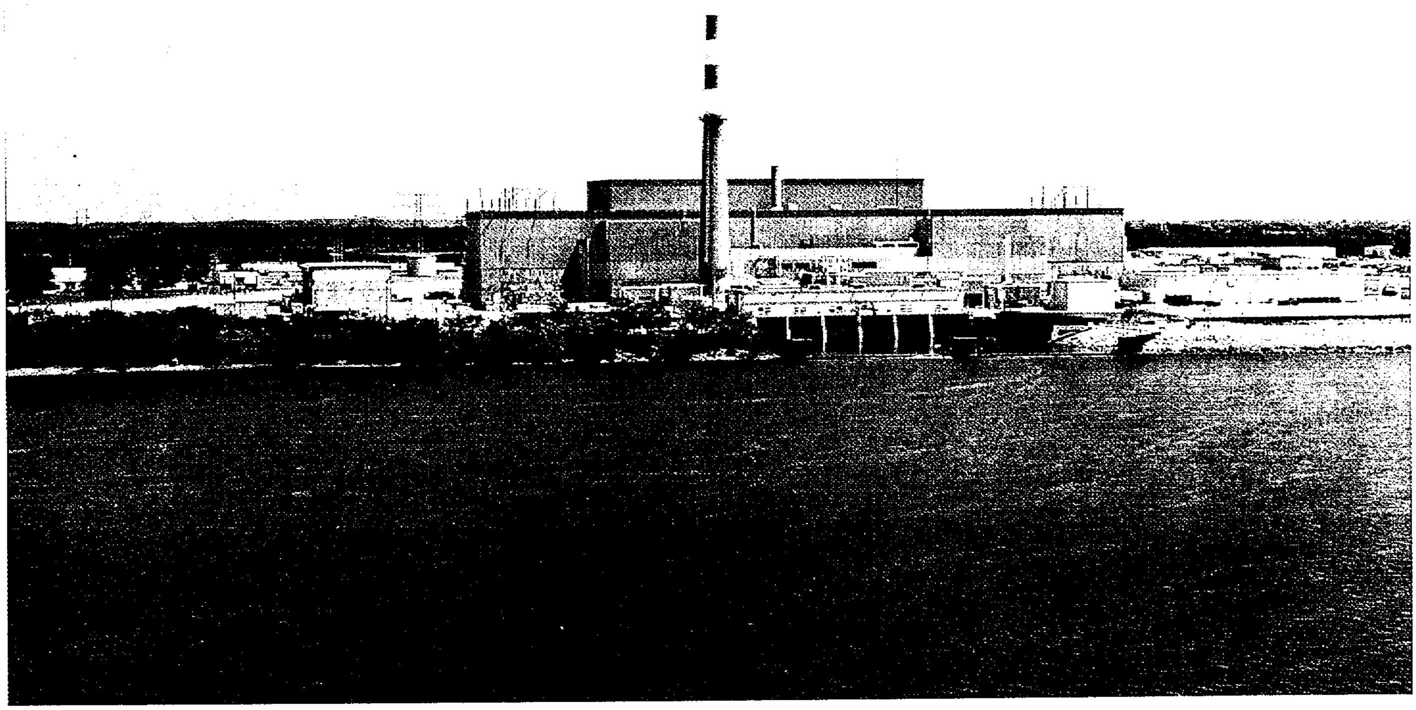
Improved Technical Specifications



Quad Cities Station

Volume 11:
Chapters 4.0 and 5.0

ComEd

4.0 DESIGN FEATURES

4.1 Site Location

4.1.1 Site and Exclusion Area

The site consists of approximately 784 acres on the east bank of the Mississippi River opposite the mouth of the Wapsipinicon River, approximately three miles north of the village of Cordova, Rock Island County, Illinois. The exclusion area shall not be less than 380 meters from the centerline of the chimney.

4.1.2 Low Population Zone

The low population zone shall be a three mile radius from the centerline of the chimney.

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 724 fuel assemblies. Each assembly shall consist of a matrix of Zircaloy clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. The assemblies may contain water rods or a water box. Limited substitutions of Zircaloy or ZIRLO filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with NRC staff approved codes and methods and have been shown by tests or analyses to comply with all safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

4.2.2 Control Rod Assemblies

The reactor core shall contain 177 cruciform shaped control rod assemblies. The control material shall be boron carbide and hafnium metal as approved by the NRC.

(continued)

4.0 DESIGN FEATURES (continued)

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1.2 of the UFSAR; and

b. A nominal 6.22 inch center to center distance between fuel assemblies placed in the storage racks.

4.3.2 Drainage

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 666 ft 8.5 inches.

4.3.3 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 3657 fuel assemblies for Unit 1 and 3897 fuel assemblies for Unit 2.

4.0 5.0 DESIGN FEATURES

4.1 5.1 SITE

Site and Exclusion Area

4.1.1 5.1.A The site consists of approximately 784 acres on the east bank of the Mississippi River opposite the mouth of the Wapsipinicon River, approximately three miles north of the village of Cordova, Rock Island County, Illinois. The Exclusion Area shall not be less than 380 meters from the centerline of the chimney.

Low Population Zone

4.1.2 5.1.B The Low Population Zone shall be a three mile radius from the centerline of the chimney.

~~**Radioactive Gaseous Effluents**~~

~~5.1.C Information regarding radioactive gaseous effluents shall be located in the OFFSITE DOSE CALCULATION MANUAL.~~

A.3

~~**Radioactive Liquid Effluents**~~

~~5.1.D Information regarding radioactive liquid effluents shall be located in the OFFSITE DOSE CALCULATION MANUAL.~~

FIGURE 5.1.A-1

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LA.2

5.2 CONTAINMENT -

Configuration

5.2.A The primary containment is a steel lined concrete structure consisting of a drywell and suppression chamber. The drywell is a steel structure composed of a spherical lower portion, a cylindrical middle portion, and a hemispherical top head. The drywell is attached to the suppression chamber through a series of downcomer vents. The drywell has a minimum free air volume of 158,236 cubic feet. The suppression chamber has an air region of 120,800 to 117,300 cubic feet and a water region of 111,500 to 115,000 cubic feet.

Design Temperature and Pressure

5.2.B The primary containment is designed and shall be maintained for:

1. Maximum internal pressure: 56 psig.
2. Maximum internal temperature: drywell 281°F.
suppression pool 281°F.
3. Maximum external pressure: drywell 2 psig.
suppression pool 1 psig.

Secondary Containment

5.2.C The secondary containment consists of the Reactor Building and a portion of the main steam tunnel and has a minimum free volume of 4,716,000 cubic feet.

5.0 DESIGN FEATURES

4.2 5.3 REACTOR CORE

Fuel Assemblies

4.2.1 5.3.A The reactor core shall contain 724 fuel assemblies. Each assembly consists of a matrix of Zircaloy clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide as fuel material. The assemblies may contain water rods or water boxes. Limited substitutions of Zircaloy or ZIRLO filler rods for fuel rods, in accordance with NRC-approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff-approved codes and methods, and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in non-limiting core regions.

Control Rod Assemblies

4.2.2 5.3.B The reactor core shall contain 177 cruciform shaped control rod assemblies. The control material shall be boron carbide powder (B_4C) and/~~or~~ hafnium metal. ~~The control rod assembly shall have a nominal axial absorber length of 143 inches.~~ LA.3

5.0 DESIGN FEATURES

4.3 5.6 FUEL STORAGE

4.3.1 Criticality

4.3.1.1 5.6.A The spent fuel storage racks are designed and shall be maintained with:

4.3.1.1.a 1. A k_{eff} equivalent to ≤0.95 when flooded with unborated water, including all calculational uncertainties and biases as described in Section 9.1.2 of the UFSAR. A.1

4.3.1.1.b 2. A nominal 6.22 inch center-to-center distance between fuel assemblies placed in the storage racks.

4.3.2 Drainage

5.6.B The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 666' 8.5".

4.3.3 Capacity

5.6.C The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 3657(Unit 1)/3897(Unit 2) fuel assemblies.

ADMINISTRATIVE

A.1 In the conversion of the Quad Cities 1 and 2 current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted that do not result in technical changes (either actual or interpretation). Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the BWR Standard Technical Specifications, NUREG-1433, Rev. 1 (i.e., the Improved Standard Technical Specifications (ISTS)).

A.2 Not used.

A.3 The details of CTS 5.1.C, Radioactive Gaseous Effluents, and CTS 5.1.D, Radioactive Liquid Effluents, that these items shall be located in the OFFSITE DOSE CALCULATION MANUAL (ODCM) are duplicative of similar requirements in the definition of ODCM. The portions of the definition regarding radioactive liquid and gaseous effluents is being maintained in proposed ITS 5.5.1, Offsite Dose Calculation Manual (ODCM). Therefore, this specific requirement is being deleted and the deletion is considered administrative.

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 Not used.

LA.2 Primary containment configuration and design details in CTS 5.2.A, primary containment design temperatures and pressures in CTS 5.2.B, and secondary containment design details in CTS 5.2.C, are proposed to be relocated to UFSAR, Sections 6.2.1 and 6.2.3, where they currently exist. Any changes to these design parameters described in the UFSAR must conform to the requirements of 10 CFR 50.59. Furthermore, sufficient detail relating to these features exists in CTS and ITS LCOs to ensure any changes which may affect safety would require prior NRC review and approval. Since the features with a potential to affect safety are sufficiently addressed by LCOs, and other features, if altered in accordance with 10 CFR 50.59, would not result in a significant affect on safety, the criteria of 10 CFR 50.36(c)(4) for including as a Design

TECHNICAL CHANGES - LESS RESTRICTIVE

LA.2 (cont'd) Feature are not met. Therefore, removing these details from the Technical Specifications, while maintaining the detail in the UFSAR, will not impact safe operation of the facility, and is not required to be in the ITS to provide adequate protection of the public health and safety.

LA.3 The nominal active control rod assembly absorber length described in CTS 5.3.B is proposed to be relocated to the UFSAR, Section 4.2.2, where it is currently described (by reference). Any changes to this design parameter referenced in the UFSAR must conform to the requirements of 10 CFR 50.59.

Furthermore, sufficient detail relating to this feature exists in a CTS and ITS LCO (e.g., SHUTDOWN MARGIN) to ensure changes that may impact safety would require prior NRC review and approval. Since this feature with a potential to impact safety is sufficiently addressed by an LCO, the criteria of 10 CFR 50.36(c)(4) for including as a Design Feature are not met. Therefore, allowing the removal of this detail from Technical Specifications, while maintaining the information in the UFSAR, will not impact safe operation of the facility, and is not required to be in the ITS to provide adequate protection of the public health and safety.

"Specific"

None

RELOCATED SPECIFICATIONS

None

<CTS>

4.0 DESIGN FEATURES

<5.1.A> <5.1.B>

4.1 Site Location ~~[Text description of site location.]~~ Insert site Location 1

4.2 Reactor Core

4.2.1 Fuel Assemblies

<5.3.A> 2

724 clad The assemblies may contain water rods or a water box.

The reactor shall contain ~~[560]~~ fuel assemblies. Each assembly shall consist of a matrix of [Zircaloy ~~or ZIRLO~~] fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material ~~[, and water rods]~~. Limited substitutions of ~~zirconium alloy or stainless steel~~ filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with NRC staff approved codes and methods and have been shown by tests or analyses to comply with all safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions. 1

Zircaloy or ZIRLO

4.2.2 Control Rod Assemblies

<5.3.B>

177

The reactor core shall contain ~~[137]~~ cruciform shaped control rod assemblies. The control material shall be [boron carbide, and hafnium metal] as approved by the NRC. 1

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

~~a. Fuel assemblies having a maximum [k-infinity of [1.31] in the normal reactor core configuration at cold conditions] [average U-235 enrichment of [4.5] weight percent];~~ 3

<5.6.A.1> ~~b.~~ a. $k_{eff} \le 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in [Section 9.1.2 of the UFSAR]; and 1

(continued)

Insert Site Location

4.1.1 Site and Exclusion Area

<5.1.A> The site consists of approximately 784 acres on the east bank of the Mississippi River opposite the mouth of the Wapsipinicon River, approximately three miles north of the village of Cordova, Rock Island County, Illinois. The exclusion area shall not be less than 380 meters from the centerline of the chimney.

