal sampling system and/or correlations of sampling or letdown line dose rates to coolant concentrations. This capability will be described in emergency plan implementing procedures and implemented with the implementation of the License amendment. The capability for classifying fuel damage events is considered a regulatory commitment.~~

~~3. [Licensee] has established the capability to monitor radioactive iodines that have been released to offsite environs. This capability is described in our emergency plan implementing procedures. The capability to monitor radioactive iodines is considered a regulatory commitment..~~
~~----------~~

~~This program provides controls that ensure the capability to obtain and analyze reactor coolant, radioactive gases, and particulates in plant gaseous effluents~~

5.5 Programs and Manuals

~~5.5.3 Post Accident Sampling (continued)~~ 2

~~and containment atmosphere samples under accident conditions. The program shall include the following:~~ 4

~~a. Training of personnel,~~

~~b. Procedures for sampling and analysis, and~~

~~c. Provisions for maintenance of sampling and analysis equipment.]~~

6.8.4.f 5.5.4 Radioactive Effluent Controls Program 4

3

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

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

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to ten times the concentration values in Appendix B, Table 2, Column 2 to 10 CFR 20.1001-20.2402,

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

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

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

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,

5.5 Programs and Manuals

6.8.4.f 5.5.~~4~~ 3 Radioactive Effluent Controls Program (continued) 4

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents from the site to areas at or beyond the site boundary 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,

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.

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

6.8.4.q 5.5.~~5~~ 4 Component Cyclic or Transient Limit 1 4 2

This program provides controls to track the U FSAR, Section ~~[ ]~~ 5.2, cyclic and transient occurrences to ensure that components are maintained within the design limits.

~~[5.5.6 Pre-Stressed Concrete Containment Tendon Surveillance Program~~ 2 4

~~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 Section XI, Subsection IWL of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10CFR 50.55a, except where an alternative, exemption, or relief has been authorized by the NRC.~~

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

5.5 Programs and Manuals (RCP) 1

5.5.~~7~~ 5 Reactor Coolant Pump Flywheel Inspection Program 4

DOC M01

This program shall provide for the inspection of each ~~reactor coolant pump~~ RCP flywheel per the recommendation of Regulatory position c.4.b of Regulatory Guide 1.14, Revision 1, August 1975. Insert 1 1 5

5.5.~~8~~ 6 Steam Generator (SG) Program 4

6.8.4.l An SG Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the SG Program shall include the following:

6.8.4.l.a a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging ~~[or repair]~~ of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected, plugged, ~~[or repaired]~~ to confirm that the performance criteria are being met. 2

6.8.4.l.b b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.

6.8.4.l.b.1 1. Structural integrity performance criterion: All in-service SG tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down), all anticipated transients included in the design specification, and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

6.8.4.l.b.2 2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for

CTS

5

Insert 1

DOC M01

, except for an RCP flywheel consisting of ASTM A-516-69 (i.e., SA-516) Grade 65 material.

For an RCP flywheel composed of ASTM A-516-69 Grade 65 material, the inspection shall consist of either a 100% volumetric inspection of the upper flywheel over the volume from the inner bore of the flywheel to the circle of one-half the outer radius or a surface examination (magnetic particle testing and/or penetrant testing) of exposed surfaces defined by the volume of the disassembled flywheel at least once every 10 years.

5.5 Programs and Manuals

5.5.~~8~~ 6 Steam Generator (SG) Program (continued) 4

6.8.4.l all SGs and leakage rate for an individual SG. Leakage is not to exceed ~~[1 gpm] per~~ 0.5 gpm total through all SGs and 0.25 gpm through any one SG. ~~[, except for specific types of degradation at specific locations as described in paragraph c of the SG Program.]~~

6.8.4.l.b.3 3. The operational LEAKAGE performance criterion is specified in LCO 3.4.13, "RCS Operational LEAKAGE."

6.8.4.l.c c. Provisions for SG tube plugging ~~[or repair]~~ criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding ~~[~~40%~~]~~ of the nominal tube wall thickness shall be plugged ~~[or repaired]~~.

~~REVIEWER'S NOTE~~
~~Alternate tube plugging [or repair] criteria currently permitted by plant technical specifications are listed here. The description of these alternate tube plugging [or repair] criteria should be equivalent to the descriptions in current technical specifications and should also include any allowed accident induced leakage rates for specific types of degradation at specific locations associated with tube plugging [or repair] criteria.~~ 6

~~[The following alternate tube plugging [or repair] criteria may be applied as an alternative to the 40% depth based criteria:~~ 2

~~1. . . .]~~

~~REVIEWER'S NOTE~~
~~The bracketed phrase in Paragraph d regarding exempt portions of the tube is only applicable to SGs with Alloy 600 thermally treated tubing.~~ 6

6.8.4.l.d d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet ~~[except for any portions of the tube that are exempt from inspection by alternate repair criteria]~~, and that may satisfy the applicable tube plugging ~~[or repair]~~ criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. A degradation assessment shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

5.5 Programs and Manuals

5.5.~~8~~ 6 Steam Generator (SG) Program (continued)

4

~~-----REVIEWER'S NOTE-----~~
~~Plants are to include the appropriate Frequency (e.g., select the appropriate Item 2.) for their SG design. The first Item 2 is applicable to SGs with Alloy 600 mill annealed tubing. The second Item 2 is applicable to SGs with Alloy 600 thermally treated tubing. The third Item 2 is applicable to SGs with Alloy 690 thermally treated tubing.~~ 6
~~----------~~

6.8.4.l.d.1 1. Inspect 100% of the tubes in each SG during the first refueling outage following SG installation.

~~[2. After the first refueling outage following SG installation, inspect 100% of the tubes in each SG at least every 24 effective full power months, which defines the inspection period.]~~

~~[2. After the first refueling outage following SG installation, inspect 100% of the tubes in each SG at least every 54 effective full power months, which defines the inspection period. If none of the SG tubes have ever experienced cracking other than in regions that are exempt from inspection by alternate repair criteria and the SG inspection was performed with enhanced probes, the inspection period may be extended to 72 effective full power months. Enhanced probes have a capability to detect flaws of any type equivalent to or better than array probe technology. The enhanced probes shall be used from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet except any portions of the tube that are exempt from inspection by alternate repair criteria. If there are regions where enhanced probes cannot be used, the tube inspection techniques shall be capable of detecting all forms of existing and potential degradation in that region.]~~ 2

6.8.4.l.d.2
DOC L01 ~~[~~2. After the first refueling outage following SG installation, inspect 100% of the tubes in each SG at least every 96 effective full power months, which defines the inspection period. ~~]~~

~~-----REVIEWER'S NOTE-----~~
~~The bracketed phrases in Paragraph 3 are only applicable to SGs with Alloy 600 thermally treated tubing.~~ 6
~~----------~~

6.8.4.l.d.3 3. If crack indications are found in any SG tube ~~[excluding any region that is exempt from inspection by alternate repair criteria]~~, then the next
DOC L01 inspection for each ~~affected and potentially affected~~ SG for the degradation mechanism that caused the crack indication shall be at the next refueling outage ~~[, but may be deferred to the following refueling~~ 2 1

5.5 Programs and Manuals

5.5.~~8~~ 6 Steam Generator (SG) Program (continued) 4

6.8.4.l.d.3 ~~outage if the 100% inspection of all SGs was performed with enhanced probes as described in paragraph d.2]~~. If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack. 2

6.8.4.l.e e. Provisions for monitoring operational primary to secondary LEAKAGE.

~~[f. Provisions for SG tube repair methods. SG tube repair methods shall provide the means to reestablish the RCS pressure boundary integrity of SG tubes without removing the tube from service. For the purposes of these Specifications, tube plugging is not a repair. All acceptable tube repair methods are listed below.~~

~~-----REVIEWER'S NOTE-----~~

~~Tube repair methods currently permitted by plant technical specifications are to be listed here. The description of these tube repair methods should be equivalent to the descriptions in current technical specifications. If there are no approved tube repair methods, this section should not be used.~~ 2 6

~~-----~~

~~1. . . .]~~

6.8.4.c 5.5.~~9~~ 7 Secondary Water Chemistry Program 4

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation ~~and low pressure turbine disc stress corrosion cracking~~. The program shall include: 1

6.8.4.c (i) a. Identification of a sampling schedule for the critical variables and control points for these variables,

6.8.4.c (ii) b. Identification of the procedures used to measure the values of the critical variables,

6.8.4.c (iii) c. Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in leakage,

6.8.4 (iv) d. Procedures for the recording and management of data,

6.8.4 (v) e. Procedures defining corrective actions for all off control point chemistry conditions, and

5.5 Programs and Manuals

5.5.~~9~~ 7 Secondary Water Chemistry Program (continued) 4

6.8.4 (vi) f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.

5.5.~~10~~ 8 Ventilation Filter Testing Program (VFTP) 4

6.8.4 k

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] at the system flowrate specified below [± 10%]~~. Insert 2 2 1

6.8.4 k.1

a. Demonstrate for each of the ESF systems that an inplace test of the high efficiency particulate air (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%~~]~~. 1

ESF Ventilation System	Flowrate
~~[ ]~~	~~[ ]~~ Insert 3

6.8.4 k.2

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%~~]~~. 1 2

Insert 4

6.8.4 k.3

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]~~ 3, shows the methyl iodide penetration ~~less than the value specified below~~ ≤ 2.5% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F) and ~~the~~ a relative humidity ~~specified below~~ of 70%. 1

ESF Ventilation System	~~Penetration~~	~~RH~~	~~Face Velocity~~
~~[ ]~~ CREVS SBVS ECCS Area Ventilation System	~~[See Reviewer's Note]~~	~~[See Reviewer's Note]~~	~~[See Reviewer's Note]~~

~~------------------------------------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.~~ 6

1

Insert 2

6.8.4.k 4.9.12.b and the Fuel Pool Area Ventilation System in accordance with Regulatory Guide 1.52, Revision 3, ASME N510-1989, ASTM D3803-1989 and, ANSI N510-1975, as described herein.

6.8.4.k The tests described in Specification 5.5.8.a through 5.5.8.d shall be performed at the frequencies specified in Regulatory Guide 1.52, Revision 3.

4.9.12.b 4.9.12.c 4.6.6.1.c 4.7.8.1.c The tests described in Specification 5.5.8.e through 5.5.8.k shall be performed at a frequency in accordance with the Surveillance Frequency Control Program.

4.9.12.b The tests described in Specification 5.5.8.g through 5.5.8.i shall be performed after any structural maintenance on the HEPA filter or charcoal adsorber housings; and following painting, fire or chemical release in any ventilation zone communicating with the system.

4.9.12.d 4.9.12.e The tests described in Specification 5.5.8.g and 5.5.8.h shall be performed after each complete or partial replacement of a HEPA filter bank.

Insert 3

6.8.4.k..1

Control Room Emergency Ventilation System (CREVS)	2000
Shield Building Ventilation System (SBVS)	6000
Emergency Core Cooling System (ECCS) Area Ventilation System	30,000

Insert 4

6.8.4.k.2

ESF Ventilation System	Flowrate
CREVS	2000
SBVS	6000
ECCS Area Ventilation System	30,000

5.5 Programs and Manuals

5.5.~~10~~ 8 Ventilation Filter Testing Program (continued) 4

~~ASTM D 3803-1989 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.~~

~~Allowable Penetration = [(100% - Methyl Iodide Efficiency * for Charcoal Credited in Licensee's Accident Analysis) / 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 ≥ 2 for systems with or without humidity control.~~ 6

~~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.~~

6.8.4 k.4 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 below ~~[~~± 10%~~]~~. 1

ESF Ventilation System	Delta P	Flowrate
~~[ ]~~	~~[ ]~~	~~[ ]~~ Insert 5

2

6.8.4 k.5 f ~~[e~~. Demonstrate that the ~~heaters for each of the ESF systems dissipate the value specified below [± 10%]~~ when tested in accordance with Insert 6 ~~[~~ASME N510-1989~~]~~. 1

~~ESF Ventilation System~~	~~Wattage]~~
~~[ ]~~	~~[ ]~~

Insert 7 1

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.

Insert 5

6.8.4.k.4

ESF Ventilation System	Delta P (in wg)	Flowrate
CREVS	< 4.15	2000
SBVS	≤ 6.15	6000
ECCS Area Ventilation System	< 4.15	30,000

4.6.6.1.c.1
4.7.8.1.c.1

e. Demonstrate for the SBVS and ECCS Area Ventilation System that the air flow distribution across HEPA filters and charcoal adsorbers is uniform within 20% when tested in accordance with ASME N510-1989.

Insert 6

6.8.4.k.5

SBVS main heaters dissipate 30 kW ± 10% and the SBVS auxiliary heaters dissipate ≥ 1.25 kW and ≤ 1.75 kW

CTS

1

Insert 7

4.9.12.b.2
4.9.12.d

g. Demonstrate for the Fuel Pool Area Ventilation System that an inplace test of the HEPA filters shows a removal efficiency of ≥ 99% of dioctyl phthalate when tested in accordance with ANSI N510-1975 at a system flowrate of 10,350 cfm ± 10%.

4.9.12.b.1
4.9.12.e

h. Demonstrate for the Fuel Pool Area Ventilation System that an inplace test of the charcoal adsorbers shows a removal efficiency ≥ 99% of a halogenated hydrocarbon refrigerant test gas when tested in accordance with ANSI N510-1975 at a system flowrate of 10,350 cfm ± 10%.

4.9.12.b.3

i. Demonstrate for the Fuel Pool Area Ventilation System that a laboratory test of a sample of the charcoal adsorber, when obtained as described herein, shows a removal efficiency of ≥ 85% for radioactive methyl iodide when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F) and relative humidity of 95%.

A carbon sample shall be obtained from either one test canister or two carbon samples removed from one of the charcoal adsorbers. Carbon samples not obtained from test canisters shall be prepared by emptying:

4.9.12.b.3 a)

1. one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

4.9.12.b.3 b)

2. a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

4.9.12.c.1

j. Demonstrate for the Fuel Pool Area Ventilation System that the pressure drop across the combined HEPA filters and the charcoal adsorbers is < 4.15 inches water gauge when tested at a system flowrate of 10,350 cfm ± 10%.

4.9.12.c.2

k. Demonstrate for the Fuel Pool Area Ventilation System that the air flow distribution across HEPA filters and charcoal adsorbers is uniform within 20% when tested in accordance with ANSI N510-1975.

5.5 Programs and Manuals

5.5.~~11~~ 9 Explosive Gas and Storage Tank Radioactivity Monitoring Program (4)

3/4.11.2.5
3/4.11.2.6

This program provides controls waste gas decay tanks and 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 gaseous radioactivity quantities shall be determined following the methodology in [Branch Technical Position (BTP) ETSB 11-5, "Postulated Radioactive Release due to Waste Gas System Leak or Failure"]. The liquid radwaste quantities shall be determined in accordance with [Standard Review Plan, Section 15.7.3, "Postulated Radioactive Release due to Tank Failures"].~~ (2) (7)
The program shall include:

3.11.2.5
3.11.2.6

a. The limits for concentrations of hydrogen and oxygen in the ~~[Waste Gas Holdup System]~~ waste gas decay tanks and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion), and (2)

4.11.2.5
4.11.2.6

b. A surveillance program to ensure that the quantity of radioactivity contained in ~~[~~each gas storage tank ~~and fed into the offgas treatment system]~~ is less than the amount that would result in a whole body exposure of ≥ 0.5 rem to any individual in an unrestricted area, in the event of ~~[~~an uncontrolled release of the tanks' contents~~]~~, and (2)

~~c. A surveillance program to ensure that the quantity of radioactivity contained in all outdoor liquid radwaste tanks that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the [Liquid Radwaste Treatment System] is less than the amount that would result in concentrations less than the limits of 10 CFR 20, Appendix B, Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tanks' contents.~~ (7)

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

5.5 Programs and Manuals

5.5.~~12~~ 10 Diacritics none — **Diesel Fuel Oil Testing Program** 4

6.8.4.n 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:

6.8.4.n (i) 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 or a water and sediment content within limits,

6.8.4.n (ii) b. Within 31 days following addition of the new fuel oil to storage tanks, verify that the properties of the new fuel oil, other than those addressed in a., above, are within limits for ASTM 2D fuel oil, and

6.8.4.n (iii) c. Total particulate concentration of the fuel oil is ≤ 10 mg/l when tested ~~every 31 days~~ at a Frequency in accordance with the Surveillance Frequency Control Program. 3

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program test frequencies.

6.8.4.j 5.5.~~13~~ 11 **Technical Specifications (TS) Bases Control Program** 4

This program provides a means for processing changes to the Bases of these Technical Specifications.

6.8.4.j.1 a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

6.8.4.j.2 b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

6.8.4.j.2.a 1. A change in the TS incorporated in the license or 1

6.8.4.j.2.b 2. A change to the ~~updated~~ U FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

6.8.4.j.3 c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the U FSAR. 1

5.5 Programs and Manuals

6.8.4.j 5.5.~~13~~ 11 Technical Specifications (TS) Bases Control Program (continued) 4

6.8.4.j.4

d. Proposed changes that meet the criteria of 5.5.~~13~~11.b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

5.5.~~14~~ 12 Safety Function Determination Program (SFDP) 4

6.8.4.s

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.

A loss of safety function exists when, assuming no concurrent single failure, no concurrent loss of offsite power, or no concurrent loss of onsite diesel generator(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.

5.5 Programs and Manuals

6.8.4.s 5.5.~~14~~ 12 Safety Function Determination Program (SFDP) (continued) 4

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. 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.

6.8.4.h 5.5.~~15~~ 13 Containment Leakage Rate Testing Program 4

~~[OPTION A]~~

~~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 A, as modified by approved exemptions.~~

~~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.~~ 8

~~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 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.~~

6.8.4.h 5.5.~~15~~ 13 Containment Leakage Rate Testing Program (continued) 4 8

~~[OPTION B]~~

6.8.4.h 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:~~ NEI 94-01, Revision 2-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J."

~~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.~~ 1

~~2. The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR50, 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. . . .]~~

6.8.4.h b. The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a is ~~[45~~ 42.8 psig~~]~~. The containment design pressure is ~~[50~~ 44 psig~~]~~. 2

6.8.4.h c. The maximum allowable containment leakage rate, L_a at P_a, shall be [0.50]% of containment air weight per day. 2

6.8.4.h d. Leakage rate acceptance criteria are:

6.8.4.h.a 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 $\leq 0.75\ L_a$ for Type A tests.

6.8.4.h.b 2. Air lock testing acceptance criteria are:

a) Overall personnel air lock leakage rate is $\leq$ ~~[~~0.05 L_a~~]~~ when tested at $\geq P_a$. 1

b) For each door, leakage rate is ~~$\leq$~~ < ~~[~~0.01 L_a~~]~~ when pressurized to ~~[~~$\geq$ ~~10 psig]~~ 1.0 P_a. 2

c) For each emergency air lock door, leakage rate is < 0.01 L_a when pressurized to $\geq$ 10 psig. 1

5.5 Programs and Manuals

5.5.~~15~~ 13 Containment Leakage Rate Testing Program (continued) 4

6.8.4.h e. The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

6.8.4.h 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.~~ 8

~~2. The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR50, 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.~~

5.5 Programs and Manuals

~~5.5.15 Containment Leakage Rate Testing Program (continued)~~

~~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].~~

~~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.~~ 8

5.5.~~16~~ 14 Battery Monitoring and Maintenance Program 4

~~-----REVIEWER'S NOTE-----~~
~~This program and the corresponding requirements in LCO 3.8.4, LCO 3.8.5, and LCO 3.8.6 require providing the information and verifications requested in the Notice of Availability for TSTF-500, Revision 2, "DC Electrical Rewrite - Update to TSTF-360," (76FR54510).~~ 6
~~----------~~

DOC M02

This Program provides controls for battery restoration and maintenance. The program shall be in accordance with IEEE Standard (Std) 450-~~2002~~ 2010, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications," as endorsed by Regulatory Guide 1.129, Revision ~~2 (RG), with RG exceptions and program provisions as identified below:~~ 3.

