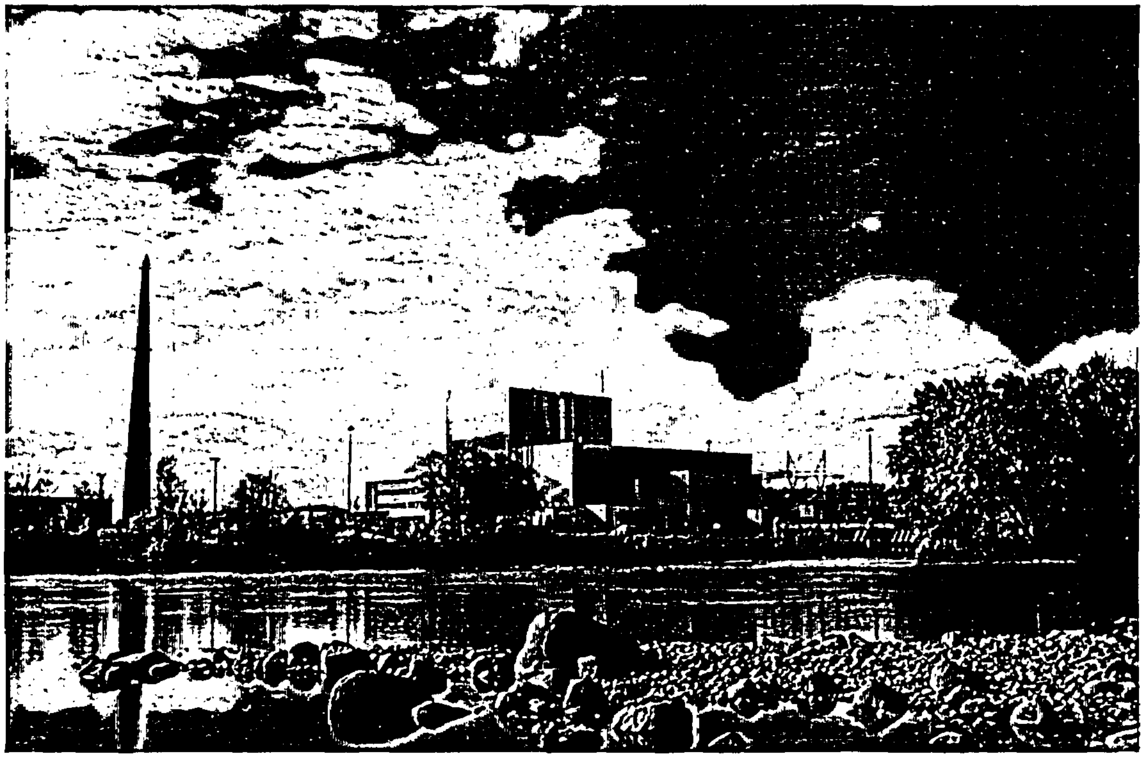
IMPROVED TECHNICAL SPECIFICATIONS



MONTICELLO NUCLEAR GENERATING PLANT

VOLUME 17

ITS Chapter 5.0,
Administrative Controls

NMC
Committed to Nuclear Excellence

ATTACHMENT 1

VOLUME 17

MONTICELLO
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)

6.0 ADMINISTRATIVE CONTROLS

6.1 Organization

5.1 Responsibility

See ITS 5.2

5.1.1 A. The plant manager shall be responsible for overall unit safe operation and shall have control over those onsite activities necessary for the safe operation and maintenance of the plant. During periods when the plant manager is unavailable, this responsibility may be delegated to other qualified supervisory personnel.

Add proposed second paragraph of ITS 5.1.1

M.1

5.1.2 The Shift Supervisor (or, a designated individual during periods of absence from the control room and shift supervisor's office) shall be responsible for the control room command function.

LA.1

M.2

B. Offsite and Onsite Organizations

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

1. Lines of authority, responsibility and communication shall be established and defined for the highest management levels through intermediate levels 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 department 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 are documented in corporate and plant procedures, or the Updated Safety Analysis Report or the Operational Quality Assurance Plan.

See ITS 5.2

2. A corporate officer with direct responsibility for the plant 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.

3. 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.

DISCUSSION OF CHANGES
ITS 5.1, RESPONSIBILITY

ADMINISTRATIVE CHANGES

A.1 In the conversion of the Monticello 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-1433, Rev. 3, "Standard Technical Specifications General Electric Plants, BWR/4" (ISTS).

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

MORE RESTRICTIVE CHANGES

M.1 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. The CTS does not include this requirement. This changes the CTS by adding an approval requirement for the plant manager or his designee.

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. This change is acceptable because having the plant manager or his designee approve actions affecting nuclear safety is consistent with the CTS 6.1.A (ITS 5.2.1.b) requirement that 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. This change is designated more restrictive because it adds a requirement for the plant manager or his designee to the CTS.

M.2 CTS 6.1.A allows a designated individual to assume the responsibility for the control room command function when the shift supervisor is absent from the control room and shift supervisor's office. ITS 5.1.2 provides the allowance for the designated individual to assume the responsibility for the control room command function, but provides additional requirements for the designated individual. In MODE 1, 2, or 3, ITS 5.1.2 requires the designated individual hold an active Senior Operator license. In MODE 4 or 5, ITS 5.1.2 requires the designated individual hold an active Senior Operator license or Operator license. This changes the CTS by adding qualification requirements for the designated individual that assumes the control room command function.

The purpose of the ITS 5.1.2 requirement is to ensure that the control room command function is maintained. This change is acceptable because the additional requirements ensure that the designated individual assuming the control room command function meets the appropriate qualification requirements. This change is designated as more restrictive because it adds qualification requirements for the designated individual that assumes the control room command function to the CTS.

DISCUSSION OF CHANGES
ITS 5.1, RESPONSIBILITY

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA.1 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 6.1.A uses the title "Shift Supervisor." ITS 5.1.2 uses the generic title "shift supervisor." This changes the CTS by moving the specific Monticello organizational title to the USAR or Operational Quality Assurance Plan (OQAP) and replacing it with a generic title.

The removal of this detail, which is related to meeting Technical Specification requirements, from the Technical Specifications is acceptable because this type of information is not necessary to be included in Technical Specifications to provide adequate protection of public health and safety. The allowance to relocate the specific Monticello organizational title out of the Technical Specifications is consistent with the NRC letter from C. Grimes to the Owners Groups Technical Specification Committee Chairmen, dated November 10, 1994. The various requirements of the shift supervisor are still retained in the ITS. Also, this change is acceptable because the removed information will be adequately controlled in the USAR or OQAP. Any changes to the USAR are made under 10 CFR 50.59 or 10 CFR 50.71(e) and any changes to the OQAP are made under 10 CFR 50.54(a), which ensure changes are properly evaluated. This change is designated as a less restrictive removal of detail change because information related to meeting Technical Specification requirements are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

None

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

CTS

5.0 ADMINISTRATIVE CONTROLS

6.1.A 5.1 Responsibility

---REVIEWER'S NOTES---

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

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

DOC M.1 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. 2

shift supervisor 3 complex 4

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

shift supervisor

Senior Operator

JUSTIFICATION FOR DEVIATIONS
ITS 5.1, RESPONSIBILITY

1. This 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. Grammatical error corrected.

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

4. The term "control room" in ISTS 5.1.2 has been changed to "control room complex" to be consistent with the current licensing basis. Currently, CTS 6.1.A discusses absence from both the control room and the shift supervisor's office, which is not in the control room proper. Therefore, the term "complex" shall be used and includes both the control room proper and the shift supervisor's office.

5. Typographical error corrected. The terms in 10 CFR 55.4 and 10 CFR 50.54(m) are "Senior Operator" and "Operator," not "Senior Reactor Operator" and "Reactor Operator."

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.

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ATTACHMENT 2

ITS 5.2, Organization

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

6.0 ADMINISTRATIVE CONTROLS

5.2 6.1 Organization

5.2.1.b A. The plant manager shall be responsible for overall unit safe operation and shall have control over those onsite activities necessary for the safe operation and maintenance of the plant. During periods when the plant manager is unavailable, this responsibility may be delegated to other qualified supervisory personnel.

The Shift Supervisor (or, a designated individual during periods of absence from the control room and shift supervisor's office) shall be responsible for the control room command function.

See ITS 5.1

5.2.1 B. Offsite and Onsite Organizations

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

5.2.1.a 1. Lines of authority, responsibility and communication shall be established and defined for the highest management levels through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organization charts, function descriptions of department 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 are documented in ~~corporate and plant procedures, or~~ the Updated Safety Analysis Report or the Operational Quality Assurance Plan.

M.1

5.2.1.c 2. A corporate officer with direct responsibility for the plant 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.d 3. 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.

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A.1

ITS

5.2.2 C. Plant Staff

1. Each ~~on-duty~~ shift shall be composed ~~of at least~~ the minimum shift crew composition ~~shown~~ in Table 6.1.1. LA.1

2. At least one licensed ~~operator~~ shall be in the control room ~~when~~ fuel is in the reactor.

3. At least ~~two~~ licensed operators shall be ~~present~~ in the control room during cold startup, ~~scheduled~~ reactor shutdown, and during recovery from reactor ~~trips~~. A.2

L.1 Add proposed 2 hour absence allowance

5.2.2.c 4. An individual qualified in radiation protection procedures shall be onsite when fuel is in the reactor.

5. All alterations ~~of the reactor~~ core shall 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~~. A.2

5.2.2.e 6. The operations manager shall be formerly licensed as a Senior Reactor Operator or hold a current Senior Reactor Operator License.

5.2.2.e 7. At least one member of plant management holding a current Senior Reactor Operator License shall be assigned to the plant operations group on a long term basis (approximately two years). This individual will not be assigned to a rotating shift.

8. Licensed reactor operators and senior reactor operators shall complete qualification training in accordance with a Commission-approved training program that is based on a systems approach to training and uses a simulation facility that is acceptable to the Commission. This program has been accredited by the National Nuclear Accrediting Board. See ITS 5.3

D. Each member of the site staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for (1) the radiation protection manager ~~or designated health physicist 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, (3) the operations manager who shall meet the requirement of ANSI N18.1-1971 except that NRC license requirements are as specified in Specification 6.1.C.7, and (4) licensed reactor operators and senior reactor operators shall meet the requirements of Specification 6.1.C.8. The training program shall be under the direction of a designated member of Nuclear Management Company, LLC management. A.3 See ITS 5.3

5.2.2.f

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ITS

E. (Deleted)

5.2.2.d F. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety-related functions; e.g., senior reactor operators, reactor operators, health physicists, auxiliary operators, and key maintenance personnel. Procedures shall include the following provisions:

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

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

   b. Overtime should be limited for all nuclear plant staff personnel so that total work time does not exceed 16 hours in any 24-hour period, nor more than 24 hours in any 48-hour period, not more than 84 hours in any seven day period, all excluding shift turnover time. Individuals should not be required to work more than 15 consecutive days without two consecutive days off.

   c. A break of at least eight hours including shift turnover time should be allowed between work periods.

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

L.2

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ITS

e. ~~Shift Technical Advisor (STA) and Shift Emergency Coordinator (SEC) onsite rest time periods shall not be considered as hours worked when determining the total work time for which the above limitations apply.~~ L.2

5.2.2.d 2. Any deviation from the above guidelines shall be authorized by the plant manager or designee, or higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation. During plant emergencies the Emergency Director shall have this authority. Controls shall be included in the procedures such that individual overtime shall be reviewed monthly to assure that excessive hours have not been assigned. Routine deviation from the above guidelines is not allowed.

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TABLE 6.1.1
MINIMUM SHIFT CREW COMPOSITION (Note 1)

CATEGORY	APPLICABLE PLANT CONDITIONS	
	SHUTDOWN OR REFUELING MODE AND <212°F	STARTUP OR RUN MODE (Note 4) OR ≥212°F
No. Licensed Senior Operators (LSO)	1 (Note 2)	2 (Note 3, 5)
Total No. Licensed Operators (LSO & LO)	2	4
5.2.2.a Total No. Licensed and Unlicensed Operators	3	6

LA.1

1 2 A.4

Notes:

5.2.2.b 1. Shift crew composition may be one less than the minimum requirements for a period of time not to exceed two hours in order to accommodate an unexpected absence of one duty shift crew member provided immediate action is taken to restore the shift crew composition to within the minimum requirements specified.

2. Does not include the licensed Senior Reactor Operator, or Senior Reactor Operator Limited to Fuel Handling, supervising alterations of the reactor core.

3. One LSO shall be in the control room or the shift supervisor's office at all times when the reactor is in the Startup or Run Mode or reactor coolant temperature is greater than or equal to 212°F. At least 50% of the time, an LSO shall actually be in the control room proper when the reactor is in the Startup or Run Mode or reactor coolant temperature is greater than or equal to 212°F.

4. Except for momentary switching to Startup Mode for testing.

LA.1

5. One LSO position shall be filled by an individual who meets the qualifications of a Shift Technical Advisor as defined in Section 6.1.D(2). If a qualified individual to staff the combined LSO/STA position is not available, a dedicated Shift Technical Advisor shall be on duty, in addition to two licensed senior operators.

A.5

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A.1

6.2 (Deleted)

6.3 (Deleted)

6.4 (Deleted)

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DISCUSSION OF CHANGES
ITS 5.2, ORGANIZATION

ADMINISTRATIVE CHANGES

A.1 In the conversion of the Monticello 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-1433, Rev. 3, "Standard Technical Specifications General Electric Plants, BWR/4" (ISTS).

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

A.2 CTS 6.1.C.2 states "At least one licensed operator shall be in the control room when fuel is in the reactor." CTS 6.1.C.3 states "At least two licensed operators shall be present in the control room during cold startup, scheduled reactor shutdown, and during recovery from reactor trips." CTS 6.1.C.5 states "All alterations of the reactor core shall 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.

The purpose of CTS 6.1.C.2, 6.1.C.3, and 6.1.C.5 is to provide additional requirements as to the physical location at which the required licensed operators must be. 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 the unit's technical specifications, each licensee shall have a person holding a senior operator license for the nuclear power unit in the control room at all times. In addition to this senior operator, for each fueled nuclear power unit, a licensed operator or senior operator shall be 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) and the Monticello Operating License requires compliance with all NRC regulations. This change is designated as administrative because it does not result in technical changes to the CTS.

