ENCLOSURE 2

VOLUME 16

TURKEY POINT NUCLEAR GENERATING STATION UNIT 3 AND UNIT 4

IMPROVED TECHNICAL SPECIFICATIONS CONVERSION

ITS CHAPTER 5.0 ADMINISTRATIVE CONTROLS

Revision 0

LIST OF ATTACHMENTS

1. **ITS 5.1 – Responsibility**
2. **ITS 5.2 – Organization**
3. **ITS 5.3 – Unit Staff Qualifications**
4. **ITS 5.4 – Procedures**
5. **ITS 5.5 – Programs and Manuals**
6. **ITS 5.6 – Reporting Requirements**
7. **ITS 5.7 – High Radiation Area**

ATTACHMENT 1

ITS Section 5.1 – Responsibility

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

~~SECTION 6.0~~

A01

~~ADMINISTRATIVE CONTROLS~~

ITS

ADMINISTRATIVE CONTROLS

5.1 6.1 RESPONSIBILITY

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

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

6.1.2 The Shift Manager (SM) shall be responsible for the control room command function. During any absence of the SM 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 SM 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.

6.2 ORGANIZATION

ONSITE AND OFFSITE ORGANIZATION

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

See ITS 5.2

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).

b. The Chief Nuclear 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.

c. The plant manager shall be responsible for overall plant safe operation and shall have control over those onsite activities 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 Health Physics Supervisor 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.

DISCUSSION OF CHANGES
ITS SECTION 5.1, RESPONSIBILITY

ADMINISTRATIVE CHANGES

A01 In the conversion of the Turkey Point Nuclear Generating Station (PTN) Unit 3 and Unit 4, 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-Westinghouse 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 does not contain any information concerning the Plant Manager's (or his designee's) role in plant activities that affect nuclear safety. ITS 5.1 contains a requirement that directs 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.

CTS 6.1 provides the responsibilities for the plant manager and shift manager. The proposed change adds an additional 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. It is being added to ensure the plant manager is cognizant of those activities associated with 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 TS.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

None

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

5.0 ADMINISTRATIVE CONTROLS

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.~~ 1

~~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.~~

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.

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.

5.1.2 The ~~[~~Shift ~~Supervisor (SS)]~~ Manager (SM) shall be responsible for the control room command function. During any absence of the ~~[SS]~~ SM from the control room while ~~the~~ 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 ~~[SS]~~ SM from the control room while ~~the unit is~~ 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. 2 3

JUSTIFICATION FOR DEVIATIONS
ITS CHAPTER 5.1, RESPONSIBILITY

1. This "Reviewers Note" is being deleted. The Reviewer's Note is for the NRC reviewer during the NRC review and will not be part of the plant specific Improved Technical Specifications (ITS).

2. The Improved Standard Technical Specifications (ISTS) contain bracketed information and/or values that are generic to all Westinghouse 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. 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.

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 Specification.

ATTACHMENT 2

ITS Section 5.2 – Organization

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

ADMINISTRATIVE CONTROLS

See ITS 5.1

6.1 RESPONSIBILITY

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

6.1.2 The Shift Manager (SM) shall be responsible for the control room command function. During any absence of the SM 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 SM 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.

6.2 ORGANIZATION

5.2.1 ONSITE AND OFFSITE ORGANIZATION

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

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

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)~~. M01 A02

requirements including the plant-specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications

A specified corporate officer

5.2.1.c b. ~~The Chief Nuclear 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. LA01

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

, carry out health physics, or A03

5.2.1.d 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 Health Physics Supervisor 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 CONTROLS

PLANT STAFF

5.2.2 6.2.2 The plant organization shall be subject to the following:

LA02

a. ~~Each on-duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1;~~

~~b. DELETED~~

~~c. At least two licensed Operators shall be present in the control room during reactor startup, scheduled reactor shutdown and during recovery from reactor trips. In addition, while either unit is in MODE 1, 2, 3, or 4, at least one licensed Senior Operator shall be in the control room;~~

5.2.2.c d. A Health Physics Technician* shall be on site when fuel is in the reactor;

~~e. All CORE ALTERATIONS shall be observed and directly supervised by either a licensed Senior Operator or licensed Senior Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation; and~~

~~f. DELETED~~

5.2.2.d h. The Assistant Operations Manager - Line shall hold a Senior Reactor Operator License.

OR

5.2.2.d i. The Operations Manager shall ~~either~~:

1. hold ~~or have held~~ a Senior Reactor Operator License on the Turkey Point Plant~~; or,~~

L01

~~2. have held a Senior Reactor Operator License on a similar plant (i.e., another pressurized water reactor); or~~

~~3. have completed the Turkey Point Plant Senior Management Operations Training Course. (i.e., certified at an appropriate simulator for equivalent senior operator knowledge level.)~~

5.2.2.c * The Health Physics Technician composition may be less than the minimum requirements for a period of time not to exceed 2 hours, in order to accommodate unexpected absence, provided immediate action is taken to fill the required positions.

ADMINISTRATIVE CONTROLS

~~TABLE 6.2-1~~

~~MINIMUM SHIFT CREW COMPOSITION~~

LA02

	~~POSITION~~	~~NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION~~		
		~~BOTH UNITS IN MODE 1, 2, 3, or 4~~	~~BOTH UNITS IN MODE 5 or 6 OR DEFUELED~~	~~ONE UNIT IN MODE 1, 2, 3, or 4 AND ONE UNIT IN MODE 5 or 6 or DEFUELED~~
	~~SM~~	~~1~~	~~1~~	~~1~~
	~~SRO~~	~~1~~	~~none**~~	~~1~~
	~~RO~~	~~3*~~	~~2*~~	~~3*~~
5.2.2.a	AO	3*	3*	3*
5.2.2.e	STA	1***	none	1***

LA02

~~SM - Shift Manager with a Senior Operator license~~
~~SRO - Individual with a Senior Operator license~~
~~RO - Individual with an Operator license~~
5.2.2.a AO - Auxiliary Operator
5.2.2.e STA - Shift Technical Advisor

10 CFR 50.54(m)(2)(i) and 5.2.2.a and 5.2.2.e

LA02

5.2.2.b The shift crew composition may be one less than the minimum requirements of ~~Table 6.2-1~~ for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements ~~of Table 6.2-1~~. ~~This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.~~

L02

5.2.2.a * At least one of the required individuals must be assigned to the designated position for each unit.

LA02

~~** At least one licensed Senior Operator or licensed Senior Operator Limited to Fuel Handling must be present during CORE ALTERATIONS on either unit, who has no other concurrent responsibilities.~~

5.2.2.e ***The STA position may be filled by the SM or an individual with a Senior Operator license who meets the 1985 NRC Policy Statement on Engineering Expertise on Shift.

6.2.3 SHIFT TECHNICAL ADVISOR FUNCTION

5.2.2.e

6.2.3.1 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 and the opposite unit. This individual shall meet the qualifications specified by the 1985 NRC Policy Statement on Engineering Expertise on Shift.

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for

See ITS 5.3

6.3.1.1 The Health Physics Supervisor who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975.

6.3.1.2 The Operations Manager whose requirement for a Senior Reactor Operator License is as stated in Specification 6.2.2.i.

6.3.1.3 The licensed operators, who shall comply only with the requirements of 10 CFR 55.

6.3.1.4 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

6.3.2 When the Health Physics Supervisor does not meet the above requirements, compensatory action shall be taken which the Plant Nuclear Safety Committee determines and the NRC office of Nuclear Reactor Regulation concurs that the action meets the intent of Specification 6.3.1.

6.3.3 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 6.3.1.3, perform the functions described in 10 CFR 50.54(m)

6.4 DELETED

6.5 DELETED

6.6 DELETED

6.7 DELETED

DISCUSSION OF CHANGES
ITS SECTION 5.2, ORGANIZATION

ADMINISTRATIVE CHANGES

A01 In the conversion of the Turkey Point Nuclear Generating Station (PTN) Unit 3 and Unit 4, 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-Westinghouse 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.a requires the organizational charts to be documented in the Quality Assurance (QA) Topical Report and updated in accordance with 10 CFR 50.54(a)(3). ITS 5.2.1.a contains the requirement for the organizational charts to be documented in the QA Topical Report but does not specifically require it to be updated in accordance with 10 CFR 50.54(a)(3). This changes the CTS by not stating in ITS Section 5.0 to update the organizational charts located in the QA Topical Report per the Code of Federal Regulations.

This change is acceptable because the Code of Federal Regulations (CFR) is law and requires changes to the QA program be performed in accordance with 10 CFR 50.54. Thus, the requirement shall be followed whether documented in the ITS Administrative Control Section or not. This change is acceptable because the organizational charts will continue to be located in the QA Topical Report and will continue to be updated per the 10 CFR 50.54. This change is considered administrative since there will be no changes in the requirement to update the QA programs in accordance with the code of federal regulations.

A03 CTS 6.2.1.d and CTS 6.2.1.e allow the individuals that train the operating staff (6.2.1.d), the individuals who carry out QA functions (6.2.1.d) and health physics individuals (6.2.1.e) to report to the appropriate onsite manager but have sufficient organizational freedom to be independent from operating pressure or, in the case of health physics supervisor, have direct access to the onsite individual having responsibility for overall unit management. In addition, CTS 6.2.1.e states that 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 states that 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. The ITS retains the requirement to comply with 10 CFR 50.54 and the Nuclear Fleet Industrial Safety Program states that each employee shall have "Stop Work Authority" if the individual observes unsafe working conditions or behaviors. This change is considered administrative since there will be no changes in the requirement for these individuals to have sufficient organizational freedom to be independent from operating pressure and to stop work if unsafe working conditions or behaviors are observed.

DISCUSSION OF CHANGES
ITS SECTION 5.2, ORGANIZATION

MORE RESTRICTIVE CHANGES

M01 CTS 6.2.1.a, regarding documentation and updating of the relationships between operating organization positions, requires that the 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 QA Topical Report and updated in accordance with 10 CFR 50.54(a)(3). ITS 5.2.1.a states "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…" This changes the CTS by adding more detailed requirements.

This change is acceptable because specifying the relationship of the specific organizational titles to the generic titles used in the Technical Specifications and industry standards in the QA Topical Report continues to ensure that organizational positions and associated responsibilities will be maintained. This change adds this requirement to the Technical Specifications. This change is designated as more restrictive because it requires additional information to be maintained in the QA Topical Report.

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.b uses the title "Chief Nuclear Officer." ITS 5.2.1.b uses the generic title "specified corporate officer." This changes the CTS by not listing the specific organizational titles in ITS Section 5.2.

The removal of the specific title from the Technical Specifications Administrative Controls Section is acceptable because this type of information is not necessary to be included in the Technical Specifications in order to provide adequate protection of public health and safety. This change is acceptable because these types of details will be adequately controlled in the TRM. The TRM is controlled under 10 CFR 50.59, which ensures changes are evaluated. This change is designated as a less restrictive removal of detail change because specific unit staffing requirements are being removed from the Technical Specifications.

DISCUSSION OF CHANGES
ITS SECTION 5.2, ORGANIZATION

LA02 *(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)* The following changes are being proposed in CTS Section 6.2.2.

- CTS 6.2.2.a contains a Table with the minimum shift crew composition. ITS does not contain this Table but refers to the minimum shift crew requirements Table in 10 CFR 50.54. This changes the CTS by moving the CTS minimum shift crew composition table out of Technical Specifications.

- CTS Table 6.2-1 includes a note providing an allowance for the shift crew composition to be one less than the minimum requirements of Table 6.2-1. ITS replaces CTS Table 6.2-1 staffing requirements with 10 CFR 50.54(m)(2)(i) and ITS 5.2.2.a and ITS 5.2.2.e. This changes the CTS by changing the CTS minimum shift crew composition reference from the table to 10 CFR 50.54 and the appropriate ITS specifications.

- CTS 6.2.2.c contains requirements to have at least two licensed Operators in the control room during reactor startup, scheduled reactor shutdown, and during recovery from reactor trips and to have one licensed senior reactor operator in the control room during MODE 1, 2, 3, and 4. ITS 5.2.2 does not contain this requirement. This changes the CTS by moving this requirement out of Technical Specifications.

- CTS 6.2.2.e requires that all CORE ALTERATIONS shall be observed and directly supervised by either a licensed senior reactor operator or licensed senior reactor operator limited to fuel handling who has no other concurrent responsibilities during this operation. ITS 5.2.2 does not contain this requirement. This changes the CTS by moving this requirement out of Technical Specifications.

The removal of these requirements from the Technical Specifications Administrative Controls Section is acceptable because this type of information is not necessary to be included in the Technical Specifications in order to provide adequate protection of public health and safety. The ITS retains the requirement to comply with 10 CFR 50.54 which contains these requirements. Also, this change is acceptable because these types of details will be adequately controlled in the TRM. The TRM is controlled under 10 CFR 50.59, which ensures changes are evaluated. This change is designated as a less restrictive removal of detail change because specific unit staffing requirements are being removed from the Technical Specifications.