~~a. The program allows the following RG 1.129, Revision 2 exceptions:~~ 9

~~1. Battery temperature correction may be performed before or after conducting discharge tests.~~

~~2. RG 1.129, Regulatory Position 1, Subsection 2, "References," is not applicable to this program.~~

5.5 Programs and Manuals

5.5.~~16~~ 14 Battery Monitoring and Maintenance Program (continued) 4

~~3. In lieu of RG 1.129, Regulatory Position 2, Subsection 5.2, "Inspections," the following shall be used: "Where reference is made to the pilot cell, pilot cell selection shall be based on the lowest voltage cell in the battery."~~

~~4 In Regulatory Guide 1.129, Regulatory Position 3, Subsection 5.4.1, "State of Charge Indicator," the following statements in paragraph (d) may be omitted: "When it has been recorded that the charging current has stabilized at the charging voltage for three consecutive hourly measurements, the battery is near full charge. These measurements shall be made after the initially high charging current decreases sharply and the battery voltage rises to approach the charger output voltage."~~ 9

~~5. In lieu of RG 1.129, Regulatory Position 7, Subsection 7.6, "Restoration," the following may be used: "Following the test, record the float voltage of each cell of the string."~~

~~b.~~ The program shall include the following provisions:

Table 4.8-2 a. ~~1.~~ Actions to restore battery cells with float voltage < ~~[~~2.13~~]~~ V;

Table 4.8-2 b. ~~2.~~ Actions to determine whether the float voltage of the remaining battery cells is ≥ ~~[~~2.13~~]~~ V when the float voltage of a battery cell has been found to be < ~~[~~2.13~~]~~ V; 2

c. ~~3.~~ Actions to equalize and test battery cells that had been discovered with electrolyte level below the top of the plates; 9

d. ~~4.~~ Limits on average electrolyte temperature, battery connection resistance, and battery terminal voltage; and

~~5. A requirement to obtain specific gravity readings of all cells at each discharge test, consistent with manufacturer recommendations.~~

5.5 Programs and Manuals

6.8.4.m 5.5.~~17~~ 15 **Control Room Envelope (CRE) Habitability Program** 4

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Emergency ~~Air Cleanup~~ Ventilation System (CRE~~AC~~VS), CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of ~~[5 rem whole body or its equivalent to any part of the body] [~~5 rem total effective dose equivalent (TEDE)~~]~~ for the duration of the accident. The program shall include the following elements: 1 2

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

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

c. Requirements for (i) determining the unfiltered air inleakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

~~[The following are exceptions to Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0:~~

~~1. ;and]~~ 2

d. Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by one train of the CREACS, operating at the flow rate required by the VFTP, at a Frequency ~~of [18] months on a STAGGERED TEST BASIS~~ in accordance with the Surveillance Frequency Control Program. The results shall be trended and used as part of the ~~[18]~~ 36 month assessment of the CRE boundary. 3 2

5.5 Programs and Manuals

6.8.4.m 5.5.~~17~~ 15 Control Room Envelope (CRE) Habitability Program (continued) 4

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

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

~~[5.5.18 Setpoint Control Program~~

~~-----REVIEWER'S NOTE-----~~
~~Adoption of a Setpoint Control Program requires changes to other technical specifications. See TSTF-493, Revision 4, "Clarify Application of Setpoint Methodology for LSSS Functions," Option B, for guidance (Agencywide Documents Access and Management System (ADAMS) Accession Number ML101160026).~~
~~----------~~

~~This program shall establish the requirements for ensuring that setpoints for automatic protective devices are initially within and remain within the assumptions of the applicable safety analyses, provides a means for processing changes to instrumentation setpoints, and identifies setpoint methodologies to ensure instrumentation will function as required. The program shall ensure that testing of automatic protective devices related to variables having significant safety functions as delineated by 10 CFR 50.36(c)(1)(ii)(A) verifies that instrumentation will function as required.~~ 10

~~a. The program shall list the Functions in the following specifications to which it applies:~~

~~1. LCO 3.3.1, "Reactor Protective System (RPS) Instrumentation - Operating [(Analog)] [(Digital)];"~~

5.5 Programs and Manuals

~~5.5.18 Setpoint Control Program (continued)~~

~~2. LCO 3.3.2, "Reactor Protective System (RPS) Instrumentation - Shutdown [(Analog)] [(Digital)];"~~
~~3. LCO [3.3.3, "Control Element Assembly Calculators (CEACs) (Digital)];"~~
~~4. [LCO 3.3.4,"Engineered Safety Features Actuation System (ESFAS) Instrumentation (Analog);"] [LCO 3.3.5, "Engineered Safety Features Actuation System (ESFAS) Instrumentation (Digital);"]~~
~~5. [LCO 3.3.6, "Diesel Generator (DG) - Loss of Voltage Start (LOVS) (Analog);"] [LCO 3.3.7, "Diesel Generator (DG) - Loss of Voltage Start (LOVS) (Digital);"]~~
~~6. [LCO 3.3.7, "Containment Purge Isolation Signal (CPIS) (Analog);"] [LCO 3.3.8, "Containment Purge Isolation Signal (CPIS) (Digital);"]~~
~~7. [LCO 3.3.8, "Control Room Isolation Signal (CRIS) (Analog);"] [LCO 3.3.9, "Control Room Isolation Signal (CRIS) (Digital);"];~~
~~8. [LCO 3.3.9, "Chemical and Volume Control System (CVCS) Isolation Signal (Analog);"]~~
~~9. [LCO 3.3.10, "Fuel Handling Isolation Signal (FHIS) (Digital);"]~~
~~10. LCO 3.3.13, "[Logarithmic] Power Monitoring Channels [(Analog)."] [(Digital)."]~~

~~b. The program shall require the [Limiting Trip Setpoint (LTSP)], [Nominal Trip Setpoint (NTSP)], Allowable Value (AV), As-Found Tolerance (AFT), and As-Left Tolerance (ALT) (as applicable) of the Functions described in paragraph a. are calculated using the NRC approved setpoint methodology, as listed below. In addition, the program shall contain the value of the [LTSP], [NTSP], AV, AFT, and ALT (as applicable) for each Function described in paragraph a. and shall identify the setpoint methodology used to calculate these values.~~

~~------------------------------------ Reviewer's Note ------------------------------------~~
~~List the NRC safety evaluation report by letter, date, and ADAMS accession number (if available) that approved the setpoint methodologies.~~
~~---~~

~~1. [Insert reference to NRC safety evaluation that approved the setpoint methodology.]~~

~~c. The program shall establish methods to ensure that Functions described in paragraph a. will function as required by verifying the as-left and as-found settings are consistent with those established by the setpoint methodology.~~

~~5.5 Programs and Manuals~~

~~5.5.18 Setpoint Control Program (continued)~~

~~d. ---------------------------------REVIEWER'S NOTE-----------------------------------~~
~~A license amendment request to implement a Setpoint Control Program must list the instrument functions to which the program requirements of paragraph d. will be applied. Paragraph d. shall apply to all Functions in the Reactor Protection System and Engineered Safety Feature Actuation System specifications unless one or more of the following exclusions apply:~~

~~1. Manual actuation circuits, automatic actuation logic circuits or to instrument functions that derive input from contacts which have no associated sensor or adjustable device, e.g., limit switches, breaker position switches, manual actuation switches, float switches, proximity detectors, etc. are excluded. In addition, those permissives and interlocks that derive input from a sensor or adjustable device that is tested as part of another TS function are excluded.~~

~~2. Settings associated with safety relief valves are excluded. The performance of these components is already controlled (i.e., trended with as-left and as-found limits) under the ASME Code for Operation and Maintenance of Nuclear Power Plants testing program.~~

~~3. Functions and Surveillance Requirements which test only digital components are normally excluded. There is no expected change in result between SR performances for these components. Where separate as-left and as-found tolerance is established for digital component SRs, the requirements would apply.~~
~~--~~

~~The program shall identify the Functions described in paragraph a. that are automatic protective devices related to variables having significant safety functions as delineated by 10 CFR 50.36(c)(1)(ii)(A). The [LTSP] of these Functions are Limiting Safety System Settings. These Functions shall be demonstrated to be functioning as required by applying the following requirements during CHANNEL CALIBRATIONS and CHANNEL FUNCTIONAL TESTS that verify the [LTSP or NTSP].~~

~~1 The as-found value of the instrument channel trip setting shall be compared with the previous as-left value or the specified [LTSP or NTSP].~~

~~2. If the as-found value of the instrument channel trip setting differs from the previous as-left value or the specified [LTSP or NTSP] by more than the pre-defined test acceptance criteria band (i.e., the specified AFT), then the instrument channel shall be evaluated before declaring the SR met and returning the instrument channel to service. This condition shall be entered in the plant corrective action program.~~

5.5 Programs and Manuals

~~5.5.18~~ ~~Setpoint Control Program (continued)~~

~~3. If the as-found value of the instrument channel trip setting is less conservative than the specified AV, then the SR is not met and the instrument channel shall be immediately declared inoperable.~~

~~4. The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the [LTSP or NTSP] at the completion of the surveillance test; otherwise, the channel is inoperable (setpoints may be more conservative than the [LTSP or NTSP] provided that the as-found and as-left tolerances apply to the actual setpoint used to confirm channel performance).~~ 10

~~e. The program shall be specified in [insert the facility FSAR reference or the name of any document incorporated into the facility FSAR by reference].]~~

~~[~~5.5.~~19~~ Surveillance Frequency Control Program 2 10

16

6.8.4.o

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

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

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

c. The provisions of Surveillance Requirements 3.0.2 and 3.0.3 are applicable to the Frequencies established in the Surveillance Frequency Control Program. ~~]~~ 2

~~[~~5.5.~~20~~ Risk Informed Completion Time Program 2 10

17

6.8.4.r

This program provides controls to calculate a Risk Informed Completion Time (RICT) and must be implemented in accordance with NEI 06-09-A, Revision 0, "Risk-Managed Technical Specifications (RMTS) Guidelines." The program shall include the following:

a. The RICT may not exceed 30 days;

5.5 Programs and Manuals

6.8.4.r 5.5.~~20~~ 17 Risk Informed Completion Time Program (continued) 10

~~REVIEWER'S NOTE~~
~~The Risk Informed Completion Time is only applicable in MODES supported by the licensee's PRA. Licensees applying the RICT Program to MODES other than MODES 1 and 2 must demonstrate that they have the capability to calculate a RICT in those MODES or that the risk indicated by their MODE 1 and 2 PRA model is bounding with respect to the lower MODE conditions.~~ 6

b. A RICT may only be utilized in ~~MODE 1, 2~~ MODES 1 and 2 ~~[, and 3, and MODE 4 while relying on steam generators for heat removal]~~; 2

c. When a RICT is being used, any change to the plant configuration~~, as defined in NEI 06-09-A, Appendix A,~~ within the scope of the Risk Informed Completion Time Program must be considered for the effect on the RICT. 1

1. For planned changes, the revised RICT must be determined prior to implementation of the change in configuration.

2. For emergent conditions, the revised RICT must be determined within the time limits of the Required Action Completion Time (i.e., not the RICT) or 12 hours after the plant configuration change, whichever is less.

3. Revising the RICT is not required if the plant configuration change would lower plant risk and would result in a longer RICT.

d. Use of a RICT is not permitted for entry into a configuration which represents a loss of a specified safety function or inoperability of all required trains of a system required to be OPERABLE. 1

~~d.~~ e. ~~For emergent conditions, i~~If the extent of condition evaluation for inoperable structures, systems, or components (SSCs) is not complete prior to exceeding the Completion Time, the RICT shall account for the increased possibility of common cause failure (CCF) by either: 1

1. Numerically accounting for the increased possibility of CCF in the RICT calculation; or

2. Risk Management Actions (RMAs) not already credited in the RICT calculation shall be implemented that support redundant or diverse SSCs that perform the function(s) of the inoperable SSCs, and, if practicable, reduce the frequency of initiating events that challenge the function(s) performed by the inoperable SSCs.

5.5 Programs and Manuals

~~5.5.20 Risk Informed Completion Time Program (continued)~~

~~e. The risk assessment approaches and methods shall be acceptable to the NRC. The plant PRA shall be based on the as-built, as-operated, and maintained plant; and reflect the operating experience at the plant, as specified in Regulatory Guide 1.200, Revision 2. Methods to assess the risk from extending the Completion Times must be PRA methods used to support this license amendment, or other methods approved by the NRC for generic use; and any change in the PRA methods to assess risk that are outside these approval boundaries require prior NRC approval.]~~ 1

~~[5.5.21 Spent Fuel Storage Rack Neutron Absorber Monitoring Program~~

~~This Program provides controls for monitoring the condition of the neutron absorber used in the spent fuel pool storage racks to verify the Boron-10 areal density is consistent with the assumptions in the spent fuel pool criticality analysis. The program shall be in accordance with NEI 16-03-A, "Guidance for Monitoring of Fixed Neutron Absorbers in Spent Fuel Pools," Revision 0, May 2017 [, with the following exceptions:~~ 2

~~1. . . .]]~~

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

The following programs shall be established, implemented, and maintained.

Definition 1.18

5.5.1 Offsite Dose Calculation Manual (ODCM)

a. The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the radiological environmental monitoring program and

b. The ODCM shall also contain the radioactive effluent controls and radiological environmental monitoring activities, and descriptions of the information that should be included in the Annual Radiological Environmental Operating, and Radioactive Effluent Release Reports required by Specification ~~[~~5.6.1~~]~~ and Specification ~~[~~5.6.2~~]~~. 2

Licensee initiated changes to the ODCM:

6.14.1 DOC M03

a. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

1. Sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s) and

2. A determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.1302, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations,

6.14.2

b. Shall become effective after the approval of the plant manager, and

6.14.3

c. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

CTS

5.5 Programs and Manuals

6.8.4.a 5.5.2 Primary Coolant Sources Outside Containment

High Pressure Safety Injection, Containment

Safety

Reactor Coolant Sampling

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: 2

a. Preventive maintenance and periodic visual inspection requirements and

b. Integrated leak test requirements for each system at ~~least once per [18] months~~.

a Frequency in accordance with the Surveillance Frequency Control Program

The provisions of SR 3.0.2 are applicable. 3

~~[5.5.3 Post Accident Sampling~~ 2

~~-----REVIEWER'S NOTE-----~~

~~This program may be eliminated based on the implementation of Topical Report CE NPSD-1157, Rev. 1, "Technical Justification for the Elimination of the Post-Accident Sampling System from the Plant Design and Licensing Basis for CEOG Utilities," and the associated NRC Safety Evaluation dated May 16, 2000, and implementation of the following commitments:~~

~~1. [Licensee] has developed contingency plans for obtaining and analyzing highly radioactive samples of reactor coolant, containment sump, and containment atmosphere. The contingency plans will be contained in emergency plan implementing procedures and implemented with the implementation of the License amendment. Establishment of contingency plans is considered a regulatory commitment.~~

~~2. The capability for classifying fuel damage events at the Alert level threshold has been established for [PLANT] at radioactivity levels of 300 mCi/cc dose equivalent iodine. This capability may utilize the normal sampling system and/or correlations of sampling or letdown line dose rates to coolant concentrations. This capability will be described in emergency plan implementing procedures and implemented with the implementation of the License amendment. The capability for classifying fuel damage events is considered a regulatory commitment.~~

~~3. [Licensee] has established the capability to monitor radioactive iodines that have been released to offsite environs. This capability is described in our emergency plan implementing procedures. The capability to monitor radioactive iodines is considered a regulatory commitment..~~

~~-----~~

~~This program provides controls that ensure the capability to obtain and analyze reactor coolant, radioactive gases, and particulates in plant gaseous effluents~~

5.5 Programs and Manuals

~~5.5.3 Post Accident Sampling (continued)~~ 2

~~and containment atmosphere samples under accident conditions. The program shall include the following:~~ 4

~~a. Training of personnel,~~

~~b. Procedures for sampling and analysis, and~~

~~c. Provisions for maintenance of sampling and analysis equipment.]~~

6.8.4.f 5.5.4 Radioactive Effluent Controls Program 4

3

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

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

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to ten times the concentration values in Appendix B, Table 2, Column 2 to 10 CFR 20.1001-20.2402,

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

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

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

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,

5.5 Programs and Manuals

6.8.4.f 5.5.~~4~~ 3 Radioactive Effluent Controls Program (continued) 4

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents from the site to areas at or beyond the site boundary 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,

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.

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

U 1

6.8.4.r 5.5.~~5~~ 4 Component Cyclic or Transient Limit 4

3.9 2

This program provides controls to track the U FSAR, Section ~~[ ]~~ 3.9, cyclic and transient occurrences to ensure that components are maintained within the design limits.

~~[5.5.6 Pre-Stressed Concrete Containment Tendon Surveillance Program~~ 2

~~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 Section XI, Subsection IWL of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10CFR 50.55a, except where an alternative, exemption, or relief has been authorized by the NRC.~~ 4

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

5.5 Programs and Manuals (RCP) 1

5.5.~~7~~ 5 <u>Reactor Coolant Pump Flywheel Inspection Program</u> 4

6.8.4.o
DOC L02

RCP 1

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. Insert 1 5

5.5.~~8~~ 6 <u>Steam Generator (SG) Program</u> 4

6.8.4.l

An SG Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the SG Program shall include the following:

6.8.4.l.a

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging ~~[or repair]~~ of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected~~,~~ or plugged, ~~[or repaired]~~ to confirm that the performance criteria are being met. 2

6.8.4.l.b

b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.

6.8.4.l.b.1

1. Structural integrity performance criterion: All in-service SG tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down), all anticipated transients included in the design specification, and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

6.8.4.l.b.2

2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for

CTS

5

Insert 1

6.8.4.o
DOC A14

, except for an RCP flywheel consisting of ASTM A-516-69 (i.e., SA-516) Grade 65 material.

For an RCP flywheel composed of ASTM A-516-69 Grade 65 material, the inspection shall consist of either a 100% volumetric inspection of the upper flywheel over the volume from the inner bore of the flywheel to the circle of one-half the outer radius or a surface examination (magnetic particle testing and/or penetrant testing) of exposed surfaces defined by the volume of the disassembled flywheel at least once every 10 years.

5.5 Programs and Manuals

5.5.~~8~~ 6 **Steam Generator (SG) Program** (continued)

4

6.8.4.l all SGs and leakage rate for an individual SG. Leakage is not to exceed ~~[1 gpm] per~~ 0.5 gpm total through all SGs and 0.25 gpm through any one SG. ~~[, except for specific types of degradation at specific locations as described in paragraph c of the SG Program.]~~

6.8.4.l.b.3 3. The operational LEAKAGE performance criterion is specified in LCO 3.4.13, "RCS Operational LEAKAGE."

6.8.4.l.c c. Provisions for SG tube plugging ~~[or repair]~~ criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding ~~[~~40%~~]~~ of the nominal tube wall thickness shall be plugged ~~[or repaired]~~.

~~REVIEWER'S NOTE~~

~~Alternate tube plugging [or repair] criteria currently permitted by plant technical specifications are listed here. The description of these alternate tube plugging [or repair] criteria should be equivalent to the descriptions in current technical specifications and should also include any allowed accident induced leakage rates for specific types of degradation at specific locations associated with tube plugging [or repair] criteria.~~ 6

2

~~[The following alternate tube plugging [or repair] criteria may be applied as an alternative to the 40% depth based criteria:~~

~~1. . . .]~~

~~REVIEWER'S NOTE~~

~~The bracketed phrase in Paragraph d regarding exempt portions of the tube is only applicable to SGs with Alloy 600 thermally treated tubing.~~ 6

6.8.4.l.d d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet ~~[except for any portions of the tube that are exempt from inspection by alternate repair criteria]~~, and that may satisfy the applicable tube plugging ~~[or repair]~~ criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. A degradation assessment shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

5.5 Programs and Manuals

5.5.~~8~~ 6 **Steam Generator (SG) Program** (continued)

4

6.8.4.l

~~-----REVIEWER'S NOTE-----~~
~~Plants are to include the appropriate Frequency (e.g., select the appropriate Item 2.) for their SG design. The first Item 2 is applicable to SGs with Alloy 600 mill annealed tubing. The second Item 2 is applicable to SGs with Alloy 600 thermally treated tubing. The third Item 2 is applicable to SGs with Alloy 690 thermally treated tubing.~~ 6
~~----------~~

6.8.4.l.d.1

1. Inspect 100% of the tubes in each SG during the first refueling outage following SG installation.

~~[2. After the first refueling outage following SG installation, inspect 100% of the tubes in each SG at least every 24 effective full power months, which defines the inspection period.]~~

~~[2. After the first refueling outage following SG installation, inspect 100% of the tubes in each SG at least every 54 effective full power months, which defines the inspection period. If none of the SG tubes have ever experienced cracking other than in regions that are exempt from inspection by alternate repair criteria and the SG inspection was performed with enhanced probes, the inspection period may be extended to 72 effective full power months. Enhanced probes have a capability to detect flaws of any type equivalent to or better than array probe technology. The enhanced probes shall be used from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet except any portions of the tube that are exempt from inspection by alternate repair criteria. If there are regions where enhanced probes cannot be used, the tube inspection techniques shall be capable of detecting all forms of existing and potential degradation in that region.]~~ 2

6.8.4.l.d.2
DOC L01

~~[~~2. After the first refueling outage following SG installation, inspect 100% of the tubes in each SG at least every 96 effective full power months, which defines the inspection period. ~~]~~

~~-----REVIEWER'S NOTE-----~~
~~The bracketed phrases in Paragraph 3 are only applicable to SGs with Alloy 600 thermally treated tubing.~~ 6
~~----------~~

6.8.4.l.d.3

DOC L01

3. If crack indications are found in any SG tube ~~[excluding any region that is exempt from inspection by alternate repair criteria]~~, then the next inspection for each affected and potentially affected SG for the degradation mechanism that caused the crack indication shall be at the next refueling outage ~~[, but may be deferred to the following refueling~~ 2 1

5.5 Programs and Manuals

5.5.~~8~~ 6 Steam Generator (SG) Program (continued) 4

6.8.4.l.d.3 ~~outage if the 100% inspection of all SGs was performed with enhanced probes as described in paragraph d.2]~~. If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack. 2

6.8.4.l.e e. Provisions for monitoring operational primary to secondary LEAKAGE.