A.3 CTS 6.1.D 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 transients 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. This changes the CTS by referencing the Commission Policy Statement on Engineering Expertise on Shift for qualification requirements instead of listing the specific qualification requirements.

The purpose of the CTS 6.1.D STA requirements is to specify the minimum qualification requirements for the STA. This change is acceptable because the

DISCUSSION OF CHANGES
ITS 5.2, ORGANIZATION

qualification requirements included in the Commission Policy Statement on Engineering Expertise on Shift (Generic Letter 86-04, dated February 13, 1986) encompass the current STA qualification requirements. This change is designated as administrative because it does not result in technical changes to the CTS.

A.4 CTS Table 6.1.1 requires the total number of licensed and non-licensed operators during MODES 4 and 5 (i.e., SHUTDOWN or REFUELING MODE and < 212°F) to be 3 and requires the total number of licensed and unlicensed operators during MODES 1, 2, and 3 (i.e., STARTUP or RUN MODE or ≥ 212°F) to be 6. ITS 5.2.2.a requires the total number of non-licensed operators to be 1 in MODES 4 and 5 and to be 2 in MODES 1, 2, and 3. This changes the CTS by specifically stating the total number of non-licensed operators required in MODES 1, 2, 3, 4, and 5.

The purpose of CTS Table 6.1.1, in part, is to specify the non-licensed operator requirements. CTS Table 6.1.1 requires the total number of licensed operators in MODES 4 and 5 to be 2 and the total number of licensed operators in MODES 1, 2, and 3 to be 4. Thus, the total number of non-licensed operators required in MODES 4 and 5 is 1 and in MODES 1, 2, and 3 is 2. Therefore, this change is acceptable since the total number of required non-licensed operators is unchanged. This change is designated as administrative because it does not result in technical changes to the CTS.

A.5 CTS Table 6.1.1 Note 5 states "One LSO position shall be filled by an individual who meets the qualifications of a Shift Technical Advisor as defined in Section 6.1.D(2). If a qualified individual to staff the combined LSO/STA position is not available, a dedicated Shift Technical Advisor shall be on duty, in addition to two licensed senior operators." 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 6.1.1 Note 5 is to provide allowances for the LSO to meet the requirements of the STA, and if the LSO is not filling the STA role, then to describe when the STA must be on duty (i.e., during operations in MODES 1, 2, and 3). These issues are adequately addressed in the "Commission Policy Statement on Engineering Expertise on Shift," published in Generic Letter 86-04, dated February 13, 1986, and need not be retained in the ITS. The ITS already requires this the STA to meet this policy statement (ITS 5.2.2.f). 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.

MORE RESTRICTIVE CHANGES

M.1 CTS 6.1.B.1, regarding documentation of the relationships between operating organization positions, states that the documentation be in "corporate and plant procedures," or in the Updated Safety Analysis Report (USAR) or Operational Quality Assurance Plan (OQAP). ITS 5.2.1.a states that the documentation shall be in the USAR or OQAP. This changes the CTS by requiring that this specific information be located only in the USAR or OQAP.

The purpose of CTS 6.1.B.1 is to list appropriate places to locate and maintain this information. This change is acceptable because specifying this information only in the USAR or OQAP continues to ensure that organizational positions and associated responsibilities will be maintained. These locations are the two locations specified in NUREG-1433, Revision 3. This change is designated as more restrictive because it requires this information to be maintained only in the USAR or OQAP.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA.1 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 6.1.C.1 and Table 6.1.1, including Notes 2, 3, and 4, provide minimum shift crew composition requirements. ITS 5.2.2 only 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 6.1.1 to ITS 5.2.2.a. 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. Also, this change is acceptable because these details will be adequately controlled in the TRM. Any changes to the TRM are made under 10 CFR 50.59, which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because details for meeting Technical Specification and regulatory requirements are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L.1 *(Category 1 - Relaxation of LCO Requirement)* CTS 6.1.C.4 requires an individual qualified in radiation protection procedures to be onsite when fuel is in the reactor. ITS 5.2.2.c includes the same requirement, but allows the position to

DISCUSSION OF CHANGES
ITS 5.2, ORGANIZATION

be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is take to fill the required position. This changes the CTS by allowing the radiation protection technician position to be vacant for a short time due to unexpected circumstances.

The purpose of CTS 6.1.C.4 is to ensure an individual, trained in radiation protection procedures, is onsite to provide expertise to the plant with regard to the radiation protection field. However, under unusual circumstances, such as an unexpected and sudden illness of the onsite individual, a radiation protection technician may not be available. This change allows a short time, 2 hours, to not meet the requirement, provided immediate action is taken to fill the position (e.g., call in a replacement radiation protection technician). This allowance is similar to that allowed in CTS Table 6.1.1 Note 1 for an unexpected absence in the shift operating crew requirements. Therefore, since the time allowed is short, and immediate action to rectify the problem is required, this change is considered acceptable. This change is designated as less restrictive because a 2 hour allowance is provided to not meet the radiation protection technician position requirement.

L.2 *(Category 1 - Relaxation of LCO Requirement)* CTS 6.1.F provides specific details concerning working hour limits for unit staff who perform safety related functions. These details include the normal working hours in a week, the number of hours allowed to work in a continuous period, the number of hours allowed to work in a 24 hour and 48 hour period, the number of hours for a work period break, and that overtime should be evaluated on an individual basis, not an entire staff basis, except during an extended shutdown. ITS 5.2.2.d requires procedures to be developed and implemented to limit the number of working hours for personnel who perform safety related functions, but does not include these specific details. This changes the CTS by deleting these working hour-related details.

The purpose of CTS 6.1.F is to provide guidance concerning working hour limitations for personnel who perform safety related functions. The details associated with the involved Specification are not required to be in the ITS to provide adequate protection of the public health and safety because overtime limitations are adequately addressed by Monticello commitments to NUREG-0737, and by miscellaneous IE Circulars and Generic Letters. In addition, specific controls for working hours of plant staff are described in plant procedures, as required by the CTS and maintained in the ITS, and require a deliberate decision making process to minimize the potential for impaired personnel performance. Established procedure control processes provide sufficient control for changes to these procedures. This approach provides an effective level of control and provides an appropriate change control process. The level of safety of plant operation is unaffected by the change because there is no change in the overall operational requirements. Therefore, this change is acceptable. This change is designated as less restrictive because a working hour details that are currently included in the CTS are not included in the ITS; they will be controlled in plant procedures.

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

CTS

5.0 ADMINISTRATIVE CONTROLS

6.1 ## 5.2 Organization

6.1.B 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.

6.1.B.1 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 [F] U SAR/~~QA Plan~~ or Operational Quality Assurance Plan. 1

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

6.1.B.2 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.

6.1.B.3 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.

6.1.C 5.2.2 Unit Staff

The unit staff organization shall include the following:

Table 6.1.1 a. A non-licensed operator shall be assigned to each reactor containing fuel and an additional non-licensed operator shall be assigned for each control room from which a reactor is operating in MODES 1, 2, or 3.

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

CTS

5.2 Organization

5.2.2 Unit Staff (continued)

Table 6.1.1 Note 1

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

6.1.C.4

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

6.1.F

d. Administrative procedures 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]). non-licensed 1

6.1.F.1

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

6.1.F.2

Any deviation from the above guidelines shall be authorized in advance by the plant manager or the plant manager's 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.

6.1.F.2

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

Senior Operator

6.1.C.6, 6.1.C.7

e. The operations manager ~~or assistant operations manager~~ shall hold a[n] ~~SRO~~ license[.] INSERT 1 4 3

6.1.D

f. 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.

4 **INSERT 1**

or shall formerly have held a Senior Operator license. If the operations manager does not hold a Senior Operator license, another member of plant management shall hold a Senior Operator license and shall be assigned to the plant operations group on a long term basis (approximately 2 years). This individual shall not be assigned to a rotating shift.

JUSTIFICATION FOR DEVIATIONS
ITS 5.2, ORGANIZATION

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

2. The ISTS Reviewer's Note has been deleted since it is not intended to be included in the ITS. The requirements for non-licensed operators for two unit sites addressed in the ISTS Reviewer's Note are not adopted, since Monticello is a single unit site.

3. Typographical error corrected. The term in 10 CFR 55.4 and 10 CFR 50.54(m) is "Senior Operator" not "SRO" (i.e., Senior Reactor Operator).

4. ISTS 5.2.2.e provides a requirement for the operations manager or the assistant operations manager to hold a Senior Operator license. This requirement is revised in ITS 5.2.2.e to reflect the Monticello CTS 6.1.C.6 and 6.1.C.7 requirements. CTS 6.1.C.6 requires the operations manager to hold either a Senior Operator license or have formerly held a Senior Operator license. CTS 6.1.C.7 requires a plant management individual in the plant operations group (i.e., the operations department) to hold a Senior Operator license. This individual can either be the operations manager or the assistant operations manager, which are the only two individuals in the operations department who are considered members of plant management. Thus, the operations manager, if the individual holds a Senior Operator license, meets the requirements of CTS 6.1.C.6 and CTS 6.1.C.7. However, if the operations manager is only a former Senior Operator license holder, then the assistant operations manager must hold a Senior Operators license.

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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.

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ATTACHMENT 3

ITS 5.3, Unit Staff Qualifications

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

ITS

5.3 Unit Staff Qualifications

C. Plant Staff

1. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.1.1.
2. At least one licensed operator shall be in the control room when fuel is in the reactor.
3. At least two licensed operators shall be present in the control room during cold startup, scheduled reactor shutdown, and during recovery from reactor trips.
4. An individual qualified in radiation protection procedures shall be onsite when fuel is in the reactor.
5. All alterations of the reactor core shall 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.
6. The operations manager shall be formerly licensed as a Senior Reactor Operator or hold a current Senior Reactor Operator License.
7. At least one member of plant management holding a current Senior Reactor Operator License shall be assigned to the plant operations group on a long term basis (approximately two years). This individual will not be assigned to a rotating shift.

See ITS 5.2

A.2

8. Licensed reactor operators and senior reactor operators shall complete qualification training in accordance with a Commission-approved training program that is based on a systems approach to training and uses a simulation facility that is acceptable to the Commission. This program has been accredited by the National Nuclear Accrediting Board.

LA.1

5.3.1 D. Each member of the site staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for (1) the radiation protection manager or designated health physicist 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, (3) the operations manager who shall meet the requirement of ANSI N18.1-1971 except
5.3.1 that NRC license requirements are as specified in Specification 6.1.C.7, and (4) licensed reactor operators and senior reactor operators shall meet the requirements of Specification 6.1.C.8. The training program shall be under the direction of a designated member of Nuclear Management Company, LLC management.

See ITS 5.2

A.2

LA.1

Add proposed Specification 5.3.2

A.3

Attachment 1, Volume 17, Rev. 0, Page 34 of 143

DISCUSSION OF CHANGES
ITS 5.3, UNIT STAFF QUALIFICATIONS

ADMINISTRATIVE CHANGES

A.1 In the conversion of the Monticello 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-1433, Rev. 3, "Standard Technical Specifications General Electric Plants, BWR/4" (ISTS).

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

A.2 CTS 6.1.C.8 states "Licensed reactor operators and senior operators shall complete qualification training in accordance with a Commission-approved training program that is based on a systems approach to training and uses a simulation facility that is acceptable to the Commission." CTS 6.1.D, in part, states "licensed reactor operators and senior reactor operators shall meet the requirements of Specification 6.1.C.8." The ITS does not include these requirements. This changes the CTS by deleting these requirements.

The purpose of CTS 6.1.C.8 and 6.1.D part (4) is to provide training requirements for the licensed Senior Operators and Operators. 10 CFR 55 specifies these training requirements. This change is acceptable because the requirements deleted from the Technical Specifications are already required by 10 CFR 55 and the Monticello Operating License requires compliance with all NRC regulations. This change is designated as administrative because it does not result in technical changes to the CTS.

A.3 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 ITS 5.3.1 and be capable of performing the functions described in 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

DISCUSSION OF CHANGES
ITS 5.3, UNIT STAFF QUALIFICATIONS

REMOVED DETAIL CHANGES

LA.1 *(Type 6 - Removal of LCO, SR, or other TS requirements to the TRM, UFSAR, ODCM, QAPD, or IIP)* CTS 6.1.C.8 states that the licensed Senior Operator and Operator training program be accredited by the National Nuclear Accrediting Board. CTS 6.1.D states that the training program be under the direction of a designated member of Nuclear Management Company, LLC management. These requirements are not retained in the ITS. This changes the CTS by moving the requirements for the training program to the USAR.

The removal of these details from the Technical Specifications is acceptable because this type of information is not necessary to provide adequate protection of public health and safety. These training provisions are adequately addressed by other proposed ITS Chapter 5.0 provisions and by regulations. ITS 5.3, "Unit Staff Qualifications," provides requirements to ensure adequate, competent staff in accordance with ANSI N18.1-1971 and Regulatory Guide 1.8, 1975. ITS 5.2 details organization requirements. ITS 5.2.2.a, 5.2.2.b, and 10 CFR 50.54 state minimum shift crew requirements. Training and requalification of NRC licensed positions is contained in 10 CFR 55. Placement of training requirements in the USAR will ensure that training programs are properly maintained in accordance with Monticello commitments and applicable regulations. Also, this change is acceptable because the removed information will be adequately controlled in the USAR. Any changes to the USAR are made under 10 CFR 50.59 or 10 CFR 50.71(e), which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because a requirement is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

None

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

CTS

5.0 ADMINISTRATIVE CONTROLS

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

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

INSERT 1

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

2 **INSERT 1**

ANSI N18.1-1971 for comparable positions, except for the radiation protection manager. The radiation protection manager shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975. In addition, 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 ISTS Reviewer's Note has been deleted since it is not intended to be included in the ITS.

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," not "SRO" (i.e., Senior Reactor Operator) and "RO" (i.e., Reactor Operator).

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

5.4 6.5 Procedures

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

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

5.4.1.b 2. The emergency operating procedures required to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1, as stated in Generic Letter 82-33;

5.4.1.c 3. Quality assurance for effluent and environmental monitoring;

5.4.1.d 4. Fire Protection Program implementation; and

5.4.1.e 5. All programs specified in Specification 6.8.