DISCUSSION OF CHANGES
ITS SECTION 5.2, ORGANIZATION

LESS RESTRICTIVE CHANGES

L01 *(Category 1 – Relaxation of LCO/TS Requirements)* CTS 6.2.2.h requires the Assistant Operations Manager to hold a senior reactor operator license and CTS 6.2.2.i requires the Operations Manager to hold or have held a senior reactor operator license at PTN or another pressurized water reactor (PWR) or have completed the PTN Senior Management Operations Training Course. ITS 5.2.2.d requires either the Assistant Operations Manager or the Operations Manager to hold a senior reactor operator license. This changes the CTS by allowing the flexibility of either the Assistant Operations Manager or the Operations Manager to hold a senior reactor operator license instead of specifically requiring the Assistant Operations Manager to hold a senior reactor operator license.

The purpose of CTS 6.2.2.h and 6.2.2.i is to ensure that the Operations Management, specifically the Assistant Operations Manager, has a senior reactor operator license at PTN. This proposed change will allow flexibility by allowing either the Operations Manager or the Assistant Operations Manager to hold a senior reactor operator license at PTN. This change is acceptable because allowing this flexibility does not affect the operations of the plant. A member of the Operations Management will still have a senior reactor operators license. This change is designated as less restrictive because less stringent requirements are being applied in the ITS than were applied in the CTS.

L02 The CTS Table 6.2-1 lists the minimum shift crew composition and provides a provision stating that the shift crew composition may be one less than the minimum requirements of Table 6.2-1 for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 6.2-1. The provision further states that this provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent. ITS 5.2.2.b provides a similar provision but excludes the restriction that eliminates the use of this provision due to an oncoming shift crewman being late or absent. This changes the CTS by allowing the crew composition to be one less than required due to an oncoming shift crew member being late or absent.

ITS 5.2.2.b specifically addresses the unexpected absence of on-duty shift crewmembers. As the oncoming shift has not yet come on-duty, ITS 5.2.2.b does not apply to oncoming crewmembers being late or absent. ITS 5.2.2.b specifically states that the 2-hour caveat applies to "the unexpected absence of on-duty shift crew members." Therefore, STS 5.2.2.b effectively excludes the absence or tardiness of the oncoming shift from the 2-hour caveat. 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

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 Topical Report ~~Plan]~~, 1

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

6.2.1.d 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:

Table 6.2.1 a. Three Auxiliary Operators shall be assigned at all times with at least one of the required individuals assigned to the Auxiliary Operator position for each unit. ~~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.~~ 3

~~-----REVIEWER'S NOTE-----~~
~~Two unit sites with both units shutdown or defueled require a total of three non-licensed operators for the two units.~~ 2
~~-----~~

5.2 Organization

5.2.2 Unit Staff (continued)

Table 6.2-1

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.d
6.2.2.d Footnote *

Health Physics

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. 3

6.2.2.h

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

Table 6.2-1,
6.2.3.1

With either unit in MODE 1, 2, 3, or 4, an individual shared between Unit 3 and Units 4

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. 3

3

This position may be filled by the SM or an individual with a Senior Operator license who meets the 1985 NRC Policy Statement on Engineering Expertise on Shift.

JUSTIFICATION FOR DEVIATIONS
ITS SECTION 5.2, ORGANIZATION

1. The Improved Standard Technical Specifications (ISTS) contain bracketed information and/or values that are generic to all Westinghouse 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.

2. This "Reviewers Note" is being deleted. The Reviewer's Note is for the NRC reviewer during the NRC review and will not be part of the plant specific Improved Technical Specifications (ITS).

3. 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.

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 Specification.

ATTACHMENT 3

ITS Section 5.3 – Unit Staff Qualifications

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

See ITS 5.2

6.2.3 SHIFT TECHNICAL ADVISOR FUNCTION

6.2.3.1 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 and the opposite unit. This individual shall meet the qualifications specified by the 1985 NRC Policy Statement on Engineering Expertise on Shift.

Unit

5.3 6.3 ~~FACILITY~~ STAFF QUALIFICATIONS

5.3.1 6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 ~~for comparable positions, except for~~ 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. A01

5.3.1.a 6.3.1.1 The Health Physics Supervisor who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975.

5.3.1.b 6.3.1.2 The Operations Manager whose requirement for a Senior Reactor Operator License is as stated in Specification 6.2.2.i.

5.3.1.c 6.3.1.3 The licensed operators, who shall comply only with the requirements of 10 CFR 55.

5.3.1.d 6.3.1.4 The Multi-Discipline Supervisors who shall meet or exceed the following requirements:

5.3.1.d.1 a. Education: Minimum of a high school diploma or equivalent

5.3.1.d.2 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

5.3.1.d.3 c. Training: Complete the Multi-Discipline Supervisor training program

On-site Review Group A02

5.3.3 6.3.2 When the Health Physics Supervisor does not meet the above requirements, compensatory action shall be taken which the ~~Plant Nuclear Safety Committee~~ determines and the NRC office of Nuclear Reactor Regulation concurs that the action meets the intent of Specification 6.3.1.

5.3.2 6.3.3 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 6.3.1.3, perform the functions described in 10 CFR 50.54(m)

~~6.4 DELETED~~

~~6.5 DELETED~~

~~6.6 DELETED~~

~~6.7 DELETED~~

DISCUSSION OF CHANGES
ITS SECTION 5.3, UNIT STAFF QUALIFICATIONS

ADMINISTRATIVE CHANGES

A01 In the conversion of the Turkey Point Nuclear Generating Station (PTN) Unit 3 and Unit 4, 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-Westinghouse Plants" (ISTS).

A02 CTS 6.3.2 states, in part, that compensatory action shall be taken which the Plant Nuclear Safety Committee determines. ITS 5.3.3 states, in part, that that compensatory action shall be taken which the On-Site Review Group determines. This changes the CTS by changing the generic committee's name that determines the compensatory action to the plant specific committee name. This change is considered administrative as no technical changes are made to the CTS.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

None

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

5.0 ADMINISTRATIVE CONTROLS

5.3 Unit Staff Qualifications

~~REVIEWER'S NOTE~~
~~Minimum qualifications for members of the unit staff shall be specified by use of an overall qualification statement referencing an ANSI Standard acceptable to the NRC staff or by specifying individual position qualifications. Generally, the first method is preferable; however, the second method is adaptable to those unit staffs requiring special qualification statements because of unique organizational structures.~~ 1

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~~]~~. 2 3

ANSI N18.1-1971

INSERT 1

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).

5.3.3 When the Health Physics Supervisor does not meet the above requirements, compensatory action shall be taken which the On-Site Review Group determines and the NRC office of Nuclear Reactor Regulation concurs that the action meets the intent of Specification 5.3.1. 3

3 **INSERT 1**

a. The Health Physics Supervisor who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975.

b. The Operations Manager whose requirement for a Senior Reactor Operator License is as stated in Specification 5.2.2.d.

c. The licensed operators, who shall comply only with the requirements of 10 CFR 55.

d. The Multi-Discipline Supervisors who shall meet or exceed the following requirements:

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

   2. 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

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

3 **INSERT 1**

a. The Health Physics Supervisor who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975.

b. The Operations Manager whose requirement for a Senior Reactor Operator License is as stated in Specification 5.2.2.d.

c. The licensed operators, who shall comply only with the requirements of 10 CFR 55.

d. The Multi-Discipline Supervisors who shall meet or exceed the following requirements:

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

   2. 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

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

JUSTIFICATION FOR DEVIATIONS
ITS CHAPTER 5.3, UNIT STAFF QUALIFICATIONS

1. This "Reviewers Note" is being deleted. The Reviewer's Note is for the NRC reviewer during the NRC review and will not be part of the plant specific Improved Technical Specifications (ITS).

2. The Improved Standard Technical Specifications (ISTS) contains bracketed information and/or values that are generic to all Westinghouse 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 ISTS was changed to be consistent with the Turkey Point Nuclear Generating Station (PTN) Unit 3 and Unit 4, Current Technical Specifications (CTS). This change was made to be consistent with the requirements of the Reviewer's Note that stated "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."

4. 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.

Specific No Significant Hazards Considerations (NSHCs)

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

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

ATTACHMENT 4

ITS Section 5.4 – Procedures

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

ITS

A01

ADMINISTRATIVE CONTROLS

5.4

6.8 PROCEDURES ~~AND PROGRAMS~~ A01

5.4.1

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

5.4.1.a a. The applicable procedures required by the Quality Assurance Topical Report.

5.4.1.b b. The emergency operating procedures required to implement the requirements of NUREG-0737 and Supplement 1 to NUREG-0737 as stated in Generic Letter No. 82-33;

~~c. Process Control Program implementation;~~ LA01

5.4.1.d d. Offsite Dose Calculation Manual implementation;

5.4.1.d e. Quality Control Program for effluent monitoring ~~using the guidance in Regulatory Guide 1.21, Revision 1, June 1974~~; LA02

~~f. DELETED~~ A01

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

5.4.1.d h. Diesel Fuel Oil Testing Program implementation. M01

All programs specified in Specification 5.5

6.8.2 DELETED

6.8.3 DELETED

DISCUSSION OF CHANGES
ITS SECTION 5.4, PROCEDURES

ADMINISTRATIVE CHANGES

A01 In the conversion of the Turkey Point Nuclear Generating Station (PTN) Unit 3 and Unit 4, 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-Westinghouse 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 ITS 5.4.1.e 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 and the Diesel Fuel Oil Testing Program. 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.e 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, NQAP, CLRT Program, IST Program, or ISI Program*) CTS 6.8.1.c requires that written procedures for the PROCESS CONTROL PROGRAM (PCP) be established, implemented, and maintained. ITS 5.4.1 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, NQAP, CLRT Program, IST Program, or ISI Program*) CTS 6.8.1.e and g 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.

<u>LESS RESTRICTIVE CHANGES</u>

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

Quality Assurance Topical Report

a. The applicable procedures recommended ~~in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978,~~ 1

6.8.1.b

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

6.8.1.e
6.8.1.g

c. Quality assurance for effluent and environmental monitoring,

~~d. Fire Protection Program implementation,~~ and 3

6.8.1.b

d.

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

JUSTIFICATION FOR DEVIATIONS
ITS CHAPTER 5.4, PROCEDURES

1. Changes are made (additions, deletions, and/or changes) to the Improved Standard Technical Specifications (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 all Westinghouse 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 5.4.1.d is deleted consistent with Turkey Point Nuclear Generating Station (PTN) Unit 3 and Unit 4 Current Licensing Basis (CTS). PTN Units 3 and 4 transitioned to a risk-informed, performance-based Fire Protection Program (FPP) in accordance with National Fire Protection Association Standard NFPA 805 pursuant to 10 CFR 50.48(c). This transition was approved May 28, 2015 as issued in License Amendments 262 and 257, for Unit 3 and Unit 4 respectively. As stated in the NRC Safety Evaluation supporting the license amendments, the administrative requirement that procedures be established, implemented, and maintained for FPP implementation is contained in regulations 10 CFR 50.48(a), 10 CFR 50.48(c), and NFPA 805 Chapter 3. The NRC staff concluded that maintaining a procedure requirement for Fire Protection Program implementation in the 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 Improved Technical Specifications (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 Specification.

ATTACHMENT 5

ITS Section 5.5 – Programs and Manuals

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

FINAL DRAFT 083021PLANT SYSTEMS

3/4.7.5 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION (continued)

See ITS 3.7.10

b. With the Control Room Emergency Ventilation System inoperable due to an inoperable CRE boundary during MODES 1, 2, 3 or 4, immediately initiate action to implement mitigating actions. Within 24 hours, verify mitigating actions ensure CRE occupant radiological and chemical hazards will not exceed limits, and CRE occupants are protected from smoke hazards, and restore CRE boundary to OPERABLE status within 90 days.

With the above requirements not met, be in at least HOT STANDBY within the next 6 hours for one Unit, or 12 hours for both Units, and in COLD SHUTDOWN within the following 30 hours.

With the Control Room Emergency Ventilation System inoperable due to an inoperable CRE boundary during MODES 5, 6 or during the movement of irradiated fuel assemblies, immediately suspend all movement of irradiated fuel.

SURVEILLANCE REQUIREMENTS

4.7.5 The Control Room Emergency Ventilation System shall be demonstrated OPERABLE:

See ITS 3.7.11

a. In accordance with the Surveillance Frequency Control Program by verifying that the control room air temperature is less than or equal to 120°F;

See ITS 3.7.10

b. 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 minutes**;

5.5.8 c. In accordance with the Surveillance Frequency Control Program or (1) after 720 hours of system operation, or (2) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (3) following exposure of the filters to effluents from painting, fire, or chemical release in any ventilation zone communicating with the system that may have an adverse effect on the functional capability of the system, or (4) after complete or partial replacement of a filter bank by:

See ITS 3.7.10

**As the mitigation actions of TS 3.7.5 Action a.5 include the use of the compensatory filtration unit, the unit shall meet the surveillance requirements of TS 4.7.5.b, by manual initiation from outside the control room and TS 4.7.5.c, d and f.

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

5.5.8.a, 5.5.8.b
1) Verifying that the air cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of greater than or equal to 99.95% DOP and 99% halogenated hydrocarbon removal at a system flow rate of 1000 cfm ±10%**.

