ATTACHMENT 1

VOLUME 16

DAVIS-BESSE
IMPROVED TECHNICAL SPECIFICATIONS CONVERSION

ITS CHAPTER 5.0
ADMINISTRATIVE CONTROLS

Revision 0

LIST OF ATTACHMENTS

ATTACHMENT 1

ITS 5.1, RESPONSIBILITY

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

ITS

Add proposed approval requirement for proposed tests, experiments, or modifications to systems or equipment that affects nuclear safety. M01

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

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

Add proposed ITS 5.1.2 M01

6.2 ORGANIZATION

6.2.1 OFFSITE AND ONSITE ORGANIZATIONS

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

a. Lines of authority, responsibility, and communication shall be established and defined for the highest management levels through intermediate levels up to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of 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 Updated Safety Analysis Report.

See ITS 5.2

b. 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.

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

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

6.2.2 FACILITY STAFF

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

b. At least one licensed Operator shall be in the control panel area when fuel is in the reactor.

DISCUSSION OF CHANGES
ITS 5.1, RESPONSIBILITY

ADMINISTRATIVE CHANGES

A01 In the conversion of the Davis-Besse 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-1430, Rev. 3.1, "Standard Technical Specifications-Babcock and Wilcox 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 5.1.1 requires that the plant manager or his designee approve, prior to implementation, each proposed test, experiment, or modification to systems or equipment that affects nuclear safety. ITS 5.1.2 provides the requirement that a designated individual assume the responsibility for the control room command function. In MODES 1, 2, 3, and 4, ITS 5.1.2 requires the designated individual hold an active Senior Operator license. In MODE 5 or 6, ITS 5.1.2 requires the designated individual hold an active Senior Operator license or Operator license. This changes the CTS by adding an approval requirement for the plant manager or his designee and by adding requirements for the designated individual that assumes the control room command function.

The purpose of the ITS 5.1.1 requirement is to provide additional assurance that the plant manager has direct responsibility for overall unit operation. The purpose of the ITS 5.1.2 requirement is to ensure that the control room command function is maintained. The change to ITS 5.1.1 is acceptable because the additional requirements ensure that the plant manager will be responsible for overall unit safe operation and shall have control over those activities necessary for safe operation and maintenance of the plant. The change to ITS 5.1.2 is acceptable because the requirement ensures that the designated individual assuming control room functions meets the appropriate qualification requirements. These changes are designated as more restrictive because they add additional requirements for the plant manager or his designee and they add control room command requirements.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

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DISCUSSION OF CHANGES
ITS 5.1, RESPONSIBILITY

LESS RESTRICTIVE CHANGES

None

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

CTS

5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

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

~~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.~~

~~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.~~

--- 1

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

The plant manager or his designee shall approve, prior to implementation, each proposed test, experiment or modification to systems or equipment that affect s nuclear safety. 4

DOC M01 5.1.2 The ~~[S~~hift ~~Supervisor (SS)]~~ manager shall be responsible for the control room command function. During any absence of the ~~[SS]~~ shift manager from the control room while the unit is in MODE 1, 2, 3, or 4, an individual with an active Senior ~~Reactor~~ Operator ~~(SRO)~~ license shall be designated to assume the control room command function. During any absence of the ~~[SS]~~ shift manager from the control room while the unit is in MODE 5 or 6, an individual with an active ~~SRO~~ Senior Operator license or ~~Reactor~~ Operator license shall be designated to assume the control room command function. 2 2 3 2 3

JUSTIFICATION FOR DEVIATIONS
ITS 5.1, RESPONSIBILITY

1. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed in to what is needed to meet this requirement. This is not meant to be retained in the final version of the plant specific submittal.

2. The brackets have been removed and the proper plant specific information/value has been provided.

3. Typographical error corrected. The terms in 10 CFR 55.4 and 10 CFR 50.54(m) are "Senior Operator" and "Operator."

4. Grammatical error corrected.

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Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 5.1, RESPONSIBILITY

There are no specific NSHC discussions for this Specification.

ATTACHMENT 2

ITS 5.2, ORGANIZATION

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

ITS

A01

ITS 5.2

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

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

See ITS 5.1

5.2 ### 6.2 ORGANIZATION

5.2.1 #### 6.2.1 OFFSITE AND ONSITE ORGANIZATIONS

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

5.2.1.a a. Lines of authority, responsibility, and communication shall be established and defined for the highest management levels through intermediate levels up to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of 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 Updated Safety Analysis Report.

5.2.1.c b. 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.

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

5.2.1.d d. The individuals who train the operating staff and those who carry out health physics and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

5.2.2 #### 6.2.2 FACILITY STAFF

a. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1. LA01

b. At least one licensed Operator shall be in the control panel area when fuel is in the reactor. A02

ITS A01 ITS 5.2

6.0 ADMINISTRATIVE CONTROLS

6.2.2 (Continued)

c. At least two licensed Operators, one of which has a Senior Reactor Operator license, shall be present in the control room while in MODES 1, 2, 3, or 4. A02

5.2.2.c d. An individual qualified in radiation protection procedures shall be on site when fuel is in the reactor#.

e. All CORE ALTERATIONS shall be directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation. A02

f. Deleted

5.2.2.e g. The operations manager shall either hold or have held a senior reactor operator's license on a pressurized water reactor. The assistant operations manager shall hold a senior reactor operator license for the Davis-Besse Nuclear Power Station.

5.2.2.d 6.2.3 FACILITY STAFF OVERTIME

Administrative controls shall be developed and implemented to limit the working hours of personnel who perform safety-related functions (e.g., senior reactor operators, reactor operators, auxiliary operators, health physicists, and key maintenance personnel). The controls shall include guidelines on working hours that ensure that adequate shift coverage is maintained without routine heavy use of overtime for individuals.

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

Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the plant manager or his/her designee(s) to ensure that excessive hours have not been assigned.

5.2.2.c # The individual qualified in radiation protection procedures 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 position.

ITS

A01

ITS 5.2

6.0 ADMINISTRATIVE CONTROLS

TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION#

	LICENSE CATEGORY	APPLICABLE MODES 1, 2, 3 & 4	APPLICABLE MODES 5 & 6
	SOL	2**	1*
	OL	2	1
5.2.2.a	Non-Licensed	2	1
5.2.2.f	Shift Technical Advisor	1**	None Required

LA01

M01

5.2.2.b # Shift crew composition may be less than the minimum requirements 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.

* Does not include the licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling supervising CORE ALTERATIONS. LA01

** One of the two required individuals filling the SOL positions may also assume the STA function provided the individual meets the qualifications for the combined SRO/STA position specified for Option 1 of the Commission's Policy Statement on Engineering Expertise on Shift. If this option is used for a shift, then the separate STA position may be eliminated for that shift. A03

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 (1) the radiation protection manager who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975, (2) the Shift Technical Advisor who shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design, and response and analysis of the plant for transients and accidents, and (3) the operations manager whose requirement for a senior reactor operator license is as stated in Specification 6.2.2.g.

See ITS 5.3

A04

See ITS 5.3

6.4 Deleted

6.5 REVIEW AND AUDIT

6.5.1 Deleted

6.5.2 Deleted

DISCUSSION OF CHANGES
ITS 5.2, ORGANIZATION

ADMINISTRATIVE CHANGES

A01 In the conversion of the Davis-Besse 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-1430, Rev. 3.1, "Standard Technical Specifications-Babcock and Wilcox 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 6.2.2.b states "At least one licensed Operator shall be in the control panel area when fuel is in the reactor." CTS 6.2.2.c states "At least two licensed Operators, one of which has a Senior Operator license, shall be present in the control room while in MODES 1, 2, 3, or 4." CTS 6.2.2.e requires all CORE ALTERATIONS to be directly supervised by a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation. The ITS does not include these requirements. This changes the CTS by deleting these requirements.

10 CFR 50.54(m)(2)(iii) states "When a nuclear power unit is in an operational mode other than cold shutdown or refueling, as defined by a unit's technical specifications, each licensee shall have a person holding a senior operator license for the nuclear power unit in the control room at all times. In addition to this senior operator, for each fueled nuclear power unit, a licensed operator or senior operator shall be at the controls at all times. 10 CFR 50.54(m)(2)(iv) states "Each licensee shall have present, during alteration of the core of a nuclear power unit (including fuel loading or transfer), a person holding a senior operator license or a senior operator license limited to fuel handling to directly supervise the activity and, during this time, the licensee shall not assign other duties to this person." This change is acceptable because the requirements deleted from the Technical Specifications are already required by 10 CFR 50.54(m)(2)(iii) and 10 CFR 50.54(m)(2)(iv). This change is designated as administrative because it does not result in technical changes to the CTS.

A03 CTS Table 6.2-1 footnote ** states "One of the two required individuals filling the SOL positions may also assume the STA function provided the individual meets the qualifications for the combined SRO/STA position specified for Option 1 of the Commission Policy Statement on Engineering Expertise on Shift. If this option is used for a shift, then the separate STA position may be eliminated for that shift." ITS 5.2.2, in part, requires the STA to meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift; it does not include this specific information. This changes the CTS by deleting this specific information.

The purpose of CTS Table 6.2-1 footnote ** is to provide the allowance for one of the personnel meeting the Senior Operator License requirement to also meet the requirements of the STA, as allowed in Option 1 of the Commission's Policy Statement on Engineering Expertise on Shift. The ITS already requires the STA to meet this policy statement (ITS 5.2.2.f), as described in DOC M01. Furthermore, this allowance is adequately addressed in the Commission Policy

DISCUSSION OF CHANGES
ITS 5.2, ORGANIZATION

Statement on Engineering Expertise on Shift, published in Generic Letter 86-04, dated February 13, 1986, and need not be specifically retained in the ITS. This change is considered acceptable since it is removing redundant requirements. This change is designated as administrative because it does not result in technical changes to the CTS.

A04 CTS 6.3.1 provides, in part, qualification requirements for the Shift Technical Advisor (STA), and requires the STA to have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design, and response and analysis of the plant for transient and accidents. ITS 5.2.2.f requires this individual to meet the qualification requirements of the Commission Policy Statement on Engineering Expertise on Shift for qualification requirements instead of listing the specific qualification requirements.

The purpose of the CTS 6.3.1 STA requirements is to specify the minimum qualification requirements for the STA. This change is acceptable because the qualification requirements included in the Commission Policy Statement on Engineering Expertise on Shift encompass the current STA qualification requirements. This change is designated as administrative because it does not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

M01 CTS Table 6.2-1 requires the minimum shift crew to include one STA when the unit is in MODE 1, 2, 3, or 4. ITS 5.2.2 requires that an individual 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, when the unit is in MODE 1, 2, 3, or 4. It furthermore states that the individual shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift. This changes the CTS by detailing the specific responsibilities of the STA.

The purpose of the CTS Table 6.2-1 and Footnote ** STA requirements is to ensure that appropriate engineering expertise is available on shift. This change is acceptable because it clarifies STA requirements consistent with the Commission Policy Statement on Engineering Expertise on Shift. This change is designated as more restrictive because it provides specific details of the responsibilities of the STA.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 4 – Removal of LCO, SR, or other TS Requirement to the TRM, UFSAR, ODCM, QAPM, IST Program, or IIP)* CTS 6.2.2.a and Table 6.2-1, including footnote *, provide minimum shift crew composition requirements. ITS 5.2.2 only

DISCUSSION OF CHANGES
ITS 5.2, ORGANIZATION

includes the minimum shift crew composition requirements that are not already included in 10 CFR 50.54. This changes the CTS by moving the minimum shift crew composition requirements addressed by 10 CFR 50.54 to the Technical Requirements Manual (TRM).

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 minimum shift crew composition requirements for licensed operators and senior operators are also contained in 10 CFR 50.54(k), (l), and (m) and do not need to be repeated in the Technical Specifications. The minimum shift crew composition requirements for non-licensed operators are transferred from CTS Table 6.2-1 to ITS 5.2.2.a and the minimum shift crew composition requirements for the STA are transferred from CTS Table 6.2-1 to ITS 5.2.2.f. The relocation of the details of the minimum shift crew composition requirements to the TRM is acceptable considering the controls provided by regulations and the remaining requirements in the Technical Specifications. The TRM is currently incorporated by reference into the UFSAR, thus 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 details for meeting Technical Specification and regulatory requirements are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

None

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

CTS

5.2 ADMINISTRATIVE CONTROLS

6.2 5.2 Organization

6.2.1 5.2.1 Onsite and Offsite Organizations

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

6.2.1.a a. Lines of authority, responsibility, and communication shall be defined and established throughout highest management levels, intermediate levels, and all operating organization positions. These relationships shall be documented and updated, as appropriate, in organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements , including the plant-specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications , shall be documented in the UFSAR ~~[FSAR/QA Plan]~~ . 2 1 2

6.2.1.c b. The plant manager shall be responsible for overall safe operation of the plant and shall have control over those onsite activities necessary for safe operation and maintenance of the plant . 2

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~~ . 2

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

~~-----REVIEWER'S NOTE-----~~
~~Two unit sites with both units shutdown or defueled require a total of three non-licensed operators for the two units.~~ 5

CTS

5.2 Organization

5.2.2 Unit Staff (continued)

Table 6.2-1 and Note # — 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.f for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements. ; Specifications 6 4

6.2.2.d — 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. ; 4

6.2.3 — d. Administrative ~~procedures~~ controls shall be developed and implemented to limit the working hours of personnel who perform safety related functions (e.g., licensed Senior ~~Reactor~~ Operators ~~(SROs)~~, licensed ~~Reactor~~ Operators ~~(ROs)~~, health physicists, auxiliary operators, and key maintenance personnel). 7 3 8 3

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

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

Controls shall be included in the procedures ~~to require a periodic independent review be conducted~~ INSERT 1 to ensure that excessive hours have not been assigned. ; 9 4

6.2.2.g — e. The operations manager ~~or assistant operations manager~~ shall either hold or have held ~~an~~ ~~SRO~~ Senior Operator license. INSERT 2 10 8

Table 6.2-1 — f. When the reactor is operating in MODE 1, 2, 3, or 4 ~~An~~ 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. 11

9 **INSERT 1**

such that the individual overtime shall be reviewed monthly by the plant manager or designee

10 **INSERT 2**

The assistant operations manager shall hold a Senior Operator license for the Davis-Besse Nuclear Power Station; and

JUSTIFICATION FOR DEVIATIONS
ITS 5.2, ORGANIZATION

1. The brackets have been removed and the proper plant specific information/value has been provided.

2. Typographical/grammatical error corrected.

3. Davis-Besse Nuclear Power Station includes only one unit. Therefore, the words in ITS 5.2.2.a have been modified to reflect a single unit site.

4. These punctuation corrections have been made consistent with the Writer's Guide for the Improved Standard Technical Specifications, TSTF-GG-05-01, Section 5.1.3.

5. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed in to what is needed to meet this requirement. This is not meant to be retained in the final version of the plant specific submittal.

6. The referenced requirements are Specifications, not Code of Federal Regulations (CFR) requirements. Therefore, the word "Specifications" has been added to clearly state that 5.2.2.a and 5.2.2.f are Specifications.

7. The term "procedures" has been changed to "controls" in the first paragraph of ITS 5.2.2.d to be consistent with the usage of the term in the second paragraph. This is also consistent with the current licensing basis.

8. Typographical error corrected. The terms in 10 CFR 55.4 and 10 CFR 50.54(m) are "Senior Operator" and "Operator."

9. ISTS 5.2.2.d provides requirements for working hour limitations. These requirements are revised in ITS 5.2.2.d to reflect the Davis-Besse CTS 6.2.3 requirements which were approved by the NRC in License Amendment 212, dated November 8, 1996.

10. ISTS 5.2.2.e provides license requirements for the operations manager and assistant operations manager. These requirements are revised in ITS 5.2.2.e to reflect the Davis-Besse CTS 6.2.3 requirements, which were approved by the NRC in License Amendment 272, dated February 7, 2006.

11. ISTS 5.5.2.f provides requirements for the Shift Technical Advisor (STA). These requirements are revised in ITS 5.2.2.f to reflect the Davis-Besse CTS Table 6.2-1 MODE requirements for the STA.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 5.2, ORGANIZATION

There are no specific NSHC discussions for this Specification.

ATTACHMENT 3

ITS 5.3, UNIT STAFF QUALIFICATIONS

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

A01

ITS 5.3

ITS

6.0 ADMINISTRATIVE CONTROLS

TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION#

LICENSE	APPLICABLE MODES	
CATEGORY	1, 2, 3 & 4	5 & 6
SOL	2**	1*
OL	2	1
Non-Licensed	2	1
Shift Technical Advisor	1**	None Required

See ITS 5.2

Shift crew composition may be less than the minimum requirements 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.

* Does not include the licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling supervising CORE ALTERATIONS.

** One of the two required individuals filling the SOL positions may also assume the STA function provided the individual meets the qualifications for the combined SRO/STA position specified for Option 1 of the Commission's Policy Statement on Engineering Expertise on Shift. If this option is used for a shift, then the separate STA position may be eliminated for that shift.

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 (1) the radiation protection manager who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975, (2) the Shift Technical Advisor who shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design, and response and analysis of the plant for transients and accidents, and (3) the operations manager whose requirement for a senior reactor operator license is as stated in Specification 6.2.2.g.

See ITS 5.2

Add proposed Specification 5.3.2

A02

6.4 Deleted

6.5 REVIEW AND AUDIT

6.5.1 Deleted

6.5.2 Deleted

DISCUSSION OF CHANGES
ITS 5.3, UNIT STAFF QUALIFICATIONS

ADMINISTRATIVE CHANGES

A01 In the conversion of the Davis-Besse 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-1430, Rev. 3.1, "Standard Technical Specifications-Babcock and Wilcox Plants" (ISTS).

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

A02 ITS 5.3.2 states "For the purpose of 10 CFR 55.4, a licensed Senior Operator and a licensed Operator 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)." The CTS does not include such a statement. This changes the CTS by clarifying that these individuals must meet all of the qualification requirements referenced in 10 CFR 55.4, ITS 5.3.1, and 10 CFR 50.54(m).

This change is acceptable because it clarifies the existing relationship between the Technical Specifications and regulations regarding licensed Senior Operator and Operator qualification requirements. This change 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

None

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

CTS

5.0 ADMINISTRATIVE CONTROLS

6.3 5.3 Unit Staff Qualifications

---REVIEWER'S NOTE---
~~Minimum qualifications for members of the unit staff shall be specified by use of an overall qualification statement referencing an ANSI Standard acceptable to the NRC staff or by specifying individual position qualifications. Generally, the first method is preferable; however, the second method is adaptable to those unit staffs requiring special qualification statements because of unique organizational structures.~~ 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].~~ INSERT 1 2

DOC A02 5.3.2 For the purpose of 10 CFR 55.4, a licensed Senior ~~Reactor~~ Operator ~~(SRO)~~ and a licensed ~~Reactor~~ Operator ~~(RO)~~ are those individuals who, in addition to meeting the requirements of ~~TS~~ Specification 5.3.1, perform the functions described in 10 CFR 50.54(m). 3 4

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2 **INSERT 1**

ANSI N18.1-1971 for comparable positions, except for the radiation protection manager and the operations manager. The radiation protection manager shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975. The operations manager shall be qualified as required by Specification 5.2.2.e.