4.1.2 Low Population Zone

<5.1.B> The low population zone shall be a three mile radius from the centerline of the chimney.

4.0 DESIGN FEATURES

4.3 Fuel Storage (continued)

⟨5.6.A.2⟩ 3 b c. A nominal [6.5] 6.22 inch center to center distance between fuel assemblies placed in the storage racks. 1

~~4.3.1.2 The new fuel storage racks are designed and shall be maintained with:~~

~~a. Fuel assemblies having a maximum [k-infinity of [1.31] in the normal reactor core configuration at cold conditions] [average U-235 enrichment of [4.5] weight percent];~~

~~b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in [Section 9.1 of the FSAR];~~

~~c. $k_{eff} \leq 0.98$ if moderated by aqueous foam, which includes an allowance for uncertainties as described in [Section 9.1 of the FSAR]; and~~

~~d. A nominal [6.50] inch center to center distance between fuel assemblies placed in storage racks.~~

⟨5.6.B⟩

4.3.2 Drainage

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation [185 ft] 666ft 8.5 inches

⟨5.6.C⟩

4.3.3 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than [2845] 3657 fuel assemblies for Unit 1 and 3897 fuel assemblies for Unit 2 1

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

2. This change has been made to reflect plant specific information/requirements.

3. The ISTS 4.3.1.1.a k-infinity requirement for spent fuel storage and the ISTS 4.3.1.2 new fuel storage requirements are not included in the Quad Cities 1 and 2 ITS. This change is consistent with the current licensing bases as provided in Quad Cities 1 and 2 Amendments 156 and 152, respectively (NRC SER dated June 14, 1995). These amendments deleted these requirements from the CTS, therefore there is no reason to add them in at this time. Subsequent requirements have been renumbered as applicable to reflect this change.

ADMINISTRATIVE CHANGES
("A.x" Labeled Comments/Discussions)

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   The proposed change involves reformatting, renumbering, and rewording the existing Technical Specifications. The reformatting, renumbering, and rewording process involves no technical changes to the existing Technical Specifications. As such, this change is administrative in nature and does not impact initiators of analyzed events or assumed mitigation of accident or transient events. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The proposed change will not impose any new or eliminate any old requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change will not reduce a margin of safety because it has no impact on any safety analyses assumptions. This change is administrative in nature. Therefore, the change does not involve a significant reduction in a margin of safety.

"GENERIC" LESS RESTRICTIVE CHANGES:
RELOCATING DETAILS TO TECHNICAL SPECIFICATION BASES, UFSAR, TRM, OR OTHER PLANT CONTROLLED DOCUMENTS
("LA.x" Labeled Comments/Discussions)

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   The proposed change relocates certain details from the Technical Specifications to the Bases, UFSAR, TRM, or other plant controlled documents. The Bases, UFSAR, TRM, and other plant controlled documents containing the relocated information will be maintained in accordance with 10 CFR 50.59. In addition to 10 CFR 50.59 provisions, the Technical Specification Bases are subject to the change control provisions in the Administrative Controls Chapter of the ITS. The UFSAR is subject to the change control provisions of 10 CFR 50.71(e), and the plant procedures and other plant controlled documents are subject to controls imposed by plant administrative procedures, which endorse applicable regulations and standards. Since any changes to the Bases, UFSAR, TRM, or other plant controlled documents will be evaluated per the requirements of the Bases Control Program in Chapter 5.0 of the ITS or 10 CFR 50.59, no increase (significant or insignificant) in the probability or consequences of an accident previously evaluated will be allowed. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. The proposed change will not impose or eliminate any requirements, and adequate control of the information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the details to be transposed from the Technical Specifications to the Bases, UFSAR, TRM, or other plant controlled

"GENERIC" LESS RESTRICTIVE CHANGES:
RELOCATING DETAILS TO TECHNICAL SPECIFICATION BASES, UFSAR, TRM, OR OTHER PLANT CONTROLLED DOCUMENTS
("LA.x" Labeled Comments/Discussions)

3. (continued)

documents are the same as the existing Technical Specifications. Since any future changes to these details in the Bases, UFSAR, TRM, or other plant controlled documents will be evaluated per the requirements of 10 CFR 50.59, no reduction (significant or insignificant) in a margin of safety will be allowed. Based on 10 CFR 50.92, the existing requirement for NRC review and approval of revisions, to these details proposed for relocation, does not have a specific margin of safety upon which to evaluate. However, since the proposed change is consistent with the BWR ISTS, NUREG-1433, Rev. 1, approved by the NRC Staff, revising the Technical Specifications to reflect the approved level of detail ensures no significant reduction in the margin of safety.

There were no plant specific less restrictive changes identified for this Specification.

ENVIRONMENTAL ASSESSMENT
ITS: CHAPTER 4.0 - DESIGN FEATURES

In accordance with the criteria set forth in 10 CFR 50.21, ComEd has evaluated this proposed Technical Specification change for identification of licensing and regulatory actions requiring environmental assessment, determined it meets the criteria for a categorical exclusion set forth in 10 CFR 51.22(c)(9) and as such, has determined that no irreversible consequences exist in accordance with 10 CFR 50.92(b). This determination is based on the fact that this change is being proposed as an amendment to a license issued pursuant to 10 CFR which changes a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or which changes an inspection or a surveillance requirement, and the amendment meets the following specific criteria:

1. The amendment involves no significant hazards consideration.

   As demonstrated in the No Significant Hazards Consideration, this proposed amendment does not involve any significant hazards consideration.

2. There is no significant change in the type or significant increase in the amounts of any effluents that may be released offsite.

   The proposed change will not result in changes in the operation or configuration of the facility. There will be no change in the level of controls or methodology used for processing of radioactive effluents or handling of solid radioactive waste, nor will the proposal result in any change in the normal radiation levels within the plant. Therefore, there will be no change in the types or significant increase in the amounts of any effluents released offsite resulting from this change.

3. There is no significant increase in individual or cumulative occupational radiation exposure.

   The proposed change will not result in changes in the operation or configuration of the facility which impact radiation exposure. There will be no change in the level of controls or methodology used for processing of radioactive effluents or handling of solid radioactive waste, nor will the proposal result in any change in the normal radiation levels within the plant. Therefore, there will be no increase in individual or cumulative occupational radiation exposure resulting from this change.

Therefore, based upon the above evaluation, ComEd has concluded that no irreversible consequences exist with the proposed change.

5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

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

5.1.2 A Senior Reactor Operator (SRO) shall be responsible for the control room command function while either unit is in MODE 1, 2, or 3. While both units are in MODE 4 or 5 or defueled, an individual with an active SRO license or Reactor Operator license shall be designated to assume the control room command function.

5.0 ADMINISTRATIVE CONTROLS

5.2 Organization

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.

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 Quality Assurance Manual.

b. The station 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.

c. A 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.

d. The individuals who train the operating staff, or perform radiation protection, or 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.

5.2.2 Unit Staff

The unit staff organization shall include the following:

(continued)

5.2 Organization

5.2.2 Unit Staff (continued)

a. A total of three non-licensed operators for the two units is required in all conditions. At least one of the required non-licensed operators shall be assigned to each unit.

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

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

d. 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.

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

f. The operations manager or shift operations supervisor shall hold an SRO license.

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

5.0 ADMINISTRATIVE CONTROLS

5.3 Unit Staff Qualifications

5.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971, except for the radiation protection manager or lead radiation protection technician, who shall meet or exceed the qualifications for "Radiation Protection Manager" in Regulatory Guide 1.8, September 1975.

5.0 ADMINISTRATIVE CONTROLS

5.4 Procedures

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

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

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1, as stated in Generic Letter 82-33, Section 7.1;

c. Fire Protection Program implementation; and

d. All programs specified in Specification 5.5.

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

The following programs shall be established, implemented and maintained.

5.5.1 Offsite Dose Calculation Manual (ODCM)

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

b. The ODCM shall also contain the radioactive effluent controls and radiological environmental monitoring activities, and descriptions of the information that should be included in the Annual Radiological Environmental Operating, and Radioactive Effluent Release Reports required by Specification 5.6.2 and Specification 5.6.3.

d. Licensee initiated changes to the ODCM:

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

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

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

2. Shall become effective after the approval of the station manager; and

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

(continued)

5.5 Programs and Manuals

5.5.1 Offsite Dose Calculation Manual (ODCM) (continued)

shall indicate the date (i.e., month and year) the change was implemented.

5.5.2 Primary Coolant Sources Outside Containment

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

a. Preventive maintenance and periodic visual inspection requirements; and

b. Integrated leak test requirements for each system at 24 month intervals.

The provisions of SR 3.0.2 are applicable to the 24 month Frequency for performing integrated system leak test activities.

5.5.3 Post Accident Sampling

This program provides controls that ensure the capability to obtain and analyze reactor coolant, radioactive iodines, 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.

(continued)

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program

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

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

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

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

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

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

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

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

(continued)

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

1. For noble gases: a dose rate $\leq$ 500 mrems/yr to the whole body and a dose rate $\leq$ 3000 mrems/yr to the skin, and

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

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

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

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

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluents Control Program Surveillance Frequencies.

5.5.5 Component Cyclic or Transient Limit

This program provides controls to track the UFSAR Section 3.9, cyclic and transient occurrences to ensure that components are maintained within the design limits.

5.5.6 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 pumps and valves.

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

(continued)

5.5 Programs and Manuals

5.5.6 Inservice Testing Program (continued)

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice testing activities	Required Frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days
Every 48 months	At least once per 1461 days

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

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

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

5.5.7 Ventilation Filter Testing Program (VFTP)

The VFTP shall establish the required testing of Engineered Safety Feature (ESF) filter ventilation systems. Tests described in Specification 5.5.7.a and 5.5.7.b shall be performed once per 24 months; after each complete or partial replacement of the HEPA filter bank or charcoal adsorber bank; after any structural maintenance on the HEPA filter bank or charcoal adsorber bank housing; and, following significant painting, fire, or chemical release in any ventilation zone communicating with the subsystem while it is in operation.

(continued)

5.5 Programs and Manuals

5.5.7 Ventilation Filter Testing Program (VFTP) (continued)

Tests described in Specification 5.5.7.c shall be performed once per 24 months; after 1440 hours of adsorber operation for the Standby Gas Treatment System; after 720 hours of adsorber operation for the Control Room Emergency Ventilation System; after any structural maintenance on the charcoal adsorber bank housing; and, following significant painting, fire, or chemical release in any ventilation zone communicating with the subsystem while it is in operation.

Tests described in Specification 5.5.7.d and 5.5.7.e shall be performed once per 24 months.

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

a. Demonstrate for each of the ESF systems that an inplace test of the HEPA filters shows a penetration and system bypass specified below when tested in accordance with Regulatory Guide 1.52, Revision 2, and ANSI/ASME N510-1980 at the system flowrate specified below:

ESF Ventilation System	Penetration	Flowrate
Standby Gas Treatment (SGT) System	< 1.0%	≥ 3600 cfm and ≤ 4400 cfm
Control Room Emergency Ventilation (CREV) System	< 0.05%	≥ 1800 scfm and ≤ 2200 scfm

b. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass specified below when tested in accordance with Regulatory Guide 1.52, Revision 2, and ANSI/ASME N510-1980 at the system flowrate specified below:

(continued)

5.5 Programs and Manuals

5.5.7 Ventilation filter Testing Program (VFTP) (continued)

ESF Ventilation System	Penetration	Flowrate
Standby Gas Treatment (SGT) System	< 1.0%	≥ 3600 cfm and ≤ 4400 cfm
Control Room Emergency Ventilation (CREV) System	< 0.05%	≥ 1800 scfm and ≤ 2200 scfm

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

ESF Ventilation System	Penetration	RH
Standby Gas Treatment (SGT) System	2.5%	70%
Control Room Emergency Ventilation (CREV) System	0.5%	70%

d. Demonstrate for each of the ESF systems that the pressure drop across the combined HEPA filters and the charcoal adsorbers is less than the value specified below when tested at the system flowrate specified as follows:

(continued)

5.5 Programs and Manuals

5.5.7 Ventilation Filter Testing Program (VFTP) (continued)

ESF Ventilation System	Delta P	Flowrate
Standby Gas Treatment (SGT) System	< 6 inches water guage	≥ 3600 cfm and ≤ 4400 cfm
Control Room Emergency Ventilation (CREV) System	< 6 inches water guage	≥ 1800 scfm and ≤ 2200 scfm

e. Demonstrate that the heaters for each of the ESF systems dissipate the value, corrected for voltage variations at the 480 V bus, specified below when tested in accordance with ANSI/ASME N510-1989:

ESF Ventilation System	Wattage
Standby Gas Treatment (SGT) System	≥ 27 kW and ≤ 33 kW
Control Room Emergency Ventilation (CREV) System	≥ 10.8 kW and ≤ 13.2 kW

5.5.8 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the Off-Gas System and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks

The program shall include:

(continued)

5.5 Programs and Manuals

5.5.8 Explosive Gas and Storage Tank Radioactivity Monitoring Program (continued)

a. The limits for concentrations of hydrogen in the Off-Gas System and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion); and

b. 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.