6.6 (Deleted)

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DISCUSSION OF CHANGES
ITS 5.4, PROCEDURES

ADMINISTRATIVE CHANGES

A.1 In the conversion of the Monticello 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-1433, Rev. 3, "Standard Technical Specifications General Electric Plants, BWR/4" (ISTS).

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

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

None

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

CTS

5.0 ADMINISTRATIVE CONTROLS

6.5 5.4 Procedures

6.5.A 5.4.1 Written procedures shall be established, implemented, and maintained covering the following activities:

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

6.5.A.2 b. The emergency operating procedures required to implement the requirements of NUREG-0737 and to NUREG-0737, Supplement 1, as stated in Generic Letter 82-33; 2 3 1

6.5.A.3 c. Quality assurance for effluent and environmental monitoring; 1

6.5.A.4 d. Fire Protection Program implementation; and 1

6.5.A.5 e. All programs specified in Specification 5.5.

JUSTIFICATION FOR DEVIATIONS
ITS 5.4, PROCEDURES

1. These punctuation corrections have been made consistent with the Writer's Guide for the Improved Standard Technical Specifications, NEI 01-03, Section 5.1.3.

2. Grammatical errors corrected.

3. The brackets are removed and the proper plant specific information/value is provided.

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)

6.8 Programs and Manuals

5.5.1 A. Offsite Dose Calculation Manual (ODCM)

5.5.1.a 1. 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

5.5.1.b 2. 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 6.7.C.1 and Specification 6.7.A.4.

5.5.1.c 3. Licensee initiated changes to the ODCM:

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

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

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

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

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

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5.5.2 B. Primary Coolant Sources Outside Containment

5.5.2 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 Core Spray, High Pressure Coolant Injection, Residual Heat Removal, Reactor Core Isolation Cooling, Combustible Gas Control, process sampling, and Standby Gas Treatment. The program shall include the following:

5.5.2.a 1. Preventive maintenance and periodic visual inspection requirements; and

24 months L.5

5.5.2.b 2. Integrated leak test requirements for each system at ~~refueling cycle intervals or less~~.

5.5.2 The provisions of Specification 4.0.B are applicable.

~~A program acceptable to the Commission was described in a letter dated December 31, 1979, from L O Mayer, NSP, to Director of Nuclear Reactor Regulation, "Lessons Learned Implementation."~~ A.2

C. (Deleted)

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5.5.3 D. Radioactive Effluent Controls Program

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

5.5.3.a 1. 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;

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

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 the site to unrestricted areas, conforming to 10 CFR 50, Appendix I;

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 monthly;

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

5.5.3.g 7. Limitations on the dose rate resulting from radioactive material released in gaseous effluents from the site to areas at or beyond the site boundary shall be in accordance with the following:

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

5.5.3.g.2 b. 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;

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5.5.3.h 8. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from the site to areas at or beyond the site boundary, conforming to 10 CFR 50, Appendix I.

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 >8 days in gaseous effluents released from the site to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

5.5.3.j 10. Limitations on the annual dose or dose commitment to any member of the public, beyond the site boundary, due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190; and

5.5.3.k 11. Limitations on venting and purging of the containment through the Standby Gas Treatment System to maintain releases as low as reasonably achievable.

5.5.3 The provisions of Specifications 4.0.B, 4.0.D and 4.0.E are applicable to the Radioactive Effluent Controls Program surveillance frequency.

6.8.E and 6.8.F - RESERVED

Add proposed ITS 5.5.4

M.1

5.5.5 G. Inservice Testing Program

5.5.5 This program provides controls for inservice testing of Quality Group A, B, and C pumps and valves which shall be performed in accordance with the requirements of ASME Code Class 1, 2, and 3 pumps and valves, respectively

LA.1

5.5.5.a 1. Testing frequencies specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda are as follows:

A.8

ASME Boiler and Pressure Vessel Code and Applicable Addenda Terminology for Inservice Testing Activities	Required Frequencies for Performing Inservice Testing Activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Biquarterly	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

Every 48 months	At least once per 1461 days
Every 5 years	At least once per 1827 days
Every 8 years	At least once per 2922 days
Every 10 years	At least once per 3653 days

A.8

5.5.5.b 2. The provisions of Surveillance Requirement 4.0.B are applicable to the Frequencies for performing inservice testing activities;

5.5.5.c 3. The provisions of Surveillance Requirement 4.0.D and 4.0.E are applicable to inservice testing activities; and

5.5.5.d 4. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any TS. A.8

6.8.H - RESERVED

5.5.7 I. Explosive Gas and Storage Tank Radioactivity Monitoring Program

5.5.7 This program provides controls for potentially explosive gas mixtures contained in the Offgas Treatment System, the quantity of radioactivity contained in gas storage tanks or fed into the offgas treatment system, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks. ~~The quantity of radioactivity after 12 hours holdup contained in each gas storage tank shall be limited to ≤22,000 curies of noble gases (considered as dose equivalent Xe-133). The quantity of liquid radioactive material contained in each outside temporary tank shall be limited to ≤10 curies, excluding tritium and dissolved or entrained noble gases.~~ LA.2

5.5.7 The program shall include:

5.5.7.a 1. The limits for concentrations of hydrogen and oxygen in the Offgas Treatment 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);

5.5.7.b 2. 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

5.5.7.c 3. A surveillance program to ensure that the quantity of radioactivity contained in all outdoor liquid radwaste tanks that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the Liquid Radwaste Treatment System is less than the amount that would result in concentrations less than the limits of 10 CFR 20, Appendix B, Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tanks' contents.

5.5.7 The provisions of Specifications 4.0.B, 4.0.D and 4.0.E are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

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6.8.J - RESERVED

5.5.9 K. Technical Specifications (TS) Bases Control Program

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

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

5.5.9.b 2. Changes to Bases may be made without prior NRC approval provided the changes do not involve either of the following:

a. a change in the TS incorporated in the license; or

b. a change to the USAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

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

5.5.9.d 4. Proposed changes to the Bases that involve changes as described in a. or b. of Specification 6.8.K.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).

6.8.L - RESERVED

Add proposed ITS 5.5.10

M.1

5.5.11 M. Primary Containment Leakage Rate Testing Program

5.5.11.a 1. This program shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR Part 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 exception: NEI 94-01, Rev. 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J.".

Section 9.2.3: The first Type A test after the March 1993 Type A test shall be performed no later than March 2008.

5.5.11.b 2. The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a, is 42 psig. The containment design pressure is 56 psig.

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5.5.11.c 3. The maximum allowable containment leakage rate, L_a, at P_a, shall be 1.2% of containment air weight per day.

5.5.11.d 4. Leakage rate acceptance criteria are:

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

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

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

5.5.11.d.2.b) 2) For each door, leakage rate is $\leq 0.007\ L_a$ when pressurized to ≥ 10 psig.

5.5.11.f 5. The provisions of SRs 4.0.D and 4.0.E are applicable to the Primary Containment Leakage Rate Testing Program.

~~6. Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J.~~ A.3

Add proposed ITS 5.5.12 M.1

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A.1

ITS

ITS

See ITS 3.6.4.3

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS

b. If both standby gas treatment system circuits are not operable, within 36 hours the reactor shall be placed in a condition for which the standby gas treatment system is not required in accordance with Specification 3.7.C.2.(a) through (d).

2. Performance Requirements

Add proposed ITS 5.5.6 program statement

a. Periodic Requirements

5.5.6.a (1) The results of the in-place DOP tests at 3500 cfm (±10%) on HEPA filters shall show ≤1% DOP penetration.

5.5.6.b (2) The results of in-place halogenated hydrocarbon tests at 3500 cfm (±10%) on charcoal banks shall show ≤1% penetration.

5.5.6.c (3) The results of laboratory carbon sample analysis shall show ≤5% methyl iodide penetration when tested in accordance with ASTM D3803-1989 at 30°C, 95% relative humidity.

A.4

L.1

2. Performance Requirement Tests

5.5.6 a. At least once per 720 hours of system operation; or once per operating cycle, but not to exceed 18 months, whichever occurs first; or following painting, fire, or chemical release in any ventilation zone communicating with the system while the system is operating that could contaminate the HEPA filters or charcoal adsorbers, perform the following:

24 months A.5

24 A.9

5.5.6.a (1) In-place DOP test the HEPA filter banks.

5.5.6.b (2) In-place test the charcoal adsorber banks with halogenated hydrocarbon tracer.

5.5.6.c (3) Remove one carbon test sample from the charcoal adsorber in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978. Subject this sample to a laboratory analysis to verify methyl iodide removal efficiency.

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3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS

b. The system shall be shown to be operable with:

5.5.6.d (1) Combined filter pressure drop ≤6 inches water.

5.5.6.e (2) Inline heater power output ≥ 18kW.

c. The system shall be shown to be operable with automatic initiation upon receipt of the following inputs:

(a) Low Low Reactor Water Level, or

(b) High drywell pressure, or

(c) Reactor building ventilation plenum high radiation, or

(d) Refueling floor high radiation

3. Post Maintenance Requirements

5.5.6 a. After any maintenance or testing that could affect the HEPA filter or HEPA filter mounting frame leak tight integrity,

5.5.6.a the results of the in-place DOP tests at 3500 cfm (±10%) on HEPA filters shall show ≤1% DOP penetration.

5.5.6 b. After any maintenance or testing that could affect the charcoal adsorber leak tight integrity,

5.5.6.b the results of in-place halogenated hydrocarbon tests at 3500 cfm(±10%) on charcoal adsorber banks shall show ≤1% penetration.

INSERT A — A.9

~~b. At least once per operating cycle, but not to exceed 18 months, the following conditions shall be demonstrated for each standby gas treatment system:~~

~~(1) Pressure drop across the combined filters of each standby gas treatment system circuit shall be measured at 3500 cfm (±10%) flow rate.~~

~~(2) Operability of inline heater at nominal rated power shall be verified.~~

See ITS 3.6.4.3

c. At least once per operating cycle, automatic initiation of each standby gas treatment system circuit shall be demonstrated.

3. Post Maintenance Testing

5.5.6 a. After any maintenance or testing that could affect the leak tight integrity of the HEPA filters,

5.5.6.a perform in-place DOP tests on the HEPA filters.

5.5.6 b. After any maintenance or testing that could affect the leak tight integrity of the charcoal adsorber banks,

5.5.6.b perform halogenated hydrocarbon tests on the charcoal absorbers.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test Frequencies — A.4

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ITS

A.9 **INSERT A**

5.5.6 b. Once per quarter demonstrate that the pressure
5.5.6.d drop across the combined filters of each standby gas treatment system circuit shall be measured at 3500 cfm (± 10%) flow rate.

24 months A.5

5.5.6 c. Once per ~~operating cycle~~ the operability of inline
5.5.6.e heater at nominal rated power shall be verified for each standby gas treatment system.

A.1

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3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
3. a. The inerting and deinerting operations permitted by TS 3.7.A.5.b shall be via the 18-inch purge and vent valves (equipped with 40-degree limit stops) aligned to the Reactor Building plenum and vent. All other purging and venting, when primary containment integrity is required, shall be via the 2-inch purge and vent valve bypass line and the Standby Gas Treatment System.	3. Whenever containment purge and vent valves are isolated to meet the requirements of TS 3.7.D.3.b, the position of the deactivated and isolated valves outside primary containment shall be recorded monthly.**
b. In the event one or more penetration flow paths with one or more containment purge and vent valves not within purge and vent valve leakage limits, reactor operation in the run mode may continue provided that within the subsequent 24 hours, restore the valve(s) to within leakage limits, or at least one valve in each line having a purge and vent valve not within leakage limits is deactivated in the isolated position. This requirement may be satisfied by use of one closed and deactivated automatic valve, closed manual valve, or blind flange. (Deactivated means electrically or pneumatically disarm or otherwise secure the valve.)	See ITS 3.6.1.3
4. If Specification 3.7.D.1, 3.7.D.2 and 3.7.D.3 cannot be met, initiate normal orderly shutdown and have reactor in the Cold Shutdown condition within 24 hours.	5.5.11.e 4. The seat seals of the drywell and suppression chamber 18-inch purge and vent valves shall be replaced at least once every six operating cycles. If periodic Type C leakage testing of the valves identifies a common mode test failure attributable to seat seal degradation, then the seat seals of all drywell and suppression chamber 18-inch purge and vent valves shall be replaced. A.6
	** Isolated valves in high radiation areas may be verified by use of administration means. See ITS 3.6.1.3

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A.1

ITS

Add proposed ITS 5.5.11 generic program statement A.7

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
b. For the diesel generators to be considered operable, there shall be a minimum of 38,300 gallons of diesel fuel (7 days supply for 1 diesel generator at full load @ 2500 KW) in the diesel oil storage tank.	b. 1) Once a month the quantity of diesel fuel available shall be logged. See ITS 3.8.1
	2) During the monthly generator test, the diesel fuel oil transfer pump and diesel oil service pump shall be operated.
	5.5.8.a, 5.5.8.b, 5.5.8.c 3) Once a month a sample of diesel fuel shall be taken and checked for quality. L.2 M.2
c. When a diesel generator is required to be operable, maintain air pressure for both associated air starting receivers ≥ 165 psig.	c. Verify each required operable diesel generator air start receiver pressure is ≥ 165 psig once per month.
1) With one diesel generator starting air receiver pressure < 165 psig, restore both starting air receivers pressure to ≥ 165 psig within 7 days, or declare the associated diesel generator inoperable.	See ITS 3.8.3
2) With both diesel generator starting air receivers pressure < 165 psig but ≥ 125 psig, restore one starting air receiver to ≥ 165 psig and enter TS LCO 3.9.B.3.c.1, or restore both starting air receivers pressure to ≥ 165 psig within 48 hours. If neither action can be accomplished within 48 hours, declare the associated diesel generator inoperable.	The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program test Frequencies. A.7
3) With both diesel generator starting air receivers pressure < 125 psig, immediately declare the associated diesel generator inoperable.	