JUSTIFICATION FOR DEVIATIONS
ITS 5.3, UNIT STAFF QUALIFICATIONS

1. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed in to what is needed to meet this requirement. This is not meant to be retained in the final version of the plant specific submittal.
2. The brackets are removed and the proper plant specific information/value is provided.
3. Typographical error corrected. The terms in 10 CFR 55.4 and 10 CFR 50.54(m) are "Senior Operator" and "Operator."
4. Change made to be consistent with the terminology used in other Specifications.

Specific No Significant Hazards Considerations (NSHCs)

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

There are no specific NSHC discussions for this Specification.

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ATTACHMENT 4

ITS 5.4, PROCEDURES

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

ITS

A01

ITS 5.4

6.0 ADMINISTRATIVE CONTROLS

6.6 Deleted

6.7 Deleted

5.4 6.8 PROCEDURES AND PROGRAMS

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

5.4.1.a a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, February, 1978.

b. Refueling operations. A02

c. Surveillance and test activities of safety related equipment. A02

d. Physical Security Plan implementation. A03

e. Davis-Besse Emergency Plan implementation. A03

Add proposed Specification 5.4.1.b M01

5.4.1.d f. Fire Protection Program implementation.

5.4.1.c g. The radiological environmental monitoring program.

Add proposed Specification 5.4.1.e M02

h. Deleted.

i. Offsite Dose Calculation Manual implementation. A04

6.8.2 Each procedure of 6.8.1 above, and changes thereto, shall be reviewed and approved prior to implementation as set forth in 6.5.3 above. LA01

6.8.3 Deleted

6.8.4 The following programs shall be established, implemented and maintained:

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 makeup, letdown, seal injection, seal return, low pressure injection, containment spray, high pressure injection, waste gas, primary sampling and reactor coolant drain systems. The program shall include the following: See ITS 5.5

(i) Preventive maintenance and/or periodic visual inspection requirements, and

DISCUSSION OF CHANGES
ITS 5.4, PROCEDURES

ADMINISTRATIVE CHANGES

A01 In the conversion of the Davis-Besse 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-1430, Rev. 3.1, "Standard Technical Specifications-Babcock and Wilcox Plants" (ISTS).

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

A02 CTS 6.8.1.b requires written procedures be established, implemented, and maintained covering refueling operations. CTS 6.8.1.c requires written procedures be established, implemented, and maintained covering surveillance and test activities of safety related equipment. ITS 5.4.1.a requires written procedures to be established, implemented and maintained to the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. This changes the CTS by deleting the specific wording of CTS 6.1.8.b and 6.8.1.c.

This change is acceptable because the recommendations of Regulatory Guide 1.33, Appendix A, February 1978 already require procedures for refueling operations and for surveillance tests for safety related activities. This change is designated as administrative because it does not result in technical changes to the CTS.

A03 CTS 6.8.1.d and CTS 6.8.1.e require written procedures be established, implemented, and maintained for the Physical Security Plan and the Davis-Besse Emergency Plan. The ITS does not contain these requirements. This changes the CTS by deleting the specific reference to the Security Plan and the Emergency Plan.

This change is acceptable because the requirements for implementation of the Security and Emergency Plans are contained in 10 CFR 50.54(p) and 10 CFR 50.54(q), respectively. This change is designated as administrative because it does not result in technical changes to the CTS.

A04 CTS 6.8.1.i requires written procedures be established, implemented and maintained for the Offsite Dose Calculation Manual (ODCM). ITS 5.4.1 requires procedures for various activities, but does not specifically list the ODCM. This changes the CTS by removing the explicit requirements for written procedures for implementation of the ODCM.

This change is acceptable because implementing procedures for ODCM are required by ITS 5.4.1.e. ITS 5.4.1.e (added as described in DOC M02) requires that written procedures be established, implemented, and maintained for all programs in ITS 5.5. ITS 5.5.1 covers the programmatic requirements for the ODCM. Therefore, it is not necessary to specifically identify each program in ITS 5.4.1. This change is designated as administrative because it does not result in technical changes to the CTS.

DISCUSSION OF CHANGES
ITS 5.4, PROCEDURES

MORE RESTRICTIVE CHANGES

M01 ITS 5.4.1.b requires written procedures be established, implemented, and maintained for the emergency operating procedures required to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1, as stated in Generic Letter 82-33. The CTS does not include this requirement. This changes the CTS by adopting a new requirement for emergency operating procedures.

The purpose of ITS 5.4.1.b is to ensure that written procedures are established, implemented, and maintained covering the emergency operating procedures to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1, as stated in Generic Letter 82-33. This change is acceptable because it is consistent with an existing requirement to comply with NUREG-0737 and NUREG-0737, Supplement 1, as stated in Generic Letter 82-33, for emergency operating procedures. This change is designated as more restrictive because it imposes a new requirement for procedures within the Technical Specifications.

M02 ITS 5.4.1.e requires written procedures be established, implemented, and maintained for all programs specified in Specification 5.5. The CTS does not include this requirement for any program except the ODCM. This changes the CTS by adopting a new requirement for procedures to address all programs described in ITS 5.5.

The purpose of ITS 5.4.1.e is to ensure that written procedures are established, implemented, and maintained covering all programs specified in ITS 5.5. This change is considered acceptable because it requires written procedures, including proper procedure control to address programs required by ITS 5.5. This change is designated as more restrictive because it imposes new 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, QAPM, IST Program, or IIP)* CTS 6.8.2 requires that each procedure of CTS 6.8.1, and changes to these documents, be reviewed and approved prior to implementation as set forth in CTS 6.5.3. ITS 5.4 does not include this requirement. This changes the CTS by moving these details of procedure and administrative policy reviews to the QAPM.

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. ITS 5.4.1 still retains the requirement for written procedures required by the Technical Specifications to be established, implemented, and maintained.

DISCUSSION OF CHANGES
ITS 5.4, PROCEDURES

Regulations provide an adequate level of control for the affected review requirement. The requirements for establishment, maintenance, and implementation of procedures related to activities affecting quality are contained in 10 CFR 50, Appendix B, Criterion II and Criterion V and Regulatory Guide 1.33, Revision 2, February 1978. In accordance with these requirements, the QAPM includes adequate detail with respect to administrative control of procedures related to activities affecting quality and nuclear safety, including the review requirements associated with maintenance of these procedures. Furthermore, CTS 6.5.3 is being moved to the QAPM, as described in CTS 6.0 DOC LA01 (in this chapter). Also, this change is acceptable because these types of procedural details will be adequately controlled in the QAPM. Any changes to the QAPM are made under 10 CFR 50.54(a), which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because references for meeting Technical Specification and regulatory 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 a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978, ; 2

DOC M01 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], ; 1 2

6.8.1.g c. Quality assurance for effluent and environmental monitoring, ; 2

6.8.1.f d. Fire Protection Program implementation, ; and 2

DOC M02 e. All programs specified in Specification 5.5.

JUSTIFICATION FOR DEVIATIONS
ITS 5.4, PROCEDURES

1. The brackets have been removed and the proper plant specific information/value has been provided.

2. These punctuation corrections have been made consistent with the Writer's Guide for the Improved Standard Technical Specifications, TSTF-GG-05-01, Section 5.1.3.

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Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 5.4, PROCEDURES

There are no specific NSHC discussions for this Specification.

ATTACHMENT 5

ITS 5.5, PROGRAMS AND MANUALS

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

ITS

A01

ITS 5.5

6.0 ADMINISTRATIVE CONTROLS

6.6 Deleted

6.7 Deleted

6.8 PROCEDURES AND PROGRAMS

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

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, February, 1978.

b. Refueling operations.

c. Surveillance and test activities of safety related equipment.

d. Physical Security Plan implementation.

e. Davis-Besse Emergency Plan implementation.

f. Fire Protection Program implementation.

g. The radiological environmental monitoring program.

h. Deleted.

i. Offsite Dose Calculation Manual implementation.

6.8.2 Each procedure of 6.8.1 above, and changes thereto, shall be reviewed and approved prior to implementation as set forth in 6.5.3 above.

See ITS 5.4

6.8.3 Deleted

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

5.5.2 a. Primary Coolant Sources Outside Containment

A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include makeup, letdown, seal injection, seal return, low pressure injection, containment spray, high pressure injection, waste gas, primary sampling and reactor coolant drain systems. The program shall include the following:

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

A01

6.0 ADMINISTRATIVE CONTROLS

6.8.4.a (Continued)

The provisions of SR 3.0.2 are applicable.

A02

5.5.2.b (ii) Integrated leak test requirements for each system at refueling cycle intervals or less.

b. ~~In-Plant Radiation Monitoring~~

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

~~1) Training of personnel,~~

~~2) Procedures for monitoring, and~~

~~3) Provisions for maintenance of sampling and analysis equipment.~~

LA01

c. Deleted

5.5.3 d. Radioactive Effluent Controls Program

A program shall be provided conforming with 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to MEMBERS OF THE PUBLIC from radioactive effluents as low as reasonably achievable. The program (1) shall be contained in the ODCM, (2) shall be implemented by operating procedures, and (3) shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

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

5.5.3.b 2) Limitations on the concentrations of radioactive material released in liquid effluents to UNRESTRICTED AREAS conforming to 10 CFR Part 20, Appendix B, Table II, Column 2,

5.5.3.c 3) Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM.

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

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ITS 5.5

ITS

6.0 ADMINISTRATIVE CONTROLS

6.8.4.d (Continued)

5.5.3.e 5) Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days.

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

5.5.3.g 7) Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the SITE BOUNDARY conforming to the doses associated with 10 CFR Part 20, Appendix B, Table II, Column 1,

5.5.3.h 8) Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50,

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

5.5.3.j 10) Limitations on the annual dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources conforming to 40 CFR Part 190.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluent Controls Program Surveillance Frequencies. A02

e. Radiological Environmental Monitoring Program

A program shall be provided to monitor the radiation and radionuclides in the environs of the plant. The program shall provide (1) representative measurements of radioactivity in the highest potential exposure pathways, and (2) verification of the accuracy of the effluent monitoring program and modeling of environmental exposure pathways. The program shall (1) be contained in the ODCM, (2) conform to the guidance of Appendix I to 10 CFR Part 50, and (3) include the following: LA02

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

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

A01

ITS 5.5

ITS

6.0 ADMINISTRATIVE CONTROLS

6.8.4.e (Continued)

3) ~~Participation in an Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.~~ LA02

5.5.10 f. Ventilation Filter Testing Program (VFTP):

A program shall be established to implement the following required testing of safety related filter ventilation systems in accordance with Regulatory Guide 1.52, Revision 2*, ANSI/ASME N510-1980, and ASTM D 3803-1989.

5.5.10.a 1) Demonstrate for each of the safety related systems that an in-place test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass < 1% when tested in accordance with Regulatory Guide 1.52, Revision 2 and ANSI/ASME N510-1980 at the system flowrate specified below, +/- 10%.

Safety Related Ventilation System	Flowrate
Station ~~Shield Building~~ Emergency Ventilation System	8000 cfm
Control Room Emergency Ventilation System	3300 cfm

A03

5.5.10.b 2) Demonstrate for each of the safety related systems that an in-place test of the charcoal adsorber shows a penetration and system bypass < 1% when tested in accordance with Regulatory Guide 1.52, Revision 2 and ANSI/ASME N510-1980 at the system flowrate specified below, +/-10%.

Safety Related Ventilation System	Flowrate
Station ~~Shield Building~~ Emergency Ventilation System	8000 cfm
Control Room Emergency Ventilation System	3300 cfm

A03

~~* The periodic testing for the Shield Building Emergency Ventilation System and the Control Room Emergency Ventilation System are performed once each REFUELING INTERVAL. The need for testing following painting, a fire, or a chemical release in any ventilation zone communicating with the Shield Building Emergency Ventilation System or the Control Room Emergency Ventilation System is as specified by the VFTP. The method of testing is based on Regulatory Guide 1.52, Revision 2, except for charcoal laboratory testing which will be performed in accordance with ASTM D 3803-1989.~~ A04

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6.0 ADMINISTRATIVE CONTROLS

6.8.4.f (Continued)

5.5.10.c 3) Demonstrate for each of the safety related systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 2, shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D 3803-1989 at a temperature of 30° C and the relative humidity (RH) specified below.

Safety Related Ventilation System	Penetration	RH
Station ~~Shield Building~~ Emergency Ventilation System	≤ 2.5%	95%
Control Room Emergency Ventilation System	≤ 2.5%	70%

A03

5.5.10.d 4) Demonstrate for each of the safety related 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 ANSI/ASME N510-1980 at the system flowrate specified below, +/- 10%.

Safety Related Ventilation System	Delta P	Flowrate
Station ~~Shield Building~~ Emergency Ventilation System	6 inches Water Gauge	8000 cfm
Control Room Emergency Ventilation System	4.4 inches Water Gauge	3300 cfm

A03

5.5.10 The provisions of SR 4.0.2 and SR 4.0.3 are applicable to the VFTP test frequencies.

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6.8.4 (Continued)

5.5.8 g. Steam Generator (SG) Program

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

5.5.8.a 1) 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 a SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging or repair of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected, plugged, or repaired to confirm that the performance criteria are being met.

5.5.8.b 2) 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.

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

5.5.8.b.2 b. 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 1 gpm per SG, except during a SG tube rupture.

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

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6.0 ADMINISTRATIVE CONTROLS

6.8.4.g (Continued)

5.5.8.c 3) Provisions for SG tube repair criteria :

5.5.8.c.1 a. Tubes found by inservice inspection to contain flaws, in a region of the tube that contains no repair, with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged or repaired.

5.5.8.c.2 b. Sleeves found by inservice inspection to contain flaws, in a region of the sleeve that contains no sleeve joint, with a depth equal to or exceeding 40% of the nominal sleeve wall thickness shall be plugged.

5.5.8.c.3 c. Tubes with a flaw, in either the parent tube or the sleeve, within a sleeve-to-tube joint shall be plugged.

5.5.8.c.4 d. Tubes with a flaw in a repair roll shall be plugged.

5.5.8.d 4) Provisions for SG tube inspections: Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. The tube-to-tubesheet weld is not part of the tube. For tubes that have undergone repair rolling, the tube and tube roll, outboard of the new roll area in the tube sheet, can be excluded from inspections because it is no longer part of the pressure boundary once the repair roll is installed. For tubes that have undergone sleeving repairs, the segment of the parent tube between the bottom of the upper-most sleeve roll and the top of the middle sleeve roll can be excluded from inspection because it is no longer part of the pressure boundary once the sleeve is installed. In addition to meeting the requirements of 4.a through 4.e below, the inspection scope, inspection methods and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

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

5.5.8.d.2 b. Inspect 100% of the tubes at sequential periods of 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. No SG shall operate for more than 24 effective full power months or one interval between refueling outages (whichever is less) without being inspected.

6.0 ADMINISTRATIVE CONTROLS

6.8.4.g.4 (Continued)

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

5.5.8.d.4 d. During each periodic SG tube inspection, inspect 100% of the tubes that have been repaired by the repair roll process. This special inspection shall be limited to the repair roll joint and the roll transitions of the repair roll.

5.5.8.d.5 e. Inspect peripheral tubes in the vicinity of the secured internal auxiliary feedwater header between the upper tube sheet and the 15th tube support plate during each periodic SG tube inspection. The tubes selected for inspection shall represent the entire circumference of the steam generator and shall total at least 150 peripheral tubes.

5.5.8.e 5) Provisions for monitoring operational primary to secondary leakage.

5.5.8.f 6) Provisions for SG tube repair methods: Steam generator 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.

5.5.8.f.1 a. Sleeving in accordance with Topical Report BAW-2120P.

5.5.8.f.2 b. Repair rolling in accordance with Topical Report BAW-2303P, Revision 4. The new roll area must be free of flaws in order for the repair to be considered acceptable.

5.5.8.g 7) Special visual inspections: Visual inspections of the secured internal auxiliary feedwater header, header to shroud attachment welds, and the external header thermal sleeves shall be performed on each SG through the auxiliary feedwater injection penetrations. These inspections shall be performed during the third period of each ten-year Inservice Inspection Interval (ISI).

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ITS 5.5

6.0 ADMINISTRATIVE CONTROLS

5.5.1 **6.15 OFFSITE DOSE CALCULATION MANUAL (ODCM)**

5.5.1.c Changes to the ODCM:

5.5.1.c.1 a. Shall be documented and records of reviews performed shall be retained as required by the USAR Chapter 17 Quality Assurance Program. This documentation shall contain: LA03

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

5.5.1.c.1.b) 2) A determination that the change will maintain the level of radioactive effluent control required by 10 CFR 20.1302, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50 and not adversely impact the accuracy or reliability of effluent, dose or setpoint calculations.

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

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

ITS

A01

ITS 5.5

6.0 ADMINISTRATIVE CONTROLS

6.16 CONTAINMENT LEAKAGE RATE TESTING PROGRAM

5.5.15.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 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:

5.5.15.a.1 1) A reduced duration Type A test may be performed using the criteria and Total Time method specified in Bechtel Topical Report BN-TOP-1, Revision 1.

5.5.15.a.2 2) The fuel transfer tube blind flanges (containment penetrations 23 and 24) will not be eligible for extended test frequencies. Their Type B test frequency will remain at 30 months. However, As-found testing will not be required.

5.5.15.b b. The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 38 psig.

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

5.5.15.d d. Leakage rate acceptance criteria are:

5.5.15.d.1 1) Containment leakage rate acceptance criterion is < 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are ≤ 0.75 L_a for Type A tests, < 0.60 L_a for all penetrations and valves subject to Type B and Type C tests~~,~~ and ≤ 0.03 L_a for all penetrations that are secondary containment bypass leakage paths; See ITS 3.6.3

2) A single penetration leakage rate of ≤ 0.15 L_a for each containment purge penetration; See ITS 3.6.3

5.5.15.d.2 3) Air lock acceptance criteria are:

5.5.15.d.2.a) a) Overall air lock leakage rate is ≤ 0.015 L_a when tested at ≥ P_a,

5.5.15.d.2.b) b) For each door, seal leakage rate is ≤ 0.01 L_a when the volume between the door seals is pressurized to ≥ 10 psig.

e. ~~The provisions of Specification 4.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.~~ A05

5.5.15.e f. The provisions of Specification 4.0.3 are applicable to the Containment Leakage Rate Testing Program.

ITS A01 ITS 5.5

6.0 ADMINISTRATIVE CONTROLS

5.5.13 6.17 TECHNICAL SPECIFICATIONS (TS) BASES CONTROL PROGRAM

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

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

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

5.5.13.b.1 1) A change in the TS incorporated in the license or

5.5.13.b.2 2) A change to the USAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

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

5.5.13.d d. Proposed changes that meet the criteria of 6.17b.1 and 6.17b.2 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).

Add proposed ITS 5.5.5 and ITS 5.5.14 M01

Add proposed ITS 5.5 16 and ITS 5.5.17 A06

ITS

A01

DEFINITIONS

1.29 Deleted

1.30 Deleted

1.31 Deleted

OFFSITE DOSE CALCULATION MANUAL (ODCM)

5.5.1.a, 5.5.1.b 1.32 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.8.4 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Radioactive Effluent Release Reports required by Specifications 6.9.1.10 and 6.9.1.11.