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

5.5.9 Diesel Fuel Oil Testing Program

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

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

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

   2. A flash point and kinematic viscosity within limits, and

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

(continued)

5.5 Programs and Manuals

5.5.9 Diesel Fuel Oil Testing Program (continued)

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

c. Total particulate concentration of the fuel oil in the storage tanks is $\leq$ 10 mg/l when tested every 31 days in accordance with the applicable ASTM Standard.

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

5.5.10 Technical Specifications (TS) Bases Control Program

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

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

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

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

   2. A change to the UFSAR or Bases that involves an unreviewed safety question as defined in 10 CFR 50.59.

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

d. Proposed changes that meet the criterion of Specification 5.5.10.b.1 or 5.5.10.b.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).

(continued)

5.5 Programs and Manuals

5.5.11 Safety Function Determination Program (SFDP)

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

a. The SFDP shall contain the following:

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

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

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

4. Other appropriate limitations and remedial or compensatory actions.

b. A loss of safety function exists when, assuming no concurrent single failure, and assuming no concurrent loss of offsite power or loss of onsite diesel generator(s), a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

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

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

(continued)

5.5 Programs and Manuals

5.5.11 Safety Function Determination Program (SFDP) (continued)

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

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.

5.5.12 Primary Containment Leakage Rate Testing Program

a. This program shall establish the leakage testing of the primary containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemption. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Testing Program," dated September 1995.

b. The peak calculated primary containment internal pressure for the design basis loss of coolant accident, P_a, is 48 psig.

c. The maximum allowable primary containment leakage rate, L_a, at P_a, is 1% of primary containment air weight per day.

d. Leakage rate acceptance criteria are:

1. Primary containment overall 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 $\leq 0.60\ L_a$ for the combined Type B and Type C tests, and $\leq 0.75\ L_a$ for Type A tests.

2. Air lock testing acceptance criteria is the overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

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

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

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 mrem and the associated collective deep dose equivalent (reported in man-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), or electronic dosimeter measurements. Small exposures totaling < 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% 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.

5.6.2 Annual Radiological Environmental Operating Report

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

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

(continued)

5.6 Reporting Requirements

5.6.2 Annual Radiological Environmental Operating Report (continued)

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

5.6.3 Radioactive Effluent Release Report

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

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

5.6.4 Monthly Operating Reports

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

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

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

   1. The APLHGR for Specification 3.2.1.

   2. The MCPR for Specification 3.2.2.

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

3. The LHGR for Specification 3.2.3.

4. The LHGR and transient linear heat generation rate limit for Specification 3.2.4.

5. Control Rod Block Instrumentation Setpoint for the Rod Block Monitor—Upscale Function Allowable Value for Specification 3.3.2.1.

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

1. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," (latest approved revision).

2. Commonwealth Edison Topical Report NFSR-0085, "Benchmark of BWR Nuclear Design Methods," (latest approved revision).

3. Commonwealth Edison Topical Report NFSR-0085, Supplement 1, "Benchmark of BWR Nuclear Design Methods - Quad Cities Gamma Scan Comparisons," (latest approved revision).

4. Commonwealth Edison Topical Report NFSR-0085, Supplement 2, "Benchmark of BWR Nuclear Design Methods - Neutronic Licensing Analyses," (latest approved revision).

5. Advanced Nuclear Fuels Methodology for Boiling Water Reactors, XN-NF-80-19(P)(A), Volume 1, Supplement 3, Supplement 3 Appendix F, and Supplement 4, Advanced Nuclear Fuels Corporation, November 1990.

6. Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads, XN-NF-80-19(P)(A), Volume 4, Revision 1, Exxon Nuclear Company, June 1986.

7. Exxon Nuclear Methodology for Boiling Water Reactors THERMEX: Thermal Limits Methodology Summary Description, XN-NF-80-19(P)(A), Volume 3, Revision 2, Exxon Nuclear Company, January 1987.

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

8. Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis, XN-NF-80-19(P)(A), Volume 1 and Supplements 1 and 2, Exxon Nuclear Company, March 1983.

9. Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel, XN-NF-85-67(P)(A) Revision 1, Exxon Nuclear Company, September 1986.

10. Qualification of Exxon Nuclear Fuel for Extended Burnup Supplement 1: Extended Burnup Qualification of ENC 9x9 BWR Fuel, XN-NF-82-06(P)(A) Supplement 1, Revision 2, Advanced Nuclear Fuels Corporation, May 1988.

11. Advanced Nuclear Fuels Corporation Generic Mechanical Design for Advanced Nuclear Fuels 9x9-IX and 9x9-9X BWR Reload Fuel, ANF-89-014(P)(A), Revision 1 and Supplements 1 and 2, Advanced Nuclear Fuels Corporation, October 1991.

12. Generic Mechanical Design Criteria for BWR Fuel Designs, ANF-89-98(P)(A) Revision 1, and Revision 1 Supplement 1, Advanced Nuclear Fuels Corporation, May 1995.

13. Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors, XN-NF-79-71(P)(A), Revision 2 Supplements 1, 2, and 3, Exxon Nuclear Company, March 1986.

14. ANFB Critical Power Correlation, ANF-1125(P)(A) and Supplements 1 and 2, Advanced Nuclear Fuels Corporation, April 1990.

15. Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors/Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors: Methodology for Analysis of Assembly Channel Bowing Effects/NRC Correspondence, ANF-524(P)(A), Revision 2, Supplement 1 Revision 2, Supplement 2, Advanced Nuclear Fuels Corporation, November 1990.

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

16. COTRANSA 2: A Computer Program for Boiling Water Reactor Transient Analyses, ANF-913(P)(A) Volume 1 Revision 1 and Volume 1 Supplements 2, 3, and 4, Advanced Nuclear Fuels Corporation, August 1990.

17. Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR Evaluation Model, ANF-91-048(P)(A), Advanced Nuclear Fuels Corporation, January 1993.

18. Commonwealth Edison Topical Report NFSR-0091, "Benchmark of CASMO/MICROBURN BWR Nuclear Design Methods," Revision 0, Supplements 1 and 2, December 1991, March 1992, and May 1992, respectively; SER letter dated March 22, 1993.

19. ANFB Critical Power Correlation Application for Coresident Fuel, EMF-1125(P)(A), Supplement 1, Appendix C, Siemens Power Corporation, August 1997.

20. ANFB Critical Power Correlation Determination of ATRIUM-9B Additive Constant Uncertainties, ANF-1125(P)(A), Supplement 1, Appendix E, Siemens Power Corporation, September 1998.

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

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

5.6.6 Post Accident Monitoring (PAM) Instrumentation Report

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.

5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

5.7.1 Pursuant to 10 CFR 20, paragraph 20.1601(c), in lieu of the requirements of 10 CFR 20.1601, each high radiation area, as defined in 10 CFR 20, in which the intensity of radiation is > 100 mrem/hr but < 1000 mrem/hr at 30 cm (12 in.), shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP) (or equivalent document). Individuals qualified in radiation protection procedures (e.g., radiation protection technicians) or personnel escorted by such individuals may be exempt from the RWP issuance requirement during the performance of their assigned duties, provided they are otherwise following plant radiation protection procedures for entry into high radiation areas.

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

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

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

c. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified in the RWP (or equivalent document).

5.7.2 In addition to the requirements of Specification 5.7.1, areas accessible to personnel with radiation levels > 1000 mrem/hr at 30 cm (12 in.) from the radiation source or from any surface which the radiation penetrates shall require the following:

a. Doors shall be locked to prevent unauthorized entry and shall not prevent individuals from leaving the area. In place of locking the door, direct or electronic surveillance

(continued)

5.7 High Radiation Area

5.7.2 (continued)

that is capable of preventing unauthorized entry may be used. The keys shall be maintained under the administrative control of the Shift Engineer on duty or radiation protection supervision.

b. Personnel access and exposure control requirements of activities being performed within these areas shall be specified by an approved RWP (or equivalent document).

c. Each person entering the area shall be provided with an alarming radiation monitoring device that continuously integrates the radiation dose rate (such as an electronic dosimeter). Surveillance and radiation monitoring by a radiation protection technician may be substituted for an alarming dosimeter.

5.7.3 For individual high radiation areas with radiation levels of > 1000 mrem/hr at 30 cm (12 in.), accessible to personnel, that are located within large areas where no enclosure exists for purposes of locking, and where no enclosure can be reasonably constructed around the individual area, that individual area shall be barricaded and conspicuously posted, and a flashing light shall be activated as a warning device.

5.0 ADMINISTRATIVE CONTROLS

5.1 6.1 RESPONSIBILITY

5.1.1 6.1.A The Station Manager shall be responsible for overall facility operation and shall delegate in writing the succession to this responsibility during his absence. LA.1

5.1.2 6.1.B The Shift Engineer shall be responsible for directing and commanding the safe overall operation of the facility under all conditions.

add proposed ITS 5.1.2 LA.2

ADMINISTRATIVE

A.1 In the conversion of the Quad Cities 1 and 2 current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted that do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the BWR Standard Technical Specifications, NUREG 1433, Rev. 1 (i.e., the Improved Standard Technical Specifications (ISTS)).

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 CTS 6.1.A uses the title "Station Manager." In ITS 5.1.1, this specific title is replaced with the generic title "station manager." The specific title is proposed to be relocated to the Quality Assurance (QA) Manual, which is where the description of this specific title is currently located. The allowance to relocate the specific 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 station manager are still retained in the ITS. In addition, the ITS also requires the plant specific titles to be in the QA Manual. Therefore, the relocated specific title is not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the QA Manual are controlled by the provisions of 10 CFR 50.54.

LA.2 CTS 6.1.B delineates the responsibility of the Shift Engineer for directing the control room command function and the daily operations of the facility. This requirement is relocated to the UFSAR. ITS 5.1.2 contains the requirement that a Senior Reactor Operator shall be responsible for the control room command function while either unit is in MODE 1, 2, or 3. While both units are in MODE 4 or 5 or defueled, an individual with an active SRO license or Reactor Operator (RO) license shall be designated to assume the control room command function. Since ITS 5.1.2 provides requirements for the control room command function, as a result inclusion of the detailed responsibilities of the Shift Engineer in the ITS is not required to provide adequate protection of the public health and safety. Changes to the UFSAR are controlled by the provisions of 10 CFR 50.59.

TECHNICAL CHANGES - LESS RESTRICTIVE (continued)

"Specific"

None

RELOCATED SPECIFICATIONS

None

5.2 **6.2 ORGANIZATION**

5.2.1 6.2.A 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 the safety of the nuclear power plant.