5.5.8.c
2) Verifying, ~~within 31 days after removal,~~ that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, and analyzed per ASTM D3803 - 1989 at 30°C and 95% relative humidity, meets the methyl iodide penetration criteria of less than 2.5% or the charcoal be replaced with charcoal that meets or exceeds the stated performance requirement**, and LA04

5.5.8.e
3) Verifying by a visual inspection the absence of foreign materials and gasket deterioration**.

5.5.8.d
d.1 In accordance with the Surveillance Frequency Control Program by verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches Water Gauge while operating the system at a flow rate of 1000 cfm ±10%**;

5.5.15.d
d.2 In accordance with the Surveillance Frequency Control Program, test the supply fans (trains A and B) and measure CRE pressure relative to external areas adjacent to the CRE boundary.**

e. In accordance with the Surveillance Frequency Control Program by verifying that on a Containment Phase "A" Isolation test signal the system automatically switches into the recirculation mode of operation,

f. By performing required CRE unfiltered air inleakage testing in accordance with the Control Room Envelope Habitability Program.**

See ITS 3.7.10

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies. A02

5.5.8
**As the mitigation actions of TS 3.7.5 Action a.5 include the use of the compensatory filtration unit, the unit shall meet the surveillance requirements of TS 4.7.5.b, by manual initiation from outside the control room and TS 4.7.5.c, d and f.

See ITS 3.7.10

D.C. SOURCES

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

See ITS 3.8.4

b. With one of the required battery banks inoperable, or with none of the full-capacity chargers associated with a battery bank OPERABLE, restore all battery banks to OPERABLE status and at least one charger associated with each battery bank to OPERABLE status within two hours* or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 12 hours and in COLD SHUTDOWN within the following 30 hours. This ACTION applies to both units simultaneously.

SURVEILLANCE REQUIREMENTS

4.8.2.1 Each 125-volt battery bank and its associated full capacity charger(s) shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that: See ITS 3.8.6

1) The parameters in Table 4.8-2 meet the Category A limits, and

See ITS 3.8.4

2) The total battery terminal voltage is greater than or equal to 129 volts on float charge and the battery charger(s) output voltage is ≥ 129 volts, and

3) If two battery chargers are connected to the battery bank, verify each battery charger is supplying a minimum of 10 amperes, or demonstrate that the battery charger supplying less than 10 amperes will accept and supply the D.C. bus load independent of its associated battery charger.

See ITS 3.8.6

b. In accordance with the Surveillance Frequency Control Program and within 7 days after a battery discharge with battery terminal voltage below 105 volts (108.6 volts for spare battery D-52), or battery overcharge with battery terminal voltage above 143 volts, by verifying that:

1) The parameters in Table 4.8-2 meet the Category B limits,

2) The average electrolyte temperature of every sixth cell is above 60°F, and

5.5.14.d 3) There is no visible corrosion at either terminals or connectors, or verify battery connection resistance is:

Battery	Connection	Limit (Micro-Ohms)
3B, 4A	inter-cell / termination inter-cell (brace locations) transition cables or total battery connections	≤ 29 ≤ 30 ≤ 125 ≤ 1958
Battery 3A, 4B, D-52	Connection inter-cell / termination inter-cell (brace locations) transition cables or total battery connections	Limit (Micro-Ohms) ≤ 35 ≤ 40 ≤ 125 ≤ 2463

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

5.5.14 1) ~~The cells, cell plates, and battery racks show no visual indication of physical damage or abnormal deterioration,~~ LA03

See ITS 3.8.4

*Can be extended to 24 hours if the opposite unit is in MODE 5, 6, or defueled and each of the remaining required battery chargers is capable of being powered from its associated diesel generator(s).

D.C. SOURCES

SURVEILLANCE REQUIREMENTS (Continued)

5.5.14 2) ~~The cell-to-cell and terminal connections are clean, tight, and coated with anticorrosion material,~~ LA03

3) Each 400 amp battery charger (associated with Battery Banks 3A and 4B) will supply at least 400 amperes at ≥ 129 volts for at least 8 hours, and each 300 amp battery charger (associated with Battery Banks 3B and 4A) will supply at least 300 amperes at ≥ 129 volts for at least 8 hours, and

See ITS 3.8.4

4) Battery Connection resistance is:

Battery 3B, 4A	Connection inter-cell / termination inter-cell (brace locations) transition cables or total battery connections	Limit (Micro-Ohms) ≤ 29 ≤ 30 ≤ 125 ≤ 1958
Battery 3A, 4B, D-52	Connection inter-cell / termination inter-cell (brace locations) transition cables or total battery connections	Limit (Micro-Ohms) ≤ 35 ≤ 40 ≤ 125 ≤ 2463

5.5.14.d

See ITS 3.8.4

d. In accordance with the Surveillance Frequency Control Program, during shutdown**, by verifying that the battery capacity is adequate to supply and maintain in OPERABLE status all of the actual or simulated emergency loads for the design duty cycle when the battery is subjected to a battery service test.

See ITS 3.8.6

e. At least once per 12 months, during shutdown**, by giving performance discharge tests of battery capacity to any battery that shows signs of degradation or has reached 85% [75% for Batteries 4B and D52 (Spare) when used in place of Battery 4B] of service life expected for the application. Degradation is indicated when the battery capacity drops more than 10% [7% for Batteries 4B and D52 (Spare) when used in place of Battery 4B] of rated capacity from its average on previous performance tests, or is below 90% [93% for Batteries 4B and D52 (Spare) when used in place of Battery 4B] of the manufacturer's rating.

f. In accordance with the Surveillance Frequency Control Program, during shutdown**, by verifying that the battery capacity is at least 80% [87% for Batteries 4B and D52 (Spare) when used in place of Battery 4B] of the manufacturer's rating when subjected to a performance discharge test. Once per 60-month interval this performance discharge test may be performed in lieu of the battery service test required by Specification 4.8.2.1.d.

See ITS 3.8.4

See ITS 3.8.6

**Except that the spare battery bank D-52, and any other battery out of service when spare battery bank D-52 is in service may be tested with simulated loads during operation.

TABLE 4.8-2

BATTERY SURVEILLANCE REQUIREMENTS

See ITS 3.8.6

	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 < ¼" above maximum level indication mark	>Minimum level indication mark, and < ¼" above maximum level indication mark	Above top of plates, and not overflowing
5.5.14.b Float Voltage	≥ 2.13 volts	≥ 2.13 volts(6)	≥ 2.07 volts
Specific Gravity (4)	≥ 1.200(5)	≥ 1.195	Not more than 0.020 below the average of all connected cells
		Average of all Connected cells > 1.205	Average of all connected cells ≥ 1.195(5)

See ITS 3.8.6

TABLE NOTATIONS

(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 limits within 7 days.

(3) Any Category B parameter not within its allowable value indicates an inoperable battery.

(4) Corrected for electrolyte temperature and level.

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

~~(6) Corrected for average electrolyte temperature.~~

LA01

Add proposed TS 5.5.14, Battery Monitoring and Maintenance Program

M01

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 Safety Injection System, Chemical and Volume Control System, and the Containment Spray System. The program shall include the following:

(1) Preventive maintenance and periodic visual inspection requirements, and

(2) Integrated leak test requirements for each system ~~at every 18 months~~ In accordance with the Surveillance Frequency Control Program. LA02

The provisions of Specification 4.0.2 are applicable.

~~b. DELETED~~

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:

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

(2) Identification of the procedures used to measure the values of the critical variables,

ADMINISTRATIVE CONTROLS

PROCEDURES AND PROGRAMS (Continued)

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

(4) Procedures for the recording and management of data,

(5) Procedures defining corrective actions for all off-control point chemistry conditions, and

(6) 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.~~ ~~DELETED~~

5.5.10 e. Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:

1. An API Gravity or an absolute specific gravity within limits,

2. a flash point and kinematic viscosity within limits for Grade No. 2-D fuel oil per ASTM D975, and

3. a clear and bright appearance with proper color, or a water and sediment content within limits L02

b. Other properties for Grade No. 2-D fuel oil per ASTM D975 are within limits within 30 days following sampling and addition to storage tanks; and

c. Total particulate concentration of the fuel oil is ≤ 10 mg/liter when tested ~~every 31 days~~ in accordance with the Surveillance Frequency Control Program LA02 in accordance with either ASTM D-2276 or ASTM D-5452.

Add proposed TS 5.5.4, Pre-Stressed Concrete Containment Tendon Surveillance Program M02

Add proposed TS 5.5.8, Ventilation Filter Testing Program (VFTP) M03

5.5.3 f. Radioactive Effluent Controls Program

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

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;

2. 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;

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;

4. 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;

5. Determination of cumulative dose 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.

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

7. 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:

    a. For noble gases: a dose rate less or equal to 500 mrems/yr to the whole body and a dose rate less than or equal to 3000 mrems/yr to the skin, and

    b. For iodine 131, iodine 133 tritium and all radionuclides in particulate form with half live greater than 8 days: a dose rate less than or equal to 1500 mrems/yr to any organ.

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 10 CFR §50, Appendix I;

PROCEDURES AND PROGRAMS (Continued)

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 10 CFR 50, Appendix I;

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 190.

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

~~g. DELETED~~

5.5.13 h. Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, and as modified by approved exemptions. This program shall be in accordance with Nuclear Energy Institute (NEI) 94-01, Revision 3-A, "Industry Guidance for Implementing Performance Based Option of 10 CFR 50 Appendix J," and the conditions and limitations specified in NEI 94-01, Revision 2-A, with the following deviations or exemptions:

1) A vacuum test will be performed in lieu of a pressure test for airlock door seals at the required intervals (Amendment Nos. 73 and 77, issued by NRC November 11, 1981).

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is defined here as the containment design pressure of 55 psig.

The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.20% of containment air weight per day.

Leakage Rate acceptance criteria are:

1) The As-found containment leakage rate acceptance criterion is ≤ 1.0 L_a. Prior to increasing primary coolant temperature above 200°F following testing in accordance with this program or restoration from exceeding 1.0 L_a, the As-left leakage rate acceptance criterion is ≤ 0.75 L_a, for Type A test.

2) The combined leakage rate for all penetrations subject to Type B or Type C testing is as follows:

ADMINISTRATIVE CONTROLS

PROCEDURES AND PROGRAMS (Continued)

- The combined As-left leakage rates determined on a maximum pathway leakage rate basis for all penetrations shall be verified to be less than 0.60 L_a, prior to increasing primary coolant temperature above 200°F following an outage or shutdown that included Type B and Type C testing only.

- The As-found leakage rates, determined on a minimum pathway leakage rate basis, for all newly tested penetrations when summed with the As-left minimum pathway leakage rate leakage rates for all other penetrations shall be less than 0.6 L_a, at all times when containment integrity is required.

3) Overall air lock leakage acceptance criteria is ≤ 0.05 L_a, when pressurized to P_a.

~~The provisions of Specification 4.0.2 do not apply to the test frequencies contained within the Containment Leakage Rate Testing Program.~~ Add ITS 5.5.13.e A08

5.5.11 i. Technical Specifications (TS) Bases Control Program

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

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

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

1. Change in the TS incorporated in the license or

2. A change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

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

d. Proposed changes that meet the criteria of Specification 6.8.4 i.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).

5.5.6 j. Steam Generator (SG) Program

A01

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

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

PROCEDURES AND PROGRAMS (Continued)

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

SG A01

1. Structural integrity performance criterion: All in-service ~~steam generator~~ tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cooldown), 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 0.60 gpm total through all SGs and 0.20 gpm through any one SG at room temperature conditions.

3. The operational leakage performance criterion is specified in LCO 3.4.6.2, “Reactor Coolant System Operational Leakage.”

c. Provisions for SG tube plugging 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.

The following alternate tube plugging criteria shall be applied as an alternative to the 40% depth based criteria:

1. Tubes with service-induced flaws located greater than 18.11 inches below the top of the tubesheet do not require plugging. Tubes with service-induced flaws located in the portion of the tube from the top of the tubesheet to 18.11 inches below the top of the tubesheet shall be plugged upon detection.

ADMINISTRATIVE CONTROLS

PROCEDURES AND PROGRAMS (Continued)

A03

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 criteria. ~~The portion of the tube below 18.11 inches from the top of the tubesheet is excluded from inspection.~~ 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 tube may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

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

INSERT 1 L01

2. ~~After the first refueling outage following SG installation, inspect each SG at least every 48 effective full power months or at least every other refueling outage (whichever results in more frequent inspections)*. In addition, the minimum number of tubes inspected at each scheduled inspection shall be the number of tubes in all SGs divided by the number of SG inspection outages scheduled in each inspection period as defined in a, b, and c below. If degradation assessment indicates the potential for a type of degradation to occur at a location not previously inspected with technique capable of detecting this type of degradation at this location and that may satisfy the applicable tube plugging criteria, the minimum number of locations inspected with such a capable inspection technique during the remainder of the inspection period may be prorated. The fraction of locations to be inspected for this potential type of degradation at this location at the end of the inspection period shall be no less than the ratio of the number of times the SG is scheduled to be inspected in the inspection period after the determination that a new form of degradation could potentially be occurring at this location divided by the total number of times the SG is scheduled to be inspected in the inspection period. Each inspection period defined below may be extended up to 3 effective full power months to include a SG inspection outage in an inspection period and the subsequent inspection period beings at the conclusion of the included SG inspection outage.~~

~~a) After the first refueling outage following SG installation, inspect 100% of the tubes during the next 120 effective full power months. This constitutes the first inspection period;~~

~~b) During the next 96 effective full power months, inspect 100% of the tubes. This constitutes the second inspection period; and~~

~~c) During the remaining life of the SGs, inspect 100% of the tubes every 72 effective full power months**. This constitutes the third and subsequent inspection periods.~~

A04

~~* One-time extension for Unit 3 to perform SG inspections during the Cycle 32 refueling outage in Fall 2021.~~

~~** One-time extension of the 4th inspection period for Unit 3 until the Cycle 32 refueling outage in Fall 2021.~~

L01

INSERT 1

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.