1.33 Deleted

1.34 Deleted

1.35 Deleted

1.36 Deleted

MEMBER(S) OF THE PUBLIC

1.37 MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the utility, its contractors or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational or other purposes not associated with the plant. See ITS Chapter 1.0

SITE BOUNDARY

1.38 The SITE BOUNDARY shall be that line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee.

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ITS

A01

ITS 5.5

APPLICABILITY

SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance Requirements shall be applicable during the OPERATIONAL MODES or other conditions specified for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement.

4.0.2 Each Surveillance Requirement shall be performed within the specified time interval with a maximum allowable extension not to exceed 25 percent of the specified surveillance interval.

See ITS 3.0

4.0.3 Failure to perform a Surveillance Requirement within the allowed surveillance interval, defined by Specification 4.0.2, shall constitute noncompliance with the OPERABILITY requirements for a Limiting Condition for Operation (LCO), except as noted below.

If it is discovered that a Surveillance was not performed within its specified frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified frequency, whichever is greater. This delay period is permitted to allow performance of the Surveillance. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours, and the risk impact shall be managed.

If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable ACTIONS must be entered.

When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be declared not met, and the applicable ACTIONS must be entered.

Surveillance requirements do not have to be performed on inoperable equipment.

4.0.4 Entry into an OPERATIONAL MODE or other specified applicability condition shall not be made unless the Surveillance Requirement(s) associated with the Limiting Condition for Operation have been performed within the stated surveillance interval or as otherwise specified.

5.5.7 4.0.5 Surveillance Requirements for inservice ~~inspection and~~ testing of ASME Code Class 1, 2 and 3 ~~components~~ pumps and valves shall be applicable as follows:

LA04

~~a. Inservice inspection of ASME Code Class 1, 2 and 3 components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a.~~

Inservice Testing of ASME Code Class 1, 2 and 3 pumps and valves shall be performed in accordance with the ASME Code for Operation and Maintenance of Nuclear Power Plants (ASME OM Code) and applicable Addenda as required by 10 CFR 50, Section 50.55a.

LA05

ITS

A01

ITS 5.5

APPLICABILITY

SURVEILLANCE REQUIREMENTS (Continued)

5.5.7a b. Surveillance intervals specified in ~~Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda and the~~ ASME OM Code and applicable Addenda shall be applicable as follows in these Technical Specifications: LA04

~~ASME Boiler and Pressure Vessel Code and the~~ ASME OM Code and applicable Addenda terminology for inservice ~~inspection and~~ testing criteria	Required frequencies for performing inservice ~~inspection and~~ testing activities
~~Weekly~~	~~At least once per 7 days~~
Monthly	At least once per 31 days
~~Semi-quarterly~~	~~At least once per 46 days~~
Quarterly or every 3 months	At least once per 92 days
~~Semiannually or every 6 months~~	~~At least once per 184 days~~
~~Every 9 months~~	~~At least once per 276 days~~
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

LA04 LA04 LA04 LA04 LA04 LA04

5.5.7.b c. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice ~~inspection and~~ testing activities. LA04

~~d. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements.~~ A07

5.5.7.d e. Nothing in the ~~ASME Boiler and Pressure Vessel Code or the~~ ASME OM Code shall be construed to supersede the requirements of any Technical Specification. LA04

Add proposed ITS 5.5.7.c A08

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A01

ITS 5.5

ITS

REACTOR COOLANT SYSTEM

3.4.10 STRUCTURAL INTEGRITY

ASME CODE CLASS 1, 2 and 3 COMPONENTS

LIMITING CONDITION FOR OPERATION

3.4.10.1 The structural integrity of ASME Code Class 1, 2 and 3 components shall be maintained in accordance with Specification 4.4.10.1.

APPLICABILITY: All MODES.

ACTION:

a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 50 °F above the minimum temperature required by NDT considerations.

b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200 °F.

c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the component(s) to within its limit or isolate the affected component(s) from service.

d. The provisions of Specification 3.0.4 are not applicable.

See CTS 3/4.4.10.1

SURVEILLANCE REQUIREMENTS

4.4.10.1 In addition to the requirements of Specification 4.0.5:

5.5.6 a. Inservice inspection of each reactor coolant pump flywheel shall be performed at least once every 10 years. The inservice inspection shall be either an ultrasonic examination of the volume from the inner bore of the flywheel to the circle of one-half the outer radius, or a surface examination of exposed surfaces of the disassembled flywheel. The recommendations delineated in Regulatory Guide 1.14, Revision 1, August 1975, Positions 3, 4 and 5 of Section C.4.b shall apply.

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ITS

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

5.5.4.a b. Each internals vent valve shall be demonstrated OPERABLE at least once per 24 months* ~~during shutdown~~ by: A09

5.5.4.a.1 1. Verifying through visual inspection that the valve body and valve disc exhibit no abnormal degradation,

5.5.4.a.2 2. Verifying the valve is not stuck in an open position, and

5.5.4.a.3 3. Verifying through manual actuation that the valve is fully open when a force of ≤ 400 lbs. is applied vertically upward.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Reactor Vessel Internals Vent Valves Program Surveillance Frequencies. A10

~~* An exception applies for the interval following the March 2003 verification completed during the Thirteenth Refueling Outage. Under this exception, the next performance of this surveillance requirement may be delayed until March 25, 2006.~~ A11

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ITS

A01

ITS 5.5

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

1. Verifying the fuel level in the day fuel tank. See ITS 3.8.1

2. Verifying the fuel level in the fuel storage tank. See ITS 3.8.3

3. Verifying the fuel transfer pump can be started and transfers fuel from the storage system to the day tank.

4. Verifying the diesel starts and accelerates up to 900 rpm, preceded by an engine prelube and/or appropriate other warmup procedures.

5. Verifying the generator is synchronized, loaded to ≥ 1000 kw, and operates for ≥ 60 minutes. See ITS 3.8.1

6. Verifying the diesel generator is aligned to provide standby power to the associated essential busses.

7. Verifying that the automatic load sequence timer is OPERABLE with each load sequence time within ± 10% of its required value.

A12 Add proposed ITS 5.5.11 generic program statement

5.5.12 b. At least once per 92 days by verifying that a sample of diesel fuel from the fuel storage tank is within the acceptable limits specified in Table 1 of ASTM D975-68 when checked for viscosity, water and sediment. L01 M02

c. At least once per 184 days by: See ITS 3.8.1 and ITS 3.8.2

1. Verifying the fuel level in the day fuel tank. See ITS 3.8.1

2. Verifying the fuel level in the fuel storage tank. See ITS 3.8.3

3. Verifying the fuel transfer pump can be started and transfers fuel from the storage system to the day tank.

4. Verifying the diesel starts from ambient condition and accelerates to at least 900 rpm in ≤ 10 seconds.

5. Verifying the generator is synchronized, loaded to ≥ 1000 kw, and operates for ≥ 60 minutes. See ITS 3.8.1

6. Verifying the diesel generator is aligned to provide standby power to the associated essential busses.

7. Verifying that the automatic load sequence timer is OPERABLE with each load sequence time within ± 10% of its required value.

DAVIS-BESSE, UNIT 1 3/4 8-3 Amendment No. ~~75, 97, 105,~~ 203

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program Surveillance Frequencies. A12

ITS

A01

ITS 5.5

RADIOACTIVE EFFLUENTS

LIQUID HOLDUP TANKS*

LIMITING CONDITION FOR OPERATION

Add proposed ITS 5.5.10 generic program statement

A13

5.5.11, 5.5.11.b 3.11.1 The quantity of radioactive material contained in each of the following unprotected outdoor tanks shall be limited to less than or equal to 10 curies, excluding tritium and dissolved or entrained noble gases.

a. Outside temporary tank.

~~APPLICABILITY: At all times.~~

~~ACTION:~~

~~a. With the quantity of radioactive material in any of the above listed tanks exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit, and describe the event leading to this condition in the next Radioactive Effluent Release Report.~~

~~b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.~~

LA06

SURVEILLANCE REQUIREMENTS

5.5.11.b 4.11.1 The quantity of radioactive material contained in each of the above listed tanks shall be determined to be within the above limit by analyzing a representative sample of the tank contents ~~at least once per 7 days when radioactive materials are being added to the tank.~~

LA06

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

A13

5.5.11.b *Tanks included in this specification are those outdoor tanks that are not surrounded by liners, dikes, or walls capable of holding the tank contents or that do not have tank overflows and surrounding area drains connected to the liquid radwaste treatment system.

A01

ITS 5.5

ITS

RADIOACTIVE EFFLUENTS

EXPLOSIVE GAS MIXTURE (Hydrogen rich systems not designed to withstand a hydrogen explosion)

LIMITING CONDITION FOR OPERATION

Add proposed ITS 5.5.10 generic program statement — A13

5.5.11, 5.5.11.a 3.11.2 The concentration of oxygen in the waste gas system shall be limited to less than or equal to 2% by volume whenever the hydrogen concentration exceeds 4% by volume. — LA06

~~APPLICABILITY: At all times.~~

~~ACTION:~~

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

~~b. With the concentration of oxygen in the waste gas system greater than 4% by volume and the hydrogen concentration greater than 4% by volume, immediately suspend all additions of waste gases to the system and reduce the concentration of oxygen to less than or equal to 2% by volume without delay.~~

~~c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.~~

LA06

SURVEILLANCE REQUIREMENTS

5.5.11.a 4.11.2 The concentrations of oxygen in the waste gas system shall be determined to be within the above limits by monitoring the waste gases in the waste gas system.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program Surveillance Frequencies. — A13

ITS

A01

ITS 5.5

- 6 -

2.C(4) Fire Protection

FENOC shall implement and maintain in effect all provisions of the approved Fire Protection Program as described in the Updated Safety Analysis Report and as approved in the SERs dated July 26, 1979, and May 30, 1991, subject to the following provision:

FENOC may make changes to the approved Fire Protection Program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

Not part of ITS conversion

5.5.9 (5) FENOC shall maintain in effect and implement a secondary water chemistry monitoring program to inhibit steam generator tube degradation. The program shall include:

5.5.9.a (a) Identification of a sampling schedule for the critical parameters and control points for these parameters;

5.5.9.b (b) Identification of the procedures used to quantify parameters that are critical to control points;

5.5.9.c (c) Identification of process sampling points;

5.5.9.d (d) Procedure for the recording and management of data;

5.5.9.e (e) Procedures defining corrective actions for off control point chemistry conditions; and

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

(6) Antitrust Conditions

FENOC and FirstEnergy Nuclear Generation Corp. shall comply with the antitrust conditions delineated in Condition 2.E of this license as if named therein. FENOC shall not market or broker power or energy from the Davis-Besse Nuclear Power Station, Unit No. 1. FirstEnergy Nuclear Generation Corp. is responsible and accountable for the actions of FENOC to the extent that said actions affect the marketing or brokering of power or energy from the Davis-Besse Nuclear Power Station, Unit No. 1, and in any way, contravene the antitrust license conditions contained in the license.

Not part of ITS conversion

L-6 Amendment No. ~~15~~, ~~47~~, ~~174~~, ~~226~~, 270

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DISCUSSION OF CHANGES
ITS 5.5, PROGRAMS AND MANUALS

ADMINISTRATIVE CHANGES

A01 In the conversion of the Davis-Besse 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-1430, Rev. 3.1, "Standard Technical Specifications-Babcock and Wilcox Plants" (ISTS).

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

A02 CTS 6.8.4.a specifies the requirements for the Primary Coolant Sources Outside Containment Program, however there is no statement as to whether or not the provisions of CTS 4.0.2 are applicable. CTS 6.8.4.d specifies the requirements for the Radioactive Effluent Controls Program, however there is no statement as to whether or not the provisions of CTS 4.0.2 and CTS 4.0.3 are applicable. ITS 5.5.2 states that the provisions of SR 3.0.2 are applicable to the Primary Coolant Sources Outside Containment Program Surveillance Frequency. ITS 5.5.3 states that the provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluent Controls Program Surveillance Frequencies. This changes the CTS by adding the allowances of ITS SR 3.0.2 to the Primary Coolant Sources Outside Containment Program and the allowances of ITS SR 3.0.2 and SR 3.0.3 to the Radioactive Effluent Controls Program.

These statements are needed to maintain allowances for Surveillance Frequency extensions contained in the ITS since ITS SR 3.0.2 and SR 3.0.3 are not normally applied to Frequencies identified in the Administrative Controls Chapter of the ITS. Since this change is a clarification required 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 technical changes to the CTS.

A03 CTS 6.8.4.f provides the requirements for the Ventilation Filter Testing Program. The Program uses the nomenclature "Shield Building Emergency Ventilation System." However, CTS 3.6.5.1 uses the nomenclature "emergency ventilation system" for the same ventilation system. ITS 5.5.10 uses the nomenclature "Station Emergency Ventilation System." This changes the CTS by using a common nomenclature for the Station Emergency Ventilation System.

This change is acceptable since it is only providing a common nomenclature for this ventilation filter system. Both ITS 3.7.12 and ITS 5.5.10 use the nomenclature "Station Emergency Ventilation System" for this system. This change is designated administrative since it does not result in any technical changes to the CTS.

A04 CTS 6.8.4.f footnote * states the periodic testing for the Shield Building Emergency Ventilation System and the Control Room Emergency Ventilation System are performed once each REFUELING INTERVAL. The need for testing following painting, a fire, or a chemical release in any ventilation zone communicating with the Shield Building Emergency Ventilation System or Control Room Emergency Ventilation System is as specified in the VFTP. The method

DISCUSSION OF CHANGES
ITS 5.5, PROGRAMS AND MANUALS

for testing is based on Regulatory Guide 1.52, Revision 2, except for charcoal laboratory testing which will be performed in accordance with ASTM D 3803-1989. ITS 5.5.10 does not contain this footnote. This changes the CTS by deleting this footnote.

This change is acceptable because no changes have been made to the existing requirements. The CTS 6.8.4.f footnote * restates what is already stated in the Ventilation Filter Testing Program description. This change is designated as administrative because it does not result in technical changes to the CTS.

A05 CTS 6.16, Containment Leakage Rate Testing Program, requires the performance of containment leakage rate testing in accordance with 10 CFR 50 Appendix J Option B, except as modified by NRC-approved exemptions, and Regulatory Guide 1.1.63, dated September 1995. CTS 6.16.e states that the provisions of Specification 4.0.2 do not apply to the test frequencies in the Containment Leakage Rate Testing Program. ITS 5.5.15 does not include this provision. This changes the CTS by deleting the statement that the provisions of Specification 4.0.2 are not applicable.

This change is acceptable because no changes have been made to the existing requirements. The statement associated with CTS 4.0.2 is not needed since the Frequency extension of ITS SR 3.0.2 is not applied to Frequencies identified in the Administrative Controls Section of the ITS, unless specifically identified. This change is designated as administrative because it does not result in technical changes to the CTS.

A06 ITS 5.5.16 provides the requirements for the Battery Monitoring and Maintenance Program. ITS 5.5.17 provides the requirements for the Control Room Envelope Habitability Program. The CTS does not include these two programs. This changes the CTS by including these two new programs.

ITS 5.5.16 has been added due to changes described in ITS 3.8.6, Discussion of Changes (DOC) L01, L02, L03, L07, and L08. ITS 5.5.17 has been added due to changes described in ITS 3.7.10, DOC L01. As such, the addition of these two programs in ITS 5.5 is acceptable and are designated as administrative because they do not result in technical changes to the CTS not already described in other ITS DOCs.

A07 CTS 4.0.5.d states that the performance of the above testing activities shall be in addition to other specified Surveillance Requirements. ITS 5.5.7 does not include a similar statement. This changes the CTS by deleting the statement.

CTS 4.0.5.d restates that all applicable requirements must be met. Repeating this overall requirement as a specific detail is redundant and unnecessary. Therefore, this detail can be omitted without any technical change in the requirements and is acceptable. This change is designated as administrative because it does not result in technical changes to the CTS.

A08 CTS 4.0.5 specifies the requirements for the Inservice Testing Program, however there is no statement whether the provisions of CTS 4.0.3 are applicable. ITS 5.5.7.c states that the provisions of SR 3.0.3 are applicable to the inservice

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ITS 5.5, PROGRAMS AND MANUALS

testing activities. This changes the CTS by adding the allowances of ITS SR 3.0.3 to the Technical Specification Inservice Testing Program requirements.

This statement is needed to maintain allowances for Surveillance Frequency extensions contained in the ITS since ITS SR 3.0.3 is not normally applied to Frequencies identified in the Administrative Controls Chapter of the ITS. Since this change is a clarification required 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.

A09 CTS 4.4.10.1.b requires reactor vessel internals vent valves to be tested every 24 months "during shutdown." ITS 5.5.4 requires similar testing every 24 months, but does not include the "during shutdown" requirement. This changes the CTS by deleting the "during shutdown" testing requirement.

This change is acceptable since the reactor vessel internals vent valves can only be tested when the unit is shutdown and the reactor vessel head removed. Therefore, stating that the unit must be shutdown is redundant and unnecessary. This change is designated as administrative because it does not result in any technical changes to the CTS.

A10 The internal vent valves requirements in CTS 4.4.10.1.b have been placed in a program in the proposed Administrative Controls Chapter 5.0 (ITS 5.5.4). As such a general program statement of applicability of ITS SR 3.0.2 and SR 3.0.3 is needed to clarify the allowances for Surveillance Frequency extensions do apply. This changes the CTS by specifically stating the applicability of ITS SR 3.0.2 and SR 3.0.3.

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

A11 CTS 4.4.10.b.1 requires reactor vessel internals vent valves to be tested every 24 months. CTS 4.4.10.1.b is modified by footnote *, which states that an exception applies for the interval following March 2003 verification completed during the Thirteenth Refueling Outage. Under this exception, the next performance of the surveillance requirement may be delayed until March 25, 2006. ITS 5.5.4 does not contain this footnote. This changes the CTS by deleting the footnote.

CTS 4.4.10.b footnote * is an exception for the Thirteenth Refueling Outage. Since this refueling outage has been completed and the test has been performed by March 25, 2006 as required by the footnote, there is no need to maintain the footnote. This change is designated as administrative because it does not result in technical changes to the CTS.

A12 The Surveillance associated with diesel fuel oil testing (CTS 4.8.1.1.2.b) has been placed in a program in the proposed Administrative Controls Chapter 5.0 (ITS 5.5.12). As such, a general program statement has been added as

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ITS 5.5, PROGRAMS AND MANUALS

ITS 5.5.12. Also, a statement of the applicability of ITS SR 3.0.2 and SR 3.0.3 is needed to clarify that the allowances for Surveillance Frequency extension do apply. This changes the CTS by moving the diesel fuel oil testing Surveillance 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 Surveillance. 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 LCO and SR sections of the CTS, therefore it is considered acceptable. This change is designated as administrative because it does not result in technical changes to the CTS.