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, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements ~~shall be documented in the Quality Assurance Manual.~~ Insert 5.2.1.a LA.1

5.2.1.b 2. The ~~Station 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. LA.1

5.2.1.c 3. A corporate officer ~~The Chief Nuclear Officer (CNO)~~ 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 4. The individuals who train the operating staff and those who carry out ~~health physics~~ radiation protection 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. A.2

LA.1 <u>Insert 5.2.1.a</u>

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

5.2.2 **6.2.B Unit Staff**

The unit staff shall include the following:

At least one required non-licensed operator assigned to each unit. M.1

5.2.2.a 1. Three non-licensed operators shall be on site at all times.

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

5.2.2.c 3. Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i) and 6.2.B.1 and 6.2.C for a period of time not to exceed two hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements.

5.2.2.d 4. A Radiation Protection Technician shall be on site when fuel is in the reactor. The position may be vacant for not more than two hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position. A.2

~~5. 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.~~ LA.2

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

5.2.2.f 6. The Operations Manager or Shift Operations Supervisor shall hold a Senior Reactor Operator License. LA.1

6.2.C Shift Technical Advisor

5.2.2.g The Shift Technical Advisor (STA) shall provide technical advisory support to the Unit Supervisor in the areas of thermal hydraulics, reactor engineering and plant analysis with regard to the safe operation of the facility. In addition, the STA shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift. A single STA may fulfill this function for both units.

shift manager A.3

DISCUSSION OF CHANGES
ITS: 5.2 - ORGANIZATION

ADMINISTRATIVE

A.1 In the conversion of the Quad Cities 1 and 2 current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted that do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the BWR Standard Technical Specifications, NUREG 1433, Rev. 1 (i.e., the Improved Standard Technical Specifications (ISTS)).

A.2 The title of the individual qualified to implement radiation protection procedures in CTS 6.2.B.4 has been changed from the specific title "Radiation Protection Technician" to just describe the generic function; radiation protection technician. Since the only individuals currently qualified are radiation protection technicians, this change is considered administrative. If other individuals are qualified in the future, they will meet the same qualifications. In addition, the term "health physics" in CTS 6.2.A.4 has been changed to radiation protection to be consistent. Therefore, these changes are considered administrative.

A.3 The person to whom the STA provides advisory technical support has been changed to shift manager (ITS 5.2.2.g). Currently (CTS 6.2.C), the STA is required to provide advisory technical support to the Unit Supervisor. However, the STA may provide direct technical support to the entire operating shift, but has a direct responsibility to the shift manager who is responsible for the operation of the plant. This change is considered administrative and has no adverse impact on safety.

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 CTS 6.2.B.1 non-licensed operator requirements have been revised. Proposed ITS 5.2.2.a specifies non-licensed operator staffing requirements, and requires at least one required non-licensed operator be assigned to each unit. This change does not reduce or eliminate non-licensed personnel required in the current licensing basis. This ensures both units have at least one non-licensed operator to perform required tasks. This change is consistent with the BWR ISTS, NUREG-1433, Rev. 1, and is considered more restrictive on plant operations.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 CTS 6.2.A.3 uses the title "Chief Nuclear Officer." In ITS 5.2.1.c this specific title is replaced with the generic term "a corporate officer." CTS 6.2.A.2 uses the title "Station Manager." In ITS 5.2.1.b, this specific title is replaced with the generic title "station manager.". CTS 6.2.B.6 uses the titles "Operations Manager" and "Shift Operations Supervisor." In ITS 5.2.2.f, these specific titles are replaced with the generic titles "operations manager and shift operations supervisor." The specific titles are proposed to be relocated to the Quality Assurance (QA) Manual, which is where the description of these specific titles is currently located. The allowance to relocate the specific titles out of the Technical Specifications is consistent with the NRC letter from C. Grimes to the Owners Groups Technical Specification Committee Chairmen, dated November 10, 1994. The various requirements of the individuals are still retained in the ITS. In addition, the ITS also requires the plant specific titles to be in the QA Manual. Therefore, the relocated specific titles are not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the QA Manual are controlled by the provisions of 10 CFR 50.54.

LA.2 CTS Specification 6.2.B.5 provides details with respect to the development and implementation of procedures to limit the working hours of facility staff who perform safety-related functions. These details are to be relocated the UFSAR. The relocation of the requirement to have procedures developed and implemented will have no effect on ensuring that an individual is not fatigued while performing safety-related functions. ITS 5.2.2.e includes reference to the NRC Overtime Policy Statement, which provides the programmatic requirements for the overtime policy. As such, these details are not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the UFSAR are controlled by the provisions of 10 CFR 50.59.

"Specific"

L.1 CTS 6.2.B.2 requires at least one licensed Senior Reactor Operator (SRO) to be present in the control room while the unit is in MODE(s) 1, 2, 3, or 4. The licensed operator staffing requirements of 10 CFR 50.54(m)(2)(iii) only require an SRO to be present in the control room while in an operational mode (i.e., a mode other than cold shutdown and refueling). Thus, for a Boiling Water Reactor, an SRO is only required to be present in the control room while the unit

TECHNICAL CHANGES - LESS RESTRICTIVE

L.1 (cont'd) is in MODE 1, 2, or 3. It is, therefore, proposed to delete the CTS 6.2.B.2 requirement for an SRO to be present in the control room while the unit is in MODE 4 such that the resulting requirement conforms to 10 CFR 50.54(m)(2)(iii) and is consistent with the BWR ISTS, NUREG-1433, Revision 1 (ISTS 5.2.2.b). This change is considered acceptable since the non-operational modes (MODES 4 and 5) are the safest conditions covered by the Technical Specifications. In MODE 4, all control rods are normally fully inserted and the probability and consequences of a Design Basis Accident are significantly reduced due to the limitations on pressure and temperature. In addition, pursuant to 10 CFR 50.54(m)(2), a Reactor Operator (RO) will still be required to be present at the controls (in the control room) at all times and at least one SRO, who is assigned supervisory responsibility, will be required to be on-site and readily available to the RO for consultation while the unit is in MODE 4.

RELOCATED SPECIFICATIONS

None

5.3 6.3 UNIT STAFF QUALIFICATIONS

radiation protection technician

5.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971, "Selection and Training of Nuclear Plant Personnel", dated March 8, 1971, except for the ~~Rad/Chem Superintendent~~ or ~~Lead Health Physicist~~, who shall meet or exceed the qualifications of the Radiation Protection Manager as specified in Regulatory Guide 1.8, September 1975~~, and 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.~~ LA.1 A.2

radiation protection manager LA.1

Page 1 of 1

ADMINISTRATIVE

A.1 In the conversion of the Quad Cities 1 and 2 current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted that do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the BWR Standard Technical Specifications, NUREG 1433, Rev. 1 (i.e., the Improved Standard Technical Specifications (ISTS)).

A.2 The details in CTS 6.3 for qualification requirements of the Shift Technical Advisor (STA) position are being deleted. These requirements are adequately addressed in CTS 6.2.C (proposed ITS 5.2.2.g) "specified by the Commission Policy Statement on Engineering Expertise on Shift," and therefore, it is unnecessary to restate the qualification requirements. Since the STA position requirements are retained in proposed ITS 5.2.2.g, this change is considered administrative.

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 CTS 6.3 uses the plant titles "Rad/Chem Superintendent," and "Lead Health Physicist." In ITS 5.3.1, these specific titles are replaced with the generic titles "radiation protection manager" and "lead radiation protection technician," respectively. The specific title is proposed to be relocated to the Quality Assurance (QA) Manual, which is where the description of these specific titles are currently located. The allowance to relocate the specific titles out of the Technical Specifications is consistent with the NRC letter from C. Grimes to the Owners Groups Technical Specification Committee Chairmen, dated November 10, 1994. In addition, the ITS also requires the plant specific titles to be in the QA Manual. Therefore, the relocated specific title is not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the QA Manual are controlled by the provisions of 10 CFR 50.54.

TECHNICAL CHANGES - LESS RESTRICTIVE (continued)

"Specific"

None

RELOCATED SPECIFICATIONS

None

5.4 **6.8 PROCEDURES AND PROGRAMS** <See ITS 5.5>

5.4.1 **6.8.A** Written procedures shall be established, implemented, and maintained covering the activities referenced below:

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

5.4.1.b 2. The Emergency Operating Procedures required to implement the requirements of NUREG-0737 and Supplement 1 to NUREG-0737 as stated in Section 7.1 of Generic Letter No. 82-33,

~~3. Station Security Plan implementation,~~

~~4. Generating Station Emergency Response Plan implementation,~~ A.2

~~5. PROCESS CONTROL PROGRAM (PCP) implementation,~~ LA.1

~~6. OFFSITE DOSE CALCULATION MANUAL (ODCM) implementation, and~~ A.3

5.4.1.c 7. Fire Protection Program implementation.

Add proposed ITS 5.4.1.d M.1

~~6.8.B Deleted~~

~~6.8.C Deleted~~

6.8.D The following programs shall be established, implemented, and maintained:

1. Reactor Coolant Sources Outside Primary Containment

   This program provides controls to minimize leakage from those portions of systems outside primary containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include CS, HPCI, LPCI, RCIC, process sampling (post accident sampling of reactor coolant and containment atmosphere), containment monitoring, and standby gas treatment systems. The program shall include the following:

   a. Preventive maintenance and periodic visual inspection requirements, and

   b. Leak test requirements for each system at a frequency of at least once per operating cycle.

<See ITS 5.5>

ADMINISTRATIVE

A.1 In the conversion of the Quad Cities 1 and 2 current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted that do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the BWR Standard Technical Specifications, NUREG 1433, Rev. 1 (i.e., the Improved Standard Technical Specifications (ISTS)).

A.2 Procedures required by CTS 6.8.A.3 and 6.8.A.4 to implement the Station Security Plan and the Generating Station Emergency Response Plan are also required by 10 CFR 50.54(p) and 10 CFR 50, Appendix E. Since conformance with 10 CFR Chapter 1 is a license condition and the Emergency Plan and Security Plan are required to be implemented by 10 CFR Chapter 1, specific identification of these plans is unnecessary duplication. This is a change in the presentation of the requirements only and, therefore, is considered an administrative change.

A.3 CTS 6.8.A.6, which requires written procedures for ODCM implementation, is covered by a more generic item, ITS 5.4.1.d, which requires this activity for all Programs and Manuals. Therefore, it is not necessary to specifically identify each program. Since the requirements remain, this is considered to be a change in the method of presentation only and, therefore, is considered an administrative change.

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 ITS 5.4.1.d is added to the TS that all programs specified in Specification 5.5 have written procedures. ITS 5.5 contains twelve programs that will require (by ITS 5.4.1.d) procedures to be implemented and maintained. This will ensure proper procedure control of TS required programs. This is an additional restriction on plant operation in that it will be controlled through Technical Specifications.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The requirement in CTS 6.8.A.5 that written procedures for the PROCESS CONTROL PROGRAM (PCP) be established, implemented, and maintained are proposed to be relocated to the UFSAR. The PCP implements the requirements of 10 CFR 20, 10 CFR 61, and 10 CFR 71. Compliance with these regulations is required by the Quad Cities 1 and 2 Operating Licenses, and procedures would be the method to ensure compliance with the program. As such, relocation of the procedure requirements of the PCP from the ITS does not affect the safe operation of the facility. Therefore, the relocated details are not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the UFSAR are controlled by the provisions of 10 CFR 50.59.

"Specific"

None

RELOCATED SPECIFICATIONS

None

5.0 ADMINISTRATIVE CONTROLS

5.5 6.8 PROCEDURES AND PROGRAMS

6.8.A Written procedures shall be established, implemented, and maintained covering the activities referenced below:

1. The applicable procedures recommended in Appendix A, of Regulatory Guide 1.33, Revision 2, February 1978,
2. The Emergency Operating Procedures required to implement the requirements of NUREG-0737 and Supplement 1 to NUREG-0737 as stated in Section 7.1 of Generic Letter No. 82-33,
3. Station Security Plan implementation,
4. Generating Station Emergency Response Plan implementation,
5. PROCESS CONTROL PROGRAM (PCP) implementation,
6. OFFSITE DOSE CALCULATION MANUAL (ODCM) implementation, and
7. Fire Protection Program implementation.