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ITS

ITS

Add proposed ITS 5.5.6 program statement

A.4

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
2. Performance Requirements a. Acceptance Criteria - Periodic Requirements 5.5.6.a (1) The results of the in-place DOP tests at 1000 cfm (±10%) shall show ≤1% DOP penetration on each individual HEPA filter and shall show ≤0.05% DOP penetration on the combined HEPA filters. 5.5.6.b (2) The results of in-place halogenated hydrocarbon tests at 1000 cfm (±10%) shall show ≤1% penetration on each individual charcoal adsorber and shall show ≤0.05% penetration on the combined charcoal banks. 5.5.6.c (3) The results of laboratory carbon sample analysis shall show ≤0.5% methyl iodide penetration when tested at 30°C and 95% relative humidity.	2. Performance Requirement Test 5.5.6.a, 5.5.6.b The in-place performance testing of HEPA filter banks and charcoal adsorber banks shall be conducted in accordance with Sections 10 and 11 of ASME N510-1989. 5.5.6.c The carbon sample test for methyl iodide shall be conducted in accordance with ASTM D 3803-1989. Sample removal shall be in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978. 5.5.6 a. At least once per ~~operating cycle, but not to exceed 18 months~~ 24 months, or following painting, fire, or chemical release while the system is operating that could contaminate the HEPA filters or charcoal adsorbers, perform the following: A.5 A.9 5.5.6.a (1) In-place DOP test the HEPA filter banks. 5.5.6.b (2) In-place test the charcoal adsorber banks with halogenated hydrocarbon tracer. 5.5.6.c (3) Remove one carbon test sample from each charcoal adsorber bank. Subject this sample to a laboratory analysis to verify methyl iodide removal efficiency. 5.5.6.a, 5.5.6.b (4) ~~Initiate from the control room~~ 1000 cfm (±10%) flow through both trains of the emergency filtration treatment system. L.3

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3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
b. Acceptance Criteria - System Operation Requirements 5.5.6.c The results of laboratory carbon sample analysis shall show ≤0.5% methyl iodide penetration when tested at 30°C and 95% relative humidity.	5.5.6 b. At least once per 720 hours of system operation, 5.5.6.c remove one carbon test sample from each charcoal adsorber bank. Subject this sample to a laboratory analysis to verify methyl iodide removal efficiency.

A.1

ITS | ITS

	3.0 LIMITING CONDITIONS FOR OPERATION		4.0 SURVEILLANCE REQUIREMENTS
	c. The system shall be shown to be operable with:	5.5.6	c. At least once per ~~operating cycle, but not to~~ exceed ~~18~~ 24 months, the following conditions shall be demonstrated for each emergency filtration system train: (24 months — A.5) (A.9)
5.5.6.d	(1) Combined filter pressure drop ≤8 inches water.	5.5.6.d	(1) Pressure drop across the combined filters of each train shall be measured at 1000 cfm (±10%) flow rate.
5.5.6.e	(2) Inlet heater power output 5kw±10%.	5.5.6.e	(2) Operability of inlet heater at nominal rated power shall be verified.
	(3) Automatic initiation upon receipt of a high radiation signal. — See ITS 3.7.4		(3) Verify that on a simulated high radiation signal, the train switches to the pressurization mode of operation and the control room is maintained at a positive pressure with respect to adjacent areas at the design flow rate of 1000 cfm (±10%). — See ITS 3.7.4
	3. Post Maintenance Requirements		3. Post Maintenance Testing
5.5.6 / 5.5.6.a	a. After any maintenance or testing that could affect the HEPA filter or HEPA filter mounting frame leak tight integrity, the results of the in-place DOP tests at 1000 cfm (±10%) shall show ≤1% DOP penetration on each individual HEPA filter and shall show ≤0.05% DOP penetration on the combined HEPA filters.	5.5.6 / 5.5.6.a	a. After any maintenance or testing that could affect the leak tight integrity of the HEPA filters, perform in-place DOP tests on the HEPA filters.
5.5.6 / 5.5.6.b	b. After any maintenance or testing that could affect the charcoal adsorber leak tight integrity, the results of in-place halogenated hydrocarbon tests at 1000 cfm (±10%) shall show ≤1% penetration on each individual charcoal adsorber and shall show ≤0.05% penetration on the combined charcoal adsorber banks.	5.5.6 / 5.5.6.b	b. After any maintenance or testing that could affect the leak tight integrity of the charcoal adsorber banks, perform halogenated hydrocarbon tests on the charcoal adsorbers.
			The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test Frequencies (A.4)

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DISCUSSION OF CHANGES
ITS 5.5, PROGRAMS AND MANUALS

ADMINISTRATIVE CHANGES

A.1 In the conversion of the Monticello 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-1433, Rev. 3, "Standard Technical Specifications General Electric Plants, BWR/4" (ISTS).

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

A.2 CTS 6.8.B includes the program requirements for the Primary Coolant Sources Outside Containment Program and includes a statement that a program acceptable to the Commission was described in a letter dated December 31, 1979, from L.O. Mayer, NSP, to Director of Nuclear Reactor Regulation, "Lessons Learned Implementation." ITS 5.5.2 contains the requirements for the Primary Coolant Sources Outside Containment, however the statement concerning a type of NRC-acceptable program is not included. This changes the CTS by deleting this additional statement.

The purpose of CTS 6.8.B is to define the requirements for the Primary Coolant Sources Outside Containment Program. The purpose of the letter is to describe acceptable methods in which the utility is meeting requirements in CTS 6.8.B. The requirements in CTS 6.8.B, CTS 6.8.B.1, and CTS 6.8.B.2 provide adequate information for the details of the program. These requirements are incorporated into ITS 5.5.2, ITS 5.5.2.a, and ITS 5.5.2.b. These requirements are consistent with NUREG-1433, Revision 3. This change is designated as administrative because it does not result in technical changes to the CTS.

A.3 CTS 6.8.M includes the program requirements for the Primary Containment Leakage Rate Testing Program. CTS 6.8.M.1 includes an exception from the requirements of Regulatory Guide 1.1.63, "Performance-Based Containment Leak-Test Program," dated September 1995. CTS 6.8.M.6 states that "Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J." This statement is not included in the ITS. This changes the CTS by deleting the CTS 6.8.M.6 statement.

The statement CTS 6.8.M.6 that "Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J" has been deleted because the phrase is not consistent with the allowances in CTS 6.8.M.1, which states that the 10 CFR 50, Appendix J, Option B requirements may be modified by the approved exception. This change is acceptable because the statement is inconsistent with the allowances in CTS 6.8.M.1. This change is designated as administrative because it does not result in technical changes to the CTS.

A.4 The Performance Requirements (CTS 3.7.B.2.a and CTS 3.7.B.2.b), Post Maintenance Requirements (CTS 3.7.B.3.a and CTS 3.7.B.3.b), Performance Requirement Tests (4.7.B.2.a, 4.7.B.2.b, and 4.7.B.2.c), and Post Maintenance Testing (4.7.B.3.a and 4.7.B.3.b) requirements associated with the ventilation filter testing for the Standby Gas Treatment (SGT) System and the Performance

DISCUSSION OF CHANGES
ITS 5.5, PROGRAMS AND MANUALS

Requirements (CTS 3.17.B.2.a, CTS 3.17.B.2.b, CTS 3.17.B.2.c.(1), and CTS 3.17.B.2.c.(2)), Post Maintenance Requirements (CTS 3.17.B.3.a and CTS 3.17.B.3.b), Performance Requirement Tests (CTS 4.17.B.2.a, CTS 4.17.B.2.b, CTS 4.17.B.2.c.(1), and CTS 4.17.B.2.c.(2)), and Post Maintenance Testing (CTS 4.17.B.3.a and CTS 4.17.B.3.b) requirements associated with the ventilation filter testing for the Control Room Emergency Filtration (CREF) System have been placed in a program in the proposed Administrative Controls Chapter 5.0 (ITS 5.5.6). As such, a general program statement has been added as ITS 5.5.6. 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 apply. This changes the CTS by moving the ventilation filter testing Surveillances associated with the SGT and CREF Systems 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 requirements. 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.

A.5 CTS 4.7.B.2.a requires the performance of an in-place DOP test of the SGT System HEPA filter banks, an in-place test of the SGT charcoal adsorber banks with halogenated hydrocarbon tracer, and a laboratory analysis of a carbon test sample from the SGT charcoal adsorber once per "operating cycle." CTS 4.7.B.2.c requires the performance of the SGT System heater test once per "operating cycle." CTS 4.17.B.2.a requires the performance of an in-place DOP test of the CREF System HEPA filter banks, an in-place test of the CREF charcoal adsorber banks with halogenated hydrocarbon tracer, and a laboratory analysis of a carbon test sample from the CREF charcoal adsorber once per "operating cycle." CTS 4.17.B.2.c requires the performance of the CREF System heater test and combined filter pressure drop test once per "operating cycle." ITS 5.5.6 requires the same tests, however the Surveillances are required to be performed every "24 months." This changes the CTS by changing the Frequency from "operating cycle" to "24 months."

This change is acceptable because the current "operating cycle" is "24 months." In letter L-MT-04-036, from Thomas J. Palmisano (NMC) to the USNRC, dated June 30, 2004, NMC has proposed to extend the fuel cycle from 18 months to 24 months and the same time has performed an evaluation in accordance with Generic Letter 91-04 to extend the unit Surveillance Requirements from 18 months to 24 months. CTS 4.7.B.2.a, CTS 4.7.B.2.c, CTS 4.17.B.2.a, and CTS 4.17.B.2.c were included in this evaluation. This change is designated as administrative because it does not result in any technical changes to the CTS.

A.6 CTS 4.7.D.4 requires the replacement of the seat seal of the drywell and suppression chamber 18 inch purge supply and vent valves once per "six operating cycles." ITS 5.5.11.e requires the same replacement, however the replacement is required every "9 years." In addition, a statement of the

DISCUSSION OF CHANGES
ITS 5.5, PROGRAMS AND MANUALS

applicability of ITS SR 3.0.2 has been added. This changes the CTS by changing the Frequency from "six operating cycles" to "9 years" and specifically stating the applicability of ITS SR 3.0.2.

This change is acceptable because the current "operating cycle" is "18 months" and CTS 4.0.B (ITS SR 3.0.2) is applicable to CTS 4.7.D.4. This change is designated as administrative because it does not result in any technical changes to the CTS.

A.7 The Surveillance associated with diesel fuel oil testing (CTS 4.9.B.3.b.3)) has been placed in a program in the proposed Administrative Controls Chapter 5.0 (ITS 5.5.8). As such, a general program statement has been added as ITS 5.5.8. 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 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. Other changes to the Surveillance are discussed in DOCs M.2 and DOC L.2.

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.

A.8 CTS 6.8.G requires pump and valve testing per the requirements of Section XI of the ASME Boiler and Pressure Vessel Code. ITS 5.5.6 requires pump and valve testing per the requirements of the ASME Operation and Maintenance (OM) Code. This changes the CTS by referring to the ASME OM Code instead of ASME Boiler and Pressure Code, Section XI.

In the 1987 Addenda to the 1986 edition of ASME Boiler and Pressure Vessel Code, Section XI, the requirements for Inservice Testing were removed and relocated to the ASME/ANSI OM Code. This change was endorsed in 10 CFR 50.55a. 10 CFR 50.55a(f) now addresses the requirements for inservice testing using the ASME/ANSI OM Code and 10 CFR 50.55a(g) addresses the requirements for inservice inspection using ASME Boiler and Pressure Vessel Code, Section XI. The CTS has been revised to incorporate the current Code requirements. In addition, the terms 48 months, 5 years, 8 years, and 10 years are used in the applicable ASME/ANSI OM Code. Therefore, these Frequencies have been added. The Monticello Inservice Testing Program for pumps and valves complies with the 1995 Edition, 1996 Addenda of ASME Operations and Maintenance (OM) Code. This change was submitted to the NRC in a NMC letter from Jeffrey S. Forbes (NMC) to USNRC, dated November 22, 2002. This change is designated as administrative because it does not result in technical changes to the CTS.

A.9 These changes to CTS 4.7.B.2.a, CTS 4.7.B.2.b, CTS 4.17.B.2.a, and CTS 4.17.B.2.c are provided in the Monticello ITS consistent with the Technical Specifications Change Request submitted to the USNRC for approval in NMC letter L-MT-04-036, from Thomas J. Palmisano (NMC) to USNRC, dated June 30, 2004. As such, these changes are administrative.

MORE RESTRICTIVE CHANGES

M.1 The CTS does not include program requirements for a Component Cycle or Transient Limit Program, Safety Function Determination Program, or Battery Monitoring and Maintenance Program. The ITS includes programs for these activities. This changes the CTS be adding the following programs:

ITS 5.5.4, "Component Cyclic or Transient Limit";
ITS 5.5.10, "Safety Function Determination Program (SFDP)"; and
ITS 5.5.12, "Battery Monitoring and Maintenance Program."

The Component Cyclic or Transient Limit Program is included to ensure controls are in place to track the requirements of USAR Table 4.2-1. The Safety Function Determination Program is included to support implementation of the support system OPERABILITY characteristics of the Technical Specifications. The Battery Monitoring and Maintenance Program is included to provide for battery restoration and maintenance. The specific wording associated with these programs may be found in ITS 5.5.4, ITS 5.5.10, and ITS 5.5.12. The changes are acceptable because they support implementation of the requirements of the ITS and the USAR. This change is designated as more restrictive because it imposes additional programmatic requirements in the Technical Specifications.

M.2 CTS 4.9.B.3.b.3) includes a requirement to sample and check for quality of the diesel fuel every month. Currently, this is met by performing a viscosity check and a water and sediment check of the stored fuel oil in the common storage tank. In addition, no testing is currently required on new fuel oil prior to addition to the common storage tank. ITS 5.5.8.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 within limit, a flash point and saybolt viscosity within limits, and a water and sediment content within limits. ITS 5.5.8.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 5.5.8.c requires the total particulate concentration of the stored fuel oil to be $\leq$ 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 common storage tank and providing a requirement that the total particulate concentration of the stored fuel oil be $\leq$ 10 mg/l when tested every 31 days.

The purpose of ITS 5.5.8.a and ITS 5.5.8.b are to ensure that only high quality fuel oil is added to the storage tank. The purpose of ITS 5.5.8.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.