A13 The liquid holdup tank requirements in CTS 3/4.11.1 and the explosive gas mixture requirements of CTS 3/4.11.2 have been placed in a program in the proposed Administrative Controls Chapter 5.0 (ITS 5.5.11). As such, a general program statement has been added. Also, a statement of applicability of ITS SR 3.0.2 and SR 3.0.3 is needed to clarify the allowances for Surveillance Frequency extensions do apply. This changes the CTS by moving liquid holdup tank requirements and the explosive gas mixture requirements to a program in ITS 5.5.11 and specifically stating the applicability of ITS SR 3.0.2 and SR 3.0.3 in the program.

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

MORE RESTRICTIVE CHANGES

M01 The CTS does not include program requirements for an Allowable Operating Transient Cycles Program or a Safety Function Determination Program. The ITS includes programs for these activities. This changes the CTS by adding the Allowable Operating Transient Cycles Program and Safety Function Determination Program (SFDP).

The Allowable Operating Transient Cycles Program provides controls to track the UFSAR Section 5, cyclic and transient occurrences to ensure that components are maintained within the design limits. The Safety Function Determination Program is included to support implementation of the support system OPERABILITY characteristics of the Technical Specifications. The specific wording associated with these programs are found in ITS 5.5.5 and ITS 5.5.14. 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 CTS 4.8.1.1.2.b requires verifying every 92 days that a sample of diesel fuel from the fuel oil storage tank is within the acceptable limits specified in Table 1 of

DISCUSSION OF CHANGES
ITS 5.5, PROGRAMS AND MANUALS

ASTM D975-68 when checked for viscosity, water, and sediment. In addition, no testing is currently required on new fuel oil prior to addition to the fuel oil storage tank. ITS 5.5.12.a restricts the acceptability of new fuel oil for use prior to addition to storage tanks by requiring the determination that the fuel oil has an API gravity or a absolute specific gravity within limits, a flash point and kinematic viscosity within limits, and either a clean and bright appearance with proper color or a water and sediment content within limits. ITS 5.5.12.b requires all other properties of new fuel to be verified within 31 days following addition of the new fuel oil to the storage tank. ITS 5.5.12.c requires the total particulate concentration of the stored fuel oil to be ≤ 10 mg/l when tested every 31 days. This changes the CTS by providing restrictions on the acceptability of new fuel oil prior to addition to the fuel oil storage tank and after addition to the fuel oil storage tank, and providing a requirement that the total particulate concentration of the stored fuel oil be ≤ 10 mg/l when tested every 31 days.

The purpose of ITS 5.5.12.a and ITS 5.5.12.b are to ensure that only high quality fuel oil is added to the fuel oil storage tank. The purpose of ITS 5.5.12.c is to ensure that the quality of the stored fuel oil is satisfactory for long term operation of the EDGs. The change is acceptable because the proposed Surveillances are sufficient to ensure high quality fuel oil is placed and maintained in the storage tank. 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 4 – Removal of LCO, SR, or Other TS Requirement to the TRM, UFSAR, ODCM, QAPM, IST Program, or IIP)* CTS 6.8.4.b, "In-Plant Radiation Monitoring," describes a program to ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. ITS 5.5 does not include this program. This changes the CTS by moving the requirements for the In-Plant Radiation Monitoring Program to the Technical Requirements Manual (TRM).

The removal of this requirement from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The CTS 6.8.4.b program is designed to minimize radiation exposure to plant personnel in vital areas of the plant after an accident, and has no impact on nuclear safety or the health and safety of the public. This change is acceptable because the program requirements will be adequately controlled in the TRM. The TRM is currently incorporated by reference into the UFSAR, thus 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 requirement change because requirements are being removed from the Technical Specifications.

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LA02 *(Type 4 – Removal of LCO, SR, or Other TS Requirement to the TRM, UFSAR, ODCM, QAPM, IST Program, or IIP)* CTS 6.8.4.e, "Radiological Environmental Monitoring Program," describes a program to monitor the radiation and radionuclides in the environs of the plant. ITS 5.5 does not include this program. This changes the CTS by moving the requirements for the Radiological Environmental Monitoring Program to the Offsite Dose Calculation Manual (ODCM).

The purpose of CTS 6.8.4.e is to provide representative measurements of radioactivity in the highest potential exposure pathways, and verification of the accuracy of the effluent monitoring program and modeling of environmental exposure pathways. The removal of the requirement for this program from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. ITS 5.6.1 still requires an annual report of the results of the "Radiological Environmental Monitoring Program." Also, this change is acceptable because these types of procedural details will be adequately controlled in the ODCM. Changes to the ODCM are controlled by the ODCM change control process in ITS 5.5.1, which ensures changes are properly evaluated. This change is designated as a less restrictive removal of requirement change because the requirements for a program are being removed from the Technical Specifications.

LA03 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 6.15.1.a requires changes to the ODCM to be documented and records of reviews performed to be retained as required by the UFSAR Chapter 17 Quality Assurance Program. ITS 5.5.1.c.1 requires changes to the ODCM to be documented and records of reviews performed to be retained. This changes the CTS by moving the record retention requirement reference to the Quality Assurance Program Manual (QAPM).

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. ITS 5.5.1 still retains the requirement for changes to the ODCM. Also, this change is acceptable because these types of procedural details will be adequately controlled in the QAPM. Any changes to the QAPM are made under 10 CFR 50.54(a), which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LA04 *(Type 4 - Removal of LCO, SR, or other TS requirement to the TRM, UFSAR, ODCM, QAPD, or IIP)* CTS 4.0.5 provides requirements for the Inservice Inspection Program. The ITS does not include Inservice Inspection Program requirements. In addition, since the Inservice Testing Program is the only requirement remaining, the reference to ASME Code Class 1, 2, and 3 "components" has been changed to "pumps and valves" for clarity. Pumps and valves are the only components related to the Inservice Testing Program (as

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ITS 5.5, PROGRAMS AND MANUALS

described in CTS 4.0.5.a). This changes the CTS by moving these requirements from the Technical Specifications to the Inservice Inspection Program (IIP).

The removal of these requirements 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 Technical Specifications still retain requirements for the affected components to be OPERABLE. Also, this change is acceptable because these requirements will be adequately controlled by the IIP, which is required by 10 CFR 50.55a. Compliance with 10 CFR 50.55a is required by the Davis-Besse Operating License. This change is designated as a less restrictive removal of requirement change because requirements are being removed from the Technical Specifications.

LA05 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 4.0.5.a specifies that the Inservice Testing of ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with the ASME Code for Operation and Maintenance of Nuclear Power Plants (ASME OM Code) and applicable Addenda as required by 10 CFR 50, Section 50.55a. ITS 5.5.7 states that the Inservice Testing Program provides controls for inservice testing of ASME Code Class 1, 2, and 3 pumps and valves. This changes the CTS by moving these procedural details from the Technical Specifications to the Inservice Testing Program.

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 requirements for the control for inservice testing of ASME Code Class 1, 2, and 3 pumps and valves. Also, this change is acceptable because these types of details will be adequately controlled in the plant controlled Inservice Testing Program. Changes to the Inservice Testing Program will be controlled by the provisions of 10 CFR 50.55a. This change is designated as a less restrictive removal of detail change because the details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LA06 *(Type 3 – Removing Procedural Detail for Meeting TS Requirements or Reporting Requirements)* CTS 3/4.11.1 includes the details for implementing the requirements for the liquid holdup tanks. CTS 3/4.11.2 includes the details for implementing the requirements for the explosive gas mixture. The details for implementing these requirements, including the specific limits for the explosive gas mixture, are not included in the ITS. The ITS only includes a requirement to maintain a program for these requirements. This changes the CTS by moving these procedural details for implementing the requirements, including the specific limits, from the Technical Specifications to the Technical Requirements Manual (TRM).

The removal of these details for the specific explosive gas limits, Applicability, Actions, and Surveillance Requirements from the Technical Specifications is acceptable because this type of information is not necessary to be included in the

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ITS 5.5, PROGRAMS AND MANUALS

Technical Specifications to provide adequate protection of public health and safety. ITS 5.5.11 still retains the requirement to include a program, which provides controls for potentially explosive gas mixtures contained in the Waste Gas System and the quantity of radioactivity contained in outdoor temporary liquid storage tanks. Also, this change is acceptable because these types of procedural details will be adequately controlled in the TRM. The TRM is currently incorporated by reference into the UFSAR, thus 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.

LESS RESTRICTIVE CHANGES

L01 *(Category 5 – Deletion of Surveillance Requirement)* CTS 4.8.1.1.2.b requires verifying every 92 days that a sample of diesel fuel from the fuel oil storage tank is within the acceptable limits specified in Table 1 of ASTM D975-68 when checked for viscosity, water, and sediment. ITS 5.5.12.c only requires total particulate concentration of the stored fuel oil to be tested every 31 days. This changes the CTS by deleting the quarterly viscosity, water, and sediment checks of stored fuel oil.

The purpose of CTS 4.8.1.1.2.b is to ensure that the quality of the stored diesel fuel oil is acceptable so that the emergency diesel generators can perform their safety function. This change is acceptable because the new Surveillance Requirements (added as described in Discussion of Change M02) provide an acceptable level of equipment reliability. ITS 5.5.12.a restricts the acceptance of new fuel oil for use prior to addition to storage tank until the determination that the fuel oil has an API gravity or an absolute specific gravity within limits, a flash point and kinematic viscosity within limits, and either a clear and bright appearance with proper color or a water and sediment content within limits. ITS 5.5.12.b requires all other properties of new fuel to be verified within 31 days following addition of the new fuel oil to the storage tank. ITS 5.5.12.a and ITS 5.5.12.b will ensure that the new fuel oil is of high quality. Fuel oil degradation during long term storage shows up as an increase in particulate, mostly due to oxidation. Therefore, total particulate concentration of the fuel oil is determined and compared to an acceptable limit every 31 days as required by ITS 5.5.12.c. The presence of particulate does not mean that the fuel oil will not burn properly in a diesel engine but the particulate can cause fouling of filters and fuel oil injection equipment, however, which can cause engine failure. This test is required to be performed every 31 days since fuel oil degradation trends that indicate that particulate concentration is unlikely to change significantly between the 31 day Frequency interval. In addition, ITS SR 3.8.3.5 has been added (see Discussion of Change M04 for ITS 3.8.3) to ensure that microbiological fouling does not occur. Microbiological fouling is also a major cause of fuel oil degradation. There are numerous bacteria that can grow in fuel oil and cause fouling, but all must have a water environment in order to survive. The new Surveillance has been added to ensure the removal of water from the fuel storage tank once every 31 days to eliminate the necessary environment for

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bacterial survival. This change is designated as less restrictive because a Surveillance required in the CTS will not be required in the ITS.

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

CTS

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

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

Definition 1.32 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 ; 1

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

6.15 c. Licensee initiated changes to the ODCM: 3

6.15.a 1. Shall be documented and records of reviews performed shall be retained. This documentation shall contain: 3

6.15.a.1) a) Sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s), and ; 1 3

6.15.a.2) b) 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, ; 3 1

6.15.b 2. Shall become effective after the approval of the plant manager, and ; 1 3

6.15.c 3. 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. 3

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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 INSERT 1 [Low Pressure Injection, Reactor Building Spray, Makeup and Purification, and Hydrogen Recombiner]. The program shall include the following: 2

6.8.4.a(i) a. Preventive maintenance and periodic visual inspection requirements, and ; 1

6.8.4.a(ii) b. Integrated leak test requirements for each system at least once per [18] 24 months. 2

The provisions of SR 3.0.2 are applicable.

~~[5.5.3 Post Accident Sampling~~

~~-----REVIEWER'S NOTE-----~~
~~This program may be eliminated based on the implementation of BAW-2387, "Justification for the Elimination of the Post Accident Sampling System From the Licensing Bases of Babcock and Wilcox-Designed Plants," and the associated NRC Safety Evaluation.~~

~~This program provides controls that ensure the capability to obtain and analyze reactor coolant, radioactive gases, and particulates in plant gaseous effluents and containment atmosphere samples under accident conditions. The program shall include the following:~~ 5

~~a. Training of personnel,~~

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

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

6.8.4.d 5.5.4 3 Radioactive Effluent Controls Program 5

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:

6.8.4.d.1) 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. ; 1

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2 **INSERT 1**

makeup, letdown, seal injection, seal return, low pressure injection, containment spray, high pressure injection, waste gas, primary sampling, and reactor coolant drain systems

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5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued) 5

3

6.8.4.d.2) 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; 4 1

6.8.4.d.3) 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; 1

6.8.4.d.4) 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; 1

and projected

6.8.4.d.5) 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; 4 1

6.8.4.d.6) 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; 1

6.8.4.d.7) 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:~~ 4

Appendix B, Table 2, Column 1 to 10 CFR 20;

~~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,~~ 4

6.8.4.d.8) 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; 1

6.8.4.d.9) 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 1

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5.5 Programs and Manuals

5.5.~~4~~ 3 Radioactive Effluent Controls Program (continued) 5

6.8.4.d.10) 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 ~~s~~urveillance ~~f~~requenc~~y~~ ies. 13

INSERT 2 8

5.5.5 ~~Component Cyclic or Transient Limit~~ Allowable Operating Transient Cycles Program

This program provides controls to track the U FSAR, Section ~~[ ]~~ 5, cyclic and transient occurrences to ensure that components are maintained within the design limits. 6 2

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

~~This program provides controls for monitoring any tendon degradation in pre-stressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial operations. The Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with Section XI, Subsection IWL of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR50.55a, except where an alternative, exemption, or relief has been authorized by the NRC.~~ 7

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

4.4.10.1.a 5.5.~~7~~ 6 Reactor Coolant Pump Flywheel Inspection Program 7

This program shall provide for the inspection of each reactor coolant pump flywheel ~~per the recommendation of Regulatory position c.4.b of Regulatory Guide 1.14, Revision 1, August 1975~~. INSERT 3 14

4.0.5 5.5.~~8~~ 7 Inservice Testing Program 7

4.0.5 This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 ~~components~~ pumps and valves. ~~The program shall include the following:~~ 9

4.0.5.b a. Testing frequencies applicable to the ASME Code for Operations and Maintenance of Nuclear Power Plants (ASME OM Codes) and applicable Addenda as follows:

CTS

8 **INSERT 2**

4.4.10.1.b 5.5.4 Reactor Vessel Internals Vent Valves Program

A program shall be established to implement the testing of the reactor vessel internals vent valves every 24 months as follows:

4.4.10.1.b.1 a. Verify by visual inspection that the valve body and valve disc exhibit no abnormal degradation;

4.4.10.1.b.2 b. Verify the valve is not stuck in an open position; and

4.4.10.1.b.3 c. Verify by manual actuation that the valve is fully open when a force of $\leq$ 400 lbs is applied vertically upward.

DOC A11 The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Reactor Vessel Internals Vent Valves Program test Frequencies.

14 **INSERT 3**

Inservice inspection of each reactor coolant pump flywheel shall be performed every 10 years. The inservice inspection shall be either an ultrasonic examination of the volume from the inner bore of the flywheel to the circle of one-half the outer radius, or a surface examination of exposed surfaces of the disassembled flywheel. The recommendations delineated in Regulatory Positions C.4.b(3), (4), and (5) of Regulatory Guide 1.14, Revision 1, August 1975, shall apply.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Reactor Coolant Pump Flywheel Inspection Program Surveillance Frequency.

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5.5 Programs and Manuals

5.5.8 Inservice Testing Program (continued) 7

7

4.0.5.b

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

4.0.5.c b. The provisions of SR 3.0.2 are applicable to the above required as 2 years or less TSTF -497 Frequencies and other normal and accelerated Frequencies specified in the Inservice Testing Program for performing inservice testing activities; 1

DCO A09 c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and 1

4.0.5.e d. Nothing in the ASME OM Code shall be construed to supersede the requirements of any TS.

6.8.4.g 5.5.9 Steam Generator (SG) Program 7

8

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

6.8.4.g.1) 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 2 outage during which the SG tubes are inspected, plugged, [or repaired] to confirm that the performance criteria are being met. 2

CTS

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Program (continued) 7

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6.8.4.g.2) b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.

6.8.4.g.2)a 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 cool down and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

6.8.4.g.2)b 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 [1 gpm] per SG [, except for specific types of degradation at specific locations as described in paragraph c of the Steam Generator Program]. 2

except during a steam generator tube rupture

6.8.4.g.2)c 3. The operational LEAKAGE performance criterion is specified in LCO 3.4.13, "RCS Operational LEAKAGE."

: ¶ 1.

6.8.4.g.3) 6.8.4.g.3)a c. Provisions for SG tube repair criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding [40%] of the nominal tube wall thickness shall be plugged [or repaired]. 15 2 1

;

, in a region of the tube that contains no repair.

INSERT 3A 15

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(15) **INSERT 3A**

6.8.4.g.3)b 2. Sleeves found by inservice inspection to contain flaws, in a region of the sleeve that contains no sleeve joint, with a depth equal to or exceeding 40% of the nominal sleeve wall thickness shall be plugged;

6.8.4.g.3)c 3. Tubes with a flaw, in either parent tube or the sleeve, within a sleeve to tube joint shall be plugged; and

6.8.4.g.3)d 4. Tubes with a flaw in a repair roll shall be plugged.

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5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Program (continued) 7

8

-----REVIEWER'S NOTE-----
Alternate tube repair criteria currently permitted by plant technical specifications are listed here. The description of these alternate tube 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 repair criteria. 10

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

1. . . .]

6.8.4.g.4) d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. The tube-to-tubesheet weld is not part of the tube. INSERT 4 In addition to meeting the requirements of d.1, ~~d.2, and d.3~~ through d.5 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations. 15

-----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. 10

6.8.4.g.4)a 1. Inspect 100% of the tubes in each SG during the first refueling outage following SG replacement.

6.8.4.g.4)b [2. Inspect 100% of the tubes at sequential periods of 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. No SG shall operate for more than 24 effective full power months or one refueling outage (whichever is less) without being inspected.] 2 2

Insert Page 5.5-6

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15

INSERT 4

6.8.4.g.4) For tubes that have undergone repair rolling, the tube and tube roll, outboard of the new roll area in the tube sheet, can be excluded from inspections because it is no longer part of the pressure boundary once the repair roll is installed. For tubes that have undergone sleeving repairs, the segment of the parent tube between the upper-most sleeve roll and the top of the middle sleeve roll can be excluded from inspection because it is no longer part of the pressure boundary once the sleeve is installed.

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5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Program (continued) 7

8

[2. Inspect 100% of the tubes at sequential periods of 120, 90, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 48 effective full power months or two refueling outages (whichever is less) without being inspected.] 2 15

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

6.8.4.g.4)c 3. If crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

INSERT 4A 15

6.8.4.g.5) e. Provisions for monitoring operational primary to secondary LEAKAGE.

6.8.4.g.6) f. Provisions for SG tube repair methods. Steam generator 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. 2

-----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. 10

1. [...] 15

INSERT 5

CTS

15 INSERT 4A

6.8.4.g.4)d 4. During each periodic SG tube inspection, inspect 100% of the tubes that have been repaired by the repair roll process. This special inspection shall be limited to the repair roll joint and the roll transitions of the roll repair.