<See ITS 5.4>

6.8.B Deleted

6.8.C Deleted

5.5 6.8.D The following programs shall be established, implemented, and maintained:

5.5.2 1. Reactor Coolant Sources Outside Primary Containment

M.1

This program provides controls to minimize leakage from those portions of systems outside primary containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include CS, HPCI, LPCI, RCIC, Reactor Water Cleanup, process sampling ~~(post accident sampling of reactor coolant and containment atmosphere)~~, containment monitoring, and standby gas treatment systems. The program shall include the following: LA.7

5.5.2.a a. Preventive maintenance and periodic visual inspection requirements, and

5.5.2.b b. Leak test requirements for each system at a frequency of at least once per ~~operating cycle~~ 24 months LD.1

The provisions of SR 3.0.2 are applicable to the 24 month Frequency for performing integrated system leak test activities. A.2

A.1

LA.1

2. ~~In-Plant Radiation Monitoring~~

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

~~a. Training of personnel,~~

~~b. Procedures for monitoring, and~~

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

5.5.3 3. Post Accident Sampling

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

5.5.3.a a. Training of personnel,

5.5.3.b b. Procedures for sampling and analysis,

5.5.3.c c. Provisions for maintenance of sampling and analysis equipment.

5.5.4 **4. Radioactive Effluent Controls Program**

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

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

5.5.4.b b. Limitations on the instantaneous concentrations of radioactive material released in liquid effluents to unrestricted areas conforming to ten (10) times the concentration values in 10 CFR Part 20, Appendix B, Table 2, Column 2 to 10 CFR Part 20.1001 - 20.2402,

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

or dose commitment A.3

5.5.4.d d. Limitations on the annual and quarterly doses to a member of the public from radioactive materials in liquid effluents released from each Unit conforming to Appendix I to 10 CFR Part 50,

to unrestricted areas

5.5.4.e e. Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days, 5.5.4.f f. Limitations on the operability and use of the liquid and gaseous effluent treatment systems to ensure that the appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a 31-day period would exceed 2 percent of the guidelines for the annual dose or dose commitment conforming to Appendix I to 10 CFR Part 50, A.3

5.5.4.g g. Limitations on the dose rate resulting from radioactive materials released in gaseous effluents from the site to areas at or beyond the site boundary shall be limited to the following:

5.5.4.g.1 a) For noble gases: less than or equal to a dose rate of 500 mrem/yr to the whole body and less than or equal to a dose rate of 3000 mrem/yr to the skin, and

5.5.4.g.2 b) For Iodine-131, Iodine-133, tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: less than or equal to a dose rate of 1500 mrem/yr to any organ.

5.5.4.h 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 Appendix I to 10 CFR Part 50,

5.5.4.i i. Limitations on the annual and quarterly doses to a member of the public from Iodine-131, Iodine-133, tritium, and all radionuclides in particulate form with halflives greater than 8 days in gaseous effluents released from each Unit conforming to Appendix I to 10 CFR Part 50,

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

← Add proposed ITS 5.5.5 — M.2

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluents Control Program Surveillance Frequencies. — A.2

5.5.12 5. **Primary Containment Leakage Rate Testing Program**

5.5.12.a A program shall be established to implement the leakage rate testing of the primary containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemption. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Testing Program," dated September 1995.

5.5.12.b The peak calculated primary containment internal pressure for the design basis loss of coolant accident, P_a, is 48 psig.

5.5.12.c The maximum allowable primary containment leakage rate, L_a, at P_a, is 1% of primary containment air weight per day.

5.5.12.d Leakage rate acceptance criteria are:

5.5.12.d.1 a. Primary containment overall 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 $\leq 0.60\ L_a$ for the combined Type B and Type C tests, and $\leq 0.75\ L_a$ for Type A tests.

5.5.12.d.2 b. Air lock testing acceptance criteria is the overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

~~The provisions of 4.0.B do not apply to the test frequencies specified in the Primary Containment Leakage Rate Testing Program.~~ A.4

5.5.12.e The provisions of 4.0.C are applicable to the Primary Containment Leakage Rate Testing Program.

4.0 - SURVEILLANCE REQUIREMENTS

A. Surveillance Requirements shall be met during the reactor OPERATIONAL MODE(s) or other conditions specified for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement.

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

C. Failure to perform a Surveillance Requirement within the allowed surveillance interval, defined by Specification 4.0.B, shall constitute noncompliance with the OPERABILITY requirements for a Limiting Condition for Operation. The time limits of the ACTION requirements are applicable at the time it is identified that a Surveillance Requirement has not been performed. The ACTION requirements may be delayed for up to 24 hours to permit the completion of the surveillance when the allowable outage time limits of the ACTION requirements are less than 24 hours. Surveillance requirements do not have to be performed on inoperable equipment.

D. Entry into an OPERATIONAL MODE or other specified applicable condition shall not be made unless the Surveillance Requirement(s) associated with the Limiting Condition for Operation have been performed within the applicable surveillance interval or as otherwise specified. This provision shall not prevent passage through or to OPERATIONAL MODE(s) as required to comply with ACTION requirements.

5.5.6 E. Surveillance Requirements for ~~inservice inspection and~~ testing of ASME Code Class 1, 2, and 3 ~~components~~ shall be applicable as follows: LA.2

1. ~~Inservice Inspection of ASME Code Class 1, 2 and 3 components and inservice testing of ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR Part 50, Section 50.55a(g) and 50.55a(f) respectively, except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50, Section 50.55a(g)(6)(i) or 50.55a(f)(6)(i), respectively.~~

pumps and valves LA.3 LA.2 LA.3

See ITS Section 3.0

4.0 - SURVEILLANCE REQUIREMENTS

5.5.6.a 2. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice ~~inspection and~~ testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice ~~inspection and~~ testing activities	Required Frequencies for performing inservice ~~inspection and~~ testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days
Every 48 months	At least once per 1461 days

LA.2

A.10

5.5.6.b 3. The provisions of Specification 4.0.B are applicable to the above required frequencies for performing inservice ~~inspection and~~ testing activities.

~~4. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements.~~

A.5

5.5.6.d 5. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

~~6. The Inservice Inspection Program for piping identified in NRC Generic Letter 88-01 shall be performed in accordance with the staff positions on schedule, methods, and personnel and sample expansion included in Generic Letter 88-01 or in accordance with alternate measures approved by the NRC staff.~~

LA.2

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

A.2

3.7 - LIMITING CONDITIONS FOR OPERATION

P. Standby Gas Treatment System

Two independent standby gas treatment subsystems shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, 3 and *.

ACTION:

1. With one standby gas treatment subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days, or:

   a. In OPERATIONAL MODE(s) 1,2 or 3, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

   b. In OPERATIONAL MODE *, suspend handling of irradiated fuel in the secondary containment, CORE ALTERATION(s), and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.C are not applicable.

2. With both standby gas treatment subsystems inoperable in OPERATIONAL MODE(s) 1,2 or 3, restore at least one subsystem to OPERABLE status within one hour, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.7 - SURVEILLANCE REQUIREMENTS

P. Standby Gas Treatment System

Each standby gas treatment subsystem shall be demonstrated OPERABLE:

1. At least once per 31 days by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the subsystem operates for at least 10 hours with the heaters operating. LD.2

5.5.7 2. At least once per ~~18~~ 24 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following significant painting, fire or chemical release in any ventilation zone communicating with the subsystem by: A.11 A.6 Add proposed ITS 5.5.7

   5.5.7.a 5.5.7.b a. Verifying that the subsystem satisfies the in-place penetration and bypass leakage testing acceptance criteria of < 1% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and ANSI/ASME N510-1980 and the system flow rate is 4000 cfm ± 10%. A.7

   5.5.7.c b. Verifying ~~within 31 days after removal~~ that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of ASTM-D-3803-89, for a methyl iodide penetration of < 2.5%, when tested at 30°C and 70% relative humidity; and LA.4

* When handling irradiated fuel in the secondary containment, during CORE ALTERATION(s), and operations with a potential for draining the reactor vessel.

<See ITS 3.6.4.3>

3.7 - LIMITING CONDITIONS FOR OPERATION

3. With both standby gas treatment subsystems inoperable in OPERATIONAL MODE *, suspend handling of irradiated fuel in the secondary containment, CORE ALTERATION(s), and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.C are not applicable.

<See ITS 3.6.4.3>

4.7 - SURVEILLANCE REQUIREMENTS

5.5.7.a 5.5.7.b c. Verifying a subsystem flow rate of 4000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1980.

5.5.7 3. After every 1440 hours of charcoal adsorber operation by verifying ~~within 31 days after removal~~ that a laboratory LA.4

5.5.7.c analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of ASTM-D-3803-89, for a methyl iodide penetration of <2.5%, when tested at 30°C and 70% relative humidity.

5.5.7 4. At least once per 18 months by: 24 LD.2

5.5.7.d a. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is <6 inches water gauge while operating the filter train at a flow rate of 4000 cfm ±10%.

b. Verifying that the filter train starts and isolation dampers open on each of the following test signals:

1) Manual initiation from the control room, and

2) Simulated automatic initiation signal.

5.5.7.e c. Verifying that the heaters dissipate 30 ±3 kw when tested in accordance with ANSI N510-1989. This reading shall include the appropriate correction for variations in voltage.

* When handling irradiated fuel in the secondary containment, during CORE ALTERATION(s), and operations with a potential for draining the reactor vessel.

3.7 - LIMITING CONDITIONS FOR OPERATION

4.7 - SURVEILLANCE REQUIREMENTS

5.5.7 5. After each complete or partial replacement of a HEPA filter bank by
5.5.7.a verifying that the HEPA filter bank satisfies the in-place penetration and leakage testing acceptance criteria of <1% in accordance with ANSI N510-1980 while operating the system at a flow rate of 4000 cfm ±10%.

5.5.7 6. After each complete or partial replacement of a charcoal adsorber bank by
5.5.7.b verifying that the charcoal adsorber bank satisfies the in-place penetration and leakage testing acceptance criteria of <1% in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 4000 cfm ±10%.

A.6

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

3.8 - LIMITING CONDITIONS FOR OPERATION

D. Control Room Emergency Ventilation System

The control room emergency ventilation system shall be OPERABLE, with the system comprised of an OPERABLE control room emergency filtration system and an OPERABLE refrigeration control unit (RCU).

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, 3, and *.

ACTION:

1. In OPERATIONAL MODE(s) 1, 2 or 3:
   a. With the control room emergency filtration system inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.
   b. With the refrigeration control unit (RCU) inoperable, restore the inoperable RCU to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.8 - SURVEILLANCE REQUIREMENTS

D. Control Room Emergency Ventilation System

The control room emergency ventilation system shall be demonstrated OPERABLE:

1. At least once per 18 months by verifying that the RCU has the capability to remove the required heat load.
2. At least once per 31 days by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 hours with the heaters operating.

LD.3

5.5.7 3. At least once per ~~18~~ 24 months or (1) after any ~~structural~~ significant maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system by:

A.11

Add proposed ITS 5.5.7

A.6

5.5.7.a 5.5.7.b a. Verifying that the system satisfies the in-place penetration and bypass leakage testing acceptance criteria of <0.05% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, ~~and the system flow rate is 2000 scfm ±10%.~~

A.7

and ANSI/ASME N510-1980

See ITS 3.7.4

* When handling irradiated fuel in the secondary containment, during CORE ALTERATION(s), and operations with a potential for draining the reactor vessel.

3.8 - LIMITING CONDITIONS FOR OPERATION

2. In OPERATIONAL MODE *, with the control room emergency filtration system or the RCU inoperable, immediately suspend CORE ALTERATION(s), handling of irradiated fuel in the secondary containment and operations with a potential for draining the reactor vessel.

3. The provisions of Specification 3.0.C are not applicable in OPERATIONAL MODE *.

4.8 - SURVEILLANCE REQUIREMENTS

5.5.7.c b. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of ASTM-D-3803-89, for a methyl iodide penetration of <0.50%, when tested at 30°C and 70% relative humidity; and LA.4

5.5.7.a
5.5.7.b c. Verifying a system flow rate of 2000 scfm ±10% during system operation when tested in accordance with ANSI N510-1980.