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RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA.1 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 6.8.G states that the Inservice Testing Program provides controls for inservice testing of Quality Group A, B, and C pumps and valves which shall be performed in accordance with the requirements of ASME Code Class 1, 2, and 3 pumps and valves, respectively. ITS 5.5.5 only 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 that the "Quality Group A, B, and C pumps and valves" corresponds to the ASME Code Class 1, 2, and 3 pumps and valves, respectively, from the Technical Specifications to the Inservice Testing Program.

The removal of these details for 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.

LA.2 *(Type 1 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 6.8.I includes limits for the liquid holdup tank and the explosive gas mixture. The specific limits 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 specific limits, from the Technical Specifications to the Technical Requirements Manual (TRM).

The removal of these details, which are related to system design, 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 5.5.7 still retains the requirement to include a program, which provides controls for potentially explosive gas mixtures contained in the Offgas Treatment System and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks. Also, this change is acceptable because the limits will be adequately controlled in the TRM. Any changes to the TRM are made under 10 CFR 50.59, which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

DISCUSSION OF CHANGES
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LESS RESTRICTIVE CHANGES

L.1 *(Category 7 – Relaxation Of Surveillance Frequency, Non-24 Month Type Change)* CTS 4.7.B.2.a, in part, requires the performance of an in-place DOP test of the SGT System HEPA filter banks, an in-place test of the SGT charcoal adsorber banks, and a laboratory analysis of a carbon test sample from the SGT charcoal adsorber at least once per 720 hours of system operation. ITS 5.5.6 does not require the in-place DOP test of the HEPA filter banks or an in-place test of the charcoal adsorber bank at least once per 720 hours of system operation. This changes the CTS by deleting the test requirements to perform an in-place DOP test of the HEPA filter banks and an in-place test of the charcoal adsorber banks every 720 hours of system operation.

The purpose of CTS 4.7.B.2.a is to prescribe testing requirements for the Standby Gas Treatment System consistent with Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants." This Regulatory Guide only requires a laboratory analysis of a carbon test sample from the charcoal adsorber to be performed after every 720 hours of system operation. The other tests (in-place DOP test of the HEPA filter banks and in-place test of the charcoal adsorber banks) are not required to be performed at this Frequency. This change acceptable and consistent with the current requirements for filter testing of the CREF System in CTS 4.17.B.2.a. This change is designated as less restrictive because Surveillances will be performed less frequently under the ITS than under the CTS.

L.2 *(Category 7 – Relaxation Of Surveillance Frequency, Non-24 Month Type Change)* CTS 4.9.B.3.b.3) requires a sample and check for quality of the diesel fuel every month. Currently, this is met by performing a viscosity check and a water and sediment check. ITS 5.5.8.c only requires total particulate concentration of the fuel oil to be tested every 31 days. This changes the CTS by deleting the monthly viscosity and water and sediment checks of stored fuel oil.

The purpose of CTS 4.9.B.3.b.3) is to ensure that the quality of the diesel fuel oil is acceptable so that the emergency diesel generators can perform their safety function. This change is acceptable because the new Surveillance Frequency has been evaluated to ensure that it provides an acceptable level of equipment reliability. ITS 5.5.8.a restricts the acceptance of new fuel oil for use prior to addition to storage tanks until the determination that the fuel oil has an API gravity within limit, a flash point and saybolt viscosity within limits, and a water and sediment content within limits. ITS 5.5.8.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 5.5.8.a and ITS 5.5.8.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 5.5.8.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

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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, SR 3.8.3.4 has been added (see Discussion of Changes 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 bacterial survival. This change is designated as less restrictive because Surveillances will be performed less frequently under the ITS than under the CTS.

L.3 *(Category 6 – Relaxation Of Surveillance Requirement Acceptance Criteria)* CTS 4.17.B.2.a.(4) requires the CREF System to be initiated "from the control room" with a flow of 1000 cfm (± 10%). ITS SR 5.5.6.a and 5.5.6.b do not specify how to initiate the system. This changes the CTS by deleting the requirement to start the system from the control room.

The purpose of CTS 4.17.B.2.a.(4), in part, is to ensure each CREF System can be started from the control room periodically (i.e., every 24 months) or following certain conditions (i.e., following painting, fire, or chemical release). This change is acceptable because it has been determined that the relaxed Surveillance Requirement acceptance criteria are not necessary for verification that the equipment used to meet the LCO can perform its required functions. This specific requirement to start the CREF System from the control room has been deleted, however the ability to start the system from the control room is also currently required by another Surveillance Requirement. CTS 4.17.B.1 states to "initiate from the control room" flow through CREF subsystem and operate for at least 10 hours. The Surveillance is required to be performed every 31 days. ITS SR 3.7.4.1 includes the same requirement, however, the statement to "initiate from the control room" is not included but has been relocated to the Bases in accordance with the Discussion of Changes for ITS 3.7.4. (DOC LA.1). Therefore, the CREF System will still be required to be started from the control room. This change is acceptable because ITS SR 3.7.4.1 will continue to periodically start the CREF System from the control room. This change is designed as less restrictive because less stringent Surveillance Requirements are being applied in the ITS than were applied in the CTS.

L.4 CTS 6.8.B includes the Primary Coolant Sources Outside Containment program requirements. The Combustible Gas Control System is included in this program. ITS 5.5.2 includes the same program requirements for the Primary Coolant Sources Outside Containment Program, except the Combustible Gas Control System is not included in the program. This changes the CTS by deleting the program requirement for the Combustible Gas Control System in the Primary Coolant Sources Outside Containment Program.

The purpose of CTS 6.8.B is to ensure controls are in place 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 practical. The Technical Specification requirements governing the

DISCUSSION OF CHANGES
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OPERABILITY of the Combustible Gas Control System have previously been removed from the Monticello Technical Specifications as documented in License Amendment 138, dated May 21, 2004. However the License Amendment did not remove the Combustible Gas Control System from the program requirements of CTS 6.8.B since the Residual Heat Removal System cooling water supply was still available to the Combustible Gas Control System (i.e., the potential for coolant leakage that could be highly radioactive during a transient or accident still existed). A plant modification has been completed that removes all communication between the Combustible Gas Control System and the containment and eliminated the Residual Heat Removal System cooling water supply lines to the Combustible Gas Control System. Thus, the potential for the Combustible Gas Control System to contain highly radioactive fluids no longer exists. Therefore, the program controls for this system in CTS 6.8.B are no longer necessary. This change is considered less restrictive because the program requirement for the Combustible Gas Control System in the Primary Coolant Sources Outside Containment Program has been deleted.

L.5 CTS 6.8.B.2 specifies that the integrated leak test requirements for each system outside containment that could contain highly radioactive fluids during a serious transient or accident must be performed at a refueling cycle interval or less. CTS 6.8.B also states that CTS 4.0.B is applicable (i.e., a 25% grace period is allowed). ITS 5.5.2.b specifies that the same test must be performed at least once per 24 months and ITS 5.5.2 states that the provisions of ITS SR 3.0.2 are applicable. This changes the CTS by extending the Frequency of the Surveillance from 18 months (i.e., the current Monticello frequency for this test, based on the previous refueling outage interval) to 24 months (i.e., a maximum of 30 months accounting for the allowable grace period specified in ITS SR 3.0.2).

The purpose of CTS 6.8.B.2 is to ensure the leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident is reduced to as low as practicable levels. This change was evaluated in accordance with the guidance provided in NRC Generic Letter No. 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991. Reviews of historical surveillance data and maintenance data sufficient to determine failure modes have shown that these tests normally pass their Surveillances at the current Frequency. An evaluation has been performed using this data, and it has been determined that the effect on safety due to the extended Surveillance Frequency will be minimal. Extending the Surveillance test interval for the System Integrity integrated leak test verification SR is acceptable because most portions of the subject systems included in this program are visually walked down, while the plant is operating, during plant testing, and/or operator/system engineer walkdowns. In addition, housekeeping/safety walkdowns also serve to detect any gross leakage. If leakage is observed from these systems, corrective actions will be taken to repair the leakage. Finally, the plant radiological surveys will also identify any potential sources of leakage. These visual walkdowns and surveys provide monitoring of the systems at a greater frequency than once per refueling cycle, and support the conclusion that the impact, if any, on safety is minimal as a result of the proposed changes. Based on the inherent system and component reliability and the testing performed during the operating cycle, the impact, if any, from this change on system availability is minimal. The review of historical

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surveillance data also demonstrated that there are no failures that would invalidate this conclusion. In addition, the proposed 24 month Surveillance Frequency, if performed at the maximum interval allowed by ITS SR 3.0.2 (30 months) does not invalidate any assumptions in the plant licensing basis. This change is designated as less restrictive because Surveillances will be performed less frequently under the ITS than under the CTS.

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

CTS

5.0 ADMINISTRATIVE CONTROLS

6.8 5.5 Programs and Manuals

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

6.8.A 5.5.1 Offsite Dose Calculation Manual (ODCM)

6.8.A.1 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

6.8.A.2 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 [5.6.2] and Specification [5.6.3]. TSTF-369 2

c. 1 2

6.8.A.3 Licensee initiated changes to the ODCM: 3

1

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

a)

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

b)

6.8.A.3.a.2) 2. A determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.1302, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations. ; 3 1

2

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

3

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

CTS

5.5 Programs and Manuals

6.8.B 5.5.2 Primary Coolant Sources Outside Containment

6.8.B This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The systems include [the ~~Low Pressure~~ Core Spray, High Pressure Coolant Injection, Residual Heat Removal, Reactor Core Isolation Cooling, ~~hydrogen recombiner,~~ process sampling, and Standby Gas Treatment]. The program shall include the following: 2 2

6.8.B.1 a. Preventive maintenance and periodic visual inspection requirements ~~and~~ ; 1

6.8.B.2 b. Integrated leak test requirements for each system at least once per [~~18~~] 24 months. 2

6.8.B 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 NEDO-32991, Revision 0, "Regulatory Relaxation For BWR Post Accident Sampling Stations (PASS)," and the associated NRC Safety Evaluation dated June 12, 2001.~~ 4

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

~~a. Training of personnel,~~

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

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

6.8.D 5.5.~~4~~ 3 Radioactive Effluent Controls Program 4

6.8.D 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.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

CTS

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

3

4

6.8.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; 1

6.8.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.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

6.8.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; 1

6.8.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.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:

6.8.D.7.a 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 ; 1

6.8.A.7.b 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; 1

6.8.A.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.A.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; 1

CTS

5.5 Programs and Manuals

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

4

6.8.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~~, and~~ ; 1

6.8.D.11 k. Limitations on venting and purging of the ~~Mark II~~ containment through the Standby Gas Treatment System to maintain releases as low as reasonably achievable ~~(in BWR/4s with Mark II containments)~~. 5 5

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

DOC M.1 5.5.~~5~~ 4 Component Cyclic or Transient Limit 6 4

U Table 4.2-1 6

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

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

~~This program provides controls for monitoring any tendon degradation in pre-stressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial operations. The Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with [Regulatory Guide 1.35, Revision 3, 1990].~~ 7

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

6.8.G 5.5.~~7~~ 5 Inservice Testing Program 7

6.8.G 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:~~ 8 8

CTS

5.5 Programs and Manuals

5.5.~~7~~ 5 Inservice Testing Program (continued) 7

6.8.G.1 a. Testing frequencies specified in ~~Section XI of~~ the ASME ~~Boiler and Pressure Vessel Code~~ Operation and Maintenance (OM) Code and applicable Addenda as follows: 13 8 8

ASME ~~Boiler and Pressure Vessel Code~~ OM Code and applicable Addenda terminology for inservice testing activities	Required Frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Biquarterly	At least once per 46 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly, every 12 months, or annually	At least once per 366 days
Biennially, every 24 months, or every 2 years	At least once per 731 days

INSERT 1 8 6 8 8

6.8.G.2 b. The provisions of SR 3.0.2 are applicable to the above required Frequencies for performing inservice testing activities. 13

6.8.G.3 c. The provisions of SR 3.0.3 are applicable to inservice testing activities. ~~, and~~ 13

6.8.G.4 d. Nothing in the ASME ~~Boiler and Pressure Vessel~~ OM Code shall be construed to supersede the requirements of any TS. 8

DOC A.4 5.5.~~8~~ 6 Ventilation Filter Testing Program (VFTP) 7

DOC A.4 A program shall ~~be~~ establish~~ed to implement~~ the ~~following~~ required testing of Engineered Safety Feature (ESF) filter ventilation systems. ~~at the frequencies specified in [Regulatory Guide], and in accordance with [Regulatory Guide 1.52, Revision 2, ASME N510-1989, and AG-1].~~ INSERT 1A 9 2 10

3.7.B.2.a.(1), 4.7.B.2.a.(1), 3.7.B.3.a, 4.7.B.3.a, 3.17.B.2.a.(1), 4.17.B.2, 4.17.B.2.a.(1), 4.17.B.2.a.(4) 3.17.B.3.a, 4.17.B.3.a

INSERT 1B from page 5.5-7

a. Demonstrate for each of the ESF systems that an inplace test of the ~~high efficiency particulate air (~~HEPA~~)~~ filters shows a penetration and system bypass ~~< [0.05]%~~ when tested in accordance with ~~[~~Regulatory Guide 1.52, Revision 2, and ~~ASME~~ ANSI N510-1989~~]~~ at the system flowrate specified below ~~[± 10%]~~. 9 2 13

specified below

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

INSERT 2 10

10

CTS

8 **INSERT 1**

DOC A.8	Every 48 months	At least once per 1461 days
	Every 5 years	At least once per 1827 days
	Every 8 years	At least once per 2922 days
	Every 10 years	At least once per 3653 days

10 **INSERT 1A**

4.7.B.2.a, 4.17.B.2.a Tests described in Specifications 5.5.6.a and 5.5.6.b shall be performed once per 24 months and following painting, fire, or chemical release in any ventilation zone communicating with the subsystem while it is in operation that could adversely affect the high efficiency particulate air (HEPA) filters or charcoal adsorber capability.

3.7.B.3.a, 4.7.B.3.a, 3.17.B.3.a, 4.17.B.3.a The test described in Specification 5.5.6.a shall be performed after any maintenance or testing that could affect the leak tight integrity of the HEPA filters.