6.8.4.g.4)e 5. Inspect peripheral tubes in the vicinity of the secured internal auxiliary feedwater header between the upper tube sheet and the 15th tube support plate during each periodic SG tube inspection. The tubes selected for inspection shall represent the entire circumference of the the steam generator and shall total at least 150 peripheral tubes.

15 INSERT 5

6.8.4.g.6)a 1. Sleeving in accordance with Topical Report BAW-2120P.

6.8.4.g.6)b 2. Repair rolling in accordance with Topical Report BAW-2303P, Revision 4. The new roll area must be free of flaws in order for the repair to be considered acceptable.

6.8.4.g.7) g. Special visual inspections: Visual inspections of the secured internal auxiliary feedwater header, header to shroud attachment welds, and the external header thermal sleeves shall be performed on each SG through the auxiliary feedwater injection penetrations. These inspections shall be performed during the third period of each 10 year Inservice Inspection Interval (ISI).

5.5 Programs and Manuals

License Condition 2.C(5)

5.5.~~10~~ 9 **Secondary Water Chemistry Program** 7

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: 16

License Condition 2.C(5)(a) — a. Identification of a sampling schedule for the critical variables and control points for these variables~~,~~ ; 1

License Condition 2.C(5)(b) — b. Identification of the procedures used to measure the values of the critical variables~~,~~ ; 1

License Condition 2.C(5)(c) — c. Identification of process sampling points~~, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in leakage~~~~,~~ ; 16 1

License Condition 2.C(5)(d) — d. Procedures for the recording and management of data~~,~~ ; 1

License Condition 2.C(5)(e) — e. Procedures defining corrective actions for all off control point chemistry conditions~~,~~ ; and 1

License Condition 2.C(5)(f) — 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.

6.8.4.f

5.5.~~11~~ 10 **Ventilation Filter Testing Program (VFTP)** 7

A program shall be established to implement the following required testing of ~~Engineered Safety Feature (ESF)~~ safety related filter ventilation systems ~~at the frequencies specified in [Regulatory Guide], and~~ in accordance with ~~[~~Regulatory Guide 1.52, Revision 2, ANSI/ASME N510-~~1989, and AG-1]~~ 1980, and ASTM D 3803-1989: 17

6.8.4.f.1) — a. Demonstrate for each of the ~~ESF~~ safety related systems that an inplace test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass < ~~[0.05]~~ 1.0% when tested in accordance with ~~[~~Regulatory Guide 1.52, Revision 2, and ANSI/ASME N510-~~1989]~~ 1980 at the system flowrate specified below ~~[± 10%]~~.

~~ESF Ventilation System~~	~~Flowrate~~
~~[ ]~~	~~[ ]~~

INSERT 6

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17 **INSERT 6**

	Safety Related Ventilation System	Flowrate (cfm)
6.8.4.f.1)	Station Emergency Ventilation System (EVS)	≥ 7200 and ≤ 8800
	Control Room Emergency Ventilation System (CREVS)	≥ 2970 and ≤ 3630

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CTS

5.5 Programs and Manuals

5.5.11 Ventilation Filter Testing Program (continued) 7

10

safety related

1.0

6.8.4.f.2) b. Demonstrate for each of the ~~ESF~~ systems that an inplace test of the charcoal adsorber shows a penetration and system bypass < ~~[0.05]~~% when tested in accordance with [Regulatory Guide 1.52, Revision 2, and ANSI/ ASME N510-~~1989~~] 1980 at the system flowrate specified below ~~[± 10%]~~.

ESF Ventilation System	Flowrate
[ ]	[ ]

INSERT 7

safety related

6.8.4.f.3) c. Demonstrate for each of the ~~ESF~~ systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in [Regulatory Guide 1.52, Revision 2], shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F) and the relative humidity (RH) specified below. 17

INSERT 8

ESF Ventilation System	Penetration	RH	Face Velocity
[ ]	[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.~~

~~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 Efficiently * for Charcoal Credited in Licensee's Accident Analysis) / Safety Factor]~~ 10

~~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.~~

~~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.~~

CTS

17 **INSERT 7**

CTS	Safety Related Ventilation System	Flowrate (cfm)
6.8.4.f.2)	Station EVS	≥ 7200 and ≤ 8800
	CREVS	≥ 2970 and ≤ 3630

17 **INSERT 8**

CTS	Safety Related Ventilation System	Penetration (%)	RH (%)
6.8.4.f.3)	Station EVS	≤ 2.5	95
	CREVS	≤ 2.5	70

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CTS

5.5 Programs and Manuals

5.5.~~11~~ 10 Ventilation Filter Testing Program (continued) (7) (10)

~~If the system has a face velocity greater than 110 percent of 0.203 m/s (40 ft/min), the face velocity should be specified.~~

~~*This value should be the efficiency that was incorporated in the licensee's accident analysis which was reviewed and approved by the staff in a safety evaluation.~~

6.8.4.f.4) d. Demonstrate for each of the ~~ESF~~ safety related 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 ANSI/ASME N510-~~1989~~ 1980] at the system flowrate specified below ~~[± 10%]~~. (17)

INSERT 9

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

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

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

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

3.11.1, 3.11.2 5.5.~~12~~ 11 [Explosive Gas and Storage Tank Radioactivity Monitoring Program (7) (2)

This program provides controls for potentially explosive gas mixtures contained in the [Waste Gas ~~Holdup~~ System], ~~[the quantity of radioactivity contained in gas storage tanks or fed into the offgas treatment system,~~ and the quantity of radioactivity contained in unprotected outdoor temporary 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) (11)

CTS

17 **INSERT 9**

	Safety Related Ventilation System	Delta P (inches wg)	Flowrate (cfm)
6.8.4.f.4)	Station EVS	< 6	≥ 7200 and ≤ 8800
	CREVS	< 4.4	≥ 2970 and ≤ 3630

CTS

5.5 Programs and Manuals

3.11.1, 3.11.2 5.5.~~12~~ 11 Explosive Gas and Storage Tank Radioactivity Monitoring Program (continued) 7

The program shall include:

3.11.2, 4.11.2 a. The limits for concentrations of hydrogen and oxygen in the [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), ; and 2 1

~~b. A surveillance program to ensure that the quantity of radioactivity contained in [each gas storage tank and fed into the offgas treatment system] is less than the amount that would result in a whole body exposure of ≥ 0.5 rem to any individual in an unrestricted area, in the event of [an uncontrolled release of the tanks' contents], and~~ 11

3.11.1, 4.11.1 ~~c.~~ b. A surveillance program to ensure that the quantity of radioactivity contained in all outdoor temporary liquid ~~radwaste~~ storage 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.~~ ≤ 10 Ci, excluding tritium and dissolved or entrained noble gases. 11 18 19

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program ~~S~~surveillance ~~F~~frequencies. 13 2

4.8.1.1.2.b 5.5.~~13~~ 12 Diesel Fuel Oil Testing Program 7

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 ; 1
2. A flash point and kinematic viscosity within limits for ASTM 2D fuel oil ; and 1
3. A clear and bright appearance with proper color , or a water and sediment content within limits ; 13 1

CTS

5.5 Programs and Manuals

4.8.1.1.2.b 5.5.~~13~~ 12 Diesel Fuel Oil Testing Program (continued) 7

b. Within 31 days following addition of the new fuel oil to storage tanks, verify that the properties of the new fuel oil, other than those addressed in a., above, are within limits for ASTM 2D fuel oil~~,~~ ; ~~and~~ 1

c. Total particulate concentration of the fuel oil is ≤ 10 mg/l when tested every 31 days.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program testing frequencies. 13

6.17 5.5.~~14~~ 13 Technical Specifications (TS) Bases Control Program 7

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

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

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

6.17.b.1) 1. A change in the TS incorporated in the license~~,~~ ; ~~or~~ 1

6.17.b.2) 2. A change to the ~~updated~~ U FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59. 20

6.17.c c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the U FSAR. 20

6.17.d d. Proposed changes that meet the criteria of Specification 5.5.~~14~~ 13.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). 21

DOC M01 5.5.~~15~~ 14 Safety Function Determination Program (SFDP) 7

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. ¶ a. The SFDP shall contain the following: 3

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CTS

5.5 Programs and Manuals

DOC M01 5.5.~~15~~ 14 Safety Function Determination Program (continued) 7

~~a~~ 1. 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~~,~~ ; 3 1

~~b~~ 2. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists~~,~~ ; 3 1

~~c~~ 3. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities~~,~~ ; and 3 1

~~d~~ 4. Other appropriate limitations and remedial or compensatory actions. 3

b. 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 3 1

~~a~~ 1. A required system redundant to the system(s) supported by the inoperable support system is also inoperable~~,~~ ; or 3 1

~~b~~ 2. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable~~,~~ ; or 3 1

~~c~~ 3. A required system redundant to the support system(s) for the supported systems ~~(a) and (b)~~ described in Specifications 5.5.14.b.1 and 5.5.14.b.2 above is also inoperable. 3 21

c. 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. 3

6.16 5.5.~~16~~ 15 Containment Leakage Rate Testing Program 7

~~[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.~~ 22

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

CTS

5.5 Programs and Manuals

5.5.~~16~~ 15 Containment Leakage Rate Testing Program (continued) 7

c. Leakage rate acceptance criteria are:

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

2. Air lock testing acceptance criteria are:

a) Overall air lock leakage rate is ≤ [0.05 L_a] when tested at ≥ P_a. 22

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

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

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

~~[OPTION B]~~ 22

6.16.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 B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September, 1995, as modified by the following exceptions:

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

INSERT 10

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. 23

[3. . . .]

CTS

23 **INSERT 10**

6.16.a.1) 1. A reduced duration Type A test may be performed using the criteria and Total Time method specified in Bechtel Topical Report BN-TOP-1, Revision 1.

6.16.a.2) 2. The fuel transfer tube blind flanges (containment penetrations 23 and 24) will not be eligible for extended test frequencies. Their Type B test frequency will remain at 30 months. However, as-found testing will not be required.

CTS

5.5 Programs and Manuals

5.5.~~16~~ 15 Containment Leakage Rate Testing Program (continued) 7

6.16.b b. The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a, is ~~[45 psig]~~ 38 psig. ~~The containment design pressure is [50 psig].~~ 23

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

6.16.d d. Leakage rate acceptance criteria are:

6.16.d.1) 1. Containment leakage rate acceptance criterion is ~~≤~~ < 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are ~~<~~ ≤ 0.60 L_a for the Type B and C tests and ≤ 0.75 L_a for Type A tests. 13 23

6.16.d.3) 2. Air lock testing acceptance criteria are:

6.16.d.3)a) a) Overall air lock leakage rate is ≤ ~~[0.05~~ 0.015 L_a] when tested at ≥ P_a. 2

6.16.d.3)b) b) For each door, leakage rate is ≤ [0.01 L_a] when the volume between the door seals is pressurized to [≥ 10 psig]. 2 23

6.16.f 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.~~ 12

~~[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:~~ 22

~~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.~~

CTS

5.5 *Programs and Manuals*

5.5.~~16~~ 15 Containment Leakage Rate Testing Program (continued) 7

2. The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR50, Appendix J, Option B, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWE, except where relief has been authorized by the NRC.

[3. . . .]

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

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

d. Leakage rate acceptance criteria are:

1. Containment leakage rate acceptance criterion is $\leq$ 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are < 0.60 L_a for the Type B and C tests and [< 0.75 L_a for Option A Type A tests] [$\leq$ 0.75 L_a for Option B Type A tests]. 22

2. Air lock testing acceptance criteria are:

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

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

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

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

CTS

5.5 Programs and Manuals

DOC A06

5.5.~~17~~ 16 **Battery Monitoring and Maintenance Program** 7

This Program provides for battery restoration and maintenance, based on [the recommendations of IEEE Standard 450-1995, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications," or of the battery manufacturer] including the following: 2 2

a. Actions to restore battery cells with float voltage < [2.13] V, and 2

b. Actions to equalize and test battery cells that had been discovered with electrolyte level below the ~~minimum established design limit~~ top of the plates TSTF-451T

INSERT 10 TSTF-448

CTS

TSTF-448 **INSERT 10**

DOC A06 5.5.~~18~~ 17 Control Room Envelope Habitability Program 7

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 whole body or its equivalent to any part of the body] ~~[5 rem total effective dose equivalent (TEDE)]~~ for the duration of the accident. The program shall include the following elements: 2

a. The definition of the CRE and the CRE boundary. ; 1

b. Requirements for maintaining the CRE boundary in its design condition including configuration control and preventive maintenance. ; 1

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 Section 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. ; 1

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

~~1. ; and]~~ 2

d. Measurements, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by one train of the CREVS, operating at the flow rate required by the VFTP, at a Frequency of ~~[18]~~ 24 months on a STAGGERED TEST BASIS. The results shall be trended and used as part of the ~~[18]~~ 24 month assessment of the CRE boundary. ; 2 2 1

CTS

TSTF-448

INSERT 10 (continued)

DOC A06

e. The quantitative limits on unfiltered air leakage 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~~ Specification 5.5.17.c. The unfiltered air inleakage limit for radiological challenges is the inleakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air inleakage limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis~~.~~ ; and 21 1

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~~ Specifications 5.5.17.c and 5.5.17.d, respectively.

JUSTIFICATION FOR DEVIATIONS
ITS 5.5, PROGRAMS AND MANUALS

1. These punctuation corrections have been made consistent with the Writer's Guide for the Improved Standard Technical Specifications, TSTF-GG-05-01, Section 5.1.3.

2. The brackets have been removed and the proper plant specific information/value has been provided.

3. This Specification has been renumbered to be consistent with the ITS format and for clarity.

4. Changes are made to be consistent with the Radioactive Effluent Controls Program limits in the Davis-Besse CTS.

5. The bracketed ISTS 5.5.3, "Post Accident Sampling," is not included in the Davis-Besse ITS. The requirement for Post Accident Sampling was deleted from the CTS in License Amendment 264, dated June 10, 2005. Subsequent programs have been renumbered, as necessary.

6. Changes made to be consistent with the current Program, which is controlled in the Davis-Besse Technical Requirements Manual.

7. ISTS 5.5.7 provides requirements for the Pre-Stressed Concrete Containment Tendon Surveillance Program. This bracketed requirement regarding Pre-Stressed Concrete Containment Tendon Surveillance Program is deleted because it is not applicable to Davis-Besse. The Davis-Besse containment does not utilize pre-stressed concrete containment tendons. Subsequent programs have been renumbered, as necessary.

8. ITS 5.5.4, "Reactor Vessel Internals Vent Valves Program," has been added to the ITS. The design of the Davis-Besse reactor vessel includes internal vent valves. This requirement is currently a Surveillance Requirement in CTS 3/4.4.10.1. It has been included as a program similar to the Reactor Coolant Pump Flywheel Inspection Program, which is also a Surveillance Requirement in CTS 3/4.4.10.1.

9. The Inservice Testing (IST) Program (ISTS 5.5.7) has been modified to state that the IST Program provides control for ASME Code Class 1, 2, and 3 "pumps and valves" in place of the current "components." 10 CFR 50.55a(f) provides the regulatory requirements for an IST Program. It specifies that ASME Code Class 1, 2, and 3 pumps and valves are the only components covered by an IST Program. 10 CFR 50.55a(g) provides regulatory requirements for an Inservice Inspection (ISI) Program. It specifies that ASME Code Class 1, 2, and 3 components are covered by the ISI Program, and that pumps and valves are covered by the IST Program in 10 CFR 50.55a(f). The ISTS does not include ISI Program requirements as these requirements have been relocated to a plant specific document. Therefore, the components to which the IST Program applies (i.e., pumps and valves) have been added for clarity. In addition, the statement "The program shall include the following:" has been deleted because not all of the statements that follow are really part of the program requirements. Furthermore, the terms weekly, semiannually, and every 9 months have been deleted since these terms are not used in the ASME OM Code.

JUSTIFICATION FOR DEVIATIONS
ITS 5.5, PROGRAMS AND MANUALS

10. The Reviewer's Note has been deleted since it is not intended to be included in the ITS.

11. The program details of the Explosive Gas and Storage Tank Radioactivity Monitoring Program are described in ISTS 5.5.12.a and 5.5.12.b (ITS 5.5.11.a and 5.5.11.b). Therefore, the sentence in the introductory paragraph that specifies a method to determine the explosive gas and storage tank radioactivity is not necessary. Additionally, the requirements specified in ISTS 5.5.12.b do not apply to Davis Besse. UFSAR Section 15 states that a waste gas decay tank release is not a credible accident and that the dose will remain within the 10 CFR 100 guidelines. This is also consistent with the CTS.

12. ISTS 5.5.16 (ITS 5.5.15) provides the requirements for the Containment Leakage Rate Testing Program. The statement in ISTS 5.5.16.f that "Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J" has been deleted. This phrase is not consistent with the allowances in ISTS 5.5.16.a (ITS 5.5.15.a), which states that the 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." These exceptions stated in ITS 5.5.15.a are modifications to the testing Frequencies required by 10 CFR 50, Appendix J. In addition, there is no need to state any specific exception to any of the other requirements of the Specifications that discuss testing frequencies, because the convention of application of requirements in the sections of ISTS 5.5 is that no other Specification requirements apply unless otherwise stated. For example, ISTS SR 3.0.2 does not apply to any of the ITS 5.5 sections, unless specifically noted. Therefore, there is no need to include a statement that ITS SR 3.0.2 does not apply to the Frequencies of ITS 5.5.15.

13. Typographical/grammatical error corrected.

14. The Davis-Besse plant-specific reactor coolant pump flywheel inspection requirements have been provided. These requirements were approved by the NRC in License Amendment 232, dated June 8, 1999. An allowance to apply the provisions of ITS SR 3.0.2 and ITS SR 3.0.3 has been provided, consistent with the current licensing basis (the program is currently a Surveillance, thus these allowances apply).

15. Changes are made to be consistent with the current Steam Generator Program requirements in the Davis-Besse CTS.

16. Changes are made to be consistent with the current Secondary Water Chemistry Program requirements in Davis-Besse License Condition 2.C(5).

17. Changes are made to be consistent with the current Ventilation Filter Testing Program requirements in the Davis-Besse CTS.

18. Changes are made to ISTS 5.5.12.c (ITS 5.5.11.b) to be consistent with the first paragraph in ISTS 5.5.12 (ITS 5.5.11) and with the current licensing basis.

JUSTIFICATION FOR DEVIATIONS
ITS 5.5, PROGRAMS AND MANUALS

19. Changes are made to be consistent with the current limits for the outdoor temporary liquid storage tanks in the Davis-Besse CTS.

20. Changes are made to the ISTS which reflect the plant specific nomenclature.

21. Editorial changes for consistency. These items and paragraphs are Specifications.

22. Davis-Besse complies with Option B of 10 CFR 50, Appendix J. Therefore, the Option A and combined Option A and B provisions have been deleted.

23. The Davis-Besse exceptions have been provided, consistent with the current licensing basis. The containment design pressure limit specified in ISTS 5.5.16.b has not been included because it currently does not exist in the Davis-Besse CTS, and because this limit does not provide any useful input to the Containment Leakage Rate Testing Program. The Davis-Besse specific limits for Types B and C leakage ($\leq 0.60\ L_a$) have been provided, consistent with current licensing basis. Furthermore, the specific manner in which the air lock door seal test is performed has been included, consistent with current licensing basis.