PROCEDURES AND PROGRAMS (Continued)

A01 excluding any region that is exempt from inspection by alternate repair criteria

3. If crack indications are found in any ~~portion of a~~ SG tube ~~not excluded above~~, 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 refueling outage (whichever results in more frequent inspections)~~. 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.

be at the next refueling outage, but may be deferred to the following refueling outage if the 100% inspection of all SGs was performed with enhanced probes as described in paragraph d.2. L01

e. Provisions for monitoring operational primary-secondary leakage.

5.5.15 k. 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.

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.

A05

d. Measurement, at designated locations, of the 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 ~~Surveillance Requirement 4.7.5.d, at a Frequency of 18 months~~. Additionally, the supply fans (trains A and B) will be tested ~~on a staggered test basis (defined in Technical Specification definition 1.29 every 36 months)~~. The results shall be trended and the CRE boundary assessed ~~every 18 months.~~

the VFTP,

at a Frequency in accordance with the Surveillance Frequency Control Program

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.

LA02

f. The provisions of Specification 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.

PROCEDURES AND PROGRAMS (Continued)

5.5.16 l. 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 Operations are met:

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

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

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

m. 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:

See ITS 3.0

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

b. 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 Vessel (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(a) subject to the use and conditions on the use of standards listed in 10 CFR 50.55a(b) and subject to Commission approval.

c. 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".

d. 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.

5.5.5 n. Reactor Coolant Pump Flywheel Inspection Program

Each Reactor Coolant Pump flywheel shall be inspected at least once every 20 years by either conducting an in-place ultrasonic examination over the volume from the inner bore of the flywheel to the circle of one-half the outer radius, or by conducting a surface examination (magnetic particle and/or liquid penetrant) of exposed surfaces of the disassembled flywheel.

PROCEDURES AND PROGRAMS (Continued)

5.5.9 o. Gas Decay Tank Explosive Gas and Radioactivity Monitoring Program

This Program provides controls for potentially explosive gas mixtures and the quantity of radioactivity contained in the Gas Decay 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 Program shall include:

1. The limits for concentrations of hydrogen and oxygen in the Gas Decay Tanks and a surveillance program to ensure that 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. A surveillance program to ensure that the quantity of radioactivity contained in each Gas Decay Tank 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.

The provision of SR 4.0.2 and SR 4.0.3 are applicable to the Gas Decay Tank Explosive Gas and Radioactivity Monitoring Program surveillance frequencies.

PROCEDURES AND PROGRAMS (Continued)

5.5.17 p. 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, “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 Risk Informed Completion Time 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.

ADMINISTRATIVE CONTROLS

PROCEDURES AND PROGRAMS (Continued)

5.5.12 q. Safety Function Determination Program (SFDP)

A06

limitations and remedial or compensatory

Condition and Required Actions

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.7, 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 ~~ACTIONS~~. This program implements the requirements of LCO 3.0.7. The SFDP shall contain the following:

A07

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,

Completion Time

A07

c. Provisions to ensure that an inoperable supported system's ~~allowed outage 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.

Conditions and Required Actions

A06

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

A06

6.8.5 DELETED

5.5.1

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.14.1 The ODCM shall contain the following:

a. 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 radioactive effluent controls and radiological environmental monitoring activities, and descriptions of the information that should be included in the Annual Radiological Environmental Operating, and Annual Radioactive Effluent Release Reports required by Specification 6.9.1.3 and Specification 6.9.1.4.

6.14.2 Licensee initiated changes to the ODCM:

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

1. Sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s), 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 Appendix I to 10 CFR 50, and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

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

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 Annual 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.

DISCUSSION OF CHANGES
ITS 5.5, PROGRAMS AND MANUALS

ADMINISTRATIVE CHANGES

A01 In the conversion of the Turkey Point Nuclear Generating Station (PTN) 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-1431, Rev. 5.0, "Standard Technical Specifications - Westinghouse Plants" (ISTS) and additional Technical Specification Task Force (TSTF) travelers included in this submittal.

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

A02 The Surveillances associated with the ventilation filter testing for the Control Room Emergency Ventilation System (CREVS) has been placed in a program in the proposed Administrative Controls Chapter 5.0 (ITS 5.5.8). As such, a general program statement has been added as ITS 5.5.8. A statement of the applicability of ITS Surveillance Requirement (SR) 3.0.2 and SR 3.0.3 is needed to clarify that the allowances for Surveillance Frequency extensions do apply (as allowed in the CTS). This changes the CTS by moving the ventilation filter testing Surveillances associated with the CREVS to a program in ITS 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 Surveillances. The addition of the ITS SR 3.0.2 and SR 3.0.3 statement is a clarification needed to maintain provisions that are currently allowed in the CTS; therefore, it is considered acceptable. This change is designated as administrative because it does not result in technical changes to the CTS.

A03 CTS 6.8.4.j.d states, in part, that the Steam Generator (SG) inspection objective is 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 criteria. That the portion of the tube below 18.11 inches from the top of the tubesheet is excluded from inspection. ITS 5.5.6.d states, in part, that the SG inspection objective is 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 criteria. This changes the CTS by replacing the specific dimension of the portion of the tube excluded from the inspection, which is listed in paragraph c.1, with a statement that excludes that part of the tube exempt from inspection by alternate repair criteria. This change is designated as administrative because it does not result in technical changes to the CTS.

A04 CTS 6.8.4.j.d.2 includes two footnotes (footnote * and footnote **) that support a one-time extension of the 4^{th} inspection period and a one-time extension to perform SG inspections for Unit 3 and expires after the Cycle 32 refueling outage in Fall of 2021. ITS 5.5.6, "Steam Generator Program," does not include these one-time extensions. This changes the CTS by deleting the allowance for these one-time extensions.

Both footnotes that modify the Frequency of the Unit 3 4th inspection period and the Unit 3 SG inspections are one-time extension that expire after the fall 2021 outage. Because the approval of this license amendment request will be after the one-time extensions have expired, the extensions are no longer in effect. This change is designated as administrative because it does not result in technical changes to the CTS.

A05 CTS 6.8.4.k.d requires, in part, that the Control Room Envelope Habitability Program include 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 SR 4.7.5.d. ITS 5.5.15.d requires, in part, that the Control Room Envelope Habitability Program include measurement, at designated locations, of the 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 Ventilation Filter Test Program (VFTP). The CTS 4.7.5.d flow rate of 1000 ± 10 cfm is being placed in the new VFTP; therefore, the flow rate value is not changed. This change is designated as administrative because it does not result in technical changes to the CTS.

A06 CTS 6.8.4.q, Safety Function Determination Program (SFDP) states that this program 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.

A07 The CTS format is to label remedial actions with their associated prescribed period for completion as "ACTION" and "allowed outage time." ITS format is to label these same remedial action and prescribed completion periods as "Condition and Required Action" and "Completion Time." This changes the CTS by providing a new label for the remedial actions and associated period for completion.

The change in the labeling 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.

A08 CTS 6.8.4.h specifies the requirements for the Containment Leakage Rate Testing Program. CTS 6.8.4.h states, in part, that Specification 4.0.2 is not applicable to this program. ITS 5.5.13, "Containment Leakage Rate Testing Program," does not contain this statement. Furthermore, ITS 5.5.13.e states, "The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate," which is not stated in CTS 6.8.4.h. This changes the CTS by removing the CTS statement regarding Specification 4.0.2 and adding the allowances of ITS SR 3.0.3 to the Technical Specification Containment Leakage Rate Testing Program.

CTS Specification 4.0.2 allows extension of an SR Frequency under certain conditions. ITS SR 3.0.3 provides guidance with respect to missed SRs. Section 3.0 contains guidance associated with Limiting Condition for Operation (LCO) Applicability and SR Applicability. The Applicability statements in Section 3.0 are applicable to all LCOs and SRs unless otherwise stated within a specific LCO or SR. The Containment Leakage Rate Testing Program is required per ITS SR 3.6.1.1; therefore, all LCO and SR Applicabilities within Section 3.0 of the Technical Specifications may be applied to this Section 5.5 program. Subsequently, the removal of the CTS 6.8.4.h Specification 4.0.2 exception is appropriate and consistent with the ITS. Given that SR 3.6.1.1 requires inspections and leak testing in accordance with the Containment Leakage Rate Testing Program and Section 3.0 is applicable to this SR, it is not necessary to retain the ITS 5.5.13.e SR 3.0.3 allowance. However, to maintain consistency with the ITS, the SR 3.0.3 allowance is being retained. Since this change is a clarification intended to maintain provisions that would be allowed in the LCO sections of the Technical Specifications, it is considered administrative in nature. This change is designated as administrative because it does not result in a technical change to the CTS.

MORE RESTRICTIVE CHANGES

M01 The 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. 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 (SE) for plant-specific adoption of TSTF–500, Revision 2 (NRC ADAMS Accession No. ML111751792). PTN has verified the applicable information specified in Section 2.2 of the TSTF-500 model application, including applicable Updated Final Safety Analysis Report (UFSAR) information. PTN 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 PTN Unit 3 and Unit 4 UFSAR. 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.

M02 The CTS does not include a requirement for the Pre-Stressed Concrete Containment Tendon Surveillance Program. The ITS includes a requirement for this program. This changes the CTS by adding the ITS 5.5.4, "Pre-Stressed Concrete Containment Tendon Surveillance Program."

The Pre-Stressed Concrete Containment Tendon Surveillance Program is included to provide controls for monitoring any tendon degradation in pre-stressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The specific wording associated with this program may be found in ITS 5.5.4. 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 The CTS does not include a requirement for the VFTP. The ITS includes a requirement for this program. This changes the CTS by adding the ITS 5.5.8, "Ventilation Filter Testing Program."

The VFTP is included to implement the required testing of Engineered Safety Feature (ESF) filter ventilation systems. The specific wording associated with this program may be found in ITS 5.5.8. 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.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 3 – Removing Procedural Details for meeting TS Requirements or Reporting Requirements)* CTS Table 4.8.2 footnote (6) states, in part the float voltage of ≥ 2.13 volts is corrected for average electrolyte temperature. ITS 5.5.14 b.1 requires a program with actions to restore battery cells with float voltage < 2.13 V and ITS 5.5.14 b.2 requires a program with actions to determine whether the float voltage of the remaining battery cells is ≥ 2.13 V when the float voltage of a battery cells has been found to be < 2.13 V. This changes the CTS by moving 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 still retains the requirement for float voltage ≥ 2.13 V. Also, this change is acceptable because these types of procedural details will be adequately controlled by the requirements of a program required by ITS Chapter 5. ITS 5.5.14, "Battery Monitoring and Maintenance Program," is controlled by Chapter 5 of the Technical Specifications. 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.

LA02 *(Type 5 – Removal of SR Frequency to the Surveillance Frequency Control Program)* CTS 6.8.4.a requires that the Primary Coolant Sources Outside Containment program include integrated leak test requirements for each system at every 18 months. ITS 5.5.2 contains a similar requirement but specifies the periodic Frequency as "In accordance with the Surveillance Frequency Control Program." CTS 6.8.4.e requires that total particulate concentration of the fuel oil is ≤ 10 mg/liter when tested every 31 days. ITS 5.5.10 also requires that total particulate concentration of the fuel oil is ≤ 10 mg/l when tested in accordance with the Surveillance Frequency Control Program (SFCP). In addition, 6.8.4.k requires measurement, at designated locations, of the 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 Surveillance

Requirement 4.7.5.d, at a Frequency of 18 months and that the supply fans (trains A and B) will be tested on a staggered test basis (defined in Technical Specification definition 1.29 every 36 months). TS 5.5.15 similarly requires measurement, at designated locations, of the CRE pressure relative to all 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 in accordance with the SFCP and that the supply fans (trains A and B) will be tested at a Frequency in accordance with the SFCP. This changes the CTS by moving the specified periodic Frequency for the aforementioned tests to the SFCP.

The removal of these details related to test 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 test requirements remain in the Technical Specifications. The control of changes to the test Frequencies is in accordance with the SFCP. The SFCP 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. This change is designated as a less restrictive removal of detail change, because the test Frequencies are being removed from the Technical Specifications.