3.7.B.3.b, 4.7.B.3.b, 3.17.B.3.b, 4.17.B.3.b The test described in Specification 5.5.6.b shall be performed after any maintenance or testing that could affect the leak tight integrity of the charcoal adsorber banks.

4.7.B.2.a, 4.17.B.2.a, 4.17.B.2.b Tests described in Specification 5.5.6.c shall be performed once per 24 months; at least once per 720 hours of system operation; following painting, fire, or chemical release in any ventilation zone communicating with the subsystem while it is in operation that could adversely affect the charcoal adsorber capability.

4.7.B.2.b, 4.17.B.2.c The tests described in Specification 5.5.6.d shall be performed once per 92 days for the Standby Gas Treatment (SGT) System and once per 24 months for the Control Room Emergency Filtration (CREF) System.

4.7.B.2.c, 4.17.B.2.c The test described in Specification 5.5.6.e shall be performed once per 24 months.

10 **INSERT 2**

ESF Ventilation System	Penetration (%)	Flowrate (cfm)
SGT System	$\leq$ 1.0	$\geq$ 3,150 and $\leq$ 3,850
CREF System	$\leq$ 1.0 for each individual HEPA filter and $\leq$ 0.05 for each pair of HEPA filters	$\geq$ 900 and $\leq$ 1,100

CTS

5.5 Programs and Manuals

5.5.8 Ventilation Filter Testing Program (continued) 6 7

3.7.B.2.a.(2), 4.7.B.2.a.(2), 3.7.B.3.b, 4.7.B.3.b, 3.17.B.2.a.(2), 4.17.B.2, 4.17.B.2.a.(2), 4.17.B.2.a.(4), 3.17.B.3.b, 4.17.B.3.b

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

specified below 2 ANSI 2 13 INSERT 3 10

ESF Ventilation System	Flowrate
[ ]	[ ]

3.7.B.2.a.(3), 4.7.B.2.a.(3), 3.17.B.2.a.(3), 4.17.B.2, 4.17.B.2.a.(3), 3.17.B.2.b, 4.17.B.2.b

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 specified below.

Regulatory Position C.6.b of 10

ESF Ventilation System	Penetration	RH	Face Velocity (fps)
[ ]	[See Reviewer's Note]	[See Reviewer's Note]	[See Reviewer's Note]

10 INSERT 4

-----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. 11

Allowable Penetration = [(100% - Methyl Iodide Efficiently * for Charcoal Credited in Licensee's Accident Analysis) / Safety Factor]

When ASTM D3803-1989 is used with 30°C (86°F) and 95% RH (or 70% RH with humidity control) is used, the staff will accept the following:

Safety factor ≥ 2 for systems with or without humidity control.

10

INSERT 3

ESF Ventilation System	Penetration (%)	Flowrate (cfm)
SGT System	≤ 1.0	≥ 3,150 and ≤ 3,850
CREF System	≤ 1.0 for each individual charcoal adsorber section and ≤ 0.05% for each pair of charcoal adsorber sections	≥ 900 and ≤ 1,100

10

INSERT 4

ESF Ventilation System	Penetration (%)	RH (%)
SGT System	≤ 5.0	95
CREF System	≤ 0.5	95

CTS

5.5 Programs and Manuals

DOC A.4 5.5.8 6 Ventilation Filter Testing Program (continued) 7

Humidity control can be provided by heaters or an NRC-approved analysis that demonstrates that the air entering the charcoal will be maintained less than or equal to 70 percent RH under worst-case design-basis conditions.

If the system has a face velocity greater than 110 percent of 0.203 m/s (40 ft/min), the face velocity should be specified. 11

*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.

3.7.B.2.b.(1), 4.7.B.2.b, 3.17.B.2.c.(1), 4.17.B.2.c.(1)

d. Demonstrate for each of the ESF systems that the pressure drop across filters 12 the combined [HEPA filters, the prefilters, and the charcoal adsorbers] is ~~less than the value~~ as 10 specified below when tested in accordance with [Regulatory Guide 1.52, Revision 2, and ~~ASME~~ N510-1989] at the system flowrate specified below [± 10%]. ANSI 2 2 13

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

INSERT 5 10

3.7.B.2.b.(2), 4.7.B.2.c, 3.17.B.2.c.(2), 4.17.B.2.c.(2)

[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]. ANSI 2 13

ESF Ventilation System	Wattage]
[ ]	[ ]

INSERT 6 10

DOC A.4 {move to page 5.5-5 as INSERT 1B}

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies. 10

7 7

6.8.I 5.5.9 Explosive Gas and Storage Tank Radioactivity Monitoring Program

Offgas Treatment System 2

6.8.I This program provides controls for potentially explosive gas mixtures contained in the [~~Waste Gas Holdup System~~], [the quantity of radioactivity contained in gas storage tanks or fed into the offgas treatment system, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks]. ~~The 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 16

10 **INSERT 5**

ESF Ventilation System	Delta P (inches water gauge)	Flowrate (cfm)
SGT System	≤ 6	$\geq$ 3,150 and $\leq$ 3,850
CREF System	≤ 8	$\geq$ 900 and $\leq$ 1,100

10 **INSERT 6**

ESF Ventilation System	Nominal Wattage (kW)
SGT System	≥ 18
CREF System	$\geq$ 4.5 and $\leq$ 5.5

CTS

5.5 Programs and Manuals

5.5.~~9~~ 7 Explosive Gas and Storage Tank Radioactivity Monitoring Program (continued) 7

The program shall include:

6.8.I.1 a. The limits for concentrations of hydrogen and oxygen in the ~~[Waste Gas Holdup System]~~ Offgas Treatment System and a ~~s~~Surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion)~~,~~ ; 2 13 1

6.8.I.2 b. A ~~s~~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 13 2 2 1

6.8.I.3 c. A ~~s~~Surveillance program to ensure that the quantity of radioactivity contained in all outdoor liquid radwaste tanks that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the [Liquid Radwaste Treatment System] is less than the amount that would result in concentrations less than the limits of 10 CFR 20, Appendix B, Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tanks' contents. 13 2

6.8.I 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

DOC A.6 5.5.~~10~~ 8 Diesel Fuel Oil Testing Program 7

DOC A.6 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:

4.9.B.3.b.3) a. Acceptability of new fuel oil for use prior to addition to the storage tank~~s~~ by determining that the fuel oil has: 6

1. An API gravity ~~or an absolute specific gravity~~ within limits~~,~~ ; 1 14

2. A flash point and ~~kinematic~~ saybolt viscosity within limits ~~for ASTM 2D fuel oil~~, and 14

3. A ~~clear and bright appearance with proper color or a~~ water and sediment content within limits~~,~~ ; 14 1

CTS

5.5 Programs and Manuals

5.5.10 8 Diesel Fuel Oil Testing Program (continued) 7

4.9.B.3.b.3) b. Within 31 days following addition of the new fuel oil to the storage tanks, verify that the properties of the new fuel oil, other than those addressed in a. Specification 5.5.8.a above, are within limits ~~for ASTM 2D fuel oil~~, ; and 6 9 14 1

4.9.B.3.b.3) c. Total particulate concentration of the fuel oil is ≤ 10 mg/l when tested every 31 days.

DOC A.6 The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program test frequencies. 13

6.8.K 5.5.11 9 Technical Specifications (TS) Bases Control Program 7

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

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

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

6.8.K.2.a 1. A change in the TS incorporated in the license ; or 1

6.8.K.2.b 2. A change to the updated U FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59. 6

6.8.K.3 c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the U FSAR. 6

6.8.K.4 d. Proposed changes that meet the criteria of Specification 5.5.11b 9 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). 7 13

DOC M.1 5.5.12 10 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 actions may be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6. ¶a. The SFDP shall contain the following: 3

CTS

5.5 Programs and Manuals

DOC M.1 5.5.~~12~~ 10 Safety Function Determination Program (continued) 7

1 ~~a~~. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected~~,~~ ; 3 1

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

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

4 ~~d~~. Other appropriate limitations and remedial or compensatory actions. 1

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 emergency 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 6

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

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

3 ~~c~~. A required system redundant to the support system(s) for the supported systems ~~(a) and (b)~~ described in Specifications 5.5.10.b.1 and 5.5.10.b.2 above is also inoperable. 3 9

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

6.8.M 5.5.~~13~~ 11 Primary 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.~~ 15

CTS

5.5 Programs and Manuals

5.5.13 Primary Containment Leakage Rate Testing Program (continued) 7
11

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

c. Leakage rate acceptance criteria are:

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

2. Air lock testing acceptance criteria are: 15

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

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

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

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

[OPTION B]

6.8.M.1 a. A program shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September, 1995], as modified by the following exceptions: 2 2

1. [. .] INSERT 7 15

6.8.M.2 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]. 56 42 2

6.8.M.3 c. The maximum allowable containment leakage rate, L_a, at P_a, shall be [ ]% of containment air weight per day. 1.2 2

6.8.M.4 d. Leakage rate acceptance criteria are:

5.5

15 2

INSERT 7

The Type A testing Frequency specified in NEI 94-01, Revision 0, Paragraph 9.2.3, as "at least once per 10 years based on acceptable performance history" is modified to be "at least once per 15 years based on acceptable performance history." This change applies only to the interval following the Type A test performed in March 1993.

CTS 5.5 Programs and Manuals

5.5.13 11 Primary Containment Leakage Rate Testing Program (continued) 7

6.8.M.4.a 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.

6.8.M.4.b 2. Air lock testing acceptance criteria are:

6.8.M.4.b.1) a) Overall air lock leakage rate is ≤ [0.05 L_a] when tested at ≥ P_a. 2

0.007

6.8.M.4.b.2) b) For each door, leakage rate is ≤ [~~0.01~~ L_a] when pressurized to [≥ 10 psig]. 2

f INSERT 8 15

6.8.M.5 ~~e~~. The provisions of SR 3.0.3 are applicable to the Primary Containment Leakage Rate Testing Program. 3

~~f. Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J.~~ 15

~~[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:~~ 15

~~1. . . .]~~

~~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:~~

CTS

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INSERT 8

4.7.D.4

e. The resilient seals of each 18 inch primary containment purge and vent valve shall be replaced at least once every 9 years. The provisions of SR 3.0.2 are applicable to this requirement. If a common mode failure attributable to the resilient seals is identified based on the results of SR 3.6.1.3.11, the resilient seals of all 18 inch primary containment purge and vent valves shall be replaced.

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CTS

5.5 Programs and Manuals

5.5.13 11 Primary Containment Leakage Rate Testing Program (continued) 7

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

2. Air lock testing acceptance criteria are: 15

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

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

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

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

DOC M.1 5.5.14 12 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] of the following: 13 2 2

a. Actions to restore battery cells with float voltage < [2.13] V; and 2 ; 1

b. Actions to equalize and test battery cells that had been discovered with electrolyte level below the minimum established design limit.

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, NEI 01-03, Section 5.1.3.

2. The brackets are removed and the proper plant specific information/value is provided.

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

4. The bracketed ISTS 5.5.3, Post Accident Sampling, is not included in the CNP Units 1 and 2 ITS. The requirements for Post Accident Sampling have been deleted from the CTS in License Amendments 136 dated June 17, 2003. This deletion was based on the Monticello implementation of NEDO-32991, Revision 0. Subsequent programs have been renumbered, as necessary.

5. The Monticello design does not include a Mark II containment, however it does require this limit. Therefore, ISTS 5.5.4.k (ITS 5.5.3.j) has been modified to reflect the current licensing requirements.

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

7. ISTS 5.5.6 provides requirements for the Pre-Stressed Concrete Containment Tendon Surveillance Program. Monticello does not have a pre-stressed concrete containment tendons in the primary containment. Therefore, this ISTS program is not included in the Monticello ITS. Subsequent programs have been renumbered, as necessary.

8. 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. Also, in the 1987 Addenda to the 1986 edition of ASME Boiler and Pressure Vessel Code, Section XI, the requirements for Inservice Testing were removed and relocated to the ASME/ANSI OM Code. This change was endorsed in 10 CFR 50.55a. 10 CFR 50.55a(f) now addresses the requirements for inservice testing using the ASME/ANSI OM Code and 10 CFR 50.55a(g) addresses the requirements for inservice inspection using ASME Boiler and Pressure Vessel Code, Section XI. The ITS has been revised to incorporate the current ASME/ANSI OM Code requirements. In addition, the terms every 12 months, 24 months,

JUSTIFICATION FOR DEVIATIONS
ITS 5.5, PROGRAMS AND MANUALS

48 months, 5 years, 8 years, and 10 years are used in the ASME/ANSI OM Code and have been added.

9. Editorial changes made for enhanced clarity or to be consistent with the Writer's Guide.

10. ISTS 5.5.8 (ITS 5.5.6) provides requirements for the Ventilation Filter Testing Program. ITS 5.5.6 is revised to reflect the Monticello licensing bases. In addition, for clarity, the ISTS discussion concerning the provisions of SR 3.0.2 and SR 3.0.3 have been moved from the end of this Specification to just after the discussion of the Frequencies, since it applies only to the Frequencies.

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

12. The Standby Gas Treatment System at Monticello does not include a prefilter. Therefore, the phrase "combined HEPA filters, the prefilters, and the charcoal adsorbers" has been changed to "combined filters" to be consistent with the current licensing basis. While the Control Room Emergency Filtration System does have prefilters, the term "combined filters" adequately covers prefilters.

13. Typographical/grammatical error corrected.

14. The following changes have been made to ISTS 5.5.10 (ITS 5.5.8):

    a. The allowance to determine absolute specific gravity instead of API gravity has been deleted, consistent with current practice;

    b. Saybolt viscosity has replaced kinematic viscosity, consistent with current practice;

    c. The type of fuel oil, Type 2D, has been deleted, consistent with current licensing basis; and

    d. The clear and bright appearance test with proper color has been deleted, consistent with current practice.

15. ISTS 5.5.13 (ITS 5.5.11) provides requirements for the Primary Containment Leakage Rate Testing Program. The requirements of the ISTS are revised to reflect the Primary Containment Leakage Rate Testing Program requirements in CTS 6.8.M and the replacement requirements for the primary containment purge and vent valves in CTS 4.7.D.4. The statement in ISTS 5.5.13.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 because the phrase is not consistent with the allowance in ISTS 5.5.13.a (ITS 5.5.11.a), which states that the 10 CFR 50, Appendix J, Option B requirements may be modified by approved exemptions and exceptions.