Specific No Significant Hazards Considerations (NSHCs)

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

There are no specific NSHC discussions for this Specification.

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ATTACHMENT 6

ITS 5.6, REPORTING REQUIREMENTS

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

6.0 ADMINISTRATIVE CONTROLS

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

5.6 6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted ~~to the appropriate Regional Office unless otherwise noted~~. in accordance with 10 CFR 50.4 A02

STARTUP REPORT

6.9.1.1 Deleted.

6.9.1.2 Deleted.

6.9.1.3 Deleted.

ANNUAL OPERATING REPORT

~~6.9.1.4 Annual reports covering the activities of the unit during the previous calendar year shall be submitted prior to March 31 of each year.~~ L01

~~6.9.1.5 Reports required on an annual basis shall include:~~

a. Deleted

b. Deleted

~~c. The results of specific activity analysis in which the primary coolant exceeded the limits of Specification 3.4.8. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded; (2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up system flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.~~ L01

MONTHLY OPERATING REPORT

6.9.1.6 Deleted

ITS

A01

ITS 5.6

6.0 ADMINISTRATIVE CONTROLS

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 before each reload cycle and any remaining part of a reload cycle for the following:

2.1.2	AXIAL POWER IMBALANCE Protective Limits for Reactor Core Specification 2.1.2
2.2.1	Trip Setpoint for Flux -- ΔFlux/Flow for Reactor Protection System Setpoints Specification 2.2.1
3.1.1.3c	Negative Moderator Temperature Coefficient Limit
3.1.3.6	Regulating Rod Insertion Limits
3.1.3.7	Rod Program
3.1.3.8	Xenon Reactivity
3.1.3.9	Axial Power Shaping Rod Insertion Limits
3.2.1	AXIAL POWER IMBALANCE
3.2.2	Nuclear Heat Flux Hot Channel Factor, F_Q
3.2.3	Nuclear Enthalpy Rise Hot Channel Factor, $F^N_{\Delta H}$
3.2.4	QUADRANT POWER TILT

A03

LCO 3.1.1, "SHUTDOWN MARGIN,"
LCO 3.1.7, "Position Indicator Channels," (SR 3.1.7.1 limits)
LCO 3.1.8, "PHYSICS TEST Exceptions – MODE 1,"
LCO 3.1.9, "PHYSICS TEST Exceptions – MODE 2," LCO 3.9.1, "Boron Concentration"

A04

5.6.3.b The analytical methods used to determine the core operating limits addressed by the individual Technical Specifications shall be: those previously reviewed and approved by the NRC, as described in BAW-10179P-A, "Safety Criteria and Methodology for Acceptable Cycle Reload Analyses", or any other new NRC-approved analytical methods used to determine core operating limits that are not yet referenced in the applicable approved revision of BAW-10179P-A. The applicable approved revision number for BAW-10179P-A at the time the reload analyses are performed shall be identified in the CORE OPERATING LIMITS REPORT. The CORE OPERATING LIMITS REPORT shall also list any new NRC-approved analytical methods used to determine core operating limits that are not yet referenced in the applicable approved revision of BAW-10179P-A.

INSERT 1

A08

5.6.3.d The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

5.6.3.e The CORE OPERATING LIMITS REPORT, including any mid-cycle revision 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.

A05

ITS

A08 **INSERT 1**

5.6.3.c As described in reference documents listed in accordance with the instructions given above, when an initial assumed power level of 102% of RATED THERMAL POWER is specified in a previously approved method, an actual value of 100.37% of RATED THERMAL POWER may be used when the input for reactor thermal power measurement of feedwater mass flow and temperature is from the Ultrasonic Flow Meter. The following NRC approved documents are applicable to the use of the Ultrasonic Flow Meter with a 0.37% measurement uncertainty:

Caldon Inc. Engineering Report-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM√™ System," Revision 0, dated March, 1997.

Caldon Inc. Engineering Report-157P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM√™ or LEFM CheckPlus™ System," Revision 5, dated October, 2001.

ITS — A01 — ITS 5.6

6.0 ADMINISTRATIVE CONTROLS

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT

5.6.1 6.9.1.10 The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted ~~before May 1~~ by May 15 of each year. The report shall include summaries, interpretations, and analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in (1) the ODCM, and (2) Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50. L02

Add proposed ITS 5.6.1, second paragraph M01

RADIOACTIVE EFFLUENT RELEASE REPORT

5.6.2 6.9.1.11 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 (1) consistent with the objectives outlined in the ODCM and the Process Control Program, and (2) in conformance with 10 CFR 50.36a and Section IV.B.1 of Appendix I to 10 CFR Part 50.

STEAM GENERATOR TUBE INSPECTION REPORT

6.9.1.12 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.g, Steam Generator (SG) Program. The report shall include:

a. The scope of inspections performed on each SG,

b. Active degradation mechanisms found,

c. Nondestructive examination techniques utilized for each degradation mechanism,

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

e. Number of tubes plugged or repaired during the inspection outage for each active degradation mechanism,

f. Total number and percentage of tubes plugged or repaired to date,

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

h. The effective plugging percentage for all plugging and tube repairs in each SG, and

i. Repair method utilized and the number of tubes repaired by each repair method.

Add proposed ITS 5.6.4 L03

Add proposed ITS 5.6.5 A06

A01

ITS 5.6

6.0 ADMINISTRATIVE CONTROLS

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the U.S. Nuclear Regulatory Commission in accordance with 10 CFR 50.4 within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specifications:

a. ECCS Actuation, Specifications 3.5.2 and 3.5.3.

b. Deleted

c. Deleted

d. Deleted

e. Deleted

f. Deleted

g. Inoperable Remote Shutdown System control circuit(s) or transfer switch(es) required for a serious control room or cable spreading room fire, Specification 3.3.3.5.2.

A07

6.10 RECORD RETENTION

Records of facility activities shall be retained as described in the USAR Chapter 17 Quality Assurance Program.

See CTS 6.0

6.11 Deleted

6.12 HIGH RADIATION AREA

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

See ITS 5.7

6.12.1 High radiation areas with dose rates not exceeding 1.0 rem/hour at 30 centimeters from the radiation source or from any surface penetrated by the radiation:

a. Each entry way 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.

DISCUSSION OF CHANGES
ITS 5.6, REPORTING REQUIREMENTS

ADMINISTRATIVE CHANGES

A01 In the conversion of the Davis-Besse 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-1430, Rev. 3.1, "Standard Technical Specifications-Babcock and Wilcox 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 requires, in addition to the requirements of 10 CFR, reports be submitted to the Regional Office. ITS 5.6 requires that the reports be submitted in accordance with 10 CFR 50.4. This changes the CTS by removing the explicit requirement to send reports to the Regional Office.

10 CFR 50.4 provides distribution requirements for written communications to the NRC. This change is acceptable because the requirements deleted from the Technical Specifications are already required by 10 CFR 50.4. This change is designated as administrative because it does not result in technical changes to the CTS.

A03 CTS 6.9.1.7 requires, in part, that core operating limits be established and documented in the COLR for the CTS 3/4.1.3.7, "Rod Program," and CTS 3/4.1.3.8, "Xenon Reactivity." ITS 5.6.3.a does not include a reference to these Specifications. This changes the CTS by eliminating the reference to Rod Program and Xenon Reactivity limits being core operating limits that are included in the COLR.

The Rod Program Specification is being relocated to the Technical Requirements Manual (TRM) (See CTS 3/4.1.3.7 DOC R01). The Xenon Reactivity program is being removed from the ITS (See CTS 3/4.1.3.8 DOC L01). This information is not included in the ITS, as stated in the various DOCs, and is therefore not included in the list of individual Specifications that address core operating limits in ITS 5.6.3. This change is acceptable because the information contained in the individual Specifications is no longer documented in the ITS. This change is designated as administrative because it does not result in technical changes to the CTS.

A04 CTS 6.9.1.7 contains a list of the core operating limits established and documented in the COLR. ITS 5.6.3.a includes additional core operating limits established and documented in the COLR. These are LCO 3.1.1, "SHUTDOWN MARGIN (SDM)"; LCO 3.1.7, "Position Indicator Channels" (SR 3.1.7.1 limits); LCO 3.1.8, "PHYSICS TEST Exceptions – MODE 1"; LCO 3.1.9, "PHYSICS TEST Exceptions – MODE 2"; and LCO 3.9.1, "Boron Concentration." These limits had been previously addressed in the CTS, but are being moved to the COLR in the ITS, and because of this are listed in ITS 5.6.3.a. This changes the CTS by adding core operating limits established and documented in the COLR (and applicable methodology) because they are being moved there as part of changes to other parts of the CTS. Technical aspects of the changes are

DISCUSSION OF CHANGES
ITS 5.6, REPORTING REQUIREMENTS

addressed in the Discussion of Changes for the respective individual ITS Specifications.

This change is acceptable because it administratively documents changes made to other parts of the CTS and the COLR. This change is designated as administrative because it does not result in technical changes to the CTS.

A05 CTS 6.9.1.7 requires the CORE OPERATING LIMITS REPORT (COLR) to be provided to the NRC document control desk with copies to the Regional Administrator and Resident Inspector. ITS 5.6.3.d requires the COLR to be provided to the NRC. This changes the CTS by removing the specifics regarding distribution of the reports to the NRC.

10 CFR 50.4 provides distribution requirements for written communications to the NRC. This change is acceptable because the requirements deleted from the Technical Specifications are already required by 10 CFR 50.4. Furthermore ITS 5.6 states that all reports in ITS 5.6 be submitted in accordance with 10 CFR 50.4. This change is designated as administrative because it does not result in technical changes to the CTS.

A06 ITS 5.6.5, "Post Accident Monitoring Report," provides the reporting requirements when Condition B of LCO 3.3.17, "Post Accident Monitoring (PAM) Instrumentation," is entered. The CTS does not include this report. This changes the CTS by adding a new PAM Report.

This change is acceptable since the PAM Report is being added as a result of changes made to CTS 3.3.3.6. The addition of the allowance to provide a report to the NRC in lieu of a unit shutdown is discussed in ITS 3.3.17, DOC L01. This change is designated as administrative because the addition of the report is a result of a change justified in another Specification of the Davis-Besse ITS submittal.

A07 CTS 6.9.2 requires special reports be submitted to the NRC and lists the CTS Specifications that require special reports to be submitted. The ITS does not require these special reports to be prepared and submitted. This changes the CTS by deleting the references to the CTS Specifications requiring special reports. Justification for disposition of each of the special report requirements is addressed by the Discussion of Changes for the respective ITS or CTS Specification.

The purpose of CTS 6.9.2 is to identify the Specifications that require special reports to be submitted. This change is acceptable because the special reports are no longer required by the respective Specifications. Justification for disposition of each of the special report requirements is addressed by the Discussion of Changes for the respective ITS or CTS Specification (ITS 3.5.2 DOC L01, ITS 3.5.3 DOC L01, and ITS 3.3.18 DOC R01). This change is designated as administrative because it does not result in technical changes to the CTS.

DISCUSSION OF CHANGES
ITS 5.6, REPORTING REQUIREMENTS

A08 This change to CTS 6.9.1.7 is provided in the Davis-Besse ITS consistent with License Amendment Request No. 05-0007, submitted to the USNRC for approval in FENOC letter Serial Number 3198, from Mark B. Bezilla (FENOC) to USNRC, dated April 12, 2007. As such, this change is administrative.

MORE RESTRICTIVE CHANGES

M01 The second paragraph of ITS 5.6.1 includes details required to be included in the Annual Radiological Environmental Operating Report. CTS 6.9.1.10 does not contain this level of detail. This changes the CTS by requiring additional detail to be included in the Annual Radiological Environmental Operating Report.

The purpose of the second paragraph of ITS 5.6.1 is to specify details to be included in the Annual Radiological Environmental Operating Report. This change is acceptable because the content requirements are consistent with the objectives outlined in the Offsite Dose Calculation Manual. This change is designated more restrictive because it adds new reporting requirements to the Technical Specifications.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

L01 *(Category 8 – Deletion of Reporting Requirements)* CTS 6.9.1.4 requires the annual reports of CTS 6.9.1.5 to be submitted prior to March 31 of each year. CTS 6.9.1.5.c requires annual reporting of information regarding any instances when the specific activity limit of Specification 3.4.8 is exceeded. ITS 5.6 does not contain any requirements for such a report. This changes the CTS by not including the requirements for the annual reporting of instances when the Technical Specification specific activity limit for the primary coolant is exceeded.

The purpose of CTS 6.9.1.4 and CTS 6.9.1.5.c is to specify the requirements for submitting information regarding any instances when the Technical Specification specific activity of Specification 3.4.8 is exceeded in an annual report. This change is acceptable because the regulations provide adequate details of reporting requirements, and the reporting of exceeding the I-131 specific activity limit does not affect continued plant operation. Operations or conditions prohibited by the plant's Technical Specifications are required to be reported in accordance with 10 CFR 50.73. Subsequent reports would be provided if necessary, without requiring a specific annual report. This change is designated

DISCUSSION OF CHANGES
ITS 5.6, REPORTING REQUIREMENTS

as less restrictive because the reports that would be submitted under the CTS will not be required under the ITS.

L02 *(Category 1 – Relaxation of LCO Requirements)* CTS 6.9.1.10 requires the Annual Radiological Environmental Operating Report to be submitted before May 1 of each year. ITS 5.6.1 requires the Annual Radiological Environmental Operating Report to be submitted by May 15 of each year. This changes the CTS by allowing additional time to submit this report each year.

The purpose of the due date for submitting the Annual Radiological Environmental Operating Report is to ensure that the report is provided in a reasonable period of time to the NRC for review. This change is acceptable because the report is still required to be submitted in a reasonable time frame. Given that the report is still required to be provided to the NRC on or before May 15 and cover the previous calendar year, report completion and submittal is clearly not necessary to assure operation in a safe manner for the interval between May 1 and May 15. Additionally, there is no requirement for the NRC to approve the reports. This change is designated as less restrictive because it allows more time to prepare and submit the report to the NRC.

L03 CTS 3/4.4.9.1 provides the requirements for the Reactor Coolant System (RCS) Pressure/Temperature (P/T) Limits. ITS 3.4.3, "RCS Pressure and Temperature (P/T) Limits," Discussion of Change LA02 describes that the specific P/T limits, including the P/T limit curves and the maximum heatup and cooldown rates, are being relocated to the PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR). ITS 5.6.4 provides the requirements for the PTLR. This changes the CTS by adding a PTLR to the Technical Specifications.

Creation of a PTLR is consistent with the guidance provided in Generic Letter (GL) 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits. GL 96-03 requires that the P/T limits are generated in accordance with the requirements of 10 CFR 50, Appendix G, documented in an NRC-approved topical report incorporated by reference in the Technical Specifications. Accordingly, the Davis-Besse heatup/cooldown curves have been generated using the NRC-approved methods described in BAW-10046A, Rev. 2, "Methods of Compliance with Fracture Toughness and Operational Requirements of 10 CFR 50 Appendix G," and meet the requirements of 10 CFR 50, Appendix G.

Technical Specifications include a Limiting Condition for Operation (LCO) that establishes P/T limits for the RCS. The limits are defined by figures and values that provide an acceptable range of operating temperatures and pressures for heatup, cooldown, criticality, and inservice leak and hydrostatic testing conditions. These parameters are generally valid for a specified number of effective full-power years or for a specified period. License amendments are generally required at the end of the effective period for P/T limit curves or when surveillance specimens are withdrawn and tested. Processing amendment requests for changes to Technical Specification that are developed using an accepted methodology places an unnecessary burden on licensee and NRC resources. An alternative approach for controlling these limits was proposed during the development of the ISTS. This approach, like the one used for the

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DISCUSSION OF CHANGES
ITS 5.6, REPORTING REQUIREMENTS

CORE OPERATING LIMITS REPORT, would relocate the P/T curves and maximum heatup and cooldown rates to a PTLR and would reference that document in the affected LCOs and Technical Specification Bases. The guidance contained in GL 96-03 specifically requires licensees wishing to implement this line item Technical Specification improvement to:

(1) Reference a methodology for developing the curves and setpoint that has been approved by the NRC;

(2) Develop a PTLR or a similar document that contains the figures, values, parameters, and any explanations derived from the methodology; and

(3) Make appropriate changes to the applicable sections of the Technical Specifications.

The following provides a description of the Davis-Besse compliance with the listed requirements of GL 96-03:

(1) The P/T limits currently contained in CTS 3.4.9.1 and to be contained in the PTLR (applicable through 21 EFPY) were generated in accordance with the methods described in BAW-10046A, Rev. 2, "Methods of Compliance with Fracture Toughness and Operational Requirements of 10 CFR 50 Appendix G," consistent with the requirements of 10 CFR 50 Appendix G, and Regulatory Guide 1.99, Revision 2. The NRC has reviewed the methods described in BAW-10046A and approved the topical report by issuance of a Safety Evaluation Report (SER) dated April 30, 1986.

(2) Davis-Besse will develop a PTLR consistent with the requirements in ITS 5.6.4 as part of implementing the entire Improved Technical Specifications Amendment. As required by ITS 5.6.4, the initial PTLR will be provided to the NRC upon issuance by Davis-Besse as part of the implementation of the ITS Amendment.

(3) Consistent with the guidance in Generic Letter 96-03 and in the format of NUREG-1430, Rev. 3.1, Davis-Besse is providing the proposed Technical Specifications changes associated with the PTLR as part of this ITS Amendment.

The ISTS 5.6.4 Reviewer's Note also states that the methodology for the calculation of the P/T limits for NRC approval should include the following provisions:

1. The methodology shall describe how the neutron fluence is calculated (reference new Regulatory Guide when issued).

2. The Reactor Vessel Material Surveillance Program shall comply with Appendix H to 10 CFR 50. The reactor vessel material irradiation surveillance specimen removal schedule shall be provided, along with how the specimen examinations shall be used to update the PTLR curves.

DISCUSSION OF CHANGES
ITS 5.6, REPORTING REQUIREMENTS

3. Low Temperature Overpressure Protection (LTOP) System lift setting limits for the Power Operated Relief Valves (PORVs), developed using NRC-approved methodologies may be included in the PTLR.

4. The adjusted reference temperature (ART) for each reactor beltline material shall be calculated, accounting for radiation embrittlement, in accordance with Regulatory Guide 1.99, Revision 2.

5. The limiting ART shall be incorporated into the calculation of the pressure and temperature limit curves in accordance with NUREG-0800 Standard Review Plan 5.3.2, Pressure-Temperature Limits.

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

7. Licensees who have removed two or more capsules should compare for each surveillance material the measured increase in reference temperature (RT_{NDT}) to the predicted increase in RT_{NDT}; where the predicted increase in RT_{NDT} is based on the mean shift in RT_{NDT} plus the two standard deviation value ($2\sigma_{\Delta}$) specified in Regulatory Guide 1.99, Revision 2. If the measured value exceeds the predicted value (increase $RT_{NDT} + 2\sigma_{\Delta}$), the licensee should provide a supplement to the PTLR to demonstrate how the results affect the approved methodology.