LA03 *(Type 3 – Removing Procedural Details for meeting TS Requirements or Reporting Requirements)* CTS SR 4.8.2.1.c states, in part, that the battery cells, cell plates, and battery racks show no visual indication of physical damage or abnormal deterioration, and that the cell-to-cell and terminal connections are clean, tight, and coated with anticorrosion material. ITS 5.5.14, "Battery Monitoring and Maintenance Program," requires the program be maintained in accordance with IEEE Standard (Std) 450-2010. This changes the CTS by removing the 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. The Battery Monitoring and Maintenance Program is established in accordance with the requirements contained in IEEE 450-2010. The IEEE standard contains the detail associated with battery preventative maintenance, including visual inspections to detect degradation. The requirements to maintain the station batteries in accordance with the standard remain in the Technical Specifications. This change is designated as a less restrictive removal of detail change, because the battery visual inspection details are being removed from the Technical Specifications.

LA04 *(Type 3 – Removing Procedural Details for meeting TS Requirements or Reporting Requirements)* CTS 4.7.5.c.2 requires that, within 31 days after removal of a carbon sample, the laboratory analysis results are shown to be within limit. ITS 5.5.8.c requires the same analysis to be performed; however, the detail of "within 31 days" after removal of a carbon sample is not included. This changes the CTS by moving these procedural details from the Technical Specifications to the Technical Requirements Manual (TRM).

The removal of these details for performing testing activities 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 to perform the testing at the appropriate Frequencies. Also, this change is acceptable because these types of procedural details will be adequately controlled in the TRM. Any changes to the TRM are made 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>

L01 *(Category 7 – Relaxation of Surveillance Frequency)* CTS 6.8.4.j.d. provides the provisions for SG tube inspections. CTS 6.8.4.j.d.2 requires, in part, each SG at least every 48 effective full power months or at least every other refueling outage (whichever results in more frequent inspections). In addition, the minimum number of tubes inspected at each scheduled inspection shall be the number of tubes in all SGs divided by the number of SG inspection outages scheduled in each inspection period as defined in CTS 6.8.4.j.d.2 parts a, b, and c. ITS 5.5.6.d.2 requires inspection of 100% of the tubes in each SG at least every 54 effective full power months (EFPMs) which defines the inspection period. ITS 5.5.6.d.2 also allows an exemption that 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 mat be extended to 72 EFPMs and provides requirements for the use of enhanced probes. CTS 6.8.4.j.d.3 requires, in part, that if cracks are found the next inspection for each affected and potentially affected SG for the degradation mechanism that caused the crack indication shall not exceed 24 EFPMs or one refueling outage (whichever results in more frequent inspections). ITS 5.5.6.d.3 requires the additional inspection on each affected and potentially affected SG at the next refueling outage, but may be deferred to the following refueling outage if the 100% inspection of all SGs was performed with enhanced probes as described in paragraph d.2. 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 54 EFPMs and modifying the inspection frequency when crack indications are discovered to the next refueling outage or the following.

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 SG 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." PTN Units 3 and 4 SGs contain Alloy 600 thermal treated

(TT) tubes. For Alloy 600TT tubing, TSTF-577, which is incorporated in Revision 5 of the ISTS, revised the frequencies related to inspection of the tubes such that both the maximum time between inspections and the time to inspect 100 percent of the tubes be 54 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 or the following. The nuclear industry's Steam Generator Task Force presented a technical basis supporting the 54 EFPM Alloy 600TT 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.

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). PTN 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 PTN and justify this change.

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

L02 *(Category 6 – Relaxation of Surveillance Requirement Acceptance Criteria)* CTS 6.8.4.e requires performance of the "clear and bright" test, used to establish the acceptability of new fuel oil for use prior to addition to storage tanks. ITS 5.5.10.a.3 requires a determination that the fuel oil has a clear and bright appearance with proper color or that water and sediment content is within limits. This changes the CTS by allowing a "water and sediment content" test to be performed to establish the acceptability of new fuel oil instead of only allowing a "clear and bright" test.

CTS 6.8.4.e requires performance of the "clear and bright" test, to establish the acceptability of new fuel oil for use prior to addition to storage tanks. ITS 5.5.10.a.3 is proposed to be expanded to allow a water and sediment content test to be performed to establish the acceptability of new fuel oil instead of the "clear and bright" test. The "clear and bright" test is a qualitative test for determining free water and particulate contamination in distillate fuels and is, therefore, subject to human interpretation. For example, if an attempt is made to use the qualitative "clear and bright" test with darker colored fuels (e.g., for high sulfur fuel oil that has been dyed in accordance with Environmental Protection Agency (EPA) mandated requirements), the presence of free water or particulate could be obscured and missed by the viewer. Therefore, ITS 5.5.10.a.3 has been expanded to allow a water and sediment content test. The water and sediment content test is a quantitative test using centrifuge methods. In ASTM D975, ASTM D2709, "Standard Method for Water and Sediment in Middle Distillate Fuels by Centrifuge," is an acceptable standard for the water and sediment content test. Therefore, since ASTM D2709 is currently used to verify the acceptability of new fuel oil for use after addition to the storage tanks, the use of these quantitative methods (i.e., water and sediment content) in lieu of ASTM D4176 (i.e., "clear and bright" test) does not introduce a different method for determining the acceptability of new fuel oil. This change is designated as less restrictive because test acceptance criteria required in the CTS will have alternative acceptance criteria allowed in ITS.

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.

6.14 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~~]~~. 1

Licensee initiated changes to the ODCM:

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

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

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

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

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The systems include ~~[Recirculation~~ Containment Spray, Safety Injection, and Chemical and Volume Control~~, gas stripper, and Hydrogen Recombiner]~~ Systems. 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.~~ In accordance with the Surveillance Frequency Control Program 3

The provisions of SR 3.0.2 are applicable.

~~[5.5.3 Post Accident Sampling~~ 4

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

~~This program may be eliminated based on the implementation of WCAP-14986, Rev. 1, "Post Accident Sampling System Requirements: A Technical Basis," and the associated NRC Safety Evaluation dated June 14, 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)~~

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

~~a. Training of personnel,~~

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

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

4

6.8.4.f 5.5.~~4~~ 3 Radioactive Effluent Controls Program

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

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

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

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

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.

~~5.5.5~~ ~~Component Cyclic or Transient Limit~~

~~This program provides controls to track the FSAR, Section [ ], cyclic and transient occurrences to ensure that components are maintained within the design limits.~~ 4

DOC M02 ~~[~~ 5.5.~~6~~ 4 Pre-Stressed Concrete Containment Tendon Surveillance Program 1

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 10 CFR 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. ~~]~~ 1

5.5 Programs and Manuals

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

INSERT 1

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

~~In lieu of Position C.4.b(1) and C.4.b(2), a qualified in-place UT examination over the volume from the inner bore of the flywheel to the circle one-half of the outer radius or a surface examination (MT and/or PT) of exposed surfaces of the removed flywheels may be conducted at 20 year intervals.~~ 2

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

~~The inspection interval and scope for RCP flywheels stated above can be applied to plants that satisfy the requirements in WCAP-15666, "Extension of Reactor Coolant Pump Motor Flywheel Examination."~~ 5

~~----------~~

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

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:

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. 1 1

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 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

2

INSERT 1

Each Reactor Coolant Pump flywheel shall be inspected at least once every 20 years by either conducting an in-place ultrasonic examination over the volume from the inner bore of the flywheel to the circle of one-half the outer radius, or by conducting a surface examination (magnetic particle and/or liquid penetrant) of exposed surfaces of the disassembled flywheel.

5.5 Programs and Manuals

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

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 0.60 gpm total through all SGs and 0.20 gpm through any one SG at room temperature conditions. ~~[1 gpm] per SG [, except for specific types of degradation at specific locations as described in paragraph c of the SG Program.]~~ 1

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

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]~~. 1 1

~~-----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.~~
~~-----~~ 5

~~[~~The following alternate tube plugging ~~[or repair]~~ criteria may be applied as an alternative to the 40% depth based criteria: 1

1. . . ~~.]~~ INSERT 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.~~
~~-----~~ 5

2 **INSERT 2**

Tubes with service-induced flaws located greater than 18.11 inches below the top of the tubesheet do not require plugging. Tubes with service-induced flaws located in the portion of the tube from the top of the tubesheet to 18.11 inches below the top of the tubesheet shall be plugged upon detection.

5.5 Programs and Manuals

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

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. 4 1

~~-----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.~~ 5
~~-----~~

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 steam SG at least every 24 effective full power months, which defines the inspection period.]~~ 1

~~[~~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 mat be extended to 72 effective full power months. Enhanced probes have a capability to detect flaws of ant 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 1

5.5 Programs and Manuals

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

be capable of detecting all forms of existing and potential degradation in that region.~~]~~ 1

~~[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.]~~ 1

~~-----REVIEWER'S NOTE-----~~
~~The bracketed phrases in Paragraph 3 are only applicable to SGs with Alloy 600 thermally treated tubing.~~
~~----------~~ 5

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 outage if 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. 1 1 1

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.~~ 1

~~-----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.~~
~~----------~~ 5

~~1. . . .]~~ 1

5.5 Programs and Manuals

6.8.4.c 5.5.~~9~~ 7 <u>Secondary Water Chemistry Program</u> 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: 2

a. Identification of a sampling schedule for the critical variables and control points for these variables,

b. Identification of the procedures used to measure the values of the critical variables,

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

d. Procedures for the recording and management of data,

e. Procedures defining corrective actions for all off control point chemistry conditions, and

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

CTS

5.5 Programs and Manuals

DOC M03 5.5.~~10~~ 8 Ventilation Filter Testing Program (VFTP) 4

INSERT 3

4.7.5.c 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 Position C.6.b of ~~[~~Regulatory Guide 1.52, Revision 2~~, ASME N510-1989, and AG-1].~~ and ASTM D3803-1989 3 2 1

a. Demonstrate for ~~each of the ESF~~ ventilation 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%~~]~~. 2 1

4.7.5.c.1

~~ESF~~ Ventilation System	Flowrate
~~[ ]~~ Control Room Emergency Ventilation System Control Room Emergency Filtration System	~~[ ]~~ 1000 cfm 1000 cfm

2 1

b. Demonstrate for ~~each of the ESF~~ ventilation systems that an inplace test of the charcoal adsorber shows a penetration and system bypass < ~~[0.05]~~ 1% when tested in accordance with ~~[Regulatory Guide 1.52, Revision 2, and ASME N510-1989]~~ ASTM D3803-1989 at the system flowrate specified below ~~[~~± 10%~~]~~. 2 1

4.7.5.c.1

~~ESF~~ Ventilation System	Flowrate
~~[ ]~~ Control Room Emergency Ventilation System Control Room Emergency Filtration System	~~[ ]~~ 1000 cfm 1000 cfm

4.7.5.c.2

c. Demonstrate for ~~each of the ESF systems~~ Control Room Emergency Ventilation System and Control Room Emergency Filtration System that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Position C.6.b of ~~[~~Regulatory Guide 1.52, Revision 2~~]~~, 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 relative humidity ~~specified below~~ of 95%. 2 1

~~ESF Ventilation System~~	~~Penetration~~	~~RH~~	~~Face Velocity (fps)~~
~~[ ]~~	~~[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.~~ 5

INSERT 3

at a Frequency in accordance with the Surveillance Frequency Control Program or (1) after 720 hours of system operation, or (2) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (3) following exposure of the filters to effluents from painting, fire, or chemical release in any ventilation zone communicating with the system that may have an adverse effect on the functional capability of the system, or (4) after complete or partial replacement of a filter bank, 3 2

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.~~ 5

~~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.

4.7.5.d.1 d. Demonstrate for ~~each of the ESF~~ ventilation 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%~~]~~. 2 1

~~ESF~~ Ventilation System	Delta P	Flowrate
~~[ ]~~ Control Room Emergency Ventilation System Control Room Emergency Filtration System	~~[ ]~~ 6" w.g 6" w.g	~~[ ]~~ 1000 1000

2

4.7.5.c.3 ~~[~~ e. ~~Demonstrate that the heaters for each of the ESF systems dissipate the value specified below [± 10%]~~ Verifying by a visual inspection the absence of foreign materials and gasket deterioration. ~~when tested in accordance with [ASME N510-1989].~~ 1

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

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

5.5 Programs and Manuals

6.8.4.o 5.5.~~11~~ 9 Gas Decay Tank Explosive Gas and ~~Storage Tank~~ Radioactivity Monitoring Program 4 2

This program provides controls for potentially explosive gas mixtures and the quantity of radioactivity contained in the Gas Decay Tanks ~~[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 1 1

The program shall include:

a. The limits for concentrations of hydrogen and oxygen in the Gas Decay Tanks ~~[Waste Gas Holdup System]~~ and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion), 1

b. A surveillance program to ensure that the quantity of radioactivity contained in Gas Decay Tank ~~[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~~ 1 1

~~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~~. 2

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Gas Decay Tank Explosive Gas and ~~Storage Tank~~ Radioactivity Monitoring Program surveillance frequencies. 2

5.5 Programs and Manuals

6.8.4.e 5.5.~~12~~10 Diesel Fuel Oil Testing Program 4

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:

1. An API gravity or an absolute specific gravity within limits,

2. A flash point and kinematic viscosity within limits for ASTM 2D fuel oil per ASTM D975, and 2

3. A clear and bright appearance with proper color or a water and sediment content within limits.

b. Within ~~31~~ 30 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 per ASTM D975, and 2 2

c. Total particulate concentration of the fuel oil is ≤ 10 mg/l when tested ~~every 31 days~~ in accordance with the Surveillance Frequency Control Program in accordance with either ASTM D-2276 or ASTM D-5452. 3 2

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.i 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.