5.5.7 4. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of ASTM-D-3803-89, for a methyl iodide penetration of <0.50%, when tested at 30°C and 70% relative humidity. LA.4 5.5.7.c

5.5.7 5. At least once per ~~18~~ 24 months by: LD.3

5.5.7.d a. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is <6 inches water gauge while operating the filter train at a flow rate of 2000 scfm ±10%.

<See ITS 3.7.4>

* When handling irradiated fuel in the secondary containment, during CORE ALTERATION(s), and operations with a potential for draining the reactor vessel.

A.1

ITS 5.5

3.8 - LIMITING CONDITIONS FOR OPERATION

4.8 - SURVEILLANCE REQUIREMENTS

<See ITS 3.7.4>

b. Verifying that the isolation dampers close on each of the following signals:

1) Manual initiation from the control room, and

2) Simulated automatic isolation signal.

c. Verifying that during the pressurization mode of operation, control room positive pressure is maintained at ≥ 1/8 inch water gauge relative to adjacent areas during system operation at a flow rate ≤ 2000 scfm.

5.5.7.e d. Verifying that the heaters dissipate 12 ± 1.2 kw when tested in accordance with ANSI N510-1989. This reading shall include the appropriate correction for variations from 480 volts at the bus.

5.5.7 6. After each complete or partial replacement of an HEPA filter bank by

5.5.7.a verifying that the HEPA filter bank satisfies the in-place penetration and leakage testing acceptance criteria of <0.05% in accordance with ANSI N510-1980 while operating the system at a flow rate of 2000 scfm ± 10%.

5.5.7 7. After each complete or partial replacement of an charcoal adsorber

5.5.7.b bank by verifying that the charcoal adsorber bank satisfies the in-place penetration and leakage testing acceptance criteria of <0.05% in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at flow rate of 2000 scfm ± 10%.

A.6

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

3.8 - LIMITING CONDITIONS FOR OPERATION

H. Offgas Explosive Mixture

5.5.8.a The concentration of hydrogen in the offgas holdup system shall be limited ~~to ≤4% by volume~~.

LA.5

~~**APPLICABILITY:**~~

~~During offgas holdup system operation.~~

~~**ACTION:**~~

~~With the concentration of hydrogen in the offgas holdup system exceeding the limit, restore the concentration to within the limit within 48 hours. The provisions of Specification 3.0.C are not applicable.~~

4.8 - SURVEILLANCE REQUIREMENTS

H. Explosive Gas Mixture

5.5.8.a The concentration of hydrogen in the offgas holdup system shall be determined to be within the above limits ~~as required by Table 3/2.H/1 of Specification 3.2.H~~.

M.2

Add proposed ITS 5.5.8 for Storage Tank Radioactivity Monitoring

A.8

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

Add proposed ITS 5.5.9

3.9 - LIMITING CONDITIONS FOR OPERATION

c. Restore the diesel generator to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

3. With one of the above offsite circuit power sources and one of the above required diesel generator power sources inoperable:

a. Demonstrate the OPERABILITY of the remaining offsite circuit power source by performing Surveillance Requirement 4.9.A.1.a within 1 hour and at least once per 8 hours thereafter.

b. If the diesel generator is inoperable due to any cause other than preplanned preventive maintenance or testing, demonstrate the OPERABILITY[a] of the remaining OPERABLE diesel generator by performing Surveillance Requirement 4.9.A.2.c[b] within 8 hours unless the absence of any potential common mode failure for the remaining diesel generator is demonstrated (if it has not been successfully tested within the past 24 hours) and within the subsequent 72 hours for each OPERABLE diesel generator.

4.9 - SURVEILLANCE REQUIREMENTS

5. Each of the required diesel generators shall be demonstrated OPERABLE by:

5.5.9 5.5.9.a a. Sampling new fuel oil prior to addition to the storage tanks in accordance with applicable ASTM standards, and

L.1 or absolute specific gravity L.1

L.1 or b. Verifying prior to addition to the storage tanks that the sample meets the applicable ASTM standards for API gravity, water and sediment, ~~and~~ the visual test for free water ~~and particulate contamination[h]~~, and

5.5.9.a.1 5.5.9.a.2 5.5.9.a.3

M.3 flash point

L.1

5.5.9.b c. Verifying within 31 days of ~~obtaining the sample~~ that the kinematic viscosity[h] within applicable ASTM limits.

addition of the new fuel oil to storage tanks

properties of the fuel oil, other than those addressed above are

L.1

6. Each of the required diesel generators shall be demonstrated OPERABLE by:

5.5.9.c a. Sampling and analyzing the bulk fuel storage tanks at least once per 31 days in accordance with applicable ASTM standards, and

5.5.9.c b. Verifying that ~~the sample meets the applicable ASTM standards for water and sediment, kinematic viscosity, and ASTM~~ particulate contaminant[h] is <10 mg/liter.

≤

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

A.9

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a A successful test of OPERABILITY per Surveillance Requirement 4.9.A.2.c under this ACTION statement satisfies the diesel generator test requirements of ACTION(s) 1 or 2 above.

b Contrary to the provisions of Specification 3.0.B, this test is required to be completed regardless of when the inoperable diesel generator is restored to OPERABILITY for failures that are potentially generic to the remaining diesel generator and for which appropriate alternative testing cannot be designed.

M.5

h ~~The particulate contamination surveillance is not required for No. 1 fuel oil. It is required for No. 2 fuel oil and for blends.~~

See ITS 3.8.1

M.2

Add proposed ITS 5.5.10

Add proposed ITS 5.5.11

1.0 DEFINITIONS

LOGIC SYSTEM FUNCTIONAL TEST (LSFT)

A LOGIC SYSTEM FUNCTIONAL TEST (LSFT) shall be a test of all required logic components, i.e., all required relays and contacts, trip units, solid state logic elements, etc, of a logic circuit, from as close to the sensor as practicable up to, but not including the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping or total system steps so that the entire logic system is tested.

MAXIMUM FRACTION OF LIMITING POWER DENSITY (MFLPD)

The MAXIMUM FRACTION OF LIMITING POWER DENSITY (MFLPD) shall be the highest value of the FLPD which exists in the core (applicable to GE fuel).

MINIMUM CRITICAL POWER RATIO (MCPR)

The MINIMUM CRITICAL POWER RATIO (MCPR) shall be the smallest CPR which exists in the core for each class of fuel.

5.5.1 OFFSITE DOSE CALCULATION MANUAL (ODCM)

5.5.1.a The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program.

5.5.1.b The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Specification 6.8 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Annual Radioactive Effluent Release Reports required by Specification 6.9.

OPERABLE - OPERABILITY

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

OPERATIONAL MODE

An OPERATIONAL MODE, i.e., MODE, shall be any one inclusive combination of mode switch position and average reactor coolant temperature as specified in Table 1-2.

PHYSICS TESTS

PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 14 of the UFSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

<See ITS Chapter 1.0>

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

5.5.1.c 6.14.A Changes to the ODCM:

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

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

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

5.5.1.c.2 2. Shall become effective after review and acceptance, including approval by the ~~Station~~ ~~Manager~~. LA.6

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

ADMINISTRATIVE

A.1 In the conversion of the Quad Cities 1 and 2 current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted that do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the BWR Standard Technical Specifications, NUREG 1433, Rev. 1 (i.e., the Improved Standard Technical Specifications (ISTS)).

A.2 A statement of applicability of SR 3.0.2 has been added to CTS 6.8.D.1 (ITS 5.5.2), a statement of applicability of SR 3.0.3 has been added to CTS 4.0.E (ITS 5.5.6.c), and a statement of applicability of SR 3.0.2 and SR 3.0.3 has been added to CTS 6.8.D.4 (ITS 5.5.4). These statements are needed to maintain allowances for Surveillance Frequency extensions contained in the ITS since these SRs are not normally applied to frequencies identified in the Administrative Controls Section of the ITS. Since this change is a clarification required to maintain provisions that would be allowed in the LCO sections of the Technical Specifications, it is considered administrative in nature.

A.3 The wording in CTS 6.8.D.4.d and CTS 6.8.D.4.f have been revised in ITS 5.5.4.d and ITS 5.5.4.f to provide clarity. These changes do not modify the Current Licensing Basis requirements and, as such, this change is considered administrative.

A.4 CTS 6.8.D.5 exempts the requirements of CTS SR 4.0.B from applying to the frequencies specified in the Primary Containment Leakage Rate Testing Program. In the ITS, the ITS 3.0 Chapter requirements only applies to ITS Sections 3.1 through 3.10. This is specifically stated in the Bases for ITS Chapter 3.0. In addition, by maintaining this requirement in the ITS, it will add confusion since only those ITS Chapter 3.0 allowances are provided when they are applicable. For example, CTS 4.0.A and 4.0.B also do not apply to the Primary Containment Leakage Rate Appendix J Testing Program, but this is not stated in CTS 6.8.D.5. Therefore, the specific statement to exempt this requirement is redundant and has been deleted.

A.5 CTS 4.0.E.4 restates that all applicable requirements must be met. Repeating this overall requirement as a specific detail is redundant and unnecessary. Therefore, this detail can be omitted without any technical change in the requirements and is considered administrative in nature.

ADMINISTRATIVE (continued)

A.6 The filter testing requirements for the Standby Gas Treatment (SGT) System (CTS 4.7.P.2, 4.7.P.3, 4.7.P.4, 4.7.P.5, and 4.7.P.6) and the Control Room Emergency Ventilation (CREV) System (CTS 4.8.D.3, 4.8.D.4, 4.8.D.5, 4.8.D.6, and 4.8.D.7) have been placed in a program in the proposed Administrative Controls Chapter 5.0. As such, a general program statement has been added as ITS 5.5.7. Also, a statement of applicability of SR 3.0.2 and SR 3.0.3 is needed to clarify that the allowances for Surveillance Frequency extensions do apply, since these SRs are not normally applied to Frequencies identified in the Administrative Controls Chapter of the Technical Specifications. Since this change is a clarification needed to maintain provisions that would be allowed in the LCO sections of the Technical Specifications, it is considered administrative.

A.7 Current Technical Specifications for in-place charcoal adsorber testing of the Standby Gas Treatment (SGT) System (CTS 4.7.P.2.a) and the Control Room Emergency Ventilation (CREV) System) (CTS 4.8.D.3.a) reference Regulatory Positions of Regulatory Guide (RG) 1.52, Revision 2, March 1978. ITS 5.5.7.a and ITS 5.5.7.b reference RG 1.52, Revision 2, and ANSI/ASME N510-1980. The changes to the references provide clarity but do not change the current testing requirements or acceptance criteria. Therefore, these changes are considered administrative.

A.8 The Offgas Explosive Mixture requirements in CTS 3.8.H has been placed in a program in the proposed Administrative Controls Chapter 5.0. As such, a general program statement has been added. In addition, a statement of applicability of SR 3.0.2 and SR 3.0.3 is needed to clarify that the allowances for Surveillance Frequency extensions do apply, since these SRs are not normally applied to Frequencies identified in the Administrative Controls Chapter of the Technical Specifications. Since this change is a clarification needed to maintain provisions that would be allowed in the LCO sections of the Technical Specifications, it is considered administrative in nature.

A.9 The diesel fuel oil testing requirements in CTS 4.9.A.5 and 4.9.A.6 have been placed in a program in the proposed Administrative Controls Chapter 5.0. As such, a general program statement has been added. Also, a statement of applicability of SR 3.0.2 and SR 3.0.3 is needed to clarify that the allowances for Surveillance Frequency extensions do apply, since these SRs are not normally applied to Frequencies identified in the Administrative Controls Chapter of the Technical Specifications. Since this change is a clarification needed to maintain provisions that would be allowed in the LCO sections of the Technical Specifications, it is considered administrative in nature.

ADMINISTRATIVE (continued)

A.10 An additional definition of a frequency "Every 48 months" is identified for the Inservice Testing Program requirements of CTS 4.0.E.2. This change includes no new requirements, but only provides a clarification of a term. Therefore, this change is considered to be administrative.