The following provides a description of the Davis-Besse compliance with the listed requirements of the Reviewer's Note:

1. The neutron fluence is calculated using the methodology described in BAW-2108, Rev. 1, "Fluence Tracking System," dated May 1992. This methodology was described to the NRC in the Davis-Besse License Amendment request for the current P/T limits curves, dated January 30, 1995 (approved in Amendment 199).

2. The Davis-Besse Reactor Vessel Material Surveillance Program complies with the requirements of Appendix H to 10 CFR 50 and is described in BAW-1543A, "Master Integrated Reactor Vessel Material Surveillance Program." This information was provided to the NRC in the Davis-Besse License Amendment request for the current P/T limits curves, dated January 30, 1995.

3. PORVs are not currently used for LTOP, thus are not required in ITS 3.4.12, "Low Temperature Overpressure Protection (LTOP)." Therefore, the PTLR will not include any PORV lift setting requirements.

4. Davis-Besse calculates ART in accordance with Regulatory Guide 1.99, Revision 2. This information was also provided to the NRC in the Davis-Besse License Amendment request for the current P/T limits curves, dated January 30, 1995.

DISCUSSION OF CHANGES
ITS 5.6, REPORTING REQUIREMENTS

5. The NRC has previously determined that the current limiting ART calculation complies with NUREG-0800, Standard Review Plan 5.3.2, as documented in the NRC SER approving Amendment 199, dated July 20, 1995.

6. The NRC has previously determined that the Davis-Besse P/T limit curves comply with 10 CFR 50 Appendix G, as documented in the NRC SER approving Amendment 199, dated July 20, 1995.

7. Davis-Besse has not tested any plant-specific capsules since the last time the P/T limit curves were adjusted and approved by the NRC in Amendment 199. Therefore, no changes need to be made at this time.

Based on the above information, the proposed addition of a PTLR to the Davis-Besse Technical Specifications is acceptable. Davis-Besse will continue to meet the requirements of 10 CFR 50, Appendix G and any changes to the Davis-Besse P/T limits will be generated in accordance with the NRC-approved methodology described in BAW-10046A, Rev. 2.

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

CTS

5.0 ADMINISTRATIVE CONTROLS

6.9 5.6 Reporting Requirements

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

6.9.1.10 5.6.1 Annual Radiological Environmental Operating Report

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

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 1

Radioactive

6.9.1.11 5.6.2 ~~Radiological~~ Effluent Release Report 6

---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

The Radioactive Effluent Release Report covering the operation of the unit in the previous year shall be submitted prior to May ~~1~~ of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR Part 50, Appendix I, Section IV.B.1. 2

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5.6 Reporting Requirements

6.9.1.7 5.6.3 CORE OPERATING LIMITS REPORT (COLR)

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

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

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

~~[Identify the Topical Report(s) by number and title or identify the staff Safety Evaluation Report for a plant specific methodology by NRC letter and date. The COLR will contain the complete identification for each of the TS referenced topical reports used to prepare the COLR (i.e., report number, title, revision, date, and any supplements).]~~ INSERT 2 1

INSERT 3 7

d ~~c~~. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling System (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met. 7

e ~~d~~. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC. 7

DOC L03 5.6.4 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

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

~~[The individual specifications that address RCS pressure and temperature limits must be referenced here.]~~ INSERT 4 1

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:

~~[Identify the Topical Report(s) by number and title or identify the NRC Safety Evaluation for a plant specific methodology by NRC letter and date. The PTLR will contain the complete identification for each of the TS referenced Topical Reports used to prepare the PTLR (i.e., report number, title, revision, date, and any supplements).]~~ INSERT 5 1

CTS

1 INSERT 1

6.9.1.7 1. SL 2.1.1.1, "Reactor Core Safety Limits";
2. LCO 3.1.1, "SHUTDOWN MARGIN (SDM)";
3. LCO 3.1.3, "Moderator Temperature Coefficient (MTC)";
4. LCO 3.1.7, "Position Indicator Channels," (SR 3.1.7.1 limits);
5. LCO 3.1.8, "PHYSICS TEST Exceptions – MODE 1";
6. LCO 3.1.9, "PHYSICS TEST Exceptions – MODE 2";
7. LCO 3.2.1, "Regulating Rod Insertion Limits";
8. LCO 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits";
9. LCO 3.2.3, "AXIAL POWER IMBALANCE Operating Limits";
10. LCO 3.2.4, "QUADRANT POWER TILT (QPT)";
11. LCO 3.2.5, "Power Peaking Factors";
12. LCO 3.3.1,"Reactor Protection System (RPS) Instrumentation," Function 8 (Flux - ΔFlux – Flow) Allowable Value; and
13. LCO 3.9.1, "Boron Concentration."

1 INSERT 2

6.9.1.7 as described in BAW-10179P-A, "Safety Criteria and Methodology for Acceptable Cycle Reload Analyses," or any other new NRC approved analytical methods used to determine core operating limits that are not yet referenced in the applicable approved revision of BAW-10179P-A. The applicable approved revision number for BAW-10179P-A at the time of the reload analyses are performed shall be identified in the CORE OPERATING LIMITS REPORT (COLR). The COLR shall also list any new NRC approved analytical methods used to determine core operating limits that are not yet referenced in the applicable approved revision of BAW-10179P-A.

CTS

7 **INSERT 3**

6.9.1.7 c. As described in reference documents listed in accordance with the instructions given above, when an initial assumed power level of 102% of RTP is specified in a previously approved method, an actual value of 100.37% of RTP may be used when the input for reactor thermal power measurement of feedwater mass flow and temperature is from the Ultrasonic Flow Meter. The following NRC approved documents are applicable to the use of the Ultrasonic Flow Meter with a 0.37% measurement uncertainty:

1. Caldon Inc. Engineering Report-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM√™ System," Revision 0, dated March, 1997.

2. Caldon Inc. Engineering Report-157P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM√™ or LEFM CheckPlus™ System," Revision 5, dated October, 2001.

1 **INSERT 4**

DOC L03 1. LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits."

1 **INSERT 5**

DOC L03 1. BAW-10046A, Rev. 2, "Methods of Compliance with Fracture Toughness and Operational Requirements of 10 CFR 50 Appendix G," June 1986.

CTS

5.6 Reporting Requirements

DOC L03

5.6.4 RCS PRESSURE AND TEMPERATURE LIMITS REPORT (continued)

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

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

The methodology for the calculation of the P-T limits for NRC approval should include the following provisions:

1. The methodology shall describe how the neutron fluence is calculated (reference new Regulatory Guide when issued).

2. The Reactor Vessel Material Surveillance Program shall comply with Appendix H to 10 CFR 50. The reactor vessel material irradiation surveillance specimen removal schedule shall be provided, along with how the specimen examinations shall be used to update the PTLR curves.

3. Low Temperature Overpressure Protection (LTOP) System lift setting limits for the Power Operated Relief Valves (PORVs), developed using NRC-approved methodologies may be included in the PTLR. 3

4. The adjusted reference temperature (ART) for each reactor beltline material shall be calculated, accounting for radiation embrittlement, in accordance with Regulatory Guide 1.99, Revision 2.

5. The limiting ART shall be incorporated into the calculation of the pressure and temperature limit curves in accordance with NUREG-0800 Standard Review Plan 5.3.2, Pressure-Temperature Limits.

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

7. Licensees who have removed two or more capsules should compare for each surveillance material the measured increase in reference temperature (RT_{NDT}) to the predicted increase in RT_{NDT}; where the predicted increase in RT_{NDT} is based on the mean shift in RT_{NDT} plus the two standard deviation value ($2\sigma_\Delta$) specified in Regulatory Guide 1.99, Revision 2. If the measured value exceeds the predicted value (increase in $RT_{NDT} + 2\sigma_\Delta$), the licensee should provide a supplement to the PTLR to demonstrate how the results affect the approved methodology.

CTS

5.6 Reporting Requirements

DOC A06 5.6.5 Post Accident Monitoring Report

When a report is required by Condition B or F of LCO 3.3.[17], "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. 1

~~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.]~~ 4

6.9.1.12 5.6.7 6 Steam Generator Tube Inspection Report 4

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

a. The scope of inspections performed on each SG, ; 5

b. Active degradation mechanisms found, ; 5

c. Nondestructive examination techniques utilized for each degradation mechanism, ; 5

d. Location, orientation (if linear), and measured sizes (if available) of service induced indications, ; 5

e. Number of tubes plugged [or repaired] during the inspection outage for each active degradation mechanism, ; 1 5

f. Total number and percentage of tubes plugged [or repaired] to date, ; 1 5

g. The results of condition monitoring, including the results of tube pulls and in-situ testing, ; 5

[h. The effective plugging percentage for all plugging [and tube repairs] in each SG, and] ; 1 5

[i. Repair method utilized and the number of tubes repaired by each repair method.] 1

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JUSTIFICATION FOR DEVIATIONS
ITS 5.6, REPORTING REQUIREMENTS

1. The brackets have been removed and the proper plant specific information/value has been provided.

2. ISTS 5.6.3 requires submittal of the Radioactive Effluent Release Report prior to May 1 of each year in accordance with 10 CFR 50.36a. 10 CFR 50.36a states that the report must be submitted within one year of the previous report. The existing Davis-Besse CTS submittal date is also in accordance with 10 CFR 50.36a; a May 1 date is not provided in the CTS. Therefore, the Davis-Besse current licensing basis reporting date has been maintained.

3. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed in to what is needed to meet this requirement. This is not meant to be retained in the final version of the plant specific submittal.

4. ISTS 5.6.6 provides requirements for the Tendon Surveillance Report. The Containment design at Davis-Besse does not include pre-stressed concrete tendons. Therefore, this report is not included in the Davis-Besse ITS, consistent with the current licensing basis. Subsequent Specifications are renumbered as a result of this deletion.

5. These punctuation corrections have been made consistent with the Writer's Guide for the Improved Standard Technical Specifications, TSTF-GG-05-01, Section 5.1.3.

6. Typographical error corrected.

7. This change to ITS 5.6.3 is provided in the Davis-Besse ITS consistent with License Amendment Request No. 05-0007, submitted to the USNRC for approval in FENOC letter Serial Number 3198, from Mark B. Bezilla (FENOC) to USNRC, dated April 12, 2007. Due to this addition, the remaining paragraphs in ITS 5.6.4 have been renumbered.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 5.6, REPORTING REQUIREMENTS

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGE L03

Davis-Besse is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1430, Rev. 3.1, "Standard Technical Specifications Babcock and Wilcox Plants." The proposed change involves making the Current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the determination of No Significant Hazards Considerations for conversion to NUREG-1430.

CTS 3/4.4.9.1 provides the requirements for the Reactor Coolant System (RCS) Pressure/Temperature (P/T) Limits. ITS 3.4.3, "RCS Pressure and Temperature (P/T) Limits," Discussion of Change LA02 describes that the specific P/T limits, including the P/T limit curves and the maximum heatup and cooldown rates, are being relocated to the PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR). ITS 5.6.4 provides the requirements for the PTLR. This changes the CTS by adding a PTLR to the Technical Specifications.

Creation of a PTLR is consistent with the guidance provided in Generic Letter (GL) 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits. GL 96-03 requires that the P/T limits are generated in accordance with the requirements of 10 CFR 50, Appendix G, documented in an NRC-approved topical report incorporated by reference in the Technical Specifications. Accordingly, the Davis-Besse heatup/cooldown curves have been generated using the NRC-approved methods described in BAW-10046A, Rev. 2, “Methods of Compliance with Fracture Toughness and Operational Requirements of 10 CFR 50 Appendix G,” and meet the requirements of 10 CFR 50, Appendix G.

Technical Specifications include a Limiting Condition for Operation (LCO) that establishes P/T limits for the RCS. The limits are defined by figures and values that provide an acceptable range of operating temperatures and pressures for heatup, cooldown, criticality, and inservice leak and hydrostatic testing conditions. These parameters are generally valid for a specified number of effective full-power years or for a specified period. License amendments are generally required at the end of the effective period for P/T limit curves or when surveillance specimens are withdrawn and tested. Processing amendment requests for changes to Technical Specification that are developed using an accepted methodology places an unnecessary burden on licensee and NRC resources. An alternative approach for controlling these limits was proposed during the development of the ISTS. This approach, like the one used for the CORE OPERATING LIMITS REPORT, would relocate the P/T curves and maximum heatup and cooldown rates to a PTLR and would reference that document in the affected LCOs and Technical Specification Bases. The guidance contained in GL 96-03 specifically requires licensees wishing to implement this line item Technical Specification improvement to:

(1) Reference a methodology for developing the curves and setpoint that has been approved by the NRC;

(2) Develop a PTLR or a similar document that contains the figures, values, parameters, and any explanations derived from the methodology; and

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 5.6, REPORTING REQUIREMENTS

(3) Make appropriate changes to the applicable sections of the Technical Specifications.

The following provides a description of the Davis-Besse compliance with the listed requirements of GL 96-03:

(1) The P/T limits currently contained in CTS 3.4.9.1 and to be contained in the PTLR (applicable through 21 EFPY) were generated in accordance with the methods described in BAW-10046A, Rev. 2, "Methods of Compliance with Fracture Toughness and Operational Requirements of 10 CFR 50 Appendix G," consistent with the requirements of 10 CFR 50 Appendix G, and Regulatory Guide 1.99, Revision 2. The NRC has reviewed the methods described in BAW-10046A and approved the topical report by issuance of a Safety Evaluation Report (SER) dated April 30, 1986.

(2) Davis-Besse will develop a PTLR consistent with the requirements in ITS 5.6.4 as part of implementing the entire Improved Technical Specifications Amendment. As required by ITS 5.6.4, the initial PTLR will be provided to the NRC upon issuance by Davis-Besse as part of the implementation of the ITS Amendment.

(3) Consistent with the guidance in Generic Letter 96-03 and in the format of NUREG-1430, Rev. 3.1, Davis-Besse is providing the proposed Technical Specifications changes associated with the PTLR as part of this ITS Amendment.

The ISTS 5.6.4 Reviewer's Note also states that the methodology for the calculation of the P/T limits for NRC approval should include the following provisions:

1. The methodology shall describe how the neutron fluence is calculated (reference new Regulatory Guide when issued).

2. The Reactor Vessel Material Surveillance Program shall comply with Appendix H to 10 CFR 50. The reactor vessel material irradiation surveillance specimen removal schedule shall be provided, along with how the specimen examinations shall be used to update the PTLR curves.

3. Low Temperature Overpressure Protection (LTOP) System lift setting limits for the Power Operated Relief Valves (PORVs), developed using NRC-approved methodologies may be included in the PTLR.

4. The adjusted reference temperature (ART) for each reactor beltline material shall be calculated, accounting for radiation embrittlement, in accordance with Regulatory Guide 1.99, Revision 2.

5. The limiting ART shall be incorporated into the calculation of the pressure and temperature limit curves in accordance with NUREG-0800 Standard Review Plan 5.3.2, Pressure-Temperature Limits.

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

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 5.6, REPORTING REQUIREMENTS

7. Licensees who have removed two or more capsules should compare for each surveillance material the measured increase in reference temperature (RT_{NDT}) to the predicted increase in RT_{NDT}; where the predicted increase in RT_{NDT} is based on the mean shift in RT_{NDT} plus the two standard deviation value ($2\sigma_{\Delta}$) specified in Regulatory Guide 1.99, Revision 2. If the measured value exceeds the predicted value (increase $RT_{NDT} + 2\sigma_{\Delta}$), the licensee should provide a supplement to the PTLR to demonstrate how the results affect the approved methodology.

The following provides a description of the Davis-Besse compliance with the listed requirements of the Reviewer's Note:

1. The neutron fluence is calculated using the methodology described in BAW-2108, Rev. 1, "Fluence Tracking System," dated May 1992. This methodology was described to the NRC in the Davis-Besse License Amendment request for the current P/T limits curves, dated January 30, 1995 (approved in Amendment 199).

2. The Davis-Besse Reactor Vessel Material Surveillance Program complies with the requirements of Appendix H to 10 CFR 50 and is described in BAW-1543A, "Master Integrated Reactor Vessel Material Surveillance Program." This information was provided to the NRC in the Davis-Besse License Amendment request for the current P/T limits curves, dated January 30, 1995.

3. PORVs are not currently used for LTOP; thus are not required in ITS 3.4.12, "Low Temperature Overpressure Protection (LTOP)." Therefore, the PTLR will not include any PORV lift setting requirements.

4. Davis-Besse calculates ART in accordance with Regulatory Guide 1.99, Revision 2. This information was also provided to the NRC in the Davis-Besse License Amendment request for the current P/T limits curves, dated January 30, 1995.

5. The NRC has previously determined that the current limiting ART calculation complies with NUREG-0800, Standard Review Plan 5.3.2, as documented in the NRC SER approving Amendment 199, dated July 20, 1995.

6. The NRC has previously determined that the Davis-Besse P/T limit curves comply with 10 CFR 50 Appendix G, as documented in the NRC SER approving Amendment 199, dated July 20, 1995.

7. Davis-Besse has not tested any plant-specific capsules since the last time the P/T limit curves were adjusted and approved by the NRC in Amendment 199. Therefore, no changes need to be made at this time.

Based on the above information, the proposed addition of a PTLR to the Davis-Besse Technical Specifications is acceptable. Davis-Besse will continue to meet the requirements of 10 CFR 50, Appendix G and any changes to the Davis-Besse P/T limits will be generated in accordance with the NRC-approved methodology described in BAW-10046A, Rev. 2.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 5.6, REPORTING REQUIREMENTS

An evaluation has been performed to determine whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The proposed change is the addition of PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR). The PTLR is created in accordance with the guidance provided by Generic Letter (GL) 96-03 and is consistent with the content of NUREG-1430, Rev. 3.1. The RCS P/T limits will continue to meet the requirements of 10 CFR 50, Appendix G, and will be generated in accordance with the NRC-approved methodology described in BAW-10046A, Rev. 2, "Methods of Compliance with Fracture Toughness and Operational Requirements of 10 CFR 50 Appendix G." Since the proposed change is essentially administrative in nature and does not involve any change to any values currently required by the Technical Specifications, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   Response: No.

   As stated above, the proposed change is essentially administrative in nature. Accident initial conditions and assumptions remain as previously analyzed, and the proposed change does not introduce any new or different accident initiators. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. **Does the proposed change involve a significant reduction in a margin of safety?**

   Response: No.

   The margin of safety is not affected by the creation of a PTLR. Operation of the unit in accordance with the limits specified in the PTLR will continue to meet the requirements of 10 CFR 50, Appendix G, and will assure that a margin of safety is not significantly decreased as a result of the change. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, it is concluded that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, there is a finding of "no significant hazards consideration."

ATTACHMENT 7

ITS 5.7, HIGH RADIATION AREA

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

ITS

A01

ITS 5.7

6.0 ADMINISTRATIVE CONTROLS

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the U.S. Nuclear Regulatory Commission in accordance with 10 CFR 50.4 within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specifications:

a. ECCS Actuation, Specifications 3.5.2 and 3.5.3.

b. Deleted

c. Deleted

d. Deleted

e. Deleted

f. Deleted

g. Inoperable Remote Shutdown System control circuit(s) or transfer switch(es) required for a serious control room or cable spreading room fire, Specification 3.3.3.5.2.