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

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

1. A change in the TS incorporated in the license or

2. A change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

5.5 Programs and Manuals

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

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

d. Proposed changes that meet the criteria of Specification 5.5.~~13b~~ 11b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e). 4

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

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

5.5 Programs and Manuals

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

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

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. 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, La, at Pa, shall be [ ]% of containment air weight per day.~~

~~c. Leakage rate acceptance criteria are:~~ 6

~~1. Containment leakage rate acceptance criterion is ≤ 1.0 La. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are < 0.60 La for the Type B and C tests and < 0.75 La for Type A tests.~~

~~2. Air lock testing acceptance criteria are:~~

~~a) Overall air lock leakage rate is ≤ [0.05 La] when tested at ≥ Pa.~~

~~b) For each door, leakage rate is ≤ [0.01 La] when pressurized to [≥ 10 psig].~~

~~d. The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.~~

~~e. Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J.~~

5.5 Programs and Manuals

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

~~[OPTION B]~~ 1

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:

Nuclear Energy Institute (NEI) 94-01, Revision 3-A, "Industry Guidance for Implementing Performance Based Option of 10 CFR 50 Appendix J," and the conditions and limitations specified in NEI 94-01, Revision 2-A, with the following deviations or

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~~. 2

A vacuum test will be performed in lieu of a pressure test for airlock door seals at the required intervals (Amendment Nos. 73 and 77, issued by NRC November 11, 1981).

[3. . . .] 1

b. The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a, is ~~[45 psig]. The~~ defined here as the containment design pressure ~~is~~ of ~~[50~~ 55 psig~~]~~. 2 1

c. The maximum allowable containment leakage rate, L_a, at P_a, shall be ~~[ ]~~ 0.20 % of containment air weight per day. 1

d. Leakage rate acceptance criteria are:

INSERT 4

~~1.~~ ~~Containment leakage rate acceptance criterion is 1.0 La. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are < 0.60 La for the Type B and C tests and ≤ 0.75 La for Type A tests.~~

~~2.~~ ~~Air lock testing acceptance criteria are:~~

~~a)~~ ~~Overall air lock leakage rate is≤ [0.05 La] when tested at ≥ Pa~~.

~~b)~~ ~~For each door, leakage rate is ≤ [0.01 La] when pressurized to [≥ 10 psig].~~ 2

2 **INSERT 4**

1) The As-found containment leakage rate acceptance criterion is ≤ 1.0 L_a. Prior to increasing primary coolant temperature above 200°F following testing in accordance with this program or restoration from exceeding 1.0 L_a, the As-left leakage rate acceptance criterion is ≤ 0.75 L_a, for Type A test.

2) The combined leakage rate for all penetrations subject to Type B or Type C testing is as follows:

   - The combined As-left leakage rates determined on a maximum pathway leakage rate basis for all penetrations shall be verified to be less than 0.60 L_a, prior to increasing primary coolant temperature above 200°F following an outage or shutdown that included Type B and Type C testing only.

   - The As-found leakage rates, determined on a minimum pathway leakage rate basis, for all newly tested penetrations when summed with the As-left minimum pathway leakage rate leakage rates for all other penetrations shall be less than 0.6 L_a, at all times when containment integrity is required.

3) Overall air lock leakage acceptance criterion is ≤ 0.05 L_a, when pressurized to P_a.

5.5 Programs and Manuals

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

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.~~ 2

~~[OPTION A/B Combined]~~

~~a. A program shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J. [Type A][Type B and C] test requirements are in accordance with 10 CFR 50, Appendix J, Option A, as modified by approved exemptions. [Type B and C][Type A] test requirements are in accordance with 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. The 10 CFR 50, Appendix J, Option B test requirements shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September, 1995, as modified by the following exceptions:~~

~~1. The visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWL, except where relief has been authorized by the NRC.~~

~~2. The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 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.~~ 6

~~[3. . . .]~~

~~b. The calculated peak containment internal pressure for the design basis loss of coolant accident, Pa, [45 psig]. The containment design pressure is [50 psig].~~

~~c. The maximum allowable containment leakage rate, La, at Pa, shall be [ ]% of containment air weight per day.~~

~~d. Leakage rate acceptance criteria are:~~

5.5 Programs and Manuals

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

~~1. Containment leakage rate acceptance criterion is ≤ 1.0 La. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are < 0.60 La for the Type B and C tests and [< 0.75 La for Option A Type A tests] [≤ 0.75 La for Option B Type A tests].~~

~~2. Air lock testing acceptance criteria are:~~

~~a) Overall air lock leakage rate is ≤ [0.05 La] when tested at ≥ Pa.~~

~~b) For each door, leakage rate is ≤ [0.01 La] 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.~~ 6

DOC M01 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).~~
~~----------~~ 5

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~~3 (RG), ~~with RG exceptions~~ and program provisions as identified below: 7 7

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

~~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.~~ 7

5.5 Programs and Manuals

DOC M01 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."~~ 7

~~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:~~ 7

~~1~~ a. Actions to restore battery cells with float voltage < ~~[~~2.13~~]~~ V; 7 1

~~2~~ b. 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; 7 1

~~3~~ c. Actions to equalize and test battery cells that had been discovered with electrolyte level below the top of the plates; 7

~~4~~ d. Limits on average electrolyte temperature, battery connection resistance, and battery terminal voltage; and 7

~~5. A requirement to obtain specific gravity readings of all cells at each discharge test, consistent with manufacturer recommendations.~~ 7

6.8.4.k 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 ~~Filtration~~ Ventilation System (~~CREFS~~ 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 whole body or its equivalent to any part of the body]~~ ~~[~~5 rem total 1

5.5 Programs and Manuals

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

effective dose equivalent (TEDE)~~]~~ for the duration of the accident. The program shall include the following elements: 1

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]~~ 1

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 ~~CREFS~~ CREVS, operating at the flow rate required by the Ventilation Filter Testing Program (VFTP), at a Frequency ~~of [18] months on a STAGGERED TEST BASIS~~ INSERT 5. The results shall be trended and used as part of the ~~[18] month~~ assessment of the CRE boundary. 2 3

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.

3 2 **INSERT 5**

in accordance with the Surveillance Frequency Control Program. Additionally, the supply fans (trains A and B) will be tested at a Frequency in accordance with the Surveillance Frequency Control Program.

5.5 Programs and Manuals

~~[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.~~

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

~~1. LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation;"~~
~~2. LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation Functions;"~~
~~3. LCO 3.3.5, "Loss of Power (LOP) Diesel Generator (DG) Start Instrumentation;"~~
~~4. LCO 3.3.6, "Containment Purge and Exhaust Isolation Instrumentation;"~~
~~5. LCO 3.3.7, "Control Room Emergency Filtration System (CREFS) Actuation Instrumentation;"~~
~~6. LCO 3.3.8, "Fuel Building Air Cleanup System (FBACS) Actuation Instrumentation;" and~~
~~7. LCO 3.3.9, "Boron Dilution Protection System (BDPS)."~~

~~b. The program shall require the 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 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.~~
~~----------~~

4

2

5.5 Programs and Manuals

~~5.5.18 Setpoint Control Program (continued)~~

~~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.~~

~~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 Trip 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 NTSP 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, CHANNEL OPERATIONAL TESTS, and TRIP ACTUATING DEVICE OPERATIONAL TESTS that verify the NTSP.~~

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

4

5.5 Programs and Manuals

~~5.5.18 Setpoint Control Program (continued)~~

~~2. If the as-found value of the instrument channel trip setting differs from the previous as-left value or the specified 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.~~

~~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 NTSP at the completion of the surveillance test; otherwise, the channel is inoperable (setpoints may be more conservative than the 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].]~~ 4

6.8.4.l ~~[~~ 5.5.~~19~~ 16 Surveillance Frequency Control Program 1 4

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. ~~]~~ 1

6.8.4.p ~~[~~ 5.5.~~20~~ 17 Risk Informed Completion Time Program 1 4

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:

5.5 Programs and Manuals

5.5.~~20~~ 17 Risk Informed Completion Time Program (continued) 4

a. The RICT may not exceed 30 days;

~~------------------------------- 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.~~
~~---~~ 5

b. A RICT may only be utilized in ~~MODE 1, 2 [, and 3, and MODE 4 while relying on steam generators for heat removal]~~ MODES 1 and 2; 1

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. 2

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.~~ ~~For emergent conditions, if the extent of condition~~ INSERT 6 ~~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.~~ 2

2 **INSERT 6**

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.

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].~~

2

~~[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:~~

~~1. . . .].]~~

1

JUSTIFICATION FOR DEVIATIONS
ITS 5.5, PROGRAMS AND MANUALS

1. The Improved Standard Technical Specifications (ISTS) contain bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

2. 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.

3. The Surveillance Frequency Control Program (SFCP) provides controls for periodic Surveillance Frequencies. Because this is a periodic surveillance, Turkey Point Nuclear Generating Station (PTN) is proposing to control this Frequency under the SFCP.

4. The bracketed ISTS 5.5.3, "Post Accident Sampling," ISTS 5.5.5, "Component Cyclic or Transient Limit," and ISTS 5.5.18, "Setpoint Control Program," are not included in the PTN Current Licensing Basis (CTS). Because these Specifications are not included in CTS, the Specifications are not being included in ITS. Subsequent programs in ITS Section 5.5 have been renumbered, as necessary.

5. 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.

6. PTN follows Option B of 10 CFR 50, Appendix J. Therefore, the ISTS 5.5.15 Option A and combined Option A and B provisions have been deleted.

7. ISTS 5.5.16 is modified in ITS 5.5.14 to reference IEEE 450-2010, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications," and Revision 3 of Regulatory Guide (RG) 1.129, "Maintenance, Testing, and Replacement of Vented Lead-Acid Storage Batteries for Nuclear Power Plants," 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. PTN batteries are lead-calcium type batteries and therefore, specific gravities do not have 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).

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 5.5, PROGRAMS AND MANUALS

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

ATTACHMENT 6

ITS Section 5.6 – Reporting Requirements

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

TABLE 3.3-5 (Continued)

ACTION STATEMENTS

See ITS 3.3.3

ACTION 31 With the number of OPERABLE accident monitoring instrumentation channel(s) less than the Total Number of Channels either restore the inoperable channel(s) to OPERABLE status within 30 days, or submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 14 days outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the system to OPERABLE status.

5.6.5

ACTION 32 With the number of OPERABLE accident monitoring instrumentation channels less than the Minimum Channels OPERABLE, either restore the inoperable channel(s) to OPERABLE status within 7 days, or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

ACTION 33 Close the associated block valve and open its circuit breaker.

See ITS 3.3.3

ACTION 34 With the number of OPERABLE Channels less than required by the Minimum Channels OPERABLE requirements, initiate the preplanned alternate method of monitoring the appropriate parameter(s), within 72 hours, and:

1) Either restore the inoperable channel(s) to OPERABLE status within 7 days of the event, or

5.6.5

2) Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 14 days outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the system to OPERABLE status.

ACTION 35 DELETED

See ITS 3.3.3

ACTION 36 With the number of OPERABLE accident monitoring instrumentation channels less than the Minimum Channel OPERABLE, either restore the inoperable channel to OPERABLE status within 30 days, or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

See ITS 3.3.3

ACTION 37 With the number of OPERABLE channels one less than the Total Number of Channels, restore the system to OPERABLE status within 30 days. If repairs are not feasible without shutting down, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 14 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

5.6.5

TABLE 3.3-5 (Continued)

ACTION STATEMENTS

See ITS 3.3.3

ACTION 38 With the number of OPERABLE channels less than the Minimum Channels OPERABLE requirements, restore the inoperable channel(s) to OPERABLE status within 7 days. If repairs are not feasible without shutting down:

1. Initiate an alternate method of monitoring the reactor vessel inventory; and

5.6.5

2. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 14 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status; and

3. Restore at least one channel to OPERABLE status at the next scheduled refueling.

See ITS 3.3.3

ACTION 39 With the number of OPERABLE channels less than the Minimum Channels OPERABLE requirement, verify position by an alternate means (e.g. administrative controls, ERDADS, alternate position indication, or visual observation) within 2 hours, and restore the inoperable channel(s) within 7 days, or comply with the provisions of Specification 3.6.4 for an inoperable containment isolation valve.

ADMINISTRATIVE CONTROLS

5.6 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 U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, DC pursuant to 10 CFR 50.4.