~~[f. Provisions for SG tube repair methods. SG tube repair methods shall provide the means to reestablish the RCS pressure boundary integrity of SG tubes without removing the tube from service. For the purposes of these Specifications, tube plugging is not a repair. All acceptable tube repair methods are listed below.~~

~~-----REVIEWER'S NOTE-----~~

~~Tube repair methods currently permitted by plant technical specifications are to be listed here. The description of these tube repair methods should be equivalent to the descriptions in current technical specifications. If there are no approved tube repair methods, this section should not be used.~~

~~----------~~

~~1. . . .]~~ 2 6

6.8.4.c 5.5.~~9~~ 7 Secondary Water Chemistry Program 4

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation ~~and low pressure turbine disc stress corrosion cracking~~. The program shall include: 1

6.8.4.c (i) a. Identification of a sampling schedule for the critical variables and control points for these variables,

6.8.4.c (ii) b. Identification of the procedures used to measure the values of the critical variables,

6.8.4.c (iii) c. Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in leakage,

6.8.4 (iv) d. Procedures for the recording and management of data,

6.8.4 (v) e. Procedures defining corrective actions for all off control point chemistry conditions, and

5.5 Programs and Manuals

5.5.~~9~~ 7 Secondary Water Chemistry Program (continued) 4

6.8.4 (vi) f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.

5.5.~~10~~ 8 Ventilation Filter Testing Program (VFTP)

6.8.4 k 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] at the system flowrate specified below [± 10%]~~. Insert 2 4 2 1

6.8.4 k.1 a. Demonstrate for each of the ESF systems that an inplace test of the high efficiency particulate air (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%~~]~~. 1

ESF Ventilation System	Flowrate
~~[ ]~~	~~[ ]~~ Insert 3

6.8.4 k.2 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%~~]~~. 1 2

Insert 4

6.8.4 k.3 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~~ 3~~]~~, 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 ~~the~~ a relative humidity ~~specified below~~ of 95%. 1

ESF Ventilation System	~~Penetration~~	~~RH~~	~~Face Velocity~~
~~[ ]~~	~~[See Reviewer's Note]~~	~~[See Reviewer's Note]~~	~~[See Reviewer's Note]~~

Insert 5 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.~~ 6

CTS

Insert 2 (1)

6.8.4.k 3, ASME N510-1989, and ASTM D3803-1989, as described herein.

6.8.4.k The tests described in Specification 5.5.8.a through 5.5.8.d shall be performed at the frequencies specified in Regulatory Guide 1.52, Revision 3.

4.6.6.1.b The tests described in Specification 5.5.8.e shall be performed at the frequencies specified in Regulatory Guide 1.52, Revision 3, except that the testing frequency of 24 months is required at a frequency in accordance with the Surveillance Frequency Control Program.

6.8.4.k.5 The tests described in Specification 5.5.8.f shall be performed at a frequency in accordance with the Surveillance Frequency Control Program.

Insert 3

6.8.4.k..1

Control Room Emergency Ventilation System (CREVS)	2000
Shield Building Ventilation System (SBVS)	6000
Emergency Core Cooling System (ECCS) Area Ventilation System	30,000

Insert 4

6.8.4.k.2

ESF Ventilation System	Flowrate
CREVS	2000
SBVS	6000
ECCS Area Ventilation System	30,000

Insert 5

6.8.4.k.3

ESF Ventilation System	Penetration
CREVS	≤ 0.175%
SBVS	≤ 2.5%
ECCS Area Ventilation System	≤ 2.5%

5.5 Programs and Manuals

5.5.~~10~~ 8 Ventilation Filter Testing Program (continued) 4

~~ASTM D 3803-1989 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.~~

~~Allowable Penetration = [(100% - Methyl Iodide Efficiency * for Charcoal Credited in Licensee's Accident Analysis) / 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 ≥ 2 for systems with or without humidity control.~~ 6

~~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.~~

6.8.4 k.4 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 below [± 10%].~~ 1 1

Insert 6

~~ESF Ventilation System — Delta P — Flowrate~~

~~[—] [—] [—]~~ 2

6.8.4 k.5 f ~~[e~~. Demonstrate that the ~~heaters for each of the ESF systems dissipate the value specified below [± 10%]~~ when tested in accordance with ~~[~~ASME N510-1989~~]~~. 1

Insert 7

~~ESF Ventilation System — Wattage]~~

~~[—] [—]~~

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.

Insert 6

6.8.4.k.4 Demonstrate for each of the ESF systems, when tested at the system flowrate specified below ± 10%, that the pressure drop across the following system components is less than the value specified below:

1. For the CREVS and ECCS Area Ventilation System: combined HEPA filters and the charcoal adsorbers; and

2. For the SBVS, combined demisters, electric heaters, HEPA filters and the charcoal adsorbers.

ESF Ventilation System	Delta P (in wg)	Flowrate
CREVS	< 7.4	2000
SBVS	≤ 8.5	6000
ECCS Area Ventilation System	< 4.35	30,000

4.6.6.1.b.2 e. Demonstrate for the SBVS that the air flow distribution across HEPA filters and charcoal adsorbers is uniform within 20% when tested in accordance with ASME N510-1989.

Insert 7

6.8.4.k.5 SBVS main heaters dissipate 30 kW ± 10% and the SBVS auxiliary heaters dissipate ≥ 1.25 kW and ≤ 1.75 kW

5.5 Programs and Manuals

5.5.~~11~~ 9 Explosive Gas and Storage Tank Radioactivity Monitoring Program 4

3/4.11.2.5
3/4.11.2.6

This program provides controls waste gas decay tanks and 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 gaseous radioactivity quantities shall be determined following the methodology in [Branch Technical Position (BTP) ETSB 11-5, "Postulated Radioactive Release due to Waste Gas System Leak or Failure"]. The liquid radwaste quantities shall be determined in accordance with [Standard Review Plan, Section 15.7.3, "Postulated Radioactive Release due to Tank Failures"].~~ 2 7

The program shall include:

3.11.2.5
3.11.2.6

a. The limits for concentrations of hydrogen and oxygen in the ~~[Waste Gas Holdup System]~~ waste gas decay tanks and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion), 2

4.11.2.5
4.11.2.6

b. A surveillance program to ensure that the quantity of radioactivity contained in ~~[~~each gas storage tank ~~and fed into the offgas treatment system]~~ is less than the amount that would result in a whole body exposure of ≥ 0.5 rem to any individual in an unrestricted area, in the event of ~~[~~an uncontrolled release of the tanks' contents~~]~~, and 2

~~c. A surveillance program to ensure that the quantity of radioactivity contained in all outdoor liquid radwaste tanks that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the [Liquid Radwaste Treatment System] is less than the amount that would result in concentrations less than the limits of 10 CFR 20, Appendix B, Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tanks' contents.~~ 7

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

5.5 Programs and Manuals

5.5.~~12~~ 10 **Diesel Fuel Oil Testing Program** 4

6.8.4.n 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:

6.8.4.n (i) 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 or a water and sediment content within limits,

6.8.4.n (ii) b. Within 31 days following addition of the new fuel oil to storage tanks, verify that the properties of the new fuel oil, other than those addressed in a., above, are within limits for ASTM 2D fuel oil, and

6.8.4.n (iii) c. Total particulate concentration of the fuel oil is ≤ 10 mg/l when tested ~~every 31 days~~ at a Frequency in accordance with the Surveillance Frequency Control Program. 3

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program test frequencies.

6.8.4.j 5.5.~~13~~ 11 **Technical Specifications (TS) Bases Control Program** 4

This program provides a means for processing changes to the Bases of these Technical Specifications.

6.8.4.j.1 a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

6.8.4.j.2 b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

6.8.4.j.2.a 1. A change in the TS incorporated in the license or 1

6.8.4.j.2.b 2. A change to the ~~updated~~ U FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

6.8.4.j.3 c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the U FSAR. 1

5.5 Programs and Manuals

6.8.4.j 5.5.~~13~~ 11 Technical Specifications (TS) Bases Control Program (continued) 4

6.8.4.j.4

d. Proposed changes that meet the criteria of 5.5.~~13~~11.b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

5.5.~~14~~ 12 Safety Function Determination Program (SFDP) 4

6.8.4.t

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.

A loss of safety function exists when, assuming no concurrent single failure, no concurrent loss of offsite power, or no concurrent loss of onsite diesel generator(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.

5.5 Programs and Manuals

6.8.4.t 5.5.~~14~~ 12 Safety Function Determination Program (SFDP) (continued) 4

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. 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.

6.8.4.h 5.5.~~15~~ 13 Containment Leakage Rate Testing Program 4

~~[OPTION A]~~

~~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 A, as modified by approved exemptions.~~

~~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.~~ 8

~~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 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.~~

5.5 Programs and Manuals

6.8.4.h 5.5.~~15~~ 13 Containment Leakage Rate Testing Program (continued) 4 8

~~[OPTION B]~~

6.8.4.h 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:~~ NEI 94-01, Revision 2-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," except that the next Type A test performed after the December 18, 2007 Type A test shall be performed no later than December 18, 2022.

~~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 CFR50, 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. . . .]~~ 1

6.8.4.h b. The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a is ~~[45~~ 43.48 psig~~]~~. The containment design pressure is ~~[50~~ 44 psig~~]~~. 2

6.8.4.h c. The maximum allowable containment leakage rate, L_a at P_a, shall be ~~[ ]~~ 0.50% of containment air weight per day. 2

6.8.4.h d. Leakage rate acceptance criteria are:

6.8.4.h.a 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 $\leq 0.75\ L_a$ for Type A tests.

6.8.4.h.b 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 ~~≤~~< ~~[~~0.01 L_a~~]~~ when pressurized to ~~[~~≥ ~~10 psig]~~ 1.0 P_a. 2

5.5 Programs and Manuals

5.5.~~15~~ 13 Containment Leakage Rate Testing Program (continued) 4

6.8.4.h e. The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

6.8.4.h 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.~~ 8

~~2. The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR50, 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.~~

5.5 Programs and Manuals

~~5.5.15 Containment Leakage Rate Testing Program (continued)~~

~~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 tests and [< 0.75 L_a for Option A Type A tests] [$\leq$ 0.75 L_a for Option B 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].~~

~~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.~~ 8

5.5.~~16~~ 14 Battery Monitoring and Maintenance Program 4

~~-----REVIEWER'S NOTE-----~~
~~This program and the corresponding requirements in LCO 3.8.4, LCO 3.8.5, and LCO 3.8.6 require providing the information and verifications requested in the Notice of Availability for TSTF-500, Revision 2, "DC Electrical Rewrite - Update to TSTF-360," (76FR54510).~~ 6
~~----------~~

DOC M02

This Program provides controls for battery restoration and maintenance. The program shall be in accordance with IEEE Standard (Std) 450-~~2002~~ 2010, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications," as endorsed by Regulatory Guide 1.129, Revision ~~2 (RG), with RG exceptions and program provisions as identified below:~~ 3.

~~a. The program allows the following RG 1.129, Revision 2 exceptions:~~ 9

~~1. Battery temperature correction may be performed before or after conducting discharge tests.~~

~~2. RG 1.129, Regulatory Position 1, Subsection 2, "References," is not applicable to this program.~~

5.5 Programs and Manuals

5.5.~~16~~ 14 Battery Monitoring and Maintenance Program (continued) 4

~~3. In lieu of RG 1.129, Regulatory Position 2, Subsection 5.2, "Inspections," the following shall be used: "Where reference is made to the pilot cell, pilot cell selection shall be based on the lowest voltage cell in the battery."~~

~~4 In Regulatory Guide 1.129, Regulatory Position 3, Subsection 5.4.1, "State of Charge Indicator," the following statements in paragraph (d) may be omitted: "When it has been recorded that the charging current has stabilized at the charging voltage for three consecutive hourly measurements, the battery is near full charge. These measurements shall be made after the initially high charging current decreases sharply and the battery voltage rises to approach the charger output voltage."~~

~~5. In lieu of RG 1.129, Regulatory Position 7, Subsection 7.6, "Restoration," the following may be used: "Following the test, record the float voltage of each cell of the string."~~

9

~~b.~~ The program shall include the following provisions:

Table 4.8-2 a. ~~1.~~ Actions to restore battery cells with float voltage < ~~[~~2.13~~]~~ V;

Table 4.8-2 b. ~~2.~~ Actions to determine whether the float voltage of the remaining battery cells is ≥ ~~[~~2.13~~]~~ V when the float voltage of a battery cell has been found to be < ~~[~~2.13~~]~~ V;

2

c. ~~3.~~ Actions to equalize and test battery cells that had been discovered with electrolyte level below the top of the plates;

d. ~~4.~~ Limits on average electrolyte temperature, battery connection resistance, and battery terminal voltage; and

~~5. A requirement to obtain specific gravity readings of all cells at each discharge test, consistent with manufacturer recommendations.~~

9

5.5 Programs and Manuals

6.8.4.m 5.5.~~17~~ 15 Control Room Envelope (CRE) Habitability Program 4

Ventilation V

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Emergency ~~Air Cleanup~~ System (CRE~~AC~~S), CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of ~~[5 rem whole body or its equivalent to any part of the body] [~~5 rem total effective dose equivalent (TEDE)~~]~~ for the duration of the accident. The program shall include the following elements: 1 2

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

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

c. Requirements for (i) determining the unfiltered air inleakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

~~[The following are exceptions to Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0:~~

~~1. ;and]~~ 2

d. Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by one train of the CREACS, operating at the flow rate required by the VFTP, at a Frequency ~~of [18] months on a STAGGERED TEST BASIS~~ in accordance with the Surveillance Frequency Control Program. The results shall be trended and used as part of the ~~[18]~~ 36 month assessment of the CRE boundary. 3 2

5.5 Programs and Manuals

6.8.4.m 5.5.~~17~~ 15 Control Room Envelope (CRE) Habitability Program (continued) 4

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

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

~~[5.5.18 Setpoint Control Program~~

~~-----REVIEWER'S NOTE-----~~
~~Adoption of a Setpoint Control Program requires changes to other technical specifications. See TSTF-493, Revision 4, "Clarify Application of Setpoint Methodology for LSSS Functions," Option B, for guidance (Agencywide Documents Access and Management System (ADAMS) Accession Number ML101160026).~~
~~----------~~

~~This program shall establish the requirements for ensuring that setpoints for automatic protective devices are initially within and remain within the assumptions of the applicable safety analyses, provides a means for processing changes to instrumentation setpoints, and identifies setpoint methodologies to ensure instrumentation will function as required. The program shall ensure that testing of automatic protective devices related to variables having significant safety functions as delineated by 10 CFR 50.36(c)(1)(ii)(A) verifies that instrumentation will function as required.~~ 10

~~a. The program shall list the Functions in the following specifications to which it applies:~~

~~1. LCO 3.3.1, "Reactor Protective System (RPS) Instrumentation - Operating [(Analog)] [(Digital)];"~~

5.5 Programs and Manuals

~~5.5.18 Setpoint Control Program (continued)~~

~~2. LCO 3.3.2, "Reactor Protective System (RPS) Instrumentation - Shutdown [(Analog)] [(Digital)];"~~
~~3. LCO [3.3.3, "Control Element Assembly Calculators (CEACs) (Digital)];"~~
~~4. [LCO 3.3.4,"Engineered Safety Features Actuation System (ESFAS) Instrumentation (Analog);"] [LCO 3.3.5, "Engineered Safety Features Actuation System (ESFAS) Instrumentation (Digital);"]~~
~~5. [LCO 3.3.6, "Diesel Generator (DG) - Loss of Voltage Start (LOVS) (Analog);"] [LCO 3.3.7, "Diesel Generator (DG) - Loss of Voltage Start (LOVS) (Digital);"]~~
~~6. [LCO 3.3.7, "Containment Purge Isolation Signal (CPIS) (Analog);"] [LCO 3.3.8, "Containment Purge Isolation Signal (CPIS) (Digital);"]~~
~~7. [LCO 3.3.8, "Control Room Isolation Signal (CRIS) (Analog);"] [LCO 3.3.9, "Control Room Isolation Signal (CRIS) (Digital);"];~~
~~8. [LCO 3.3.9, "Chemical and Volume Control System (CVCS) Isolation Signal (Analog);"]~~
~~9. [LCO 3.3.10, "Fuel Handling Isolation Signal (FHIS) (Digital);"]~~
~~10. LCO 3.3.13, "[Logarithmic] Power Monitoring Channels [(Analog)."] [(Digital)."]~~

~~b. The program shall require the [Limiting Trip Setpoint (LTSP)], [Nominal Trip Setpoint (NTSP)], Allowable Value (AV), As-Found Tolerance (AFT), and As-Left Tolerance (ALT) (as applicable) of the Functions described in paragraph a. are calculated using the NRC approved setpoint methodology, as listed below. In addition, the program shall contain the value of the [LTSP], [NTSP], AV, AFT, and ALT (as applicable) for each Function described in paragraph a. and shall identify the setpoint methodology used to calculate these values.~~

~~------------------------------------ Reviewer's Note ------------------------------------~~
~~List the NRC safety evaluation report by letter, date, and ADAMS accession number (if available) that approved the setpoint methodologies.~~
~~---~~

~~1. [Insert reference to NRC safety evaluation that approved the setpoint methodology.]~~

~~c. The program shall establish methods to ensure that Functions described in paragraph a. will function as required by verifying the as-left and as-found settings are consistent with those established by the setpoint methodology.~~

~~5.5 Programs and Manuals~~

~~5.5.18 Setpoint Control Program (continued)~~

~~d. ----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~A license amendment request to implement a Setpoint Control Program must list the instrument functions to which the program requirements of paragraph d. will be applied. Paragraph d. shall apply to all Functions in the Reactor Protection System and Engineered Safety Feature Actuation System specifications unless one or more of the following exclusions apply:~~

~~1. Manual actuation circuits, automatic actuation logic circuits or to instrument functions that derive input from contacts which have no associated sensor or adjustable device, e.g., limit switches, breaker position switches, manual actuation switches, float switches, proximity detectors, etc. are excluded. In addition, those permissives and interlocks that derive input from a sensor or adjustable device that is tested as part of another TS function are excluded.~~

~~2. Settings associated with safety relief valves are excluded. The performance of these components is already controlled (i.e., trended with as-left and as-found limits) under the ASME Code for Operation and Maintenance of Nuclear Power Plants testing program.~~

~~3. Functions and Surveillance Requirements which test only digital components are normally excluded. There is no expected change in result between SR performances for these components. Where separate as-left and as-found tolerance is established for digital component SRs, the requirements would apply.~~
~~---~~

~~The program shall identify the Functions described in paragraph a. that are automatic protective devices related to variables having significant safety functions as delineated by 10 CFR 50.36(c)(1)(ii)(A). The [LTSP] of these Functions are Limiting Safety System Settings. These Functions shall be demonstrated to be functioning as required by applying the following requirements during CHANNEL CALIBRATIONS and CHANNEL FUNCTIONAL TESTS that verify the [LTSP or NTSP].~~

~~1 The as-found value of the instrument channel trip setting shall be compared with the previous as-left value or the specified [LTSP or NTSP].~~

~~2. If the as-found value of the instrument channel trip setting differs from the previous as-left value or the specified [LTSP or NTSP] by more than the pre-defined test acceptance criteria band (i.e., the specified AFT), then the instrument channel shall be evaluated before declaring the SR met and returning the instrument channel to service. This condition shall be entered in the plant corrective action program.~~

5.5 Programs and Manuals

~~5.5.18~~ ~~Setpoint Control Program (continued)~~

~~3. If the as-found value of the instrument channel trip setting is less conservative than the specified AV, then the SR is not met and the instrument channel shall be immediately declared inoperable.~~

~~4. The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the [LTSP or NTSP] at the completion of the surveillance test; otherwise, the channel is inoperable (setpoints may be more conservative than the [LTSP or NTSP] provided that the as-found and as-left tolerances apply to the actual setpoint used to confirm channel performance).~~

~~e. The program shall be specified in [insert the facility FSAR reference or the name of any document incorporated into the facility FSAR by reference].]~~ 10

~~[~~5.5.~~19~~ 16 Surveillance Frequency Control Program 2 10

6.8.4.q

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

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

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

c. The provisions of Surveillance Requirements 3.0.2 and 3.0.3 are applicable to the Frequencies established in the Surveillance Frequency Control Program. ~~]~~ 2

~~[~~5.5.~~20~~ 17 Risk Informed Completion Time Program 2 10

6.8.4.s

This program provides controls to calculate a Risk Informed Completion Time (RICT) and must be implemented in accordance with NEI 06-09-A, Revision 0, "Risk-Managed Technical Specifications (RMTS) Guidelines." The program shall include the following:

a. The RICT may not exceed 30 days;

5.5 Programs and Manuals

6.8.4.s 5.5.~~20~~ 17 Risk Informed Completion Time Program (continued) 10

~~-------------------------------- REVIEWER'S NOTE --------------------------------~~
~~The Risk Informed Completion Time is only applicable in MODES supported by the licensee's PRA. Licensees applying the RICT Program to MODES other than MODES 1 and 2 must demonstrate that they have the capability to calculate a RICT in those MODES or that the risk indicated by their MODE 1 and 2 PRA model is bounding with respect to the lower MODE conditions.~~ 6
~~--~~

b. A RICT may only be utilized in ~~MODE 1, 2~~ MODES 1 and 2 ~~[, and 3, and MODE 4 while relying on steam generators for heat removal]~~; 2

c. When a RICT is being used, any change to the plant configuration~~, as defined in NEI 06-09-A, Appendix A,~~ within the scope of the Risk Informed Completion Time Program must be considered for the effect on the RICT. 1

1. For planned changes, the revised RICT must be determined prior to implementation of the change in configuration.

2. For emergent conditions, the revised RICT must be determined within the time limits of the Required Action Completion Time (i.e., not the RICT) or 12 hours after the plant configuration change, whichever is less.