See ITS 5.6

6.10 RECORD RETENTION

Records of facility activities shall be retained as described in the USAR Chapter 17 Quality Assurance Program.

See CTS 6.0

6.11 Deleted

5.7 6.12 HIGH RADIATION AREA

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

5.7.1 6.12.1 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:

5.7.1.a a. Each entry way 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.

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ITS

A01

ITS 5.7

6.0 ADMINISTRATIVE CONTROLS

6.12.1 (Continued)

5.7.1.b b. 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) and other appropriate radiation protection equipment and measures.

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

5.7.1.d d. Each individual (whether alone or in a group) entering such an area shall possess:

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

5.7.1.d.2 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

5.7.1.d.3 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

Add proposed Specification 5.7.1.d.4(i) — L01

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

5.7.1.d.4.(ii) be under the surveillance, as specified in the RWP or equivalent, while in the area, by means of closed circuit television, by 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. — M01

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

or personnel continuously escorted by such individuals, — L02

5.7.2 6.12.2 Locked high radiation areas with dose rates greater than 1.0 rem/hour at 30 centimeters from the radiation source or from any surface penetrated by the radiation, but less than 500 rads/hour at 1 meter from the radiation source or from any surface penetrated by the radiation:

L03

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

or continuously guarded door or gate — L03

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. — L02

ITS

A01

ITS 5.7

6.0 ADMINISTRATIVE CONTROLS

6.12.2.a (Continued)

5.7.2.a.1 1) All keys to such doors, gates, or other barriers shall be maintained under the administrative control of the shift supervisor, radiation protection manager, or his or her designee.

5.7.2.a.2 2) Doors, gates, or other barriers shall remain locked except during periods of personnel or equipment entry or exit.

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

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

5.7.2.d d. Each individual (whether alone or in a group) entering such an area shall possess:

5.7.2.d.1 1) 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

5.7.2.d.2 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

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

5.7.2.d.3.(i) (i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, by 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

5.7.2.d.3.(i) (ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, by personnel qualified in radiation protection procedures responsible for controlling personnel radiation exposure in the area and with the means to communicate with and control every individual in the area, or

ITS

A01

ITS 5.7

6.0 ADMINISTRATIVE CONTROLS

6.12.2.d (Continued)

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

or personnel continuously escorted by such individuals,

5.7.2.e e. Except for an individual qualified in radiation protection procedures, 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.

L02

5.7.2.f f. Such individual areas that are within a larger area that is controlled as a high radiation area, where no enclosure exists for the purpose of locking and where no enclosure can reasonably be constructed around the individual area, need not be controlled by a locked door or gate, but shall be barricaded and conspicuous, and a clearly visible flashing light shall be activated at the area as a warning device.

6.13 ENVIRONMENTAL QUALIFICATION

6.13.1 By no later than June 30, 1982 all safety-related electrical equipment in the facility shall be qualified in accordance with the provisions of Division of Operating Reactors "Guidelines for Evaluating Environmental Qualification of Class IE Electrical Equipment in Operating Reactors" (DOR Guidelines); or, NUREG-0588 "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment", December 1979. Copies of these documents are attached to Order for Modification of License NPF-3 dated October 24, 1980.

See CTS 6.0

6.13.2 By no later than December 1, 1980, complete and auditible records must be available and maintained at a central location which describe the environmental qualification method used for all safety-related electrical equipment in sufficient detail to document the degree of compliance with the DOR Guidelines or NUREG-0588. Thereafter, such records should be updated and maintained current as equipment is replaced, further tested, or otherwise further qualified.

6.14 Deleted

DISCUSSION OF CHANGES
ITS 5.7, HIGH RADIATION AREA

ADMINISTRATIVE CHANGES

A01 In the conversion of the Davis-Besse 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-1430, Rev. 3.1, "Standard Technical Specifications-Babcock and Wilcox 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 6.12.1.d.4 provides one of the options for an individual entering a high radiation area, and requires the individual to be under the surveillance, by means of closed circuit television, by personnel qualified in radiation protection procedures. ITS 5.7.1.d.4.(ii) includes a similar option; however it includes an additional requirement that the person have a means of communicating with the individuals in the high radiation area who are covered by such surveillance. This changes the CTS by requiring means to communicate with the associated individuals in the high radiation area.

The purpose of CTS 6.12.1.d.4 is to ensure personnel in high radiation areas are properly monitored. This change is acceptable because it provides additional guidance to ensure the personnel in the high radiation areas can be contacted if the need arises. This change is also consistent with a similar option provided for personnel entering a very high radiation area. This change is designated as more restrictive because additional requirements are added for personnel entering high radiation areas.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

L01 *(Category 1 – Relaxation of LCO Requirements)* CTS 6.12.1.d.4 states that each individual that enters a high radiation area with dose rates not exceeding 1.0 rem/hour at 30 centimeters from the radiation source or from any surface penetrated by the radiation shall possess a self-reading dosimeter and be under the surveillance, as specified in the RWP or equivalent, while in the area, by means of closed circuit television, by personnel qualified in radiation protection procedures responsible for controlling personnel radiation exposure in the area.

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DISCUSSION OF CHANGES
ITS 5.7, HIGH RADIATION AREA

ITS 5.7.1.d.4.(ii) retains this same requirement. However, ITS 5.7.1.d.4.(i) provides an additional option in lieu of that required by CTS 6.12.1.d.4. ITS 5.7.1.d.4.(i) states that each individual that enters a high radiation area with dose rates not exceeding 1.0 rem/hour at 30 centimeters from the radiation source or from any surface penetrated by the radiation shall possess a self-reading dosimeter and 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. This changes the CTS by allowing an individual to be monitored directly instead of indirectly (i.e., by closed circuit television) when entering a high radiation area with dose rates not exceeding 1.0 rem/hour at 30 centimeters from the radiation source or from any surface penetrated by the radiation.

The purpose of CTS 6.12.1.d.4 is to provide the method for monitoring the exposure of individuals in high radiation areas. This change is acceptable because it provides adequate means of monitoring the personnel in the high radiation areas, yet provides added flexibility for how to do it. Furthermore, this proposed option is currently allowed by CTS 6.12.2.d.3.(i) when entering high radiation areas with dose rates greater than 1.0 rem/hour at 30 centimeters. This change is designated as less restrictive because additional methods for monitoring personnel in high radiation areas have been provided.

L02 *(Category 1 – Relaxation of LCO Requirements)* CTS 6.12.1.e states that except for individuals qualified in radiation protection procedures, entry into such areas (a high radiation area with dose rates not exceeding 1.0 rem/hour at 30 centimeters from the radiation source or from any surface penetrated by the radiation) shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. CTS 6.12.2.e states that except for individuals qualified in radiation protection procedures, entry into such areas (a locked high radiation area with dose rates exceeding 1.0 rem/hour at 30 centimeters from the radiation source or from any surface penetrated by the radiation) shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. ITS 5.7.1.e states that except for individuals qualified in radiation protection procedures, "or personnel continuously escorted by such individuals," entry into such areas (a high radiation area with dose rates not exceeding 1.0 rem/hour at 30 centimeters from the radiation source or from any surface penetrated by the radiation) shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. Furthermore, ITS 5.7.1.e requires that these continuously escorted personnel will receive a pre-job briefing prior to entry into such areas, and that this dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry. ITS 5.7.2.e provides identical requirements for entry into a high radiation area with dose rates greater than 1.0 rem/hour at 30 centimeters from the radiation source or from any surface penetrated by the radiation, but less than 500rads/hours at 1 Meter from the radiation source or from any surface penetrated by the radiation. This changes the CTS by allowing additional personnel to enter high radiation areas prior to determining the current dose rates.

DISCUSSION OF CHANGES
ITS 5.7, HIGH RADIATION AREA

The purpose of CTS 6.12.1.e and CTS 6.12.2.e is to provide the entry requirements for high radiation areas. This will allow personnel to perform the task that requires entry into the high radiation area simultaneously with the radiation protection technicians performing the radiation survey. This allowance could reduce the total dose accumulated by site personnel for a given task. This change is acceptable because it provides adequate means to monitor the personnel and the personnel will receive a pre-job brief to ensure the dose is maintained as low as reasonably achievable. This change is designated as less restrictive because additional methods for monitoring personnel before dose rates in the area have been determined have been provided.

L03 *(Category 1 – Relaxation of LCO Requirements)* CTS 6.12.2 requires high radiation areas with dose rates greater than 1.0 rem/hour at 30 centimeters from the radiation source or from any surface penetrated by the radiation, but less than 500 rads/hours at 1 meter from the radiation source or from any surface penetrated by the radiation shall be locked and CTS 6.12.2.a states that the areas will be locked by a door, gate, or other barrier. ITS 5.7.2.a allows the areas to either be locked or continuously guarded by a door, gate, or other barrier. This changes the CTS by allowing the doors, gates, or other barriers to be guarded instead of being locked.

The purpose of CTS 6.12.2.a is to prevent unauthorized access to a high radiation area with dose rates greater than 1.0 rem/hour at 30 centimeters from the radiation source or from any surface penetrated by the radiation, but less than 500 rads/hours at 1 meter from the radiation source or from any surface penetrated by the radiation. This change is acceptable because adequate controls are maintained to prevent an unauthorized access, while allowing the reasonable flexibility in determining the proper methods to ensure unauthorized access. A guarded door, gate, or barrier provides a similar control over the area; an individual is providing the controls in lieu of a mechanical device (a lock). This change is designated as less restrictive because an additional method to prevent unauthorized access to a high radiation area is allowed in the ITS than is allowed in the CTS.

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

CTS

[High Radiation Area] 1
5.7

5.0 ADMINISTRATIVE CONTROLS

6.12 [5.7 High Radiation Area] 1

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

6.12.1 5.7.1 <u>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</u>

6.12.1.a 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.

6.12.1.b 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.

6.12.1.c 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.

(whether alone or in a group)

6.12.1.d d. Each individual ~~or group~~ entering such an area shall possess~~:~~ 2 3

one of the following

6.12.1.d.1) 1. A radiation monitoring device that continuously displays radiation dose rates in the area; ~~or~~ 5

6.12.1.d.2) 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~~ ; 4 5

6.12.1.d.3) 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

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

DOC L01 (i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, ~~or~~ 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

CTS

5.7 High Radiation Area 1

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

6.12.1.d.4) (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.

6.12.1.e 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 from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation

6.12.2.a 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 , or other barrier that prevents unauthorized entry, and, in addition: 6

6.12.2.a.1) 1. All such door and gate keys shall be maintained under the administrative control of the shift supervisor, radiation protection manager, or his or her designee ~~.~~ ; and 4

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

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

6.12.2.c 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.

CTS

5.7 High Radiation Area 1

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

(whether alone or in a group)

6.12.2.d d. Each individual ~~or group~~ entering such an area shall possess one of the following: 2 5

6.12.2.d.1) 1. 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~~ ; 5 4 5

6.12.2.d.2) 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 ; 4

6.12.2.d.3) 3. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and , 5

6.12.2.d.3)(i) (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 4

6.12.2.d.3)(ii) (ii) Be under the surveillance, as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with and control every individual in the area.

Specifications 5.7.2.d.2 and 5.7.2.d.3

6.12.2.d.4) 4. In those cases where ~~options (2) and (3)~~, above, are impractical or determined to be inconsistent with the "As Low As is Reasonably Achievable" principle, a radiation monitoring device that continuously displays radiation dose rates in the area. 7

6.12.2.e 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.

CTS

[5.7 High Radiation Area] 1

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

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

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JUSTIFICATION FOR DEVIATIONS
ITS 5.7, HIGH RADIATION AREA

1. The brackets have been removed and the proper plant specific information/value has been provided.

2. ISTS 5.7.1.d and ISTS 5.7.2.d allows only one member of a group of individuals entering a high radiation area to meet the associated monitoring requirements. ITS 5.7.1.d and ITS 5.7.2.d requires every individual, even if in a group, who enters a high radiation area to meet the associated monitoring requirements. This is consistent with the Davis-Besse current licensing basis.

3. Change made to be consistent with another similar Specification (i.e., ITS 5.7.2.d).

4. These punctuation corrections have been made consistent with the Writer's Guide for the Improved Standard Technical Specifications, TSTF-GG-05-01, Section 5.1.3.

5. Typographical/grammatical error corrected.

6. Change made to be consistent with the Davis-Besse current licensing basis. Not all entryways are doors or gates.

7. The proper Specification numbers have been provided.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 5.7, HIGH RADIATION AREA

There are no specific NSHC discussions for this Specification.

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ATTACHMENT 8

DELETED CURRENT TECHNICAL SPECIFICATIONS

CTS 6.0, ADMINISTRATIVE CONTROLS

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

6.5.3 TECHNICAL REVIEW AND CONTROL

ACTIVITIES

6.5.3.1 Activities which affect nuclear safety shall be conducted as follows:

a. Plant procedures required by Section 6.8.1 and changes thereto shall be prepared, reviewed and approved. Each such procedure or procedure change shall be reviewed by an individual/group other than the individual/group which prepared the procedure or procedure change, but who may be from the same organization as the individual/group which prepared the procedure or procedure change. Plant procedures, (including plant administrative procedures), Physical Security Plan Implementing Procedures and Davis-Besse Emergency Plan Implementing Procedures will be approved by procedurally authorized individuals.

b. Temporary approval of changes to plant procedures cited in Section 6.8.1 which clearly do not change the intent of the approved procedures, can be made by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License. For changes to plant procedures, which may involve a change in intent of the approved procedures, the person authorized in Section 6.5.3.1a to approve the procedure shall approve the change

LA01

c. Proposed changes or modifications to plant structures, systems and components shall be reviewed as designated by procedurally authorized individuals. Each such modification shall be reviewed by an individual/group other than the individual/group which designed the modification, but who may be from the same organization as the individual/group which designed the modifications. Implementation of modifications to plant structures, systems and components shall be approved by proecdurally authorized individuals.

d. Proposed tests and experiments which affect plant nuclear safety and are not addressed in the Safety Analysis Report shall be reviewed by an individual/group other than the individual/group which prepared the proposed test or experiment and shall be approved by procedurally authorized individuals.

e. Individuals responsible for reviews performed in accordance with Section 6.5.3.1 a, b, c and d above shall meet or exceed the appropriate qualification requirements of Section 4.2, 4.3.1, 4.4 or 4.6 of ANSI 18.1, 1971, and be previously designated by procedurally authorized individuals. Each such review shall include a determination of whether an additional, cross disciplinary, review is necessary. If deemed necessary, such review shall be performed by the review personnel of the appropriate discipline.

f. Each review will include a determination of whether prior NRC approval is required pursuant to 10 CFR 50.59.

6.0 ADMINISTRATIVE CONTROLS

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the U.S. Nuclear Regulatory Commission in accordance with 10 CFR 50.4 within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specifications:

a. ECCS Actuation, Specifications 3.5.2 and 3.5.3.

b. Deleted

c. Deleted

d. Deleted

e. Deleted

f. Deleted

g. Inoperable Remote Shutdown System control circuit(s) or transfer switch(es) required for a serious control room or cable spreading room fire, Specification 3.3.3.5.2.

See ITS 5.6

~~6.10 RECORD RETENTION~~

~~Records of facility activities shall be retained as described in the USAR Chapter 17 Quality Assurance Program.~~

LA02

6.11 Deleted

6.12 HIGH RADIATION AREA

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

See ITS 5.7

6.12.1 High radiation areas with dose rates not exceeding 1.0 rem/hour at 30 centimeters from the radiation source or from any surface penetrated by the radiation:

a. Each entry way 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.

6.0 ADMINISTRATIVE CONTROLS

6.12.2.d (Continued)

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

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

f. Such individual areas that are within a larger area that is controlled as a high radiation area, where no enclosure exists for the purpose of locking and where no enclosure can reasonably be constructed around the individual area, need not be controlled by a locked door or gate, but shall be barricaded and conspicuous, and a clearly visible flashing light shall be activated at the area as a warning device.

See ITS 5.7

6.13 ENVIRONMENTAL QUALIFICATION

6.13.1 By no later than June 30, 1982 all safety-related electrical equipment in the facility shall be qualified in accordance with the provisions of Division of Operating Reactors "Guidelines for Evaluating Environmental Qualification of Class IE Electrical Equipment in Operating Reactors" (DOR Guidelines); or, NUREG-0588 "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment", December 1979. Copies of these documents are attached to Order for Modification of License NPF-3 dated October 24, 1980.

6.13.2 By no later than December 1, 1980, complete and auditible records must be available and maintained at a central location which describe the environmental qualification method used for all safety-related electrical equipment in sufficient detail to document the degree of compliance with the DOR Guidelines or NUREG-0588. Thereafter, such records should be updated and maintained current as equipment is replaced, further tested, or otherwise further qualified.

A01

6.14 Deleted

DISCUSSION OF CHANGES
CTS 6.0, ADMINISTRATIVE CONTROLS

ADMINISTRATIVE CHANGES

A01 CTS 6.13 requires that by June 30, 1982, all safety-related electrical equipment be environmentally qualified in accordance with the Division of Operating Reactors (DOR) Guidelines or NUREG-0588. It further requires that complete and auditable environmental qualification records be available and maintained at a central location by December 1, 1980. ITS Chapter 5.0 does not retain these requirements. This changes the CTS by deleting the requirement related to complying with the 10 CFR 50.69 requirements.

These requirements have already been satisfactorily met by Davis-Besse, therefore this historical requirement is not needed to be maintained in the ITS. Environmental qualification requirements are adequately addressed in the Davis-Besse procedures implementing the requirements of 10 CFR 50.49, and need not be repeated in the ITS. This change is designated as an administrative change and is acceptable since the requirements that have been fulfilled and it does not result in a technical change to the CTS.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 4 – Removal of LCO, SR, or other TS Requirement to the TRM, UFSAR, ODCM, QAPM, IST Program, or IIP)* CTS 6.5.3, Technical Review and Control, explains how activities that affect nuclear safety (i.e., changes to procedures, changes or modifications to plant structures, systems, and components, and proposed tests and experiments) shall be conducted. ITS Chapter 5.0 does not retain these requirements. This changes the CTS by moving the Technical Review and Control requirements to the Quality Assurance Program Manual (QAPM).

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. This change is acceptable because these types of procedural details will be adequately controlled in the QAPM. Any changes to the QAPM are made under 10 CFR 50.54(a), which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because requirements are being removed from the Technical Specifications.

DISCUSSION OF CHANGES
CTS 6.0, ADMINISTRATIVE CONTROLS

LA02 *(Type 4 – Removal of LCO, SR, or other TS Requirement to the TRM, UFSAR, ODCM, QAPM, IST Program, or IIP)* CTS 6.10, Record Retention, requires that records of activities be retained as described in the USAR Chapter 17 Quality Assurance Program. ITS Chapter 5.0 does not retain this requirement. This changes the CTS by moving the record retention requirements to Quality Assurance Program Manual (QAPM).

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. This change is acceptable be these types of procedural details will be adequately controlled in the QAPM. Any changes to the QAPM are made under 10 CFR 50.54(a), which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because requirements are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

None

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Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
CTS 6.0, ADMINISTRATIVE CONTROLS

There are no specific NSHC discussions for this Specification.