~~6.9.1 .1 Deleted~~

~~6.9.1.2 Deleted~~

ADMINISTRATIVE CONTROLS

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT*

5.6.1 6.9.1.3 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 (1) the Offsite Dose Calculation Manual (ODCM), and in (2) 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.6.2 6.9.1.4 RADIOACTIVE EFFLUENT RELEASE REPORT**

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

~~6.9.1.5 DELETED~~

5.6.1 Note *A single submittal may be made for a multiple unit station.

5.6.2 Note **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; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

ADMINISTRATIVE CONTROLS

5.6.4 ~~PEAKING FACTOR LIMIT REPORT~~

5.6.4.a ~~6.9.1.6 The W(Z) function(s) for Base-Load Operation corresponding to a ± 2% band about the target flux difference and/or a ± 3% band about the target flux difference, the Load-Follow function $F_z(Z)$ and the augmented surveillance turnon power fraction P_T shall be provided to the U.S. Nuclear Regulatory Commission, whenever P_T is <1.0. In the event, the option of Baseload Operation (as defined in Section 4.2.2.3) will not be exercised, the submission of the W(Z) function is not required. Should these values (i.e., W(Z), $F_z(Z)$ and P_T) change requiring a new submittal or an amended submittal to the Peaking Factor Limit Report, the Peaking Factor Limit Report shall be provided to the NRC Document Control desk with copies to the Regional Administrator and the Resident Inspector within 30 days of their implementation, unless otherwise approved by the Commission.~~

5.6.4.b ~~The analytical methods used to generate the Peaking Factor limits shall be those previously reviewed and approved by the NRC. If changes to these methods are deemed necessary they will be evaluated in accordance with 10 CFR 50.59 and submitted to the NRC for review and approval prior to their use if the change is determined to involve an unreviewed safety question or if such a change would require amendment of previously submitted documentation.~~

LA01

5.6.3 CORE OPERATING LIMITS REPORT

5.6.3.a 6.9.1.7 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT (COLR) before each reload cycle or any remaining part of a reload cycle for the following:

TS 2.1.1 1. Reactor Core Safety Limits ~~for Specification 2.1.1~~.

LCO 3.3.1 2. Overtemperature ΔT, Note 1 of Table ~~2.2-1 for Specification 2.2.1~~, determination of values K_1, K_2, K_3, T´, P´, τ_1, τ_2, τ_3, τ_4, τ_5, τ_6, and the breakpoint and slope values for the f_1 (ΔI).

LCO 3.3.1 3. Overpower ΔT, Note 3 of Table ~~2.2-1 for Specification 2.2.1~~, determination of values for K_4, K_5, K_6, T”, τ_7 and f_2 (ΔI).

LCO 3.1.1 4. Shutdown Margin - T_{avg} >200°F ~~for Specification 3/4.1.1.1~~.

LCO 3.1.1 5. Shutdown Margin - T_{avg} ≤200°F ~~for Specification 3/4.1.1.2~~.

LCO 3.1.3 6. Moderator Temperature Coefficient ~~for Specification 3/4.1.1.3~~.

LCO 3.2.3 7. Axial Flux Difference ~~for Specification 3.2.1~~.

LCO 3.1.6 8. Control Rod Insertion Limits ~~for Specification 3.1.3.6~~.

LCO 3.2.1 9. Heat Flux Hot Channel Factor - $F_Q(Z)$ ~~for Specification 3/4.2.2~~.

LCO 3.1.7 10. All Rods Out position ~~for Specification 3.1.3.2~~.

LCO 3.2.2 11. Nuclear Enthalpy Rise Hot Channel Factor ~~for Specification 3/4.2.3~~.

LCO 3.4.1 12. DNB Parameters ~~for Specification 3.2.5~~, determination of values for Reactor Coolant System T_{avg} and Pressurizer Pressure.

A01

5.6.3.b The analytical methods used to determine the AFD limits shall be those previously reviewed and approved by the NRC in:

1. WCAP-10216-P-A, RELAXATION OF CONSTANT AXIAL OFFSET CONTROL F_Q SURVEILLANCE TECHNICAL SPECIFICATION,” June 1983.

2. WCAP-8385, “POWER DISTRIBUTION CONTROL AND LOAD FOLLOWING PROCEDURES - TOPICAL REPORT,” September 1974.

CORE OPERATING LIMITS REPORT (Continued)

5.6.3.b The analytical methods used to determine F_Q (Z), $F_{\Delta}H$ and the K(Z) curve shall be those previously reviewed and approved by the NRC in:

1. WCAP-9220-P-A, Rev. 1, "Westinghouse ECCS Evaluation Model - 1981 Version," February 1982.

2. WCAP-10054-P-A, (proprietary), "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," August 1985.

3. WCAP-10054-P-A, Addendum 2, Revision 1 (proprietary), "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model," July 1997.

4. WCAP-16009-P-A, "Realistic Large-break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)", January 2005.

5. USNRC Safety Evaluation Report, Letter from R. C. Jones (USNRC) to N. J. Liparulo (W), "Acceptance for Referencing of the Topical Report WCAP-12945(P) 'Westinghouse Code Qualification Document for Best Estimate Loss of Coolant Analysis,' " June 28, 1996.**

6. Letter dated June 13, 1996, from N. J. Liparulo (W) to Frank R. Orr (USNRC), "Re-Analysis Work Plans Using Final Best Estimate Methodology."**

7. WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Reference Core Report," S. L. Davidson and T. L. Ryan, April 1995.

8. WCAP-12610-P-A & CENPD-404-P-A, Addendum 1-A, "Optimized ZIRLO™," July 2006.

5.6.3.b The analytical methods used to determine Overtemperature ΔT and Overpower ΔT shall be those previously reviewed and approved by the NRC in:

1. WCAP-8745-P-A, "Design Basis for the Thermal Overtemperature ΔT and Overpower ΔT Trip Functions, " September 1986

2. WCAP-9272-P-A, " Westinghouse Reload Safety Evaluation Methodology," July 1985

5.6.3.b The analytical methods used to determine Safety Limits, Shutdown Margin - T_{avg} > 200°F, Shutdown Margin - $T_{avg} \leq$ 200°F, Moderator Temperature Coefficient, DNB Parameters, Rod Bank Insertion Limits and the All Rods Out position shall be those previously reviewed and approved by the NRC in:

1. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985.

5.6.3.b The analytical methods used to support the suspension of the measurement of the Moderator Temperature Coefficient in accordance with Surveillance Requirement 4.1.1.3.b shall be those previously reviewed and approved by the NRC in:

1. WCAP-13749-P-A, "Safety Evaluation Supporting the Conditional Exemption of the Most Negative EOL Moderator Temperature Coefficient Measurement," March 1997.

2. WCAP-11596-P-A, "Qualification of the Phoenix-P/ANC Nuclear Design System for Pressurized Water Reactor Cores," June 1988.

3. WCAP-16045-P-A, "Qualification of the Two-Dimensional Transport Code PARAGON," August 2004.

4. WCAP-16045-P-A, Addendum 1-A, "Qualification of the NEXUS Nuclear Data Methodology," August 2007.

**As evaluated in NRC Safety Evaluation dated December 20, 1997.

ADMINISTRATIVE CONTROLS

5.6.3.c The ability to calculate the COLR nuclear design parameters are demonstrated in:

1. Florida Power & Light Company Topical Report NF-TR-95-01, "Nuclear Physics Methodology for Reload Design of Turkey Point & St. Lucie Nuclear Plants."

Topical Report NF-TR-95-01 was approved by the NRC for use by Florida Power & Light Company in:

1. Safety Evaluation by the Office of Nuclear Reactor Regulations Related to Amendment No. 174 to Facility Operating License DPR-31 and Amendment No. 168 to Facility Operating License DPR-41, Florida Power & Light Company Turkey Point Units 3 and 4, Docket Nos. 50-250 and 50-251.

5.6.3.c The AFD, $F_Q(Z)$, $F_{\Delta}H$, K(Z), Safety Limits, Overtemperature ΔT, Overpower ΔT, Shutdown Margin - T_{avg} > 200°F, Shutdown Margin -T_{avg} ≤ 200°F, Moderator Temperature Coefficient, DNB Parameters, and Rod Bank Insertion Limits shall be determined such that all applicable limits of the safety analyses are met.

5.6.3.d The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC ~~Document Control Desk with copies to the Regional Administrator and Resident Inspector, unless otherwise approved by the Commission~~. A02

ADMINISTRATIVE CONTROLS

STEAM GENERATOR TUBE INSPECTION REPORT

5.6.6 6.9.1.8 A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with Specification 6.8.4.j, Steam Generator (SG) Program. The report shall include:

5.6.6.a a. The scope of inspections performed on each SG,

M01

5.6.6.b b. ~~Degradation mechanisms found~~ The nondestructive examination techniques utilized for tubes with increased degradation susceptibility,

5.6.6.c c. ~~Nondestructive examination techniques utilized for each degradation mechanism~~ INSERT 1,

5.6.6.d d. ~~Location, orientation (if linear), and measured sizes (if available) of service induced indications~~ INSERT 2,

~~e. Number of tubes plugged during the inspection outage for each degradation mechanism~~,

5.6.6.e ~~f~~ e. The number and percentage of tubes plugged to date, and the effective plugging percentage in each steam generator, A01

5.6.6.f ~~g~~ f. The results of ~~condition monitoring, including the results of tube pulls and in-situ testing~~ any SG secondary side inspections, and A01

5.6.6.g ~~h~~ g. The primary to secondary leakage rate observed in each SG (if it is not practical to assign the leakage to an individual SG, the entire primary to secondary leakage should be conservatively assumed to be from one SG) during the cycle preceding the inspection which is the subject of the report, A01

5.6.6.h ~~i~~ h. The calculated accident induced leakage rate from the portion of the tubes below 18.11 inches from the top of the tubesheet for the most limiting accident in the most limiting SG. In addition, if the calculated accident induced leakage rate from the most limiting accident is less than 1.82 times the maximum operational primary to secondary leakage rate, the report should describe how it was determined, and A01

5.6.6.i ~~j~~ i. The results of monitoring for tube axial displacement (slippage). If slippage is discovered, the implications of the discovery and corrective action shall be provided. A01

~~SPECIAL REPORTS~~

5.6 6.9.2 ~~Special~~ reports shall be submitted to the Regional Administrator of the Regional Office of the NRC within the time period specified for each report as stated in the Specifications within Sections 3.0, 4.0, or 5.0. A01

~~6.10 DELETED~~

M01 **INSERT 1**

c. For each degradation mechanism found:

1. The nondestructive examination techniques utilized;

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;

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

4. The number of tubes plugged during the inspection outage.

M01 **INSERT 2**

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;

ADMINISTRATIVE CHANGES

A01 In the conversion of the Turkey Point Nuclear Generating Station (PTN) 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-1431, Rev. 5.0, "Standard Technical Specifications - Westinghouse 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 6.9.1.7 requires changes to the Core Operating Limits Report (COLR) to be submitted to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector, unless otherwise approved by the Commission. ITS 5.6.3.d requires changes to the COLR to be provided to the NRC. This changes the CTS by not specifically requiring COLR changes to be sent to the NRC Document Control Desk, Regional Administrator, and Resident Inspector.

The CTS requirement for the COLR that specifically lists addressees for changes is not required. 10 CFR 50.4, "Written Communications," specifies NRC addressees for specific correspondence which is required to be met. In addition, this requirement, "The following reports shall be submitted in accordance with 10 CFR 50.4," is listed at the beginning and applies to all Sections in ITS 5.6. This change is listed as Administrative because it is deletes duplicate information located in 10 CFR 50.4 requirements.

MORE RESTRICTIVE CHANGES

M01 CTS 6.9.1.8 provides requirements for the Steam Generator (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 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 SG 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 Technical Specification Task Force (TSTF) traveler 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 the changes are acceptable because the changes 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.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 5 – Removal of Cycle-Specific Parameter Limits from the Technical Specifications to the Core Operating Limits Report)* CTS 6.9.1.6 provides the requirements for a Peaking Factor Limit Report. ITS 5.6 does not include this report. This changes the CTS by relocating the Peaking Factor Limit Report by attachment to the Core Operating Limits Report (COLR).