3. Revising the RICT is not required if the plant configuration change would lower plant risk and would result in a longer RICT.

~~d~~ e. ~~For emergent conditions, i~~ If the extent of condition evaluation for inoperable structures, systems, or components (SSCs) is not complete prior to exceeding the Completion Time, the RICT shall account for the increased possibility of common cause failure (CCF) by either: 1

1. Numerically accounting for the increased possibility of CCF in the RICT calculation; or

2. Risk Management Actions (RMAs) not already credited in the RICT calculation shall be implemented that support redundant or diverse SSCs that perform the function(s) of the inoperable SSCs, and, if practicable, reduce the frequency of initiating events that challenge the function(s) performed by the inoperable SSCs.

d. Use of a RICT is not permitted for entry into a configuration which represents a loss of a specified safety function or inoperability of all required trains of a system required to be OPERABLE. 1

5.5 Programs and Manuals

~~5.5.20 Risk Informed Completion Time Program (continued)~~

~~e. The risk assessment approaches and methods shall be acceptable to the NRC. The plant PRA shall be based on the as-built, as-operated, and maintained plant; and reflect the operating experience at the plant, as specified in Regulatory Guide 1.200, Revision 2. Methods to assess the risk from extending the Completion Times must be PRA methods used to support this license amendment, or other methods approved by the NRC for generic use; and any change in the PRA methods to assess risk that are outside these approval boundaries require prior NRC approval.]~~ 1

~~[5.5.21 Spent Fuel Storage Rack Neutron Absorber Monitoring Program~~

~~This Program provides controls for monitoring the condition of the neutron absorber used in the spent fuel pool storage racks to verify the Boron-10 areal density is consistent with the assumptions in the spent fuel pool criticality analysis. The program shall be in accordance with NEI 16-03-A, "Guidance for Monitoring of Fixed Neutron Absorbers in Spent Fuel Pools," Revision 0, May 2017 [, with the following exceptions:~~ 2

~~1. . . .]]~~

JUSTIFICATION FOR DEVIATIONS
ITS 5.5, PROGRAMS AND MANUALS

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. ISTS 5.5.2.b requires integrated leak test requirements for each system at least once per [18] months. ISTS 5.5.12.c requires total particulate sampling every 31 days. ISTS 5.5.17.d requires control room envelope pressure measurement testing relative to external adjacent areas to be performed at a frequency of [18] months on a STAGGERED TEST BASIS. St. Lucie Plant (PSL) controls periodic Frequencies for Surveillances in accordance with the Surveillance Frequency Control Program per Unit 1 CTS 6.8.4.o and Unit 2 CTS 6.8.4.q. Therefore, ITS 5.5.2.b, 5.5.10.c, and 5.5.15.d will be performed at a Frequency in accordance with the Surveillance Frequency Control Program.

4. ISTS 5.6.3, Post Accident Sampling, and ISTS 5.5.6, Pre-Stressed Concrete Containment Tendon Surveillance Program are not included in the ITS. The Post Accident Sampling requirements were deleted from the Unit 1 and Unit 2 CTS in March 27, 2001 by Amendments 174, and 114, respectively (ADAMS Accession No. ML011140017). Additionally, the containment design at PSL Unit 1 and Unit 2 does not include tendons. Subsequent Specifications are renumbered as a result of this deletion.

5. ISTS 5.5.7, Reactor Coolant Pump Flywheel Inspection Program, is changed to the PSL Unit 2 current licensing basis for flywheels composed of ASTM A-516-69 Grade 65 material and retains the ISTS requirement for flywheels composed of materials other than ASTM A-516-69 Grade 65. By letter dated October 9, 2019 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19282D338), as supplemented by letters dated April 30, 2020, and June 26, 2020 (ADAMS Accession Nos. ML20121A170 and ML20178A463, respectively), FPL submitted to the NRC, a license amendment request (LAR) to revise Technical Specification (TS) 6.8.4.o, “Reactor Coolant Pump Flywheel Inspection Program,” for PSL Unit 2. As discussed in Section 3.1 of this safety evaluation (SE), the licensee proposed to modify the St. Lucie Unit 2 Reactor Coolant Pump (RCP) Flywheel Inspection Program requirements to be consistent with the conclusions and limitations specified in the NRC’s May 21, 1997, SE of Topical Report SIR-94-080, Revision 1, “Relaxation of Reactor Coolant Pump Flywheel Inspection Requirements” (ADAMS Accession No. ML20013C086). The proposed change to Unit 2 CTS 6.8.4.o. was approved in NRC SE, dated November 18, 2020 (ADAMS Accession No. ML20259A298). Unit 1 is incorporating an RCP Flywheel Inspection Program as a more restrictive programmatic requirement and, therefore, is including the same frequency requirement in Unit 1 ITS as Unit 2 ITS to allow RCP flywheel exchanges between Unit 1 and Unit 2.

6. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

7. The program details of the Explosive Gas and Storage Tank Radioactivity Monitoring Program are described in ISTS 5.5.11.a and 5.5.11.b (ITS 5.5.9.a and 5.5.9.b). Therefore, the sentence in the introductory paragraph that specifies a method to determine the explosive gas and storage tank radioactivity is not necessary. Additionally, the requirements specified in ISTS 5.5.11.c do not apply to PSL Unit 1 or Unit 2. UFSAR Chapter 15 states that a waste gas decay tank release is not a credible accident and that the dose will remain within the regulatory guidelines. This is also consistent with the CTS.

8. PSL Unit 1 and Unit 2 comply with Option B of 10 CFR 50, Appendix J. Therefore, the Option A and combined Option A and B provisions of ISTS 5.5.15 are not included in the ITS.

9. ISTS 5.5.16 is modified in ITS 5.5.14 to reference IEEE 450-2010 and Revision 3 of Regulatory Guide (RG) 1.129 instead of IEEE-450-2002 and Revision 2 of RG 1.129. RG 1.129, Revision 3 endorses the use of IEEE 450-2010 and eliminates the need for the exceptions specified in ISTS 5.5.16.a. Therefore, the exceptions of ISTS 5.5.16.a are not included in ITS 5.5.14. Section 5.4.2 of IEEE 450-2010 states, in part, that specific gravity readings are not recommended to be taken on a regular basis. PSL batteries are lead-calcium type batteries and therefore, specific gravities are not required to be obtained at each discharge test. Therefore, ISTS 5.5.16.b.5 is not included in ITS 5.5.14. Use of IEEE 450-2010 and RG 1.129, Revision 3 in the Battery Monitoring and Maintenance Program has been previously approved in Donald C. Cook Nuclear Plant Amendments 343 and 325 dated February 5, 2019 for Units 1 and 2, respectively (NRC ADAMS Accession No. ML18346A358).

10. PSL Unit 1 and Unit 2 CTS do not use a Setpoint Control Program for the control of instrumentation setpoints for automatic protective devices. Therefore, ISTS 5.5.18, Setpoint Control Program, is not included in the ITS. Subsequent Specifications are renumbered as a result of this deletion.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 5.5, PROGRAMS AND MANUALS

10 CFR 50.92 EVALUATION
FOR
MORE RESTRICTIVE CHANGE M01 – UNIT 1
LESS RESTRICTIVE CHANGE L02 – UNIT 2

Florida Power & Light Company (FPL) is converting the St. Lucie Plant (PSL) Unit 1 and Unit 2 Technical Specifications to the Improved Technical Specifications (ITS) as outlined in NUREG-1432, "Standard Technical Specifications, Combustion Engineering Plants," Revision 5. The proposed change involves making the Unit 1 Current Technical Specifications (CTS) more restrictive and the Unit 2 CTS less restrictive. Below is the description of the change and the determination of No Significant Hazards Considerations for conversion to NUREG-1432.

Unit 1 CTS does not provide administrative controls for reactor RCP flywheel inspections. ITS 5.5.5, “Reactor Coolant Pump (RCP) Flywheel Inspection Program” is added to the Unit 1 ITS equivalent to the proposed Unit 2 program requirements. Unit 2 CTS requires administrative controls for RCP flywheel inspections. ITS 5.5.5 includes the same requirements and, in addition, includes program requirements for the Unit 2 CTS that were issued prior to Amendment No. 205, which are also consistent with the requirements specified in NUREG-1432, with a distinction on applicability of the two sets of requirements. The proposed program for both units will provide for the inspection of each RCP flywheel per the recommendation of Regulatory position c.4.b of Regulatory Guide 1.14, Revision 1, August 1975 except for an RCP flywheel consisting of ASTM A 516 69 (i.e., SA-516) Grade 65 material. For an RCP flywheel composed of ASTM A 516-69 Grade 65 material, the inspection shall consist of either a 100% volumetric inspection of the upper flywheel over the volume from the inner bore of the flywheel to the circle of one-half the outer radius or a surface examination (magnetic particle testing and/or penetrant testing) of exposed surfaces defined by the volume of the disassembled flywheel at least once every 10 years. This changes the CTS by adding an additional administrative program to the Unit 1 CTS and restoring program requirements that were issued prior to Amendment No. 205 to the Unit 2 CTS, in addition to the current requirements.

The purpose of the RCP flywheel inspection program is to implement requirements to minimize the potential for failures of the flywheels of RCP motors in light-water-cooled power reactors. At PSL, RCP flywheels are interchanged between Unit 1 and Unit 2 during refurbishment and replacement. Between 2011 and 2017, the Unit 2 RCP flywheels, which consist of ASTM A-543 (three flywheels Class 1, Type B and one flywheel Grade 70) material, were exchanged with the Unit 1 RCP flywheels, which consist of ASTM A-516-69, Grade 65 material. Unit 2 CTS was modified to relax the RCP flywheel inspection requirements similar to the conclusions specified in the NRC Safety Evaluation of Topical Report SIR-94-080, Revision 1. The relaxed inspection requirements were approved in Unit 2 CTS Amendment No. 205, dated November 18, 2020 (ADAMS Accession No. ML20259A298). Previously, PSL Unit 2 RCP flywheels were inspected per NRC Regulatory Guide 1.14, “Reactor Coolant Pump Flywheel Integrity,” Revision 1 and the requirements were previously contained in the CTS pre-Amendment 205 requirements. NRC Regulatory Guide 1.14 describes a method acceptable to the NRC staff of implementing the RCP flywheel inspection program requirements and include a recommended inspection schedule. Since the PSL RCP flywheels consisting of ASTM A 543 material were previously inspected per Regulatory Guide 1.14, Revision 1, August 1975 and the requirements were previously approved for use for Unit 2, the pre-Amendment 205 requirements are proposed to be restored in support of future RCP flywheel exchanges between Unit 1 and Unit 2.

FPL has evaluated whether or not a significant hazards consideration is involved with these proposed Technical Specification changes 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 any accident previously evaluated?

Response: No.

The proposed license amendment would add RCP flywheel inspection program requirements to the Unit 1 Technical Specifications and revise the Unit 2 RCP flywheel inspection program by restoring previously NRC approved inspection requirements in addition to the current requirements. Adding an administrative program and restoring previously NRC approved inspection requirements neither affects the design of any plant structure, system or component (SSC), nor the manner in which the SSCs are operated and controlled. No changes are proposed to the facility or to any accident analysis assumptions, inputs or expected outcomes. Providing RCP flywheel inspection frequencies based on the flywheel metallurgical properties in the applicable Unit 2 administrative program and adding an equivalent administrative program to Unit 1 is acceptable because, for the flywheel metallurgical properties that satisfy the criteria specified in the NRC safety evaluation for topical report SIR-94-080, the increase inspection frequency has been determined acceptable by the NRC and approved for use at PSL Unit 2. Flywheels with other metallurgical properties will continue to be inspected using the methods and inspection schedule in accordance with the methods described in Regulatory Guide 1.14, "Reactor Coolant Pump Flywheel Integrity," Revision 1. Compliance with the SIR-94-080 NRC safety evaluation and the NRC regulatory guide cannot adversely affect the likelihood or outcome of any design basis accident.

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

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

Response: No

The proposed addition of RCP flywheel inspection program requirements to the Unit 1 Technical Specifications and revision of the Unit 2 RCP flywheel inspection program requirements does not affect RCP capability to provide adequate core cooling in the event of a power loss and thereby cannot create new inputs or assumptions associated with any accident analyses. The proposed change neither installs new plant equipment nor modifies the manner in which existing equipment is operated and controlled, and therefore cannot introduce new failure modes. Providing RCP flywheel inspection frequencies based on the flywheel metallurgical properties in the applicable Unit 2 administrative program and adding an equivalent administrative program to Unit 1 is acceptable because, for the flywheel metallurgical properties that satisfy the criteria specified in the NRC safety evaluation for topical report SIR-94-080, the increase inspection frequency has been determined acceptable by the NRC and approved for use at PSL Unit 2. Flywheels with other metallurgical properties will continue to be inspected using the methods and inspection schedule in accordance with NRC Regulatory Guide 1.14. Compliance with the SIR-94-080 NRC safety evaluation and the NRC regulatory guide cannot create new or different kinds of accidents.

Therefore, operation of the facility in accordance with this proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

The proposed addition of RCP flywheel inspection program requirements to the Unit 1 Technical Specifications and revision of the Unit 2 RCP flywheel inspection program requirements do not involve changes to any safety analyses, safety limits or limiting safety system settings. The proposed change does not adversely impact plant operating margins or the reliability of equipment credited in safety analyses. Providing RCP flywheel inspection frequencies based on the flywheel metallurgical properties in the applicable Unit 2 administrative program and adding an equivalent administrative program to Unit 1 is acceptable because, for the flywheel metallurgical properties that satisfy the criteria specified in the NRC safety evaluation for topical report SIR-94-080, the increase inspection frequency has been determined acceptable by the NRC and approved for use at PSL Unit 2. Flywheels with other metallurgical properties will continue to be inspected using the methods and inspection schedule in accordance with NRC Regulatory Guide 1.14. Compliance with the SIR-94-080 NRC safety evaluation and the NRC regulatory guide cannot result in a reduction in the margin of safety.

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

Based upon the above analysis, FPL concludes that the proposed change does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of no significant hazards consideration is justified.

10 CFR 50.92 EVALUATION
FOR
REMOVE DETAIL CHANGE LA01

Florida Power & Light Company (FPL) is converting the St. Lucie Plant (PSL) Unit 1 and Unit 2 Technical Specifications to the Improved Technical Specifications (ITS) as outlined in NUREG-1432, "Standard Technical Specifications, Combustion Engineering Plants," Revision 5. The proposed change involves making the Unit 1 Current Technical Specifications (CTS) more restrictive and the Unit 2 CTS less restrictive. Below is the description of the change and the determination of No Significant Hazards Considerations for conversion to NUREG-1432.

CTS contains periodic testing frequencies in administrative program requirements for: leakage testing of primary coolant sources outside containment; testing of the Shield Building Ventilation System filter heaters; total particulate concentration testing of the diesel generator fuel oil; measurement, and at designated locations, of the control room envelope (CRE) pressure relative to external areas adjacent to the CRE boundary during the pressurization mode of operation of the Control Room Emergency Ventilation System. ITS specifies the periodic Frequency for these tests as "In accordance with the Surveillance Frequency Control Program." This changes the CTS by moving the specified periodic Frequency for these tests to the Surveillance Frequency Control Program.

The purpose of these Surveillances is to assure that the necessary quality of systems and components is maintained. The removal of these details related to surveillance requirement frequencies from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The existing frequency is removed from Technical Specifications and placed under licensee control pursuant to the methodology described in Nuclear Energy Institute (NEI) 04-10, "Risk-Informed Technical Specifications Initiative 5b, Risk-Informed Method for Control of Surveillance Frequencies," Revision 1, dated April 2007 (ADAMS Accession No. ML071360456). The surveillance test requirements remain in the Technical Specifications. The control of changes to the surveillance frequencies is in accordance with the Surveillance Frequency Control Program, which is retained in the ITS. The Surveillance Frequency Control Program provides the necessary administrative controls to require that surveillances related to testing, calibration and inspection are conducted at a frequency to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met. The proposed change to relocate periodic frequencies in the administrative controls section of Technical Specifications has been previously approved for Wolf Creek Generating Station Unit 1 in Amendment 227, dated April 8, 2021 (NRC ADAMS Accession No. ML21053A117), River Bend Station Unit 1 in Amendment 196, dated April 29, 2019 (NRC ADAMS Accession No. ML19066A008), and Grand Gulf Nuclear Station Unit 1 in Amendment 219, dated June 11, 2019 (NRC ADAMS Accession No. ML19094A799). PSL Unit 1 and Unit 2 adopted a Surveillance Frequency Control Program in Amendment Nos. 223 (Unit 1) and 173 (Unit 2) (ADAMS Accession No. ML15127A066). This change is acceptable because the testing frequencies will be adequately controlled in accordance with the Surveillance Frequency Control Program requirements retained in ITS, which ensure changes are properly evaluated.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 5.5, PROGRAMS AND MANUALS

FPL has reviewed the proposed no significant hazards consideration (NSHC) determination published in *Federal Register 74 FR 32000* dated July 6, 2009. FPL has concluded that the proposed NSHC presented in the Federal Register notice is applicable to the proposed relocation of the periodic testing frequencies specified herein.

As required by 10 CFR 50.91(a), the FPL analysis of the issue of no significant hazards consideration is presented below:

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

Response: No.

The proposed change relocates the specified frequencies for periodic surveillance requirements to licensee control under a new Surveillance Frequency Control Program. Surveillance frequencies are not an initiator to any accident previously evaluated. As a result, the probability of any accident previously evaluated is not significantly increased. The systems and components required by the technical specifications for which the surveillance frequencies are relocated are still required to be operable, meet the acceptance criteria for the surveillance requirements, and be capable of performing any mitigation function assumed in the accident analysis. As a result, the consequences of any accident previously evaluated are not significantly increased.

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

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

Response: No

No new or different accidents result from utilizing the proposed change. The changes do not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. In addition, the changes do not impose any new or different requirements. The changes do not alter assumptions made in the safety analysis. The proposed changes are consistent with the safety analysis assumptions and current plant operating practice.

Therefore, operation of the facility in accordance with this proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

The design, operation, testing methods, and acceptance criteria for systems, structures, and components (SSCs), specified in applicable codes and standards (or alternatives approved for use by the NRC) will continue to be met as described in the plant licensing basis (including the updated final safety analysis report and bases to technical specifications), since these are not affected by changes to the surveillance frequencies.

Similarly, there is no impact to safety analysis acceptance criteria as described in the plant licensing basis. To evaluate a change in the relocated ·surveillance frequency, FPL will perform a probabilistic risk evaluation using the guidance contained in NRC approved NEI 04-10, Rev. 1 in accordance with the Technical Specification Surveillance Frequency Control Program. NEI 04-10, Rev. 1, methodology provides reasonable acceptance guidelines and methods for evaluating the risk increase of proposed changes to surveillance frequencies consistent with Regulatory Guide 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications."

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based upon the above analysis, FPL concludes that the proposed change does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of no significant hazards consideration is justified.

ATTACHMENT 6

ITS 5.6 Section, Reporting Requirements

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

A loss of safety function exists when, assuming no concurrent single failure, no concurrent loss of offsite power, or no concurrent loss of onsite diesel generator(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:

See ITS 5.5

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 ACTIONS of the LCO in which the loss of safety function exists are required to be entered. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate ACTIONS to enter are those of the support system.

6.9 REPORTING REQUIREMENTS

~~ROUTINE REPORTS~~

5.6 ~~6.9.1~~ In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the NRC. in accordance with 10 CFR 50.4 A02

~~STARTUP REPORT~~

~~6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following:~~

~~(1) receipt of an operating license,~~

~~(2) amendment of the license involving a planned increase in power level,~~ L01

~~(3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and~~

~~(4) modifications that may have significantly altered the nuclear, thermal or hydraulic performance of the plant.~~

~~6.9.1.2 The startup report shall address each of the tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.~~

~~6.9.1.3 Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial operation), supplementary reports shall be submitted at least every three months until all three events have been completed.~~ L01

~~**ANNUAL REPORTS** 1/~~

~~6.9.1.4 Annual reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.~~

~~6.9.1.5 Annual reports shall include the results of specific activity analysis in which the primary coolant exceeded the limits of Specification 3.4.8. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded; (2) Results of the last~~ L02

~~1/ A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.~~

~~isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while the limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than the limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up system flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.~~ L02

~~**MONTHLY OPERATING REPORTS**~~ A01

~~6.9.1.6 Deleted~~

ADMINISTRATIVE CONTROLS

~~ANNUAL~~ RADIOACTIVE EFFLUENT RELEASE REPORT* A01

5.6.2 ~~6.9.1.7~~ The ~~Annual~~ Radioactive Effluent Release Report covering the operation of the unit ~~during the previous 12 months of operation~~ shall be submitted ~~within 60 days after January 1 of each year~~ prior to May 1 of each year in accordance with 10 CFR 50.36a L03. 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 (1) consistent with the objectives outlined in the ODCM and PCP and (2) in conformance with 10 CFR 50.36a and Section IV.B.1 of Appendix I to 10 CFR Part 50.