A.11 CTS 4.7.P.2 and 4.8.D.3 requires certain SGT and CREV System filter testing following painting, fire, or chemical release in any ventilation zone communicating with the subsystems. ITS 5.5.7 only requires testing if the painting, fire, or chemical release is significant. Current Quad Cities 1 and 2 practice is that not all painting, fire, or chemical release results in the need to perform certain ventilation filter tests. Only painting, fire, or chemical release that could affect the ventilation filter subsystems, i.e., that which is significant, would require performance of the tests. The word "significant" was added for clarity and consistency with current practice to avoid a misinterpretation that any painting, fire, or chemical release (such as using a small can of paint to do touch-up work in the reactor building) would result in the need to perform the tests. This clarification is administrative, and is consistent with the most recently approved BWR/5 ITS Amendment, WNP-2. In addition, the NRC, in a letter to Entergy Operations dated September 11, 1997, supported the clarification that not all painting, fires, or chemical releases required the ventilation filter subsystems to be tested.

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 CTS 6.8.D.1 (proposed ITS 5.5.2) is revised to include RWCU in the systems addressed by the Reactor Coolant Sources Outside Primary Containment Program. This will ensure the RWCU System leakage is controlled. This change is considered more restrictive on plant operations since the requirement is now controlled by the Technical Specifications.

M.2 Four new programs are included in the proposed Technical Specifications. These programs are:

ITS 5.5.5	Component Cyclic or Transient Limit
ITS 5.5.8	Storage Tank Radioactive Monitoring Program
ITS 5.5.10	Technical Specification (TS) Bases Control
ITS 5.5.11	Safety Function Determination Program (SFDP)

TECHNICAL CHANGES - MORE RESTRICTIVE

M.2 (cont'd) The Component Cyclic or Transient Limit Program is provided to control the tracking of UFSAR cyclic and transient occurrences. The Storage Tank Radioactive Monitoring Program is provided to control the quantity of radioactivity contained in unprotected outdoor liquid storage tanks. The TS Bases Control Program is provided to specifically delineate the appropriate methods and reviews necessary for a change to the Technical Specification Bases. The Safety Function Determination Program is included to support implementation of the support system OPERABILITY characteristics of the Technical Specifications. The specific wording associated with these three programs may be found in ITS 5.5.5, 5.5.8, 5.5.10, and 5.5.11.

M.3 CTS 4.9.A.5.b requirements for new fuel oil testing prior to addition to the storage tanks do not include flash point checks. ITS 5.5.9.a.2 includes a requirement to verify the new fuel oil flash point is within the requirements of the applicable ASTM standard. This will ensure the new fuel oil has a proper flash point prior to addition to the storage tanks. In addition, the Frequency of the CTS 4.9.5.A.5.c requirement to verify kinematic viscosity within 31 days of obtaining the sample is being changed to prior to addition to the storage tanks. This will ensure the kinematic viscosity of new fuel is within the limits prior to adding the new fuel to the storage tanks, in lieu of the current requirement which could allow the new fuel to be added with the kinematic viscosity not within the limit. In addition, ITS 5.5.9.b includes the requirement to verify, within 31 days of adding new fuel to the storage tanks, that properties other than those specifically addressed are within ASTM limits. These changes are consistent with BWR ISTS, NUREG-1433, Rev. 1, impose additional operational requirements and are considered more restrictive.

M.4 Not used.

M.5 CTS 4.9.A.5.b and CTS 4.9.A.6.b footnote h allows No. 1 fuel oil to be exempted from the particulate contamination testing requirements. This allowance is not consistent with the ASTM standard and has been deleted. The requirement to monitor total particulate concentration in the storage tanks is incorporated in ITS 5.5.9.c. This change is consistent with the BWR ISTS, NUREG-1433, Rev. 1, imposes additional operational requirements, and is considered more restrictive.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The details contained in CTS 6.8.D.2, "In-Plant Radiation Monitoring," are proposed to be relocated to the UFSAR. This program is required as a result of a license condition for Quad Cities 1 and 2 (Operating License Amendments 62 and 56, respectively dated February 6, 1981). This program contains controls to ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program is designed to minimize radiation exposure to plant personnel post-accident and has no impact on nuclear safety or the health and safety of the public. The training aspect of the program is accomplished as part of the continual training program for personnel in the cognizant organizations, as well as during the training for those individuals responsible for implementing the Radiological Emergency Planning procedures. Provisions for monitoring and performing maintenance of the sampling and analysis equipment are addressed in chemistry and radiation protection procedures. Therefore, the relocated details are not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the UFSAR are controlled by the provisions of 10 CFR 50.59.

LA.2 Details of the Inservice Inspection (ISI) Program in CTS 4.0.E are proposed to be relocated to the plant controlled ISI Program. The ISI Program is required by 10 CFR 50.55a to be performed in accordance with ASME Section XI. Compliance with 10 CFR 50.55a is required by the Quad Cities 1 and 2 Operating Licenses. The ISI Program, outside of the CTS, implements the applicable provisions of ASME Section XI. Generic Letter 88-01 provides an ISI Program for piping in accordance with the NRC staff positions on schedule, methods, personnel, and sample expansion or in accordance with alternate measures approved by the NRC staff. Quad Cities 1 and 2 commitments to Generic Letter 88-01 are documented in the NRC SER dated August 21, 1990, and do not need to be repeated in the ITS. Regulations and Quad Cities 1 and 2 commitments to the NRC contain the necessary programmatic requirements for ISI without repeating them in the ITS. Therefore, the relocated details are not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the plant controlled ISI Program will be controlled by the provisions of 10 CFR 50.55a. In addition, since the Inservice Testing Program is the only requirement remaining, the reference to ASME Code Class 1, 2, and 3 "components" has been changed to "pumps and valves" for clarity. Pumps and valves are the only components related to the Inservice Testing Program (as described in CTS 4.0.E.1).

TECHNICAL CHANGES - LESS RESTRICTIVE (continued)

LA.3 Details of the Inservice Testing Program (IST) in the CTS 4.0.E are proposed to be relocated to the plant controlled IST Program. The relocated requirements are duplicated in 10 CFR 50.55a, which requires the implementation of ASME, Section XI and applicable addenda, for inservice testing of ASME Code Class 1, 2, and 3 pumps and valves. Compliance with 10 CFR 50.55a is required by the Quad Cities 1 and 2 Operating Licenses. Therefore, it is not necessary to retain the details proposed to be relocated in the ITS, since these details are not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the plant controlled IST program will be controlled by the provisions of 10 CFR 50.55a.

LA.4 Details of the methods for implementing CTS 4.7.P.2.b, 4.7.P.3, 4.8.D.3.b, and 4.8.D.4 are relocated to the Technical Requirements Manual (TRM). The requirements of ITS 5.5.7 are adequate to ensure the required ventilation filter testing is performed. Proposed SR 3.6.4.3.2 of ITS 3.6.4.3, "Standby Gas Treatment (SGT) System," which requires ventilation filter testing of the SGT System to be performed in accordance with the VFTP, and proposed SR 3.7.4.2 of ITS 3.7.4, "Control Room Emergency Ventilation (CREV) System", which requires ventilation filter testing of the CREV System to be performed in accordance with the VFTP, and the requirements of ITS 5.5.7 provide adequate regulatory controls over the testing requirements proposed to be relocated. As a result, the requirements proposed to be relocated are not required to be included in the Technical Specifications to ensure required ventilation filter testing is adequately performed. Therefore, the relocated details are not required to be in the ITS to provide adequate protection of the public health and safety. The TRM will be incorporated by reference into the Quad Cities 1 and 2 UFSAR at ITS implementation. Changes to the TRM will be controlled by the provisions of 10 CFR 50.59.

LA.5 The details for implementing the requirements contained in CTS 3.8.H are proposed to be relocated to the Technical Requirements Manual (TRM). The requirements of ITS 5.5.8 are adequate to ensure the explosive gas mixtures in the offgas system are maintained within limits. ITS 5.5.8 provides regulatory control over the limitations and surveillances proposed to be relocated. The details proposed to be relocated are not required to be included in the ITS to ensure the explosive gas mixtures in the offgas system are maintained within limits. Therefore, the relocated details are not required to be in the ITS to provide adequate protection of the public health and safety. The TRM will be incorporated by reference into the Quad Cities 1 and 2 UFSAR at ITS implementation. Changes to the TRM will be controlled by the provisions of 10 CFR 50.59.

TECHNICAL CHANGES - LESS RESTRICTIVE (continued)

LA.6 CTS 6.14.A.2 uses the title "Station Manager." In ITS 5.5.1.c.2, this specific title is replaced with the generic title "station manager." The specific title is proposed to be relocated to the Quality Assurance (QA) Manual, which is where the description of this specific title is currently located. The allowance to relocate the specific 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 station manager are still retained in the ITS. In addition, the ITS also requires the plant specific titles to be in the QA Manual. Therefore, the relocated specific title is not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the QA Manual are controlled by the provisions of 10 CFR 50.54.

LA.7 The details in CTS 6.8.D.1, the Reactor Coolant Sources Outside Primary Containment Program, that the process sampling system includes the post accident sampling of reactor coolant and containment atmosphere is proposed to be relocated to the UFSAR. The requirements of ITS 5.5.2 that the Primary Coolant Sources Outside Containment Program must include the "process sampling" system is sufficient to ensure the requirements are met. Therefore, the relocated details are not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the UFSAR are controlled by the provisions of 10 CFR 50.59.

LD.1 The Frequency for performing CTS 6.8.D.1.b (ITS 5.5.2.b) has been extended from 18 months to 24 months. This requirement establishes a program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The proposed change will allow this Surveillance to extend the Surveillance Frequency from the current 18 month Surveillance Frequency (i.e., a maximum of 22.5 months accounting for the allowable grace period specified in CTS 4.0.B and proposed SR 3.0.2) to a 24 month Surveillance Frequency (i.e., a maximum of 30 months accounting for the allowable grace period specified in CTS 4.0.B and proposed SR 3.0.2). This proposed 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 maintenance and surveillance data have shown that this test normally passes the Surveillance 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. This conclusion is based upon the fact that most portions of the subject systems included in this

TECHNICAL CHANGES - LESS RESTRICTIVE

LD.1 (cont'd) 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 that once per refueling cycle, and support the conclusion that the impact, if any, on safety is minimal as a result of the proposed changes.

The review of historical maintenance and surveillance data also demonstrates that there is no adverse trend that would invalidate the conclusion that the impact on system availability, if any, is minimal from a change to CTS 6.8.D.1.b as implemented in ITS 5.5.2.b. In addition, the proposed 24 month Surveillance Frequency, if performed at the maximum interval allowed by proposed SR 3.0.2 (30 months) does not invalidate any assumptions in the plant licensing basis.

LD.2 The Frequency for performing CTS 4.7.P.2.a, 4.7.P.2.b, 4.7.P.2.c, 4.7.P.4.a, and 4.7.P.4.c has been extended from 18 months to 24 months in ITS 5.5.7. These requirements ensure that the SGT System inplace charcoal adsorbers, HEPA filters, and heaters perform their safety function. The proposed change will allow these Surveillances to extend their Surveillance Frequency from the current 18 month Surveillance Frequency (i.e., a maximum of 22.5 months accounting for the allowable grace period specified in CTS 4.0.B and proposed SR 3.0.2) to a 24 month Surveillance Frequency (i.e., a maximum of 30 months accounting for the allowable grace period specified in CTS 4.0.B and proposed SR 3.0.2). This proposed 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 maintenance and surveillance data 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. UFSAR Table 1.8-1 identifies that charcoal adsorber and HEPA filter in-place tests are in accordance with Regulatory Guide 1.52, which states that testing Frequencies be at least once per 18 months. The SGT System filters radioactive particulates and both radioactive and nonradioactive forms of iodine from the air exhausted from the reactor enclosure and/or refueling area to maintain a negative pressure during secondary containment isolation. Regulatory positions C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, state HEPA filters and carbon adsorbers should be in-place tested (1) initially, (2) at least once per 18 months

TECHNICAL CHANGES - LESS RESTRICTIVE

LD.2 (cont'd) thereafter, and (3) following painting, fire, or chemical release in any ventilation zone communicating with the system. Position C.5.d also states that carbon adsorbers should be in-place tested following removal of an adsorber sample for laboratory testing if the integrity of the adsorber section is affected. ITS 5.5.7 also requires in-place filter and charcoal adsorber testing and filter pressure drop testing after any structural maintenance on the HEPA filter or charcoal adsorber housings or following painting, fire, or chemical release in any ventilation zone communicating with the SGT System. By testing after maintenance, fire, chemical release, painting, HEPA replacement, or charcoal replacement, potential changes in HEPA filter efficiency, carbon adsorber bypass leakage, and filter pressure drop will be detected that would be detected by conducting the 18 month surveillance tests. The SGT System is normally in standby. In addition, the SGT System active components and power supplies are designed with redundancy to meet the single active failure criteria, which will ensure system availability in the event of a failure of one of the system components. Based on the fact that the SGT System is normally in standby and additional testing will be performed if potential degradation occurs and the system design, it is shown that the impact, if any, on system availability is minimal as a result of this change.