16. The program details of the Explosive Gas and Storage Tank Radioactivity Monitoring Program are described in ISTS 5.5.9 (ITS 5.5.7) parts a, b, and c. Therefore, the

sentence in the introductory paragraph that specifies a method to determine the explosive gas and storage tank radioactivity is not necessary.

Specific No Significant Hazards Considerations (NSHCs)

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

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGE L.4

Nuclear Management Company, LLC (NMC) is converting the Monticello Current Technical Specifications (CTS) to the Improved Technical Specifications (ITS) as outlined in NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4." 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-1433.

CTS 6.8.B includes the Primary Coolant Sources Outside Containment program requirements. The Combustible Gas Control System is included in this program. ITS 5.5.2 includes the same program requirements for the Primary Coolant Sources Outside Containment Program, except the Combustible Gas Control System is not included in the program. This changes the CTS by deleting the program requirement for the Combustible Gas Control System in the Primary Coolant Sources Outside Containment Program.

The purpose of CTS 6.8.B is to ensure controls are in place 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 practical. The Technical Specification requirements governing the OPERABILITY of the Combustible Gas Control System have previously been removed from the Monticello Technical Specifications as documented in License Amendment 138, dated May 21, 2004. However the License Amendment did not remove the Combustible Gas Control System from the program requirements of CTS 6.8.B since the Residual Heat Removal System cooling water supply was still available to the Combustible Gas Control System (i.e., the potential for coolant leakage that could be highly radioactive during a transient or accident still existed.). A plant modification has been completed that removes all communication between the Combustible Gas Control System and the containment and eliminated the Residual Heat Removal System cooling water supply lines to the Combustible Gas Control System. Thus, the potential for the Combustible Gas Control System to contain highly radioactive fluids no longer exists. Therefore, the program controls for this system in CTS 6.8.B are no longer necessary. This change is considered less restrictive because the program requirement for the Combustible Gas Control System in the Primary Coolant Sources Outside Containment Program has been deleted.

NMC has evaluated whether or not a significant hazards consideration is involved with these proposed Technical Specification changes by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

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

   Response: No.

   The proposed change deletes the program requirement for the Combustible Gas Control System in the Primary Coolant Sources Outside Containment Program. This change will not affect the probability of an accident since the program is not considered to be an initiator of any accident previously analyzed. The consequences of an accident are not affected by this change since the potential for the Combustible Gas Control System to contain highly radioactive fluids no longer exists. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   Response: No.

   The proposed change deletes the program requirement for the Combustible Gas Control System in the Primary Coolant Sources Outside Containment program. While the plant has already been altered (a plant modification has been completed that removes all communication between the Combustible Gas Control System and the containment and eliminated the Residual Heat Removal System cooling water supply lines to the Combustible Gas Control System), this specific Technical Specification change will not physically result in an alteration to the plant (no new or different type of equipment will be installed as a result of this Technical Specificaion change). The changes in methods governing normal plant operation are consistent with current safety analysis assumptions. 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 proposed change deletes the program requirement for the Combustible Gas Control System in the Primary Coolant Sources Outside Containment Program. The margin of safety is not affected by this change because the potential for the Combustible Gas Control System to contain highly radioactive fluids no longer exists. Therefore, the program controls for this system is no longer necessary. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

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

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGE L.5

Nuclear Management Company, LLC (NMC) is converting the Monticello Current Technical Specifications (CTS) to the Improved Technical Specifications (ITS) as outlined in NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4." 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-1433.

CTS 6.8.B.2 specifies that the integrated leak test requirements for each system outside containment that could contain highly radioactive fluids during a serious transient or accident must be performed at a refueling cycle interval or less. CTS 6.8.B also states that CTS 4.0.B is applicable (i.e., a 25% grace period is allowed). ITS 5.5.2.b specifies that the same test must be performed at least once per 24 months and ITS 5.5.2 states that the provisions of ITS SR 3.0.2 are applicable. This changes the CTS by extending the Frequency of the Surveillance from 18 months (i.e., the current Monticello frequency for this test, based on the previous refueling outage interval) to 24 months (i.e., a maximum of 30 months accounting for the allowable grace period specified in ITS SR 3.0.2).

The purpose of CTS 6.8.B.2 is to ensure the leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident is reduced to as low as practicable levels. This change was evaluated in accordance with the guidance provided in NRC Generic Letter No. 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991. Reviews of historical surveillance data and maintenance data sufficient to determine failure modes have shown that these tests normally pass their Surveillances at the current Frequency. An evaluation has been performed using this data, and it has been determined that the effect on safety due to the extended Surveillance Frequency will be minimal. Extending the Surveillance test interval for the System Integrity integrated leak test verification SR is acceptable because most portions of the subject systems included in this program are visually walked down, while the plant is operating, during plant testing, and/or operator/system engineer walkdowns. In addition, housekeeping/safety walkdowns also serve to detect any gross leakage. If leakage is observed from these systems, corrective actions will be taken to repair the leakage. Finally, the plant radiological surveys will also identify any potential sources of leakage. These visual walkdowns and surveys provide monitoring of the systems at a greater frequency than once per refueling cycle, and support the conclusion that the impact, if any, on safety is minimal as a result of the proposed changes. Based on the inherent system and component reliability and the testing performed during the operating cycle, the impact, if any, from this change on system availability is minimal. The review of historical surveillance data also demonstrated that there are no failures that would invalidate this conclusion. In addition, the proposed 24 month Surveillance Frequency, if performed at the maximum interval allowed by ITS SR 3.0.2 (30 months) does not invalidate any assumptions in the plant licensing basis. This change is designated as less restrictive because Surveillances will be performed less frequently under the ITS than under the CTS.

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

NMC has evaluated whether or not a significant hazards consideration is involved with these proposed Technical Specification changes by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

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

   Response: No.

   The proposed change involves a change in the Surveillance Frequency from 18 months to 24 months. The proposed change does not physically impact the plant, and does not impact any design or functional requirements of the associated systems. That is, the proposed change does not degrade the performance or increase the challenges of any safety systems assumed to function in the accident analyses. The proposed change does not impact the Surveillance Requirement itself, and does not change the methods used for performing the Surveillance. Additionally, the proposed change does not introduce any new accident initiators, because no accidents previously evaluated have as their initiators anything related to the Frequency of Surveillance testing. The proposed change does not affect the availability of equipment or systems required to mitigate the consequences of an accident, because of the availability of redundant systems or equipment and because other tests performed more frequently will identify potential equipment problems. Furthermore, an historical review of Surveillance test results indicates that all failures identified were unique, non-repetitive, and not related to any time-based failure modes, and indicated no evidence of any failures that would invalidate the above conclusions. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   Response: No.

   The proposed change involves a change in the Surveillance Frequency from 18 months to 24 months. The proposed change does not introduce any failure mechanisms of a different type than those previously evaluated since there are no physical changes being made to the facility. In addition, the Surveillance Requirement itself and the way Surveillance is performed will remain unchanged. Furthermore, an historical review of Surveillance test results indicates no evidence of any failures that would invalidate the above conclusions. 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.

   Although the proposed change will result in an increase in the interval between Surveillance tests, the impact on system availability is minimal based on other,

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

more frequent testing or redundant systems or equipment, and there is no evidence of any failures that would impact the availability of the systems. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

ATTACHMENT 6

ITS 5.6, Reporting Requirements

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

5.6 6.7 REPORTING REQUIREMENTS

5.6 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following identified reports shall be submitted ~~to the U.S. Nuclear Regulatory Commission, Attn: Document Control Desk, Washington, DC 20555, unless otherwise noted~~. in accordance with 10 CFR 50.4 A.2

~~A. Routine Reports~~

~~1. Startup Report~~

~~A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant. The report shall address each of the tests identified in the FSAR and shall in general include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.~~ L.1

~~Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.~~

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A.1

ITS

2. (Deleted)

3. (Deleted)

5.6.2 4. Radioactive Effluent Release Report

The Radioactive Effluent Release Report covering the operation of the unit during the previous year shall be submitted pri to May 15 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 in conformance with 10 CFR 50.36a and 10 CFR Part 50, Appendix I, Section IV.B.1.

5. (Deleted)

6. (Deleted)

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5.6.3 7. Core Operating Limits Report

5.6.3.a a. Core operating limits shall be established and documented in the Core Operating Limits Report before each reload cycle or any remaining part of a reload cycle for the following:

5.6.3.a.4 Rod Block Monitor Operability Requirements (Specification 3.2.C.2a)
Rod Block Monitor Upscale Trip Settings (Table 3.2.3, Item 4.a)

5.6.3.a.5 Recirculation System Power to Flow Map Stability Regions (Specification 3.5.F)

5.6.3.a.1 Maximum Average Planar Linear Heat Generation Rate Limits (Specification 3.11.A)

5.6.3.a.3 Linear Heat Generation Rate Limits (Specification 3.11.B)

5.6.3.a.2 Minimum Critical Power Ratio Limits (Specification 3.11.C)

~~Power to Flow Map (Bases 3.1)~~ A.3

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

5.6.3.b.1 NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel" ~~(the approved version at the time the reload analyses are performed)~~

5.6.3.b.2 NSPNAD-8608-A, "Reload Safety Evaluation Methods for Application to the Monticello Nuclear Generating Plant" ~~(the approved version at the time the reload analyses are performed)~~

5.6.3.b.3 NSPNAD-8609-A, "Qualification of Reactor Physics Methods for Application to Monticello" ~~(the approved version at the time the reload analyses are performed)~~

5.6.3.b.4 NEDO-31960, "BWR Owners' Group Long-Term Stability Solutions Licensing Methodology," ~~June 1991 (the approved version at the time the reload analyses are performed)~~

5.6.3.b.4 NEDO-31960, Supplement 1, "BWR Owners' Group Long-Term Stability Solutions Licensing Methodology," ~~March 1992 (the approved version at the time the reload analyses are performed)~~

LA.1

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

5.6.3.d d. The Core Operating Limits Report, including any mid-cycle revisions or supplements, shall be supplied upon issuance, for each reload cycle, to the NRC ~~Document Control Desk with copies to the Regional Administrator and Resident Inspector~~ A.2

A.1

ITS

B. (Deleted)

C. Environmental Reports

5.6.1 1. Annual Radiological Environmental Operating Report

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.

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2. (Deleted)

3. Other Environmental Reports (non-radiological, non-aquatic)

   a. Environmental events that indicate or could result in a significant environmental impact causally related to plant operation. The following are examples: excessive bird impaction; onsite plant or animal disease outbreaks; unusual mortality of any species protected by Endangered Species Act of 1973; increase in nuisance organisms or conditions; or excessive environmental impact caused by herbicide application to transmission corridors associated with the plant. This report shall be submitted within 30 days of the event and shall (a) describe, analyze, and evaluate the event, including extent and magnitude of the impact and plant operating characteristics, (b) describe the probable cause of the event, (c) indicate the action taken to correct the reported event, (d) indicate the corrective action taken to preclude repetition of the event and to prevent similar occurrences involving similar components or systems, and (e) indicate the agencies notified and their preliminary responses.

   b. Proposed changes, tests or experiments which may result in a significant increase in any adverse environmental impact which was not previously reviewed or evaluated in the Final Environmental Statement or supplements thereto. This report shall include an evaluation of the environmental impact of the proposed activity and shall be submitted 30 days prior to implementing the proposed change, test or experiment.

L.2

D. Special Reports

Unless otherwise indicated, special reports required by the Technical Specification shall be submitted within the time period specified for each report.

A.4

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Table 3.14.1

Instrumentation for Accident Monitoring

Function	Total No. of Instrument Channels	Minimum No. of Operable Channels	Required Conditions*
Reactor Vessel Fuel Zone Water Level	2	1	A, B
Safety/Relief Valve Position (One Channel Pressure Switch and One Channel Thermocouple Position Indication per Valve)	2	1	A, C
Drywell Wide Range Pressure	2	1	A, B
Suppression Pool Wide Range Level	2	1	A, B
Suppression Pool Temperature	2	1	A, D
Drywell High Range Radiation	2	1	A, D
Offgas Stack Wide Range Radiation	2	1	A, D
Reactor Bldg Vent Wide Range Radiation	2	1	A, D

* Required Conditions

See ITS 3.3.3.1

5.6.4 A. When the number of channels made or found to be inoperable is such that the number of operable channels is less than the total number of channels, either restore the inoperable channels to operable status within seven days, or prepare and submit a special report to the Commission pursuant to Technical Specification 6.7.D within the next ~~30~~ 14 days outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the system to operable status.

M.1

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DISCUSSION OF CHANGES
ITS 5.6, REPORTING REQUIREMENTS

ADMINISTRATIVE CHANGES

A.1 In the conversion of the Monticello 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-1433, Rev. 3, "Standard Technical Specifications General Electric Plants, BWR/4" (ISTS).

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

A.2 CTS 6.7 requires, in addition to the requirements of 10 CFR, reports be submitted to the U.S. Nuclear Regulatory Commission, Attn: Document Control Desk, Washington DC 20555, unless otherwise noted. CTS 6.7.A.7.d requires the COLR to be submitted to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector. ITS 5.6 requires that the reports be submitted in accordance with 10 CFR 50.4. 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. This change is designated as administrative because it does not result in technical changes to the CTS.

A.3 CTS 6.7.A.7.a states, in part, that core operating limits shall be established and documented in the Core Operating Limits Report (COLR) before each reload cycle or any remaining part of a reload cycle for the "Power to Flow Map (Bases 3.1)." ITS 5.6.3.a does not include reference to the "Power to Flow Map (Bases 3.1)." This changes the CTS by removing the specific reference to "Power to Flow Map (Bases 3.1)."

The purpose of this CTS 6.7.A.7.a statement is to specify the power to flow map discussed in the CTS 3.1 Bases is located in the COLR. The power to flow map is not currently discussed in the Bases of CTS 3.1. The power to flow map is referenced in ITS 3.4.1 and therefore ITS 5.6.3.a.4 cross references ITS 3.4.1. This change is acceptable because ITS 5.6.3.a references all Specifications associated with the power to flow map in the ITS (i.e., ITS 3.4.1). This change is designated as administrative because it does not result in technical changes to the CTS.