5.6.2 Note * A single submittal may be made for a multiple unit station. The submittal ~~should~~ shall combine ~~those~~ sections ~~that are~~ 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. A01

ADMINISTRATIVE CONTROLS

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT**

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible. M01

5.6.1 ~~6.9.1.8~~ The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted ~~before May 1~~ by May 15 of each year. The report shall include summaries, interpretations, and analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in ~~(1)~~ the ~~ODCM~~ Offsite Dose Calculation Manual (ODCM) and ~~(2)~~ in Sections IV.B.2, IV.B.3, and IV.C ~~of~~ Appendix I ~~to~~ 10 CFR Part 50. L03

5.6.1 Note ** A single submittal may be made for a multiple unit station.

The submittal should combine sections common to all units at the station. A01

ADMINISTRATIVE CONTROLS

~~ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT (continued)~~

LA01

~~6.9.1.9 At least once every 5 years, an estimate of the actual population within 10 miles of the plant shall be prepared and submitted to the NRC.~~

~~6.9.1.10 At least once every 10 years, an estimate of the actual population within 50 miles of the plant shall be prepared and submitted to the NRC.~~

5.6.3 ~~6.9.1.11~~ **CORE OPERATING LIMITS REPORT (COLR)**

5.6.3.a

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:

5.6.3.a Document List ITS Specifications

Specification 3.1.1.1	Shutdown Margin – T_{avg} Greater Than 200°F
Specification 3.1.1.2	Shutdown Margin – T_{avg} Less Than or Equal to 200°F
Specification 3.1.1.4	Moderator Temperature Coefficient
Specification 3.1.3.1	Full Length CEA Position – Misalignment > 15 inches
Specification 3.1.3.6	Regulating CEA Insertion Limits
Specification 3.2.1	Linear Heat Rate
Specification 3.2.3	Total Integrated Radial Peaking Factor – F_r^T
Specification 3.2.5	DNB Parameters
Specification 3.9.1	Refueling Operations – Boron Concentration

5.6.3.b

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, ~~as~~ specifically those described in the following documents, approved Revisions and Supplements as specified in the COLR.

5.6.3.b Document List

1. WCAP-11596-P-A, "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores," June 1988 (Westinghouse Proprietary)

2. NF-TR-95-01, "Nuclear Physics Methodology for Reload Design of Turkey Point & St. Lucie Nuclear Plants," Florida Power & Light Company, January 1995.

3. XN-75-27(A) [also issued as XN-NF-75-27(A)], "Exxon Nuclear Neutronic(s) Design Methods for Pressurized Water Reactors"

4. ~~DELETED~~ A01

5. XN-NF-82-21(P)(A), "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations"

ADMINISTRATIVE CONTROLS

CORE OPERATING LIMITS REPORT (continued)

5.6.3.b 6. ~~DELETED~~ A01

7. XN-75-32(P)(A), "Computational Procedure for Evaluating Fuel Rod Bowing"

8. ~~DELETED~~ A01

9. XN-NF-78-44(NP)(A), "A Generic Analysis of the Control Rod Ejection Transient for Pressurized Water Reactors"

10. XN-NF-621(P)(A), " Exxon Nuclear DNB Correlation for PWR Fuel Designs"

11. ~~DELETED~~ A01

12. XN-NF-82-06(P)(A), "Qualification of Exxon Nuclear Fuel for Extended Burnup"

13. ANF-88-133(P)(A), "Qualification of Advanced Nuclear Fuels' PWR Design Methodology for Rod Burnups of 62 GWd/MTU"

14. XN-NF-85-92 (P)(A), "Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results"

15. ~~DELETED~~ A01

16. ~~DELETED~~ A01

17. EMF-92-116(P)(A), "Generic Mechanical Design Criteria for PWR Fuel Design"

18. EMF-92-153(P)(A), "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel"

19. EMF-96-029(P)(A), Volumes 1 and 2, "Reactor Analysis System for PWRs Volume 1 – Methodology Description, Volume 2 – Benchmarking Results"

CORE OPERATING LIMITS REPORT (continued)

5.6.3.b
20. EMF-1961(P)(A), "Statistical Setpoint/Transient Methodology for Combustion Engineering Type Reactors"

21. EMF-2310(P)(A), "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors," Revision 1, as supplemented by ANP-3000(P), "St. Lucie Unit 1 EPU - Information to Support License Amendment Request," Revision 0.

22. EMF-2328(P)(A), "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based," Revision 0, as supplemented by ANP-3000(P), "St. Lucie Unit 1 EPU - Information to Support License Amendment Request," Revision 0.

23. EMF-2103(P)(A), "Realistic Large Break LOCA Methodology for Pressurized Water Reactors," Revision 0, as supplemented by ANP-2903(P), "St. Lucie Nuclear Plant Unit 1 EPU Cycle Realistic Large Break LOCA summary Report with Zr-4 Fuel Cladding," Revision 1.

24. BAW-10240(P)(A) Revision 0, "Incorporation of M5 Properties in Framatome ANP Approved Methods."

25. WCAP-16045-P-A, Revision 0, "Qualification of the Two -Dimensional Transport Code PARAGON," August 2004.

ADMINISTRATIVE CONTROLS

CORE OPERATING LIMITS REPORT (continued)

assuming operation up to RATED THERMAL POWER A01

5.6.3.c 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 SHUTDOWN MARGIN, transient analysis limits, and accident analysis limits) of the safety analysis are met.

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

5.6.5 STEAM GENERATOR TUBE INSPECTION REPORT

~~6.9.1.12~~ A report shall be submitted within 180 days after the initial entry into ~~HOT SHUTDOWN~~ MODE 4 following completion of an inspection performed in accordance with Specification ~~6.8.4.l~~ 5.5.6, Steam Generator (SG) Program. The report shall include:

5.6.5 a a. The scope of inspections performed on each SG,

Add proposed ITS 5.6.5.b M02

5.6.5 c b. ~~Active~~ degradation mechanisms found,

A03

5.6.5 c.1 c. Nondestructive examination techniques utilized for each degradation mechanism,

and voltage response for each indication. For tube wear at support structures less than 20 percent through-wall, only the total number of indications needs to be reported;

5.6.5 c.2 d. Location, orientation (if linear), and measured sizes (if available) of service induced indications,

5.6.5 c.4 e. Number of tubes plugged during the inspection outage for each ~~active~~ degradation mechanism, found

Add proposed ITS 5.6.5.d M02

5.6.5.e f. ~~Total~~ The number and percentage of tubes plugged to date, and

5.6.5.c.3 g. The results of condition monitoring~~, including the results of tube pulls and in-situ testing, and~~

5.6.5.e h. The effective plugging percentage ~~for all plugging~~ in each SG.

Add proposed ITS 5.6.5.f M02

~~SPECIAL REPORTS~~

~~6.9.2 Special reports shall be submitted to the NRC within the time period specified for each report.~~ A01

including the margin to the tube integrity performance criteria and comparison with the margin predicted to exist at the inspection by the previous forward-looking tube integrity assessment; A03

~~6.10 DELETED~~

~~DELETED~~

~~TABLE 3.3-6 (Continued)~~

~~TABLE NOTATION~~

ACTION 12 - DELETED

ACTION 13 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

See ITS 3.3.8

ACTION 14 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.6.1.

See ITS 3.4.15

ACTION 15 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, either restore the inoperable Channel(s) to OPERABLE status within 72 hours, or:

See ITS 3.3.9

5.6.4

When a report is required by Condition B or F of LCO 3.3.9, "Post Accident Monitoring (PAM) Instrumentation,"

instrumentation channels of the Function

~~1)~~ ~~Initiate the~~ preplanned alternate method of monitoring the appropriate parameter(s),and

~~2)~~ Prepare and submit a ~~Special~~ Report ~~to the Commission pursuant to Specification6.9.2~~ within 14 days following the event outlining the ~~action taken~~, the cause of the inoperability and the plans and schedule for restoring the ~~system~~ to OPERABLE status.

A01

ACTION 16 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirements, comply with the ACTION requirements of Specification 3.9.9.

See ITS 3.3.6

ACTION 17 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, within 1 hour initiate and maintain operation of the control room emergency ventilation system in the recirculation mode of operation. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

See ITS 3.3.7

~~TABLE 3.3-11 (continued)~~

~~ACTION STATEMENTS~~

ACTION 1 - With the number of OPERABLE channels less than the Total No. of Channels shown in Table 3.3-11, either restore the inoperable channel(s) to OPERABLE status within 30 days or be in HOT STANDBY in 6 hours and HOT SHUTDOWN in 12 hours.

ACTION 2 - With position indication inoperable, restore the inoperable indicator to OPERABLE status or close the associated PORV block valve and remove power from its operator within 48 hours or be in HOT STANDBY in 6 hours and HOT SHUTDOWN in 12 hours.

See ITS 3.3.9

ACTION 3 - With any individual valve position indicator inoperable, obtain quench tank temperature, level and pressure information once per shift to determine valve position.

ACTION 4 - With the number of OPERABLE Channels one less than the Total Number of Channels shown in Table 3.3-11, either restore the inoperable channel to OPERABLE status within 7 days if repairs are feasible without shutting down or When a report is required by Condition B or F of LCO 3.3.9, "Post Accident Monitoring (PAM) Instrumentation," prepare and submit a ~~Special~~ Report ~~to the Commission pursuant to the specification 6.9.2~~ within ~~30~~ 14 days following the event outlining the ~~action taken~~ preplanned alternate method of monitoring, the cause of the inoperability and the plans and schedule for restoring the ~~system~~ instrumentation channels of the Function to OPERABLE status.

5.6.4 M03

ACTION 5 - With the number of OPERABLE Channels less than the Minimum Channels OPERABLE requirements of Table 3.3-11, either restore the inoperable channel(s) to OPERABLE status within 48 hours if repairs are feasible without shutting down or:

See ITS 3.3.9

5.6.4

~~1.~~ ~~Initiate an~~ alternate method of monitoring the reactor vessel inventory; and

~~2.~~ When a report is required by Condition B or F of LCO 3.3.9, "Post Accident Monitoring (PAM) Instrumentation," Prepare and submit a ~~Special~~ Report ~~to the Commission pursuant to Specification 6.9.2~~ within ~~30~~ 14 days following the event outlining the ~~action taken~~, the cause of the inoperability and the plans and schedule for restoring the ~~system~~ instrumentation channels of the Function to OPERABLE status; and

M03

3. Restore the Channel to OPERABLE status at the next scheduled refueling.

ACTION 6 - With the number of OPERABLE accident monitoring channels less than the Minimum Channels OPERABLE requirements of Table 3.3-11, either restore the inoperable channel(s) to OPERABLE status within 48 hours or be in HOT STANDBY in 6 hours and HOT SHUTDOWN in 12 hours.

See ITS 3.3.9

ACTION 7 - With the number of OPERABLE accident monitoring channels less than the Minimum Channels OPERABLE requirements of Table 3.3-11, either restore the inoperable channel(s) to OPERABLE status within 72 hours or be in HOT STANDBY in 6 hours and HOT SHUTDOWN in 12 hours.

ADMINISTRATIVE CONTROLS

6.9 REPORTING REQUIREMENTS

~~ROUTINE REPORTS~~

5.6 ~~6.9.1~~ In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the NRC in accordance with 10 CFR 50.4. A02

~~STARTUP REPORT~~

~~6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier; and (4) modifications that may have significantly altered the nuclear, thermal or hydraulic performance of the plant.~~

~~6.9.1.2 The startup report shall address each of the tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.~~ L01

~~6.9.1.3 Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial operation), supplementary reports shall be submitted at least every three months until all three events have been completed.~~

ADMINISTRATIVE CONTROLS

~~ANNUAL REPORTS~~ ~~1/~~

~~6.9.1.4 Annual reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.~~

~~6.9.1.5 Reports required on an annual basis shall include the results of specific activity analysis in which the primary coolant exceeded the limits of Specification 3.4.8. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded; (2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while the limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than the limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up system flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.~~ L02

~~MONTHLY OPERATING REPORTS~~ A01

~~6.9.1.6 Deleted~~

~~1/ A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.~~ L02

ADMINISTRATIVE CONTROLS

~~ANNUAL~~ RADIOACTIVE EFFLUENT RELEASE REPORT* A01

5.6.2 ~~6.9.1.7~~ The ~~Annual~~ Radioactive Effluent Release Report covering the operation of the unit ~~during the previous 12 months of operation~~ shall be submitted ~~within 60 days after January 1 of each year~~ prior to May 1 of each year in accordance with 10 CFR 50.36a L03. 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 (1) consistent with the objectives outlined in the ODCM and PCP and (2) in conformance with 10 CFR 50.36a and Section IV.B.1 of Appendix I to 10 CFR Part 50.

5.6.2 Note * A single submittal may be made for a multiple unit station. The submittal shall ~~should~~ combine ~~those~~ sections ~~that are~~ 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. A01

ADMINISTRATIVE CONTROLS

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT**

5.6.1 ~~6.9.1.8~~ The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted ~~before May 1~~ by May 15 of each year. The report shall include summaries, interpretations, and analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in ~~(1)~~ the ~~ODCM~~ Offsite Dose Calculation Manual (ODCM) and ~~(2)~~ Sections IV.B.2, IV.B.3, and IV.C ~~of~~ Appendix I ~~to~~ 10 CFR Part 50. in L03

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible. M01

** A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station. A01

ADMINISTRATIVE CONTROLS

~~ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT (continued)~~

~~6.9.1.9 At least once every 5 years, an estimate of the actual population within 10 miles of the plant shall be prepared and submitted to the NRC.~~ LA01

~~6.9.1.10 At least once every 10 years, an estimate of the actual population within 50 miles of the plant shall be prepared and submitted to the NRC.~~

5.6.3 ~~6.9.1.11~~ CORE OPERATING LIMITS REPORT (COLR)

5.6.3.a 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:

5.6.3.a
Document List
ITS Specifications

Specification 3.1.1.1	Shutdown Margin – T_{avg} Greater than 200°F
Specification 3.1.1.2	Shutdown Margin – T_{avg} Less Than or Equal to 200°F
Specification 3.1.1.4	Moderator Temperature Coefficient
Specification 3.1.3.1	Movable Control Assemblies – CEA Position
Specification 3.1.3.6	Regulating CEA Insertion Limits
Specification 3.2.1	Linear Heat Rate
Specification 3.2.3	Total Integrated Radial Peaking Factors – F_r^T
Specification 3.2.5	DNB Parameters
Specification 3.9.1	Refueling Operations – Boron Concentration

5.6.3.b b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, ~~as~~ specifically those described in the following documents or any approved Revisions and Supplements thereto:

5.6.3.b
Document List

1. WCAP-11596-P-A, “Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores,” June 1988 (Westinghouse Proprietary).

2. NF-TR-95-01, “Nuclear Physics Methodology for Reload Design of Turkey Point & St. Lucie Nuclear Plants,” Florida Power & Light Company, January 1995.

3. ~~DELETED~~ A01

4. ~~DELETED~~ A01

5. CENPD-275-P, Revision 1-P-A, “C-E Methodology for Core Designs Containing Gadolinia-Urania Burnable Absorbers,” May 1988, & Revision 1-P Supplement 1-P-A, April 1999.

6. ~~DELETED~~ A01

ADMINISTRATIVE CONTROLS (Continued)

CORE OPERATING LIMITS REPORT (COLR) (Continued)

5.6.3.b b. (Continued)

7. ~~DELETED~~ A01

8. CEN-123(F)-P, "Statistical Combination of Uncertainties Methodology Part 1: CE Calculated Local Power Density and Thermal Margin/Low Pressure LSSS for St. Lucie Unit 1," December 1979.

9. ~~DELETED~~ A01

10. CEN-123(F)-P, "Statistical Combination of Uncertainties Methodology Part 3: CE Calculated Departure from Nucleate Boiling and Linear Heat Rate Limiting Conditions for Operation for St. Lucie Unit 1," February 1980.

11. CEN-191(B)-P, "CETOP-D Code Structure and Modeling Methods for Calvert Cliffs Units 1 and 2," December 1981.

12. Letter, J.W. Miller (NRC) to J.R. Williams, Jr. (FPL), Docket No. 50-389, Regarding Unit 2 Cycle 2 License Approval (Amendment No. 8 to NPF-16 and SER), November 9, 1984 (Approval of CEN-123(F)-P (three parts) and CEN-191(B)-P).

13. ~~DELETED~~ A01

14. Letter, J.A. Norris (NRC) to J.H. Goldberg (FPL), Docket No. 50-389, "St. Lucie Unit 2 – Change to Technical Specification Bases Sections '2.1.1 Reactor Core' and '3/4.2.5 DNB Parameters' (TAC No. M87722)," March 14, 1994 (Approval of CEN-371(F)-P).

15. ~~DELETED~~ A01

16. ~~DELETED~~ A01

17. ~~DELETED~~ A01

18. ~~DELETED~~ A01

ADMINISTRATIVE CONTROLS (Continued)

CORE OPERATING LIMITS REPORT (COLR) (Continued)

5.6.3.b b. (Continued)

19. CENPD-225-P-A, "Fuel and Poison Rod Bowing," June 1983.
20. CENPD-139-P-A, "C-E Fuel Evaluation Model Topical Report," July 1974.
21. CEN-161(B)-P-A, "Improvements to Fuel Evaluation Model," August 1989.
22. CEN-161(B)-P, Supplement 1-PA, "Improvements to Fuel Evaluation Model," January 1992.
23. CENPD-132, Supplement 3-P-A, "Calculative Methods for the C-E Large Break LOCA Evaluation Model for the Analysis of C-E and W Designed NSSS," June 1985.
24. CENPD-133, Supplement 5-A, "CEFLASH-4A, A FORTRAN77 Digital Computer Program for Reactor Blowdown Analysis," June 1985.
25. CENPD-134, Supplement 2-A, "COMPERC-II, a Program for Emergency Refill-Reflood of the Core," June 1985.
26. CENPD-135-P, Supplement 5, "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program," April 1977.
27. Letter, R.L. Baer (NRC) to A.E. Scherer (CE), "Evaluation of Topical Report CENPD-135, Supplement #5," September 6, 1978.
28. CENPD-137, Supplement 1-P, "Calculative Methods for the C-E Small Break LOCA Evaluation Model," January 1977.
29. CENPD-133, Supplement 3-P, "CEFLASH-4AS, A Computer Program for the Reactor Blowdown Analysis of the Small Break Loss of Coolant Accident," January 1977.
30. Letter, K. Kneil (NRC) to A.E. Scherer (CE), "Evaluation of Topical Reports CENPD-133, Supplement 3-P and CENPD-137, Supplement 1-P," September 27, 1977.
31. CENPD-138, Supplement 2-P, "PARCH, A FORTRAN-IV Digital Program to Evaluate Pool Boiling, Axial Rod and Coolant Heatup," January 1977.
32. Letter, C. Aniel (NRC) to A.E. Scherer (CE), "Evaluation of Topical Report CENPD-138, Supplement 2-P," April 10, 1978.
33. Letter, W.H. Bohlke (FPL) to Document Control Desk (NRC), "St. Lucie Unit 2, Docket No. 50-389, Proposed License Amendment, MTC Change from –27 pcm to –30 pcm," L-91-325, December 17, 1991.

ADMINISTRATIVE CONTROLS (continued)

CORE OPERATING LIMITS REPORT (COLR) (continued)

5.6.3.b b. (continued)

34. Letter, J.A. Norris (NRC) to J.H. Goldberg (FPL), "St. Lucie Unit 2 – Issuance of Amendment Re: Moderator Temperature Coefficient (TAC No. M82517)," July 15, 1992.

35. Letter, J.W. Williams, Jr. (FPL) to D.G. Eisenhut (NRC), "St. Lucie Unit No. 2, Docket No. 50-389, Proposed License Amendment, Cycle 2 Reload," L-84-148, June 4, 1984.

36. Letter, J.R. Miller (NRC) to J.W. Williams, Jr. (FPL), Docket No. 50-389, Regarding Unit 2 Cycle 2 License Approval (Amendment No. 8 to NPF-16 and SER), November 9, 1984 (Approval to Methodology contained in L-84-148).

37. ~~DELETED~~

38. ~~DELETED~~

39. ~~DELETED~~ A01

40. ~~DELETED~~

41. ~~DELETED~~

42. CEN-348(B)-P-A, Supplement 1-P-A, "Extended Statistical Combination of Uncertainties," January 1997.

43. CEN-372-P-A, "Fuel Rod Maximum Allowable Gas Pressure," May 1990.

44. ~~DELETED~~ A01

45. ~~DELETED~~

CORE OPERATING LIMITS REPORT (COLR) (continued)

5.6.3.b b. (continued)

46. ~~DELETED~~ A01

47. ~~DELETED~~

48. CEN-396(L)-P, "Verification of the Acceptability of a 1-Pin Burnup Limit of 60 MWD/KG for St. Lucie Unit 2," November 1989 (NRC SER dated October 18, 1991, Letter J.A. Norris (NRC) to J.H. Goldberg (FPL), TAC No. 75947).