The removal of these cycle-specific parameter limits from the Technical Specifications and relocation of the limits into the COLR is acceptable because these limits are developed or utilized under NRC-approved methodologies. The NRC documented in Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits from the Technical Specifications," that 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 requirements and Surveillances that verify that the cycle-specific parameter limits are being met. Limits will be used from the Peaking Factor Limit Report. The location will be in the COLR. Also, this change is acceptable because the removed information will be adequately controlled in the COLR under the requirements provided in ITS 5.6.3, "Core Operating Limits Report." ITS 5.6.3 ensures that the applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems limits, and nuclear limits such as SHUTDOWN MARGIN (SDM), transient analysis limits, and accident analysis limits) of the safety analysis are met. This change is designated as a less restrictive removal of detail change because information relating to cycle-specific parameter limits is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

None

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

5.0 ADMINISTRATIVE CONTROLS

6.9
5.6 Reporting Requirements

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

6.9.1.3 5.6.1 Annual Radiological Environmental Operating Report

6.9.1.3 Note
---------------------------------------~~REVIEWER'S~~ NOTE---------------------------------------
~~[~~ A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station. ~~]~~ 1

6.9.1.3 The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

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

6.9.1.4 5.6.2 Radioactive Effluent Release Report

6.9.1.4 Note
---------------------------------------~~REVIEWER'S~~ 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. ~~]~~ 1

6.9.1.4 The Radioactive Effluent Release Report covering the operation of the unit ~~in the previous yea~~r 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. 2

CTS

5.6 Reporting Requirements

6.9.1.7 5.6.3 CORE OPERATING LIMITS REPORT

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

~~[The individual specifications that address core operating limits must be referenced here.]~~ 1

INSERT 1 4

6.9.1.7 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 4

~~[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.]~~ 1

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

INSERT 3 4

6.9.1.7 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, LTOP arming, and PORV lift settings 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.]~~

4 **INSERT 1**

1. TS 2.1.1 Reactor Core Safety Limits
2. LCO 3.1.1 Shutdown Margin (SDM)
3. LCO 3.1.3 Moderator Temperature Coefficient
4. LCO 3.2.3 Axial Flux Difference
5. LCO 3.1.5 Shutdown Bank Insertion Limit
6. LCO 3.1.6 Control Rod Insertion Limits
7. LCO 3.2.1 Heat Flux Hot Channel Factor - $F_Q(Z)$
8. LCO 3.1.7 All Rods Out position
9. LCO 3.2.2 Nuclear Enthalpy Rise Hot Channel Factor
10. LCO 3.3.1 Overtemperature ΔT, Note 1 of Table 3.3.1-1, determination of values K_1, K_2, K_3, T', P', τ_1, τ_2, τ_3, τ_4, τ_5, τ_6, and the breakpoint and slope values for the f_1 (ΔI)
11. LCO 3.3.1 Overpower ΔT, Note 3 of Table 3.3.1-1, determination of values for K_4, K_5, K_6, T'', τ_7 and f_2 (ΔI)
12. LCO 3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

4 **INSERT 2**

The analytical methods used to determine the AFD limits shall be those previously reviewed and approved by the NRC in:

1. WCAP-10216-P-A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL F_Q SURVEILLANCE TECHNICAL SPECIFICATION," June 1983.

2. WCAP-8385, "POWER DISTRIBUTION CONTROL AND LOAD FOLLOWING PROCEDURES - TOPICAL REPORT," September 1974.

The analytical methods used to determine F_Q (Z), $F_\Delta H$ and the K(Z) curve shall be those previously reviewed and approved by the NRC in:

1. WCAP-9220-P-A, Rev. 1, "Westinghouse ECCS Evaluation Model - 1981 Version," February 1982.

2. WCAP-10054-P-A, (proprietary), "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," August 1985.

3. WCAP-10054-P-A, Addendum 2, Revision 1 (proprietary), "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model," July 1997.

4. WCAP-16009-P-A, "Realistic Large-break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," January 2005.

5. USNRC Safety Evaluation Report, Letter from R. C. Jones (USNRC) to N. J. Liparulo (W), "Acceptance for Referencing of the Topical Report WCAP-12945(P) 'Westinghouse Code Qualification Document for Best Estimate Loss of Coolant Analysis,'" June 28, 1996.**

6. Letter dated June 13, 1996, from N. J. Liparulo (W) to Frank R. Orr (USNRC), "Re-Analysis Work Plans Using Final Best Estimate Methodology."**

7. WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Reference Core Report," S. L. Davidson and T. L. Ryan, April 1995.

8. WCAP-12610-P-A & CENPD-404-P-A, Addendum 1-A, "Optimized ZIRLO™," July 2006.

4

INSERT 2 (Continue)

The analytical methods used to determine Overtemperature ΔT and Overpower ΔT shall be those previously reviewed and approved by the NRC in:

1. WCAP-8745-P-A, "Design Basis for the Thermal Overtemperature ΔT and Overpower ΔT Trip Functions," September 1986

2. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985

The analytical methods used to determine Safety Limits, Shutdown Margin, Moderator Temperature Coefficient, DNB Parameters, Rod Bank Insertion Limits and the All Rods Out position shall be those previously reviewed and approved by the NRC in:

1. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985.

The analytical methods used to support the suspension of the measurement of the Moderator Temperature Coefficient in accordance with Surveillance Requirement (SR) 3.1.3.2 shall be those previously reviewed and approved by the NRC in:

1. WCAP-13749-P-A, "Safety Evaluation Supporting the Conditional Exemption of the Most Negative EOL Moderator Temperature Coefficient Measurement," March 1997.

2. WCAP-11596-P-A, "Qualification of the Phoenix-P/ANC Nuclear Design System for Pressurized Water Reactor Cores," June 1988.

3. WCAP-16045-P-A, "Qualification of the Two-Dimensional Transport Code PARAGON," August 2004.

4. WCAP-16045-P-A, Addendum 1-A, "Qualification of the NEXUS Nuclear Data Methodology," August 2007.

**As evaluated in NRC Safety Evaluation dated December 20, 1997.

4 **INSERT 2 (Continue)**

ADMINISTRATIVE CONTROLS

The ability to calculate the COLR nuclear design parameters are demonstrated in:

1. Florida Power & Light Company Topical Report NF-TR-95-01, "Nuclear Physics Methodology for Reload Design of Turkey Point & St. Lucie Nuclear Plants."

Topical Report NF-TR-95-01 was approved by the NRC for use by Florida Power & Light Company in:

1. Safety Evaluation by the Office of Nuclear Reactor Regulations Related to Amendment No. 174 to Facility Operating License DPR-31 and Amendment No. 168 to Facility Operating License DPR-41, Florida Power & Light Company Turkey Point Units 3 and 4, Docket Nos. 50-250 and 50-251.

4

INSERT 3

The AFD, $F_Q(Z)$, $F_{\Delta}H$, K(Z), Safety Limits, Overtemperature ΔT, Overpower ΔT, Shutdown Margin, Moderator Temperature Coefficient, DNB Parameters, and Rod Bank Insertion Limits shall be determined such that all applicable limits of the safety analyses are met.

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.]~~

~~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:~~ 4

~~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. LTOP arming temperature limit development methodology.~~

CTS

5.6 Reporting Requirements

~~5.6.4 RCS PRESSURE AND TEMPERATURE LIMITS REPORT (continued)~~

~~7. The minimum temperature requirements of Appendix G to 10 CFR Part 50 shall be incorporated into the pressure and temperature limit curves.~~

~~8. 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 RT_{NDT} + $2\sigma_{\Delta}$), the licensee should provide a supplement to the PTLR to demonstrate how the results affect the approved methodology.~~ 4

5.6.5 Post Accident Monitoring Report 1

When a report is required by Condition ~~B or F~~ E or I of LCO 3.3.~~[~~3~~]~~, "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. 2

~~[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.]~~ 1

6.9.1.8 5.6.~~7~~6 Steam Generator Tube Inspection Report 2

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~~6, "Steam Generator (SG) Program." The report shall include:

6.9.1.8.a a. The scope of inspections performed on each SG,

6.9.1.8.b b. The nondestructive examination techniques utilized for tubes with increased degradation susceptibility;

6.9.1.8.c c. For each degradation mechanism found:

1. The nondestructive examination techniques utilized;

CTS

5.6 Reporting Requirements

5.6.~~7~~ 6 Steam Generator Tube Inspection Report (continued)

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;

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

4. The number of tubes plugged ~~[or repaired]~~ during the inspection outage.~~; and~~ 1 2

~~[5. The repair methods utilized and the number of tubes repaired by each repair method.]~~ 1

6.9.1.8.d 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;

6.9.1.8.e e. The number and percentage of tubes plugged ~~[or repaired]~~ to date, and the effective plugging percentage in each SG, 1

6.9.1.8.f f. The results of any SG secondary side inspections; and

6.9.1.8.g ~~[g. Insert any plant-specific reporting requirements, if applicable.]~~ 1

g. The primary to secondary leakage rate observed in each SG (if it is not practical to assign the leakage to an individual SG, the entire primary to secondary leakage should be conservatively assumed to be from one SG) during the cycle preceding the inspection which is the subject of the report,

h. The calculated accident induced leakage rate from the portion of the tubes below 18.11 inches from the top of the tubesheet for the most limiting accident in the most limiting SG. In addition, if the calculated accident induced leakage rate from the most limiting accident is less than 1.82 times the maximum operational primary to secondary leakage rate, the report should describe how it was determined, and 4

i. The results of monitoring for tube axial displacement (slippage). If slippage is discovered, the implications of the discovery and corrective action shall be provided.

JUSTIFICATION FOR DEVIATIONS
ITS 5.6, REPORTING REQUIREMENTS

1. The Improved Standard Technical Specifications (ISTS) contain bracketed information and/or values that are generic to all Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

2. 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.

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. Changes are made to be consistent with the current licensing basis.

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 Specification (CTS) Markup
and Discussion of Changes (DOCs)

6.11 DELETED

5.7 6.12 HIGH RADIATION AREA

or equivalent that includes specification of radiation dose rates in the immediate work

5.7.1 6.12.1 Pursuant to paragraph 20.1601(c) of 10 CFR Part 20, in lieu of the “control device” or “alarm signal” required by paragraph 20.1601(a), each high radiation area, as defined in 10 CFR Part 20, in which the intensity of radiation is greater than 100 mrem/hr but equal to or less than 1000 mrem/hr at 30 cm (12 in.) from the radiation source or from any surface which the radiation penetrates shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP). Individuals qualified in radiation protection procedures (e.g., Health Physics Technician) or personnel continuously escorted by such individuals may be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas with exposure rates equal to or less than 1000 mrem/hr, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas. A02

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; or

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 levels in the area have been established and personnel have been made knowledgeable of them; or

INSERT 1 L01

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

5.7.2 6.12.2 In addition to the requirements of Specification 6.12.1, areas accessible to personnel with radiation levels greater than 1000 mrem/hr at 30 cm (12 in.) and less than 500 rads/hr at 1 meter from the radiation source or from any surface which the radiation penetrates shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the shift supervisor 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 the immediate work areas and the maximum allowable stay time for individuals in that area. In lieu of the stay time specification of the RWP, direct or remote (such as closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area.

For individual high radiation areas accessible to personnel with radiation levels of greater than 1000 mrem/hr and less than 500 rads/hr that are located within large areas, such as PWR containment, where no enclosure exists for purposes of locking, and where no enclosure can be reasonably constructed around the individual area, that individual area shall be barricaded, conspicuously posted, and a flashing light shall be activated as a warning device.

~~6.13 DELETED~~

INSERT 1

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

DISCUSSION OF CHANGES
ITS 5.7, HIGH RADIATION AREA

ADMINISTRATIVE CHANGES

A01 In the conversion of the Turkey Point Nuclear Generating Station (PTN) 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-1431, Rev. 5.0, "Standard Technical Specifications - Westinghouse 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 6.12.1 states, in part, "and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP)." ITS 5.7.1.b states, in part, that access to, and activities in, each such area shall be controlled by means of a Radiation Work Permit (RWP) or equivalent that includes specification of radiation dose rates in the immediate work area (s). This changes the CTS by specifying that the RWP shall include specification of radiation dose rates in the immediate work area(s).

The purpose of CTS 6.12.1 is to specify the controls needed to access high radiation areas. This change is acceptable because the additional wording that the RWP equivalent includes a specification of radiation dose rates in the immediate work area(s) clarifies the requirements of an RWP. This is designated as administrative because it does not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

L01 *(Category 1 – Relaxation of LCO Requirements)* CTS 6.12.1 provides requirements for radiation monitoring of individuals entering a high radiation area. ITS 5.8.1.d provides similar requirements adding an option for remote monitoring by a radiation protection personnel adding 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. This changes the CTS by providing an additional device that an individual entering these high radiation areas must possess for radiation monitoring.

The purpose of CTS 6.12 is to provide appropriate alternate means for monitoring the exposure of personnel in the respective high radiation areas. This change is acceptable because the means specified provide a reliable method of monitoring personnel exposure. This change is designated as less restrictive because a new alternative for measuring personnel dose in high radiation areas has been provided.

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

5.0 ADMINISTRATIVE CONTROLS

6.12 5.7 High Radiation Area

As provided in paragraph 20.1601(c) of 10 CFR Part 20, the following controls shall be applied to high radiation areas in place of the controls required by paragraph 20.1601(a) ~~and (b)~~ of 10 CFR Part 20: 2

6.12.1 5.7.1 High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters (12 inches) from the Radiation Source or from any Surface Penetrated by the Radiation 1

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

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

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

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

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

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

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

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

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

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) (12 inches) 1

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

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

6.12.2 5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters (12 inches) from the Radiation Source of 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 1

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 designees~~ and/or health physics supervision, and 2

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

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), maximum allowable stay time and other appropriate radiation protection equipment and measures. 2

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

5.7 High Radiation Area

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters (12 inches) from the Radiation Source of 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

d. Each individual 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 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 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 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 displaces radiation dose rates in the area.

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

5.7 High Radiation Area

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters (12 inches) from the Radiation Source of 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

f. Such individual areas such as containment 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.

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JUSTIFICATION FOR DEVIATIONS
ITS 5.7, HIGH RADIATION AREA

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

2. Changes are made to be consistent with the current licensing basis.

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 Specification.