A.4 CTS 6.7.D requires special reports be submitted within the time period specified by each report. CTS Table 3.14.1 Required Condition A requires the preparation and submittal of a special report to the Commission pursuant to CTS 6.7.D. This is the only Technical Specification that currently references CTS 6.7.D. The ITS does not include a Special Report requirement; all reports have there own individual titles. This changes the CTS by deleting the reference to Special Reports. The special report requirement in CTS Table 3.14.1 is required by ITS 5.6.4, as modified by DOC M.1.

DISCUSSION OF CHANGES
ITS 5.6, REPORTING REQUIREMENTS

The purpose of CTS 6.7.D is to identify that special reports are required to be submitted. This change is acceptable because this specific CTS requirement is redundant to the actual report requirement. CTS 6.7.D simply states to follow whatever the special report in TS requires. CTS Table 3.14.1 is the only Technical Specification requirement that requires a special report to be prepared and submitted to the Commission and it is required by ITS 5.6.4, as modified by DOC M.1. This change is designated as administrative because it does not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

M.1 CTS Table 3.14.1 Required Condition A requires a report to be prepared and submitted within the next 30 days outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the inoperable Post Accident Monitoring Instrumentation to OPERABLE status. ITS 5.6.4 requires the same report to be prepared and submitted within 14 days. This changes the CTS by reducing the time required to prepare and submit a Post Accident Monitoring Report from 30 days to 14 days.

The purpose of the Post Accident Monitoring Report is to inform the NRC of inoperabilities associated with Post Accident Monitoring Instrumentation. This report can be prepared and submitted to the NRC within the proposed 14 day time period. This change is acceptable because the report can be prepared and submitted within the 14 day time period. This change is designated more restrictive because it decreases the time allowed to prepare the Post Accident Monitoring Report.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA.1 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 6.7.A.7.b specifies the revision/supplement numbers and dates (e.g., latest approved version at the time the reload analyses are performed) of the referenced methodologies used for the development of the COLR. ITS 5.6.3.b does not contain this level of detail. This changes the CTS by moving the specific methodology references for revisions/supplements and dates to the COLR.

The removal of these details, which are related to meeting Technical Specifications requirements, from the Technical Specifications is acceptable because this type of information is not necessary to be included in Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the references for the COLR and only NRC-approved methodologies may be used. The methodologies used to develop the parameters in the COLR have obtained prior approval by the NRC in accordance with Generic Letter 88-16. Also, this change is acceptable because the removed information will be adequately controlled in the COLR under the requirements provided in ITS 5.6.3, "CORE OPERATING LIMITS REPORT." ITS 5.6.3 ensures that the applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems limits, and nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analyses are met and that only NRC-approved methodologies are used. This change is designated as a less restrictive removal of detail change because information relating to the methodology used to develop cycle-specific parameter limits is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L.1 *(Category 8 – Deletion of Reporting Requirements)* CTS 6.7.A.1 contains requirements for submitting a report of plant startup and power escalation testing following: a) receipt of an operating license; b) amendment to the license involving planned increase in power level; c) installation of fuel that has a different design or has been manufactured by a different fuel supplier; and d) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant. The ITS does not contain such reporting requirements. This changes the CTS by deleting the requirements of CTS 6.7.A.1.

The purpose of CTS 6.7.A.1 is to provide a summary of plant startup and power escalation testing following the four specified conditions as verification that the plant operated as expected. This change is acceptable because the regulations provide adequate reporting requirements. If there were any plant conditions outside the expected parameters during plant startup, they would be reported to the NRC if they met the reporting requirements in the regulations. Otherwise, the reports would document that the plant operated as expected and already approved by the NRC, as required by regulations. This change is designated as less restrictive because reports that would be submitted under the CTS will not be required under the ITS.

L.2 *(Category 8 – Deletion of Reporting Requirements)* CTS 6.7.C.2 specifies requirements for other Environment Reports (non-radiological, non-aquatic). ITS 5.6 does not include this reporting requirement. This changes the CTS by deleting the requirement of other Environmental Reports (non-radiological, non-aquatic).

The purpose of the other Environmental Reports (non-radiological, non aquatic) is to ensure the NRC is informed of environmental events that indicate or could result in a significant environmental impact casually related to plant operation. In addition, the purpose of the report is to ensure that the NRC is notified of any proposed changes, tests or experiments that may result in a significant increase in any adverse environmental impact which was not previously reviewed or evaluated in the Final Environmental Statement or supplements thereto. This change is acceptable because the regulations provide adequate controls associated with reports associated with "environmental events" and "proposed changes, test, or experiments" which have a significant environmental impact.

DISCUSSION OF CHANGES
ITS 5.6, REPORTING REQUIREMENTS

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

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

CTS

5.0 ADMINISTRATIVE CONTROLS

6.7 5.6 Reporting Requirements

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

5.6.1 Occupational Radiation Exposure 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.]

A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors), for whom monitoring was performed, receiving an annual deep dose equivalent > 100 mrems and the associated collective deep dose equivalent (reported in person - rem) according to work and job functions (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance [describe maintenance], waste processing, and refueling). This tabulation supplements the requirements of 10 CFR 20.2206. The dose assignments to various duty functions may be estimated based on pocket ionization chamber, thermoluminescence dosimeter (TLD), electronic dosimeter, or film badge measurements. Small exposures totaling < 20 percent of the individual total dose need not be accounted for. In the aggregate, at least 80 percent of the total deep dose equivalent received from external sources should be assigned to specific major work functions. The report covering the previous calendar year shall be submitted by April 30 of each year. [The initial report shall be submitted by April 30 of the year following the initial criticality.] TSTF -369

6.7.C.1 5.6.2 1 Annual Radiological Environmental Operating Report TSTF -369

-----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

CTS

5.6 Reporting Requirements

6.7.C.1 5.6.[2] Annual Radiological Environmental Operating Report (continued) 1 TSTF-369

environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements [in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979]. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible. 2 TSTF-369

6.7.A.4 5.6.[3] 2 Radiological Effluent Release Report

~~-----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 during the previous year shall be submitted prior to May [1] 15 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. 3

~~5.6.4 Monthly Operating Reports~~

~~Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.~~ TSTF-369

6.7.A.7 5.6.[5] 3 CORE OPERATING LIMITS REPORT (COLR) TSTF-369

6.7.A.7.a a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

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

INSERT 1 2

CTS (2) **INSERT 1**

6.7.A.7.a 1. The APLHGR for Specification 3.2.1;

2. The MCPR for Specification 3.2.2;

3. The LHGR for Specification 3.2.3;

4. Control Rod Block Instrumentation Allowable Value for the Table 3.3.2.1-1 Rod Block Monitor Functions 1.a, 1.b, and 1.c and associated Applicability RTP levels; and

5. The Normal Region, the Stability Exclusion Region, and the Stability Buffer Region of the power to flow map, and the power distribution controls for Specification 3.4.1.

CTS

5.6 Reporting Requirements

5.6.~~5~~ 3 CORE OPERATING LIMITS REPORT (continued) TSTF-369

6.7.A.7.b b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents: INSERT 2 2

~~[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 Technical Specification referenced topical reports used to prepare the COLR (i.e., report number, title, revision, date, and any supplements).~~]~~ 2

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

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

~~5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)~~ 4 TSTF-369

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

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

~~b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:~~ 4

~~[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).]~~

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

CTS

2 **INSERT 2**

6.7.A.7.b 1. NEDE-24011-P-A,"General Electric Standard Application for Reactor Fuel";

2. NSPNAD-8608-A,"Reload Safety Evaluation Methods for Application to the Monticello Nuclear Generating Plant";

3. NSPNAD-8609-A,"Qualification of Reactor Physics Methods for Application to Monticello"; and

4. NEDO-31960,"BWR Owners' Group Long-Term Stability Solutions Licensing Methodology."

CTS

5.6 Reporting Requirements

5.6.5 RCS PRESSURE AND TEMPERATURE LIMITS REPORT (continued) 4 TSTF-369

-----REVIEWER'S NOTE-----
The methodology for the calculation of the P-T limits for NRC approval should include the following provisions:

1. The methodology shall describe how the neutron fluence is calculated (reference new Regulatory Guide when issued).
2. The Reactor Vessel Material Surveillance Program shall comply with Appendix H to 10 CFR 50. The reactor vessel material irradiation surveillance specimen removal schedule shall be provided, along with how the specimen examinations shall be used to update the PTLR curves.
3. Low Temperature Overpressure Protection (LTOP) System lift setting limits for the Power Operated Relief Valves (PORVs), developed using NRC-approved methodologies may be included in the PTLR.
4. The adjusted reference temperature (ART) for each reactor beltline material shall be calculated, accounting for radiation embrittlement, in accordance with Regulatory Guide 1.99, Revision 2. 4
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.

Table 3.14.1 Note A 5.6.7 Post Accident Monitoring Report 4 4 5 TSTF-369

When a report is required by Condition B or F of LCO 3.3.[3.1], "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status. 2

TSTF -369

~~5~~ 4

5.6 Reporting Requirements 4

Table 3.14.1 Note A 5.6.~~7~~ Post Accident Monitoring Report (continued)

~~-----REVIEWER'S NOTE-----~~
~~These reports may be required covering inspection, test, and maintenance activities. These reports are determined on an individual basis for each unit and their preparation and submittal are designated in the Technical Specifications.~~ 5

JUSTIFICATION FOR DEVIATIONS
ITS 5.6, REPORTING REQUIREMENTS

1. The bracketed Note has been deleted because Monticello is not a multiple unit station.

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

3. 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. The existing Monticello CTS submittal date for this report is not May 1 of each year. Therefore, the submittal date for this report is revised in ISTS 5.6.3 (ITS 5.6.2) to reflect the CNP CTS requirement (i.e., prior to May 15).

4. ISTS 5.6.6, "Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)," is not adopted in the ITS. CTS Figures 3.6.1, 3.6.2, 3.6.3, and 3.6.4, which provide Reactor Coolant System heatup and cooldown limitations, respectively, were adopted in ITS 3.4.3, "RCS Pressure and Temperature (P/T) Limits." Subsequent Specifications are renumbered accordingly.

5. The ISTS Reviewer's Note has been deleted since they it not intended to be included in the ITS.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 5.6, REPORTING REQUIREMENTS

There are no specific NSHC discussions for this Specification.

ATTACHMENT 7

ITS 5.7, High Radiation Area

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

5.7 6.9 High Radiation Area

5.7 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 A. 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 1. 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.

5.7.1.b 2. 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.

5.7.1.c 3. 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.

5.7.1.d 4. Each individual or group entering such an area shall possess:

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

5.7.1.d.2 b. 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 c. 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

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

5.7.1.d.4.(i) 1) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

5.7.1.d.4.(ii) 2) 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.

5.7.1.e 5. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

5.7.2 B. 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

5.7.2.a 1. Each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked or continuously guarded door or gate that prevents unauthorized entry, and, in addition:

5.7.2.a.1 a. 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.

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

5.7.2.b 2. 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.

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5.7.2.c 3. Individuals qualified in radiation protection procedures may be exempted from the requirement for an RWP or equivalent while performing radiation surveys in such areas provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

5.7.2.d 4. Each individual or group entering such an area shall possess:

5.7.2.d.1 a. 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 b. 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 c. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

5.7.2.d.3.(i) 1) Be under surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

5.7.2.d.3.(ii) 2) 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.

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

5.7.2.e 5. 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.

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5.7.2.f

6. Such individual areas that are within a larger area where no enclosure exists for the purpose of locking and where no enclosure can reasonably by 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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DISCUSSION OF CHANGES
ITS 5.7, HIGH RADIATION AREA

ADMINISTRATIVE CHANGES

A.1 In the conversion of the Monticello 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-1433, Rev. 3, "Standard Technical Specifications General Electric Plants, BWR/4" (ISTS).

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

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

None

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

CTS

5.0 ADMINISTRATIVE CONTROLS

6.9 ## 5.7 High Radiation Area

6.9 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:

6.9.A 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

6.9.A.1 a. Each entryway to such an area shall be barricaded and conspicuously posted as a high radiation area. Such barricades may be opened as necessary to permit entry or exit of personnel or equipment.

6.9.A.2 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.

6.9.A.3 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.

6.9.A.4 d. Each individual or group entering such an area shall possess one of the following: 3

6.9.A.4.a 1. A radiation monitoring device that continuously displays radiation dose rates in the area~~, or~~ ; 1

6.9.A.4.b 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~~ ; 1

6.9.A.4.c 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 ; 1

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

6.9.A.4.d.1) (i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation do~~es~~se rates in the area; who is responsible for controlling personnel exposure within the area, or 4

CTS

5.7 High Radiation Area

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

6.9.A.4.d.2) (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.

6.9.A.5 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.

6.9.B 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

6.9.B.1 a. Each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked or continuously guarded door or gate that prevents unauthorized entry, and, in addition:

6.9.B.1.a 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.

6.9.B.1.b 2. Doors and gates shall remain locked except during periods of personnel or equipment entry or exit.

6.9.B.2 b. Access to, and activities in, each such area shall be controlled by means of an RWP or equivalent that includes specification of radiation does rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

6.9.B.3 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

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)

6.9.B.4 d. Each individual or group entering such an area shall possess one of the following:

6.9.B.4.a 1. A radiation monitoring device that continuously integrates the radiation rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint~~, or~~ ; 1

6.9.B.4.b 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~~ ; 1

6.9.B.4.c 3. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

6.9.B.4.c.1) (i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

6.9.B.4.c.2) (ii) Be under the surveillance, as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with and control every individual in the area~~.~~ ; or 1

6.9.B.4.d 4. In those cases where ~~option (2) and (3)~~ Specifications 5.7.2.d.2 and 5.7.2.d.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. 2

6.9.B.5 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

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)

6.9.B.6

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

JUSTIFICATION FOR DEVIATIONS
ITS 5.7, HIGH RADIATION AREA

1. These punctuation corrections have been made consistent with the Writer's Guide for the Improved Standard Technical Specifications, NEI 01-03, Section 5.1.3.
2. The proper Specification numbers have been provided.
3. Change made to be consistent with another similar Specification (i.e., ITS 5.7.2.d).
4. Typographical error corrected.

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.