49. CENPD-269-P, Rev. 1-P, "Extended Burnup Operation of Combustion Engineering PWR Fuel," July 1984.

50. CEN-289(A)-P, "Revised Rod Bow Penalties for Arkansas Nuclear One Unit 2," December 1984 (NRC SER dated December 21, 1999, Letter K. N. Jabbour (NRC) to T.F. Plunkett (FPL), TAC No. MA4523).

51. CENPD-137, Supplement 2-P-A, "Calculative Methods for the ABB CE Small Break LOCA Evaluation Model," April 1998.

52. CENPD-140-A, "Description of the CONTRANS Digital Computer Code for Containment Pressure and Temperature Transient Analysis," June 1976.

53. ~~DELETED~~ A01

54. ~~DELETED~~

55. CENPD-387-P-A, Revision 000, "ABB Critical Heat Flux Correlations for PWR Fuel," May 2000.

56. CENPD-132, Supplement 4-P-A, "Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model," March 2001.

57. CENPD-137, Supplement 2-P-A, "Calculative Methods for the ABB CE Small Break LOCA Evaluation Model," April 1998.

58. CENPD-404-P-A, Rev. 0, "Implementation of ZIRLO^{TM} Cladding Material in CE Nuclear Power Fuel Assembly Designs," November 2001.

59. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985.

60. WCAP-10216-P-A, Revision 1A, "Relaxation of Constant Axial Offset Control; FQ Surveillance Technical Specification," February 1994.

CORE OPERATING LIMITS REPORT (COLR) (continued)

5.6.3.b b. (continued)

61. WCAP-11397-P-A, (Proprietary), 'Revised Thermal Design Procedure," April 1989.

62. WCAP-14565-P-A, (Proprietary), "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," October 1999.

63. WCAP-14565-P-A, Addendum 1, "Qualification of ABB Critical Heat Flux Correlations with VIPRE-01 Code," May 2003.

64. Letter, W. Jefferson Jr. (FPL) to Document Control Desk (USNRC), "St. Lucie Unit 2 Docket No. 50-389: Proposed License Amendment WCAP-9272 Reload Methodology and Implementing 30% Steam Generator Tube Plugging Limit," L-2003-276, December 2003 (NRC SER dated January 31, 2005, Letter B.T. Moroney (NRC) to J.A. Stall (FPL), TAC No. MC1566).

65. WCAP-14882-P-A, Rev. 0, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses", April 1999.

66. WCAP-7908-A, Rev. 0, "FACTRAN-A FORTRAN IV Code for Thermal Transients in a UO2 Fuel Rod", December 1989.

67. WCAP-7979-P-A, Rev. 0, "TWINKLE - A Multi-Dimensional Neutron Kinetics Computer Code", January 1975.

68. WCAP-7588, Rev. 1-A, "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Special Kinetics Methods", January 1975.

69. WCAP-16045-P-A, Revision 0, "Qualification of the Two-Dimensional Transport Code PARAGON," August 2004.

70. EMF-96-029(P)(A), Volumes 1 and 2, "Reactor Analysis System for PWRs, Volume 1 Methodology Description, Volume 2 Benchmarking Results," Siemens Power Corporation, January 1997.

71. XN-NF-78-44 (NP)(A), "A Generic Analysis of the Control Rod Ejection Transient for Pressurized Water Reactors," Exxon Nuclear Company, Inc.," October 1983.

72. XN-75-27(A) and Supplements 1 through 5, "Exxon Nuclear Neutronics Design Methods for Pressurized Water Reactors," Exxon Nuclear Company, Report and Supplement 1 dated April 1977, Supplement 2 dated December 1980, Supplement 3 dated September 1981 (P), Supplement 4 dated December 1986 (P), and Supplement 5 dated February 1987 (P).

73. XN-NF-82-06 (P)(A), Rev. 1 and Supplements 2, 4, and 5, "Qualification of Exxon Nuclear Fuel for Extended Burnup," Exxon Nuclear Company, Inc., October 1986.

ADMINISTRATIVE CONTROLS (continued)

CORE OPERATING LIMITS REPORT (COLR) (continued)

5.6.3.b b. (continued)

74. XN-NF-85-92(P)(A), "Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results," Exxon Nuclear Company, Inc., November 1986.

75. ANF-88-133(P)(A) and Supplement 1, "Qualification of Advanced Nuclear Fuels PWR Design Methodology for Rod Burnups of 62 GWd/MTU," Advanced Nuclear Fuels Corporation, December 1991.

76. EMF-92-116(P)(A), Rev. 0, "Generic Mechanical Design Criteria for PWR Fuel Design," Siemens Power Corporation, February, 1999.

77. BAW-10240(P)(A), Rev.0, "Incorporation of M5™ Properties in Framatome ANP Approved Methods," Framatome ANP, Inc., May 2004.

78. XN-NF-82-21(P)(A), Revision 1, "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations," Exxon Nuclear Company, September 1983.

79. EMF-92-153(P)(A), Revision 1, "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," January 2005.

80. EMF-1961(P)(A), Revision 0, "Statistical/Transient Methodology for Combustion Engineering Type Reactors," Siemens Power Corporation, July 2000.

81. EMF-2310(P)(A), Revision 1, "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors," Framatome ANP, Inc., May 2004.

82. XN-75-32(P)(A), Supplements 1, 2, 3, and 4, "Computational Procedure for Evaluating Fuel Rod Bowing," October 1983.

83. BAW-10231P-A Revision 1, "COPERNIC Fuel Rod Design Computer Code," January 2004.

84. EMF-2103(P)(A) Revision 0, "Realistic Large Break LOCA Methodology for Pressurized Water Reactors," April 2003.

85. EMF-2328 (P)(A) Revision 0, "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based," March 2001.

assuming operation up to RATED THERMAL POWER A01

5.6.3.c 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 SHUTDOWN MARGIN, transient analysis limits, and accident analysis limits) of the safety analysis are met.

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

SPECIAL REPORTS

STEAM GENERATOR TUBE INSPECTION REPORT

MODE 4

5.6.5 ~~6.9.1.12~~ A report shall be submitted within 180 days after the initial entry into ~~HOT SHUTDOWN~~ following completion of an inspection of the replacement SGs performed in accordance with Specification ~~6.8.4.l.1~~. The report shall include:

5.5.6

5.6.5 a a. The scope of inspections performed on each SG,

Add proposed ITS 5.6.5.b M02

5.6.5 c b. ~~Active~~ degradation mechanisms found,

5.6.5 c.1 c. Nondestructive examination techniques utilized for each degradation mechanism,

A03

5.6.5 c.2 d. Location, orientation (if linear), and measured sizes (if available) of service induced indications, and voltage response for each indication. For tube wear at support structures less than 20 percent through-wall, only the total number of indications needs to be reported;

5.6.5 c.4 e. Number of tubes plugged during the inspection outage for each active degradation mechanism, found

Add proposed ITS 5.6.5.d M02

The

5.6.5.e f. ~~Total~~ number and percentage of tubes plugged to date,

and

5.6.5.c.3 g. The results of condition monitoring~~, including the results of tube pulls and in-situ testing,~~

5.6.5.e h. The effective plugging percentage ~~for all plugging~~ in each SG.

~~6.9.2 Special reports shall be submitted to the NRC within the time period specified for each report.~~

A01

Add proposed ITS 5.6.5.f M02

~~6.10 DELETED~~

including the margin to the tube integrity performance criteria and comparison with the margin predicted to exist at the inspection by the previous forward-looking tube integrity assessment;

A03

~~TABLE 3.3-6 (Continued)~~

~~ACTION STATEMENTS~~

ACTION 22 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

See ITS 3.3.8

ACTION 23 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.6.1.

See ITS 3.4.15

ACTION 24 - DELETED

ACTION 25 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.9.

See ITS 3.3.6

ACTION 26 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirements, within 1 hour initiate and maintain operation of the control room emergency ventilation system in the recirculation mode of operation. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

See ITS 3.3.7

ACTION 27 - With the number of OPERABLE Channels less than required by the Minimum Channels OPERABLE requirement, either restore the inoperable Channel(s) to OPERABLE status within 72 hours, or:

See ITS 3.3.9

5.6.4

~~1) Initiate the~~ preplanned alternate method of monitoring the appropriate parameter(s), and

When a report is required by Condition B or F of LCO 3.3.9, "Post Accident Monitoring (PAM) Instrumentation,"

~~2)~~ Prepare and submit a ~~Special~~ Report ~~to the Commission pursuant to Specification 6.9.2~~ within 14 days following the event outlining the ~~action taken~~, the cause of the inoperability and the plans and schedule for restoring the ~~system~~ to OPERABLE status.

instrumentation channels of the Function

A01

INSTRUMENTATION

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.6 The accident monitoring instrumentation channels shown in Table 3.3-10 shall be OPERABLE.

See ITS 3.3.9

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a.* With the number of OPERABLE accident monitoring channels less than the Required Number of Channels shown in Table 3.3-10, either restore the inoperable channel to OPERABLE status within 7 days, or be in HOT STANDBY in 6 hours and HOT SHUTDOWN in 12 hours.

b.* With the number of OPERABLE accident monitoring channels less than the Minimum Channels OPERABLE requirements of Table 3.3-10, either restore the inoperable channels to OPERABLE status within 48 hours or be in HOT STANDBY in 6 hours and HOT SHUTDOWN in 12 hours.

When a report is required by Condition B or F of LCO 3.3.9, "Post Accident Monitoring (PAM) Instrumentation,"

5.6.4 preplanned alternate method of monitoring

c.** With the number of OPERABLE Channels one less than the Total Number of Channels shown in Table 3.3-10, either restore the inoperable channel to OPERABLE status within 7 days if repairs are feasible without shutting down or prepare and submit a ~~Special~~ Report ~~to the Commission pursuant to Specification 6.9.2~~ within ~~30~~ 14 days following the event outlining the ~~action taken~~, the cause of the inoperability and the plans and schedule for restoring the ~~system~~ instrumentation channels of the Function to OPERABLE status. M03

d.** With the number of OPERABLE Channels less than the Minimum Channels OPERABLE requirements of Table 3.3-10, either restore the inoperable channel(s) to OPERABLE status within 48 hours if repairs are feasible without shutting down or:

See ITS 3.3.9

5.6.4

When a report is required by Condition B or F of LCO 3.3.9, "Post Accident Monitoring (PAM) Instrumentation,"

~~1.~~ ~~Initiate an~~ alternate method of monitoring the reactor vessel inventory; and

~~2.~~ Prepare and submit a ~~Special~~ Report ~~to the Commission pursuant to Specification 6.9.2~~ within ~~30~~ 14 days following the event outlining the ~~action taken~~, the cause of the inoperability and the plans and schedule for restoring the ~~system~~ instrumentation channels of the Function to OPERABLE status, ~~and~~

M03

3. Restore the Channel to OPERABLE status at the next scheduled refueling.

* Action statements do not apply to Reactor Vessel Level Monitoring System, Containment Sump Water Level (narrow range) and Containment Sump Water Level (wide range) instruments.

** Action statements apply only to Reactor Vessel Level Monitoring System, Containment Sump Water Level (narrow range) and Containment Sump Water Level (wide range) instruments.

See ITS 3.3.9

DISCUSSION OF CHANGES
ITS 5.6, REPORTING REQUIREMENTS

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2, Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications-Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 CTS 6.9.1 specifies in addition to the applicable reporting requirements of 10 CFR, the reports that must be submitted to the NRC. ITS 5.6 requires that reports be submitted in accordance with 10 CFR 50.4. This changes the CTS by adding the 10 CFR 50.4 requirement.

10 CFR 50.4 provides distribution requirements for written communications to the NRC. This change is acceptable because the 10 CFR 50.4 distribution requirements already apply to each report to be submitted to the NRC. This change is designated as administrative because it does not result in technical changes to the CTS.

A03 CTS 6.9.1.12 provides requirements for the Steam Generator (SG) Tube Inspection Report and lists items to include in the report, including the requirement to provide location, orientation, and measured sizes of each degradation mechanism found, and the results of condition monitoring including the results of tube pulls and in situ testing. ITS 5.6.5 also provides requirements for the SG Tube Inspection Report and a list of items to include in the report the requirement to provide location, orientation, and measured sizes of each degradation mechanism found, and the results of condition monitoring. ITS 5.6.5.c.2 and c.3 include additional detail on what to include in the report. ITS 5.6.5.c.2 clarifies that for any tube degradation mechanisms found, the report must include voltage response for each indication and, for tube wear at support structures less than 20 percent through-wall, only the total number of indications needs to be reported. ITS 5.6.5.c.3 clarifies that the description of the condition monitoring assessment and results must include the margin to the tube integrity performance criteria and comparison with the margin predicted to exist at the inspection by the previous forward-looking tube integrity assessment. The purpose of the report is to advise the NRC of results of SG tube inspections performed in accordance with the plant's steam generator program, which monitors and manages tube degradation and degradation precursors. These changes are consistent with the ISTS and approved for use by the NRC as specified in the safety evaluation associated with TSTF-577-A, Revision 1, dated April 14, 2021 (NRC ADAMS Accession No. ML21098A188). The NRC staff reviewed the proposed changes to the ISTS SG Tube Inspection Report and determined that they are acceptable because they will provide additional detailed information to allow the NRC staff to better understand the overall condition of the SGs. This change solely provides clarification detail on reporting requirements currently listed in the CTS. These clarifications to what must be provided in the SG Tube Inspection Report for the CTS items specified herein do not result in a technical change to the CTS and, therefore, the change is designated as administrative.

MORE RESTRICTIVE CHANGES

M01 The second paragraph of ITS 5.6.1 includes details required to be included in the Annual Radiological Environmental Operating Report. CTS 6.9.1.8 does not contain this level of detail. This changes the CTS by requiring additional detail to be included in the Annual Radiological Environmental Operating Report.

The purpose of the second paragraph of ITS 5.6.1 is to specify details to be included in the Annual Radiological Environmental Operating Report. This change is acceptable because the content requirements are consistent with the objectives outlined in the Offsite Dose Calculation Manual. This change is designated more restrictive because it adds new reporting requirements to the Technical Specifications.

M02 CTS 6.9.1.12 provides requirements for the SG Tube Inspection Report and lists items to include in the report. ITS 5.6.5 also provides requirements for the SG Tube Inspection Report and a list of items to include in the report. ITS 5.6.5 additionally requires the report to include: 1) the nondestructive examination techniques utilized for tubes with increased degradation susceptibility (ITS 5.6.5.b); 2) an analysis summary of the tube integrity conditions predicted to exist at the next scheduled inspection (the forward-looking tube integrity assessment) relative to the applicable performance criteria, including the analysis methodology, inputs, and results (ITS 5.6.5.d); and 3) the results of any SG secondary side inspections (ITS 5.6.5.f). This changes the CTS by requiring additional items to be included in the SG Tube Inspection Report that is submitted to the NRC.

The purpose of the report is to advise the NRC of results of SG tube inspections performed in accordance with the plant's steam generator program, which monitors and manages tube degradation and degradation precursors. These changes are consistent with the ISTS and approved for use by the NRC as specified in the safety evaluation associated with TSTF-577-A, Revision 1, dated April 14, 2021 (NRC ADAMS Accession No. ML21098A188). The NRC staff reviewed the proposed changes to the ISTS SG Tube Inspection Report and determined that they are acceptable because they will provide additional detailed information to allow the NRC staff to better understand the overall condition of the SGs. The proposed change is designated more restrictive because additional items are required in the ITS to be included in the SG Tube Inspection Report than required in the CTS.

M03 Unit 1 CTS Table 3.3-11 Actions 4 and 5, and Unit 2 CTS 3.3.3.6 Actions c and d contain a requirement that becomes applicable if the applicable channels are not restored to operable status within the allowed time. These actions require submittal of a Special Report to the Commission within 30 days outlining the action taken, the cause of the inoperability and the plans and schedule to restore the associated parameter to OPERABLE status. ITS Specification 5.6.4 requires the same report to be submitted within 14 days. This changes the CTS by requiring a more timely NRC report to be submitted when channel(s) are not restored to OPERABLE status within the specified Completion Time.

The purpose of the CTS (and ITS) Actions is to advise the NRC regarding the failure to restore the required post accident monitoring channels to OPERABLE status in the allowed time. The CTS requirement allows the NRC report to be submitted within 30 days. The proposed change is acceptable because it provides a more timely report to the NRC. The proposed change does not impose an undue burden on plant personnel nor does the proposed change adversely affect the safe operation of the plant. The proposed change is designated more restrictive because a more timely report is required in the ITS than in the CTS.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 6.9.1.9 and CTS 6.9.1.10 require reporting for estimates of actual population within 10 miles of the plant every 5 years, and within 50 miles of the plant every 10 years, respectively. ITS 5.6 does not contain this requirement. This changes the CTS by removing the population reporting requirements from the CTS and relocating them to the UFSAR, which contains these population reporting requirements.

Unit 1 UFSAR, Section 2.1.3.9 and Unit 2 UFSAR, Section 2.1.3.3 provide requirements for periodic update of population data. The UFSAR states, in part, that commencing in April 1993 and at least every 5 years thereafter, FPL will prepare and submit to the NRC an estimate of the actual population within 10 miles of the plant. In addition, commencing in April 1993 and at least every 10 years thereafter, FPL will prepare and submit to the NRC an estimate of the actual population within 50 miles of the plant.

The removal of these details for making reports from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. Also, this change is acceptable because these types of procedural details are adequately controlled in the UFSAR. The UFSAR is controlled under 10 CFR 50.59 which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because reporting requirements are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L01 *(Category 9 – Deletion of Reporting Requirements)* CTS 6.9.1.1, CTS 6.9.1.2, and CTS 6.9.1.3 contain requirements for submitting a report of plant startup and power escalation testing following receipt of an operating license; amendments to the license involving planned increase in power level; installation of fuel that has a different design or has been manufactured by a different fuel supplier; and modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the unit. The ITS does not contain such reporting requirements. This changes the CTS by deleting the requirements of CTS 6.9.1.1, CTS 6.9.1.2, and CTS 6.9.1.3.

The purpose of CTS 6.9.1.1, CTS 6.9.1.2 and CTS 6.9.1.3 is to provide a summary of plant startup and power escalation testing following the four specified conditions as verification that the unit operated as expected. This change is acceptable because the regulations provide adequate reporting requirements. If there were any unit conditions outside the expected parameters during unit startup, they would be reported to the NRC if they met the reporting requirements in the regulations. Otherwise, the reports would document that the unit operated as expected and already approved by the NRC, as required by regulations. This change is designated as less restrictive because reports that would be submitted under the CTS will not be required under the ITS.

L02 *(Category 9 - Deletion of Reporting Requirements)* CTS 6.9.1.4 requires the annual reports of CTS 6.9.1.5 regarding any instances when the specific activity limit of Specification 3.4.8 is exceeded, to be submitted prior to March 31 of each year. ITS 5.6 does not contain any requirements for such a report. This changes the CTS by not including the requirements for the annual reporting of instances when the Technical Specification specific activity limit for the primary coolant is exceeded.

The purpose of CTS 6.9.1.4 and CTS 6.9.1.5 is to specify the requirements for submitting information regarding any instances when the Technical Specification specific activity of Specification 3.4.8 is exceeded in an annual report. This change is acceptable because the regulations provide adequate details of reporting requirements, and the reporting of exceeding the 1-131 specific activity limit does not affect continued plant operation. Operations or conditions prohibited by the plant's Technical Specifications are required to be reported in accordance with 10 CFR 50.73. Subsequent reports would be provided if necessary, without requiring a specific annual report. This change is designated as less restrictive because the reports that would be submitted under the CTS will not be required under the ITS.

L03 *(Category 1 – Relaxation of LCO Requirements)* CTS 6.9.1.7 requires the Radioactive Effluent Release Report to be submitted within 60 days after January 1 of each year. ITS 5.6.1 requires the Radioactive Effluent Release Report to be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. CTS 6.9.1.8 requires the Annual Radiological Environmental Operating Report to be submitted before May 1 of each year. ITS 5.6.1 requires the Annual Radiological Environmental Operating Report to be submitted by May 15 of each year. This changes the CTS by allowing an additional time to submit these reports each year.

The purpose of the due date for submitting the Radioactive Effluent Release Report and Annual Radiological Environmental Operating Report is to ensure that the reports are provided in a reasonable period of time to the NRC for review. This change is acceptable because the reports are still required to be submitted in a reasonable time frame. Given that the reports are still required to be provided to the NRC on or before May 1 or May 15, respectively, and cover the previous calendar year, report completion and submittal is clearly not necessary to assure operation in a safe manner for the interval between March 1 and April 30, and May 1 and May 15, respectively. Additionally, there is no requirement for the NRC to approve the reports. This change is designated as less restrictive because it allows more time to prepare and submit the reports to the NRC.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

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

6.9.1.8 5.6.1 Annual Radiological Environmental Operating Report

6.9.1.8 Footnote **

---NOTE---
~~[~~A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station.~~]~~ 2

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

DOC M01

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements ~~[in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979]~~. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible. 2

6.9.1.7 5.6.2 Radiological Effluent Release Report

6.9.1.7 Footnote *

---NOTE---
~~[~~A single submittal may be made for a multiple unit station. The submittal shall 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.~~]~~ 2

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

5.6 Reporting Requirements

6.9.1.11 5.6.3 CORE OPERATING LIMITS REPORT

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: , including approved revisions and supplements, as specified thereto 1

INSERT 1 ~~[The individual specifications that address core operating limits must be referenced here.]~~ 2

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:

~~-----------------------------REVIEWER'S NOTE-------------------------------------~~
~~Licensees that have received prior NRC approval to relocate Topical Report revision numbers and dates to licensee control need only list the number and title of the Topical Report, and the COLR will contain the complete identification for each of the Technical Specification referenced Topical Reports used to prepare the COLR (i.e., report number, title, revision, date, and any supplements). See NRC ADAMS Accession No: ML110660285 for details.~~ 3
~~---~~

INSERT 2 ~~[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.]~~ 2

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

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

~~5.6.4~~ ~~Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT~~

~~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:~~ 4

~~[The individual specifications that address RCS pressure and temperature limits must be referenced here.]~~

2

INSERT 1

LCO 3.1.1	SHUTDOWN MARGIN (SDM)
LCO 3.1.3	Moderator Temperature Coefficient (MTC)
LCO 3.1.4	Control Element Assembly (CEA) Alignment
LCO 3.1.6	Regulating Control Element Assembly (CEA) Insertion Limits
LCO 3.2.1	Linear Heat Rate (LHR)
LCO 3.2.2	Total Integrated Radial Peaking Factor (F_r^T)
LCO 3.2.4	AXIAL SHAPE INDEX (ASI)
LCO 3.4.1	RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits
LCO 3.9.1	Boron Concentration

2

INSERT 2

1. WCAP-11596-P-A, "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores," June 1988 (Westinghouse Proprietary)

2. NF-TR-95-01, "Nuclear Physics Methodology for Reload Design of Turkey Point & St. Lucie Nuclear Plants," Florida Power & Light Company, January 1995.

3. XN-75-27(A) [also issued as XN NF 75 27(A)], "Exxon Nuclear Neutronic(s) Design Methods for Pressurized Water Reactors"

4. XN-NF-82-21(P)(A), "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations

5. XN-75-32(P)(A), "Computational Procedure for Evaluating Fuel Rod Bowing"

6. XN-NF-78-44(NP)(A), "A Generic Analysis of the Control Rod Ejection Transient for Pressurized Water Reactors"

7. XN-NF-621(P)(A), " Exxon Nuclear DNB Correlation for PWR Fuel Designs"

8. XN-NF-82-06(P)(A), "Qualification of Exxon Nuclear Fuel for Extended Burnup"

9. ANF-88-133(P)(A), "Qualification of Advanced Nuclear Fuels' PWR Design Methodology for Rod Burnups of 62 GWd/MTU"

10. XN-NF-85-92 (P)(A), "Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results"

11. EMF-92-116(P)(A), "Generic Mechanical Design Criteria for PWR Fuel Design"

12. EMF-92-153(P)(A), "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel"

13. EMF-96-029(P)(A), Volumes 1 and 2, "Reactor Analysis System for PWRs Volume 1 – Methodology Description, Volume 2 – Benchmarking Results"

14. EMF-1961(P)(A), "Statistical Setpoint/Transient Methodology for Combustion Engineering Type Reactors"

15. EMF-2310(P)(A), "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors," Revision 1, as supplemented by ANP-3000(P), “St. Lucie Unit 1 EPU - Information to Support License Amendment Request,” Revision 0.

16. EMF-2328(P)(A), "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based,” Revision 0, as supplemented by ANP-3000(P), “St. Lucie Unit 1 EPU - Information to Support License Amendment Request,” Revision 0.

17. EMF-2103(P)(A), “Realistic Large Break LOCA Methodology for Pressurized Water Reactors,” Revision 0, as supplemented by ANP-2903(P), “St. Lucie Nuclear Plant Unit 1

2

INSERT 2

EPU Cycle Realistic Large Break LOCA summary Report with Zr-4 Fuel Cladding," Revision 1.

18. BAW-10240(P)(A) Revision 0, "Incorporation of M5 Properties in Framatome ANP Approved Methods."

19. WCAP-16045-P-A, Revision 0, "Qualification of the Two -Dimensional Transport Code PARAGON," August 2004.

5.6 Reporting Requirements

~~5.6.4 RCS Pressure and Temperature Limits Report (continued)~~

~~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:~~

~~-------------------------------REVIEWER'S NOTE-------------------------------------~~
~~Licensees that have received prior NRC approval to relocate Topical Report revision numbers and dates to licensee control need only list the number and title of the Topical Report, and the PTLR will contain the complete identification for each of the Technical Specification referenced Topical Reports used to prepare the PTLR (i.e., report number, title, revision, date, and any supplements). See NRC ADAMS Accession No: ML110660285 for details.~~ 4
~~---~~

~~[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.~~

~~-----------------------------------REVIEWER'S NOTE------------------------------------~~
~~The methodology for the calculation of the P-T limits for NRC approval should include the following provisions:~~

~~1. The methodology shall describe how the neutron fluence is calculated (reference new Regulatory Guide when issued).~~

~~2. The Reactor Vessel Material Surveillance Program shall comply with Appendix H to 10 CFR 50. The reactor vessel material irradiation surveillance specimen removal schedule shall be provided, along with how the specimen examinations shall be used to update the PTLR curves.~~

~~3. Low Temperature Overpressure Protection (LTOP) System lift setting limits for the Power Operated Relief Valves (PORVs), developed using NRC-approved methodologies may be included in the PTLR.~~

~~4. The adjusted reference temperature (ART) for each reactor beltline material shall be calculated, accounting for radiation embrittlement, in accordance with Regulatory Guide 1.99, Revision 2.~~

~~5. The limiting ART shall be incorporated into the calculation of the pressure and temperature limit curves in accordance with NUREG-0800 Standard Review Plan 5.3.2, Pressure-Temperature Limits.~~

~~6. The minimum temperature requirements of Appendix G to 10 CFR Part 50 shall be incorporated into the pressure and temperature limit curves.~~

5.6 Reporting Requirements

~~5.6.4 RCS Pressure and Temperature Limits Report (continued)~~

~~7. Licensees who have removed two or more capsules should compare for each surveillance material the measured increase in reference temperature (RT_{NDT}) to the predicted increase in RT_{NDT}; where the predicted increase in RT_{NDT} is based on the mean shift in RT_{NDT} plus the two standard deviation value ($2\sigma_{\Delta}$) specified in Regulatory Guide 1.99, Revision 2. If the measured value exceeds the predicted value (increase in RT_{NDT} + $2\sigma_{\Delta}$), the licensee should provide a supplement to the PTLR to demonstrate how the results affect the approved methodology.~~ 4

Table 3.3-6, Action 15
Table 3.3-11, Actions 4 and 5
DOC M03

5.6.~~5~~ 4 Post Accident Monitoring Report 1

When a report is required by Condition B or F of LCO 3.3.~~[11]~~ 9, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

~~5.6.6 Tendon Surveillance Report~~

~~[Any abnormal degradation of the containment structure detected during the tests required by the Pre-stressed Concrete Containment Tendon Surveillance Program shall be reported to the NRC within 30 days. The report shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, and the corrective action taken.]~~ 5

6.9.1.12 5.6.~~7~~ 5 Steam Generator Tube Inspection Report 1

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification ~~5.5.9~~ 5.5.6, "Steam Generator (SG) Program." The report shall include:

6.9.1.12.a a. The scope of inspections performed on each SG;

DOC M02 b. The nondestructive examination techniques utilized for tubes with increased degradation susceptibility;

c. For each degradation mechanism found:

6.9.1.12.c 1. The nondestructive examination techniques utilized;

5.6 Reporting Requirements

6.9.1.12 5.6.~~7~~ 5 **Steam Generator Tube Inspection Report** (continued) 1

6.9.1.12.d DOC A03 2. The location, orientation (if linear), measured size (if available), and voltage response for each indication. For tube wear at support structures less than 20 percent through-wall, only the total number of indications needs to be reported;

6.9.1.12.g DOC A03 3. A description of the condition monitoring assessment and results, including the margin to the tube integrity performance criteria and comparison with the margin predicted to exist at the inspection by the previous forward-looking tube integrity assessment; and

6.9.1.12.e 4. The number of tubes plugged ~~[or repaired]~~ during the inspection outage~~;~~.

~~[5. The repair methods utilized and the number of tubes repaired by each repair method.]~~ 2

DOC M02 d. An analysis summary of the tube integrity conditions predicted to exist at the next scheduled inspection (the forward-looking tube integrity assessment) relative to the applicable performance criteria, including the analysis methodology, inputs, and results;

6.9.1.12.f 6.9.1.12.h e. The number and percentage of tubes plugged ~~[or repaired]~~ to date, and the effective plugging percentage in each SG; and 2

DOC M02 f. The results of any SG secondary side inspections~~;~~.

~~[g. Insert any plant-specific reporting requirements, if applicable.]~~ 2

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

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

6.9.1.8 5.6.1 Annual Radiological Environmental Operating Report

---NOTE---

6.9.1.8 Footnote ** ~~[~~A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station.~~]~~ 2

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

DOC M01 The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements ~~[in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979]~~. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible. 2

6.9.1.7 5.6.2 Radiological Effluent Release Report

---NOTE---

6.9.1.7 Footnote * ~~[~~A single submittal may be made for a multiple unit station. The submittal shall 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.~~]~~ 2

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

5.6 Reporting Requirements

6.9.1.11 5.6.3 CORE OPERATING LIMITS REPORT

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: , including approved revisions and supplements, as specified thereto 1

INSERT 1 ~~[The individual specifications that address core operating limits must be referenced here.]~~ 2

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:

~~------------------------------REVIEWER'S NOTE-------------------------------------~~
~~Licensees that have received prior NRC approval to relocate Topical Report revision numbers and dates to licensee control need only list the number and title of the Topical Report, and the COLR will contain the complete identification for each of the Technical Specification referenced Topical Reports used to prepare the COLR (i.e., report number, title, revision, date, and any supplements). See NRC ADAMS Accession No: ML110660285 for details.~~ 3
~~---~~

INSERT 2 ~~[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.]~~ 2

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

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

~~5.6.4 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT~~

~~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:~~ 4

~~[The individual specifications that address RCS pressure and temperature limits must be referenced here.]~~

2

INSERT 1

LCO 3.1.1	SHUTDOWN MARGIN (SDM)
LCO 3.1.3	Moderator Temperature Coefficient (MTC)
LCO 3.1.4	Control Element Assembly (CEA) Alignment
LCO 3.1.6	Regulating Control Element Assembly (CEA) Insertion Limits
LCO 3.2.1	Linear Heat Rate (LHR)
LCO 3.2.2	Total Integrated Radial Peaking Factor (F_r^T)
LCO 3.2.4	AXIAL SHAPE INDEX (ASI)
LCO 3.4.1	RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits
LCO 3.9.1	Boron Concentration

2

INSERT 2

1. WCAP-11596-P-A, “Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores,” June 1988 (Westinghouse Proprietary).

2. NF-TR-95-01, “Nuclear Physics Methodology for Reload Design of Turkey Point & St. Lucie Nuclear Plants,” Florida Power & Light Company, January 1995.

3. CENPD-275-P, Revision 1-P-A, “C-E Methodology for Core Designs Containing Gadolinia-Urania Burnable Absorbers,” May 1988, & Revision 1-P Supplement 1-P-A, April 1999.

4. CEN-123(F)-P, “Statistical Combination of Uncertainties Methodology Part 1: CE Calculated Local Power Density and Thermal Margin/Low Pressure LSSS for St. Lucie Unit 1,” December 1979.

5. CEN-123(F)-P, “Statistical Combination of Uncertainties Methodology Part 3: CE Calculated Departure from Nucleate Boiling and Linear Heat Rate Limiting Conditions for Operation for St. Lucie Unit 1,” February 1980.

6. CEN-191(B)-P, “CETOP-D Code Structure and Modeling Methods for Calvert Cliffs Units 1 and 2,” December 1981.

7. Letter, J.W. Miller (NRC) to J.R. Williams, Jr. (FPL), Docket No. 50-389, Regarding Unit 2 Cycle 2 License Approval (Amendment No. 8 to NPF-16 and SER), November 9, 1984 (Approval of CEN-123(F)-P (three parts) and CEN-191(B)-P).

8. Letter, J.A. Norris (NRC) to J.H. Goldberg (FPL), Docket No. 50-389, “St. Lucie Unit 2 – Change to Technical Specification Bases Sections ‘2.1.1 Reactor Core’ and ‘3/4.2.5 DNB Parameters’ (TAC No. M87722),” March 14, 1994 (Approval of CEN-371(F)-P).

9. CENPD-225-P-A, “Fuel and Poison Rod Bowing,” June 1983.

10. CENPD-139-P-A, “C-E Fuel Evaluation Model Topical Report,” July 1974.

11. CEN-161(B)-P-A, “Improvements to Fuel Evaluation Model,” August 1989.

12. CEN-161(B)-P, Supplement 1-PA, “Improvements to Fuel Evaluation Model,” January 1992.

13. CENPD-132, Supplement 3-P-A, “Calculative Methods for the C-E Large Break LOCA Evaluation Model for the Analysis of C-E and W Designed NSSS,” June 1985.

14. CENPD-133, Supplement 5-A, “CEFLASH-4A, A FORTRAN77 Digital Computer Program for Reactor Blowdown Analysis,” June 1985.

15. CENPD-134, Supplement 2-A, “COMPERC-II, a Program for Emergency Refill-Reflood of the Core,” June 1985.

2

INSERT 2

16. CENPD-135-P, Supplement 5, “STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program,” April 1977.

17. Letter, R.L. Baer (NRC) to A.E. Scherer (CE), “Evaluation of Topical Report CENPD-135, Supplement #5,” September 6, 1978.

18. CENPD-137, Supplement 1-P, “Calculative Methods for the C-E Small Break LOCA Evaluation Model,” January 1977.

19. CENPD-133, Supplement 3-P, “CEFLASH-4AS, A Computer Program for the Reactor Blowdown Analysis of the Small Break Loss of Coolant Accident,” January 1977.

20. Letter, K. Kneil (NRC) to A.E. Scherer (CE), “Evaluation of Topical Reports CENPD-133, Supplement 3-P and CENPD-137, Supplement 1 P,” September 27, 1977.

21. CENPD-138, Supplement 2-P, “PARCH, A FORTRAN-IV Digital Program to Evaluate Pool Boiling, Axial Rod and Coolant Heatup,” January 1977.

22. Letter, C. Aniel (NRC) to A.E. Scherer (CE), “Evaluation of Topical Report CENPD-138, Supplement 2-P,” April 10, 1978.

23. Letter, W.H. Bohlke (FPL) to Document Control Desk (NRC), “St. Lucie Unit 2, Docket No. 50-389, Proposed License Amendment, MTC Change from –27 pcm to –30 pcm,” L-91-325, December 17, 1991.

24. Letter, J.A. Norris (NRC) to J.H. Goldberg (FPL), “St. Lucie Unit 2 – Issuance of Amendment Re: Moderator Temperature Coefficient (TAC No. M82517),” July 15, 1992.

25. Letter, J.W. Williams, Jr. (FPL) to D.G. Eisenhut (NRC), “St. Lucie Unit No. 2, Docket No. 50-389, Proposed License Amendment, Cycle 2 Reload,” L-84-148, June 4, 1984.

26. Letter, J.R. Miller (NRC) to J.W. Williams, Jr. (FPL), Docket No. 50-389, Regarding Unit 2 Cycle 2 License Approval (Amendment No. 8 to NPF-16 and SER), November 9, 1984 (Approval to Methodology contained in L-84-148).

27. CEN-348(B)-P-A, Supplement 1-P-A, “Extended Statistical Combination of Uncertainties,” January 1997.

28. CEN-372-P-A, “Fuel Rod Maximum Allowable Gas Pressure,” May 1990.

29. CEN-396(L)-P, “Verification of the Acceptability of a 1-Pin Burnup Limit of 60 MWD/KG for St. Lucie Unit 2,” November 1989 (NRC SER dated October 18, 1991, Letter J.A. Norris (NRC) to J.H. Goldberg (FPL), TAC No. 75947).

30. CENPD-269-P, Rev. 1-P, “Extended Burnup Operation of Combustion Engineering PWR Fuel,” July 1984.

2

INSERT 2

31. CEN-289(A)-P, “Revised Rod Bow Penalties for Arkansas Nuclear One Unit 2,” December 1984 (NRC SER dated December 21, 1999, Letter K. N. Jabbour (NRC) to T.F. Plunkett (FPL), TAC No. MA4523).

32. CENPD-137, Supplement 2-P-A, “Calculative Methods for the ABB CE Small Break LOCA Evaluation Model,” April 1998.

33. CENPD-140-A, “Description of the CONTRANS Digital Computer Code for Containment Pressure and Temperature Transient Analysis,” June 1976.

34. CENPD-387-P-A, Revision 000, “ABB Critical Heat Flux Correlations for PWR Fuel,” May 2000.

35. CENPD-132, Supplement 4-P-A, “Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model," March 2001.

36. CENPD-137, Supplement 2-P-A, “Calculative Methods for the ABB CE Small Break LOCA Evaluation Model,” April 1998.

37. CENPD-404-P-A, Rev. 0, “Implementation of ZIRLOTM Cladding Material in CE Nuclear Power Fuel Assembly Designs,” November 2001.

38. WCAP-9272-P-A, “Westinghouse Reload Safety Evaluation Methodology,” July 1985.

39. WCAP-10216-P-A, Revision 1A, “Relaxation of Constant Axial Offset Control; FQ Surveillance Technical Specification,” February 1994.

40. WCAP-11397-P-A, (Proprietary), 'Revised Thermal Design Procedure," April 1989.

41. WCAP-14565-P-A, (Proprietary), "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," October 1999.

42. WCAP-14565-P-A, Addendum 1, "Qualification of ABB Critical Heat Flux Correlations with VIPRE-01 Code," May 2003.

43. Letter, W. Jefferson Jr. (FPL) to Document Control Desk (USNRC), “St. Lucie Unit 2 Docket No. 50-389: Proposed License Amendment WCAP-9272 Reload Methodology and Implementing 30% Steam Generator Tube Plugging Limit,” L-2003-276, December 2003 (NRC SER dated January 31, 2005, Letter B.T. Moroney (NRC) to J.A. Stall (FPL), TAC No. MC1566).

44. WCAP-14882-P-A, Rev. 0, “RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses”, April 1999.

45. WCAP-7908-A, Rev. 0, “FACTRAN-A FORTRAN IV Code for Thermal Transients in a UO2 Fuel Rod”, December 1989.

2

INSERT 2

46. WCAP-7979-P-A, Rev. 0, “TWINKLE - A Multi-Dimensional Neutron Kinetics Computer Code”, January 1975.

47. WCAP-7588, Rev. 1-A, “An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Special Kinetics Methods”, January 1975.

48. WCAP-16045-P-A, Revision 0, “Qualification of the Two-Dimensional Transport Code PARAGON,” August 2004.

49. EMF-96-029(P)(A), Volumes 1 and 2, “Reactor Analysis System for PWRs, Volume 1 Methodology Description, Volume 2 Benchmarking Results,” Siemens Power Corporation, January 1997.

50. XN-NF-78-44 (NP)(A), “A Generic Analysis of the Control Rod Ejection Transient for Pressurized Water Reactors,” Exxon Nuclear Company, Inc.,” October 1983.

51. XN-75-27(A) and Supplements 1 through 5, “Exxon Nuclear Neutronics Design Methods for Pressurized Water Reactors,” Exxon Nuclear Company, Report and Supplement 1 dated April 1977, Supplement 2 dated December 1980, Supplement 3 dated September 1981 (P), Supplement 4 dated December 1986 (P), and Supplement 5 dated February 1987 (P).

52. XN-NF-82-06 (P)(A), Rev. 1 and Supplements 2, 4, and 5, “Qualification of Exxon Nuclear Fuel for Extended Burnup,” Exxon Nuclear Company, Inc., October 1986.

53. XN-NF-85-92(P)(A), “Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results,” Exxon Nuclear Company, Inc., November 1986.

54. ANF-88-133(P)(A) and Supplement 1, “Qualification of Advanced Nuclear Fuels PWR Design Methodology for Rod Burnups of 62 GWd/MTU,” Advanced Nuclear Fuels Corporation, December 1991.

55. EMF-92-116(P)(A), Rev. 0, “Generic Mechanical Design Criteria for PWR Fuel Design,” Siemens Power Corporation, February, 1999.

56. BAW-10240(P)(A), Rev.0, “Incorporation of M5™ Properties in Framatome ANP Approved Methods,” Framatome ANP, Inc., May 2004.

57. XN-NF-82-21(P)(A), Revision 1, “Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations,” Exxon Nuclear Company, September 1983.

58. EMF-92-153(P)(A), Revision 1, “HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel,” January 2005.

59. EMF-1961(P)(A), Revision 0, “Statistical/Transient Methodology for Combustion Engineering Type Reactors,” Siemens Power Corporation, July 2000.

2

INSERT 2

60. EMF-2310(P)(A), Revision 1, “SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors,” Framatome ANP, Inc., May 2004.

61. XN-75-32(P)(A), Supplements 1, 2, 3, and 4, “Computational Procedure for Evaluating Fuel Rod Bowing,” October 1983.

62. BAW-10231P-A Revision 1, “COPERNIC Fuel Rod Design Computer Code,” January 2004.

63. EMF-2103(P)(A) Revision 0, “Realistic Large Break LOCA Methodology for Pressurized Water Reactors,” April 2003.

64. EMF-2328 (P)(A) Revision 0, “PWR Small Break LOCA Evaluation Model, S-RELAP5 Based,” March 2001.

5.6 Reporting Requirements

~~5.6.4 RCS Pressure and Temperature Limits Report (continued)~~

~~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:~~

~~-------------------------------REVIEWER'S NOTE-------------------------------~~
~~Licensees that have received prior NRC approval to relocate Topical Report revision numbers and dates to licensee control need only list the number and title of the Topical Report, and the PTLR will contain the complete identification for each of the Technical Specification referenced Topical Reports used to prepare the PTLR (i.e., report number, title, revision, date, and any supplements). See NRC ADAMS Accession No: ML110660285 for details.~~
~~--~~

~~[Identify the NRC staff approval document by date.]~~

4

~~c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.~~

~~-------------------------------REVIEWER'S NOTE-------------------------------~~
~~The methodology for the calculation of the P-T limits for NRC approval should include the following provisions:~~

~~1. The methodology shall describe how the neutron fluence is calculated (reference new Regulatory Guide when issued).~~

~~2. The Reactor Vessel Material Surveillance Program shall comply with Appendix H to 10 CFR 50. The reactor vessel material irradiation surveillance specimen removal schedule shall be provided, along with how the specimen examinations shall be used to update the PTLR curves.~~

~~3. Low Temperature Overpressure Protection (LTOP) System lift setting limits for the Power Operated Relief Valves (PORVs), developed using NRC-approved methodologies may be included in the PTLR.~~

~~4. The adjusted reference temperature (ART) for each reactor beltline material shall be calculated, accounting for radiation embrittlement, in accordance with Regulatory Guide 1.99, Revision 2.~~

~~5. The limiting ART shall be incorporated into the calculation of the pressure and temperature limit curves in accordance with NUREG-0800 Standard Review Plan 5.3.2, Pressure-Temperature Limits.~~

~~6. The minimum temperature requirements of Appendix G to 10 CFR Part 50 shall be incorporated into the pressure and temperature limit curves.~~

5.6 Reporting Requirements

~~5.6.4~~ ~~RCS Pressure and Temperature Limits Report (continued)~~

~~7. Licensees who have removed two or more capsules should compare for each surveillance material the measured increase in reference temperature (RT_{NDT}) to the predicted increase in RT_{NDT}; where the predicted increase in RT_{NDT} is based on the mean shift in RT_{NDT} plus the two standard deviation value ($2\sigma_{\Delta}$) specified in Regulatory Guide 1.99, Revision 2. If the measured value exceeds the predicted value (increase in RT_{NDT} + $2\sigma_{\Delta}$), the licensee should provide a supplement to the PTLR to demonstrate how the results affect the approved methodology.~~ 4

Table 3.3-6, Action 27 3.3.3.6 Actions c and d DOC M03 5.6.~~5~~ 4 Post Accident Monitoring Report 1

9

When a report is required by Condition B or F of LCO 3.3.~~[11]~~, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

~~5.6.6~~ ~~Tendon Surveillance Report~~

~~[Any abnormal degradation of the containment structure detected during the tests required by the Pre-stressed Concrete Containment Tendon Surveillance Program shall be reported to the NRC within 30 days. The report shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, and the corrective action taken.]~~ 5

6.9.1.12 5.6.~~7~~ 5 Steam Generator Tube Inspection Report 1

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification ~~5.5.9~~ 5.5.6, "Steam Generator (SG) Program." The report shall include:

6.9.1.12.a a. The scope of inspections performed on each SG;

DOC M02 b. The nondestructive examination techniques utilized for tubes with increased degradation susceptibility;

c. For each degradation mechanism found:

6.9.1.12.c 1. The nondestructive examination techniques utilized;

5.6 Reporting Requirements

6.9.1.12 5.6.~~7~~ 5 **Steam Generator Tube Inspection Report** (continued) 1

6.9.1.12.d DOC A03 2. The location, orientation (if linear), measured size (if available), and voltage response for each indication. For tube wear at support structures less than 20 percent through-wall, only the total number of indications needs to be reported;

6.9.1.12.g DOC A03 3. A description of the condition monitoring assessment and results, including the margin to the tube integrity performance criteria and comparison with the margin predicted to exist at the inspection by the previous forward-looking tube integrity assessment; and

6.9.1.12.e 4. The number of tubes plugged ~~[or repaired]~~ during the inspection outage~~;~~.

~~[5. The repair methods utilized and the number of tubes repaired by each repair method.]~~ 2

DOC M02 d. An analysis summary of the tube integrity conditions predicted to exist at the next scheduled inspection (the forward-looking tube integrity assessment) relative to the applicable performance criteria, including the analysis methodology, inputs, and results;

6.9.1.12.f 6.9.1.12.h e. The number and percentage of tubes plugged ~~[or repaired]~~ to date, and the effective plugging percentage in each SG; and 2

DOC M02 f. The results of any SG secondary side inspections~~;~~.

~~[g. Insert any plant-specific reporting requirements, if applicable.]~~ 2

JUSTIFICATION FOR DEVIATIONS
5.6, REPORTING REQUIREMENTS

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

4. ISTS 5.6.4 provides requirements for the RCS Pressure and Temperature Limits report. PSL is not adopting a Pressure Temperature Limits Report (PTLR) and is retaining the RCS Pressure and Temperature (P/T) limits in the ITS 3.4.3, RCS Pressure and Temperature (P/T) Limits. Therefore, this report is not included in the PSL ITS, consistent with the current licensing basis. Subsequent Specifications are renumbered as a result of this deletion.

5. ISTS 5.6.6 provides requirements for the Tendon Surveillance Report. The Containment design at PSL does not include pre-stressed concrete tendons. Therefore, this report is not included in the PSL ITS, consistent with the current licensing basis. Subsequent Specifications are renumbered as a result of this deletion.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 5.6, REPORTING REQUIREMENTS

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 7

ITS Section 5.7, High Radiation Area

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

6.12 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

and b

5.7 6.12.1 ~~In lieu of the "control device" or "alarm signal"~~ required by paragraph 20.1601(a) of 10 CFR Part 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr measured at a distance of 30 cm (12 in) shall be barricaded and conspicuously
5.7.1 posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP)*. 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 which continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device which 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 level in the area has been established and personnel have been made knowledgeable of them.

c. An ~~health physics qualified~~ individual ~~(i.e.,~~ 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 facility Health Physicist~~ in the RWP. A02 LA01

LA02 A02

6.12.2 In addition to the requirements of Specification 6.12.1, areas accessible to personnel with radiation levels in excess of 1000 mrem/hr at 30 cm (12 in) and less than 500 rad/hr at 1 meter shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the ~~Shift Foreman on duty and/or health physics supervision~~. Doors shall remain locked except during periods of access by personnel under an approved RWP which shall specify the dose rate levels in
5.7.2 the immediate work area and the maximum allowable stay time for individuals in that area. For individual areas accessible to personnel with radiation levels in excess of 1000 mrem/hr at 30 cm (12 in) and less than 500 rads/hr at 1 meter that are located within large areas, such as PWR containment, where no enclosure exists for purposes of locking, and no enclosure can be reasonably constructed around the individual areas, then that area shall be roped off, conspicuously posted and a flashing light shall be activated as a warning device. In lieu of the stay time specification of the RWP, direct or remote (such as use of closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities within the area.

shift supervisor, radiation protection manager, or his or her designee.

radiation protection

radiation protection

5.7.1 * ~~Health Physics~~ personnel or personnel escorted by ~~Health Physics~~ personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they are otherwise following plant radiation protection procedures for entry into high radiation areas. A02

ADMINISTRATIVE CONTROLS

6.12 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

and b

5.7 6.12.1 ~~In lieu of the "control device" or "alarm signal"~~ required by paragraph 20.1601(a) of 10 CFR Part 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr measured at a distance of 30 cm (12 in) shall be barricaded and conspicuously
5.7.1 posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP)*. 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 which continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device which 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 level in the area has been established and personnel have bee made knowledgeable of them.

c. An ~~health physics qualified~~ individual ~~(i.e.,~~ qualified in radiation protection procedures~~)~~ with a radiation dose rate monitoring device who is responsible for providing position control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified ~~by the facility Health Physicist~~ in the RWP. A02 LA01

LA02 A02

6.12.2 In addition to the requirements of Specification 6.12.1, areas accessible to personnel with radiation levels in excess of 1000 mrem/hr at 30 cm (12 in) and less than 500 rad/hr at 1 meter shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the ~~Shift Foreman on duty and/or health physics supervision~~ shift supervisor, radiation protection manager, or his or her designee. Doors shall remain locked except during periods of access by personnel under an approved RWP which shall specify the dose rate levels in the immediate work area and the maximum allowable stay time for individuals in
5.7.2 that area. For individual areas accessible to personnel with radiation levels in excess of 1000 mrem/hr at 30 cm (12 in) and less than 500 rads/hr at 1 meter that are located within large areas, such as PWR containment, where no enclosure exists for purposes of locking, and no enclosure can be reasonably constructed around the individual areas, then that area shall be roped off, conspicuously posted and a flashing light shall be activated as a warning device. In lieu of the stay time specification of the RWP, direct or remote (such as use of closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities within the area.

A02

5.7.1 * ~~Health Physics~~ radiation protection personnel or personnel escorted by ~~Health Physics~~ radiation protection personnel shall be exempt for the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they are otherwise following plant radiation protection procedures for entry into high radiation areas.

DISCUSSION OF CHANGES
ITS 5.7, HIGH RADIATION AREA

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG - 1432, Rev. 5.0, "Standard Technical Specifications - Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 CTS 6.12 specifies “health physics” personnel when referring to radiation protection qualified individuals, supervision, and personnel exempt from radiation work permits (RWPs). PSL does not use the term “health physics” when referring to these personnel but rather uses the term “radiation protection.” ITS 5.7 replaces the term “health physics” with “radiation protection.” This changes the CTS by replacing references to health physics personnel with radiation protection personnel. Specifically, a health physics qualified individual is replaced with an individual qualified in radiation protection procedures; health physics supervision is replaced with radiation protection manager, or his or her designee; and health physics personnel is replaced with radiation protection personnel. These changes are consistent with the ISTS and are designated as administrative because they do not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 (*Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements*) CTS 6.12.1.c, states, in part, that the health physics qualified individual responsible for positive control over the activities within the high radiation area shall also perform periodic radiation surveillance at the frequency specified by the facility Health Physicist in the RWP. ITS 5.7.1.c provides a similar requirement but does not specify that the frequency of radiation monitoring specified in the RWP is specified by the facility Health Physicist. This changes the CTS by moving the procedural detail that the frequency of radiation monitoring is specified by the facility Health Physicist to the Updated Final Safety Analysis Report (UFSAR).

The removal of these details, which are related to who establishes the frequency of radiation monitoring in the RWP, from the Technical Specifications is acceptable because this type of information is not necessary to be included in Technical Specifications to provide adequate protection of public health and safety. Implementation of radiation work procedures, including RWPs, are the responsibility of the radiation protection manager (i.e., facility Health Physicist). This change is acceptable because the individual qualified in radiation protection procedures (i.e., health physics qualified individual) who is responsible for providing positive control over the activities within the area will continue to be required to perform periodic radiation surveillance at the frequency specified in the RWP independent of who is responsible for establishing the surveillance frequency. This change is also acceptable because the removed information will be adequately controlled in the UFSAR. Any changes to the UFSAR are made pursuant to 10 CFR 50.59 or 10 CFR 50.71(e). These change control processes ensure changes are properly evaluated. This change is designated as a less restrictive removal of detail change because information related to meeting Technical Specification requirements are being removed from the Technical Specifications.

LA02 (*Type 4 – Removal of LCO, SR, or other TS requirements to the TRM, UFSAR, ODCM, QAP, CLRT Program, IST Program, ISI Program or Surveillance Frequency Control Program*) CTS 6.12.2 uses the title "Shift Foreman." ITS 5.7.2 uses the generic title "shift supervisor." This changes the CTS by moving the specific PSL organizational title to the UFSAR or the Quality Assurance Topical Report (QATR) and replacing it with a generic title.

The removal of these details, which are related to meeting Technical Specification requirements, from the Technical Specifications is acceptable because this type of information is not necessary to be included in Technical Specifications to provide adequate protection of public health and safety. The allowance to relocate the specific PSL organizational titles out of the Technical Specifications is consistent with the NRC letter from C. Grimes to the Owners Groups Technical Specification Committee Chairmen, dated November 10, 1994. The various requirements of the shift supervisor are retained in the ITS. Also, this change is acceptable because the removed information will be adequately controlled in the UFSAR or the QATR. Any changes to the UFSAR are made pursuant to 10 CFR 50.59 or 10 CFR 50.71(e) and changes to the QATR are made pursuant to 10 CFR 50.54(a). These change control processes ensure changes are properly evaluated. This change is designated as a less restrictive removal of detail change because information related to meeting Technical Specification requirements are being removed from the Technical Specifications.

<u>LESS RESTRICTIVE CHANGES</u>

None

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

6.12.1 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:

Insert 1

1

~~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 or radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measurers.~~

~~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~~

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

1

Insert 1

6.12.1 including footnote*

5.7.1 Each high radiation area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr measured at a distance of 30 cm (12 in) 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). Radiation protection personnel or personnel escorted by radiation protection personnel shall be exempt for the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they are otherwise following plant radiation protection procedures for entry into 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 which continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device which 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 level in the area has been established and personnel have been made knowledgeable 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 in the RWP.

6.12.2

5.7.2 In addition to the requirements of Specification 5.7.1, areas accessible to personnel with radiation levels in excess of 1000 mrem/hr at 30 cm (12 in) and less than 500 rad/hr at 1 meter shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the shift supervisor, radiation protection manager, or his or her designee. Doors shall remain locked except during periods of access by personnel under an approved RWP which shall specify the dose rate levels in the immediate work area and the maximum allowable stay time for individuals in that area. For individual areas accessible to personnel with radiation levels in excess of 1000 mrem/hr at 30 cm (12 in) and less than 500 rads/hr at 1 meter that are located within large areas, such as PWR containment, where no enclosure exists for purposes of locking, and no enclosure can be reasonably constructed around the individual areas, then that area shall be roped off, conspicuously posted and a flashing light shall be activated as a warning device. In lieu of the stay time specification of the RWP, direct or remote (such as use of closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities within the area.

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)~~ 1

~~(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, or 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.~~

~~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.~~

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)~~ 1

~~c. Individuals qualified in radiation protection procedures my 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, 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.~~

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)~~ 1

~~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.~~

~~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.~~

5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

6.12.1 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:

Insert 1

1

~~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 or radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measurers.~~

~~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~~

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

1

Insert 1

6.12.1 including footnote*

5.7.1 Each high radiation area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr measured at a distance of 30 cm (12 in) 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). Radiation protection personnel or personnel escorted by radiation protection personnel shall be exempt for the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they are otherwise following plant radiation protection procedures for entry into 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 which continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device which 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 level in the area has been established and personnel have been made knowledgeable 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 in the RWP.

6.12.2

5.7.2 In addition to the requirements of Specification 5.7.1, areas accessible to personnel with radiation levels in excess of 1000 mrem/hr at 30 cm (12 in) and less than 500 rad/hr at 1 meter shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the shift supervisor, radiation protection manager, or his or her designee. Doors shall remain locked except during periods of access by personnel under an approved RWP which shall specify the dose rate levels in the immediate work area and the maximum allowable stay time for individuals in that area. For individual areas accessible to personnel with radiation levels in excess of 1000 mrem/hr at 30 cm (12 in) and less than 500 rads/hr at 1 meter that are located within large areas, such as PWR containment, where no enclosure exists for purposes of locking, and no enclosure can be reasonably constructed around the individual areas, then that area shall be roped off, conspicuously posted and a flashing light shall be activated as a warning device. In lieu of the stay time specification of the RWP, direct or remote (such as use of closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities within the area.

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)~~ 1

~~(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, or 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.~~

~~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.~~

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)~~ 1

~~c. Individuals qualified in radiation protection procedures my 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, 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.~~

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)~~ 1

~~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.~~

~~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.~~

JUSTIFICATION FOR DEVIATIONS
ITS 5.7, HIGH RADIATION AREA

1. ISTS 5.7 provides requirements for High Radiation Areas. The brackets are removed and the proper plant specific information/value is provided. ITS 5.5.7 is revised to reflect the St. Lucie Plant Unit 1 and Unit 2 current licensing basis and high radiation area controls The change is consistent with the requirements in CTS 6.12.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 5.7, HIGH RADIATION AREA

There are no specific No Significant Hazards Considerations for this Section.

ATTACHMENT 8

Relocated/Deleted Current Technical Specifications (CTS)

CTS 6.11, Radiation Protection Program

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

~~ADMINISTRATIVE CONTROLS~~

~~6.11 RADIATION PROTECTION PROGRAM~~

LA01

~~Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.~~

~~6.11 RADIATION PROTECTION PROGRAM~~

~~Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.~~

LA01

DISCUSSION OF CHANGES
CTS 6.11, RADIATION PROTECTION PROGRAM

ADMINISTRATIVE CHANGES

None

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 6 - Removal of LCO, SR, or other TS requirements to the TRM, UFSAR, ODCM, QAPD, or IIP)* CTS 6.11 provides requirements for the Radiation Protection Program. The ITS does not include these requirements. This changes the CTS by moving the requirements for the Radiation Protection Program to the UFSAR.

The removal of these requirements from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The Radiation Protection Program requires procedures to be prepared for personnel radiation protection consistent with 10 CFR 20. These procedures are for nuclear plant personnel and have no impact on nuclear safety or the health and safety of the public. Requirements to have procedures to implement 10 CFR 20 are contained in 10 CFR 20.1101(b). Periodic review of these procedures is addressed in 10 CFR 20.1101(c). Since the PSL Units 1 and 2 Operating Licenses require compliance with 10 CFR 20, there is no need to repeat the requirements in the ITS. As such, the relocated details are not required to be in the ITS to provide adequate protection of the public health and safety. Also, this change is acceptable because these details will be adequately controlled in the UFSAR. Any changes to the UFSAR are made under 10 CFR 50.59 or 10 CFR 50.71(e), which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because details for meeting Technical Specification and regulatory requirements are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

None

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
CTS 6.11, RADIATION PROTECTION PROGRAM

There are no specific No Significant Hazards Considerations for this Specification.

CTS 6.13, Process Control Program

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

~~ADMINISTRATIVE CONTROLS~~

~~6.13 PROCESS CONTROL PROGRAM (PCP)~~

~~Changes to the PCP:~~

LA01

~~1. Shall be documented and this documentation shall contain:~~

~~a) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and~~

~~b. A determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.~~

~~2. Shall become effective after the approval of the plant manager.~~

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

Changes to the ODCM:

1. Shall be documented and this documentation shall contain:

   a) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and

   b. A determination that the change will maintain the level of radioactive effluent control required by 10 CFR 20.1302, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50 and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

See ITS 5.5

2. Shall become effective after the approval of the plant manager.

3. Shall be submitted to the Commission in the form of a complete, legible copy of the entire ODCM as part of or concurrent with the Annual Radioactive Effluent Release Report for the period of the report in which any change to the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (e.g., month/year) the change was implemented.

~~ADMINISTRATIVE CONTROLS~~

~~6.13 PROCESS CONTROL PROGRAM (PCP)~~

~~Changes to the PCP:~~

~~1. Shall be documented and this documentation shall contain:~~ LA01

~~a) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and~~

~~b. A determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.~~

~~2. Shall become effective after the approval of the plant manager.~~

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

Changes to the ODCM:

1. Shall be documented and this documentation shall contain:

   a) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and

   See ITS 5.5

   b) A determination that the change will maintain the level of radioactive effluent control required by 10 CFR 20.1302, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50 and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

2. Shall become effective after the approval of the plant manager.

3. Shall be submitted to the Commission in the form of a complete, legible copy of the entire ODCM as part of or concurrent with the Annual Radioactive Effluent Release Report for the period of the report in which any change to the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (e.g., month/year) the change was implemented.

ADMINISTRATIVE CHANGES

None

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 6 - Removal of LCO, SR, or other TS requirements to the TRM, UFSAR, ODCM, QAPD, or IIP)* CTS Definition 1.23 contains the definition for the Process Control Program (PCP). CTS 6.13 describes the process for control of changes to the PCP. The ITS does not include these requirements. This changes the CTS by moving the requirements of the PCP to the UFSAR.

The removal of these requirements from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The PCP implements the requirements of 10 CFR 20, 10 CFR 61, and 10 CFR 71. Compliance with these regulations is required by the PSL Units 1 and 2 Operating Licenses, and procedures are the method to ensure compliance with the program. Regulations provide an adequate level of control for the affected requirements and inclusion of this requirement in the Technical Specifications is not necessary. Also, this change is acceptable because these details will be adequately controlled in the UFSAR. Any changes to the UFSAR are made under 10 CFR 50.59 or 10 CFR 50.71(e), which ensures changes are properly evaluated. This change is designated as a less restrictive removal of requirements because details for meeting Technical Specification and regulatory requirements are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

None

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
CTS 6.13, PROCESS CONTROL PROGRAM (PCP)

There are no specific No Significant Hazards Considerations for this Specification.