The review of historical maintenance and surveillance data also demonstrates that there are no failures that would invalidate the conclusion that the impact on system availability, if any, is minimal from a change to CTS 4.7.P.2.a, 4.7.P.2.b, 4.7.P.2.c, 4.7.P.4.a, and 4.7.P.4.c as implemented in ITS 5.5.7, 5.5.7.a, 5.5.7.b, 5.5.7.c, 5.5.7.d, and 5.5.7.e. In addition, the proposed 24 month Surveillance Frequencies, if performed at the maximum interval allowed by proposed SR 3.0.2 (30 months) do not invalidate any assumptions in the plant licensing basis.

LD.3 The Frequency for performing CTS 4.8.D.3.a, 4.8.D.3.b, 4.8.D.3.c, 4.8.D.5.a, and 4.8.D.5.d has been extended from 18 months to 24 months in ITS 5.5.7. These requirements ensure that in-place Control Room Emergency Ventilation System charcoal adsorbers, HEPA filters, and heaters are capable of performing their safety function. The proposed change will allow these Surveillances to extend their Surveillance Frequency from the current 18 month Surveillance Frequency (i.e., a maximum of 22.5 months accounting for the allowable grace period specified in CTS 4.0.B and proposed SR 3.0.2) to a 24 month Surveillance Frequency (i.e., a maximum of 30 months accounting for the allowable grace period specified in CTS 4.0.B and proposed SR 3.0.2). This proposed change was evaluated in accordance with the guidance provided in

TECHNICAL CHANGES - LESS RESTRICTIVE

LD.3 (cont'd) 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 maintenance and surveillance data 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. UFSAR Table 1.8-1 (Conformance with Division I NRC Regulatory Guides) identifies that charcoal adsorber and HEPA filter in-place tests are in accordance with Regulatory Guide 1.52, which states that testing Frequencies be every 18 months. The Control Room Emergency Ventilation (CREV) System provides filtration for control room air intake and recirculated air during a high radiation accident and maintains a positive pressure in the control room during control room isolation. Regulatory positions C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, require CREV System filters and charcoal adsorbers be in-place tested (1) initially, (2) at least once per 18 months thereafter, and (3) following painting, fire, or chemical release in any ventilation zone communicating with the system. Position C.5.d also states that carbon adsorbers should be in-place tested following removal of an adsorber sample for laboratory testing if the integrity of the adsorber section is affected. ITS 5.5.7 also requires in-place filter and charcoal adsorber testing and filter pressure drop testing after any structural maintenance on the HEPA filter or charcoal adsorber housings or following painting, fire, or chemical release in any ventilation zone communicating with the CREV System. By testing after maintenance, fire, chemical release, painting, HEPA replacement, or charcoal replacement, potential changes in HEPA filter efficiency, carbon adsorber bypass leakage, and filter pressure drop will be detected that would be detected by conducting the 18 month surveillance tests. The CREV System is normally in standby. Based on the fact that the CREV System is normally in standby and additional testing will be performed if potential degradation occurs and the system design, it is shown that the impact, if any, on system availability is minimal as a result of this change.

The review of historical maintenance and surveillance data also demonstrates that there are no failures that would invalidate the conclusion that the impact on system availability, if any, is minimal from a change to CTS 4.8.D.3.a, 4.8.D.3.b, 4.8.D.3.c, 4.8.D.5.a, and 4.8.D.5.d as implemented in ITS 5.5.7, 5.5.7.a, 5.5.7.b, 5.5.7.c, 5.5.7.d, and 5.5.7.e. In addition, the proposed 24 month Surveillance Frequencies, if performed at the maximum interval allowed by proposed SR 3.0.2 (30 months) do not invalidate any assumptions in the plant licensing basis.

TECHNICAL CHANGES - LESS RESTRICTIVE (continued)

"Specific"

L.1 CTS 4.9.A.5.b requires verifying new fuel oil meets the ASTM standard for API gravity. Proposed ITS 5.5.9.a.1 allows new fuel oil to meet either API gravity or absolute specific gravity. This is acceptable since both methods are considered appropriate in determining the qualifications of the new fuel.

CTS 4.9.A.5.b requires verifying new fuel oil meets the ASTM standards for water and sediment and the visual test for free water and particulate concentration. Proposed ITS 5.5.9.a.3 allows the performance of a clear and bright appearance test with proper color or a water and sediment test. The allowance to perform a water and sediment test, in lieu of the clear and bright test, is necessary since Quad Cities receives dyed fuel and the performance of a visual test in accordance with ASTM D4176 (as specified in the CTS Bases) is not considered appropriate for dyed fuel not meeting the color requirements of ASTM D4176. However, the water and sediment test is considered an appropriate test when using dyed fuel since the actual water and sediment content is determined in accordance with ASTM D1796 as specified in the CTS and proposed ITS 3.8.3 Bases.

CTS 4.9.A.5.b requires sampling and verification that new fuel oil meets ASTM standards for "water and sediment" prior to addition to the fuel oil storage tanks. Proposed ITS 5.5.9.b relaxes these requirements for new fuel by allowing "water and sediment" analyses of the stored fuel (for fuel oil that meets the color requirements of ASTM D4176) to be performed within 31 days after the addition of any new fuel oil.

CTS 4.9.A.6.b requires sampling of stored fuel oil is required every 31 days to verify particulate contaminants < 10 mg/liter, and "water and sediment" and "kinematic viscosity" within ASTM limits. Proposed ITS 5.5.9.c relaxes the requirements for bulk stored fuel oil by not including the 31 day requirement to verify "water and sediment" and "kinematic viscosity" and providing a limit for particulate contaminants of ≤ 10 mg/liter in lieu of < 10 mg/liter.

These changes are acceptable because the purpose of the fuel oil analyses is to ensure proper fuel oil quality is maintained to support the operation of the emergency DGs. The proposed "new" fuel oil requirements in ITS 5.5.9.a (prior to addition to the storage tanks) ensure the fuel oil is of the appropriate grade (API gravity or absolute specific gravity, kinematic viscosity, flash point, and appearance or water and sediment content) and that it may be added to the

TECHNICAL CHANGES - LESS RESTRICTIVE

L.1 (cont'd) stored fuel without concern for contaminating the entire stored fuel volume such that it would have an immediate detrimental impact on diesel engine combustion. The subsequent sampling of ITS 5.5.9.b (31 days after new fuel oil addition) and the normal 31 day sampling frequency of ITS 5.5.9.c evaluate properties that would not have an immediate effect on the DG operation and are typically associated with contamination or fuel oil degradation as a result of long term storage. A failure to satisfy these criteria does not mean the fuel oil will not burn properly in the DG and is reflected in the allowed outage time when outside the allowable limits. The limit of $\leq$ 10 mg/liter for particulate contaminants reflects the limit specified in ASTM standards. These changes have no impact on the safe operation of the plant and are consistent with RG 1.137, Rev. 1, and the ASTM standards.

RELOCATED SPECIFICATIONS

None

5.6 **6.9 REPORTING REQUIREMENTS**

In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following identified reports shall be submitted ~~to the Regional Administrator of the appropriate Regional Office of the NRC unless otherwise noted.~~ in accordance with 10 CFR 50.4 A.2

6.9.A. ~~Routine Reports~~ A.3

~~1. Deleted~~

2. ~~Annual Report~~

5.6.1 Annual reports covering the activities of the Unit for the previous calendar year, as described in this section shall be submitted prior to May 1 of each year.

Add proposed ITS 5.6.1 Note A.4

~~The reports required shall include:~~ A.3

5.6.1 a. Tabulation of the number of station, utility, and other personnel (including contractors) receiving exposures greater than 100 mrem/year and their associated person rem exposure according to work and job functions, e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignments to various duty functions may be estimated based on electronic or pocket dosimeter or TLD. Small exposures totaling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources should be assigned to specific major work functions.

A.5

A.6

~~b. The results of specific activity analysis in which the reactor coolant exceeded the limits of Specification 3.6.J. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded; (2) results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than the limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up system flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the reactor coolant exceeded the radioiodine limit.~~

A.4

5.6.2 3. Annual Radiological Environmental Operating Report Add proposed ITS 5.6.2 Note

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

A.4 Add proposed ITS 5.6.3 Note

A.7 in accordance with 10 CFR 50.36a

5.6.3 **4. Radioactive Effluent Release Report**

The Radioactive Effluent Release Report covering the operation of the facility during the previous calendar year shall be submitted ~~prior to April 1 of each year~~. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the facility. The material provided shall be (1) consistent with the objectives outlined in the ODCM and PCP and (2) in conformance with 10 CFR 50.36a and Section IV.B.1 of Appendix I to 10 CFR Part 50.

prior to May 1 of each year L.1

5.6.4 **5. Monthly Operating Report**

Routine reports of operating statistics and shutdown experience, including documentation of all challenges to safety valves or safety/relief valves, shall be submitted on a monthly basis ~~to the Director, Office of Resource Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Administrator of the NRC Regional Office,~~ no later than the 15th of each month following the calendar month covered by the report. A.2

5.6.5 **6. CORE OPERATING LIMITS REPORT**

5.6.5.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:

5.6.5.a.5 (1) The Control Rod Withdrawal Block Instrumentation for Table 3.2.E-1 of Specification 3.2.E.

5.6.5.a.1 (2) The Average Planar Linear Heat Generation Rate (APLHGR) Limit for Specification 3.11.A.

5.6.5.a.3 (3) The Linear Heat Generation Rate (LHGR) for Specification 3.11.D.

5.6.5.a.2 (4) The Minimum Critical Power Operating Limit ~~(including scram insertion time)~~ for Specification 3.11.C. ~~This includes rated and off-rated flow conditions.~~ LA.1

5.6.5.b b. The analytical methods used to determine the operating limits shall be those previously reviewed and approved by the NRC in the latest approved revision or supplement of topical reports:

5.6.5.b.1 (1) NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," (latest approved revision).

5.6.5.b.2 (2) Commonwealth Edison Topical Report NFSR-0085, "Benchmark of BWR Nuclear Design Methods," (latest approved revision).

5.6.5.a.4 The LHGR and transient linear heat generation rate limit for 3.2.4 A.9

5.6.5.b.3 (3) Commonwealth Edison Topical Report NFSR-0085, Supplement 1, "Benchmark of BWR Nuclear Design Methods - Quad Cities Gamma Scan Comparisons," (latest approved revision).

5.6.5.b.4 (4) Commonwealth Edison Topical Report NFSR-0085, Supplement 2, "Benchmark of BWR Nuclear Design Methods - Neutronic Licensing Analyses," (latest approved revision).

5.6.5.b.5 (5) Advanced Nuclear Fuels Methodology for Boiling Water Reactors, XN-NF-80-19(P)(A), Volume 1, Supplement 3, Supplement 3 Appendix F, and Supplement 4, Advanced Nuclear Fuels Corporation, November 1990.

5.6.5.b.6 (6) Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads, XN-NF-80-19(P)(A), Volume 4, Revision 1, Exxon Nuclear Company, June 1986.

5.6.5.b.7 (7) Exxon Nuclear Methodology for Boiling Water Reactors THERMEX: Thermal Limits Methodology Summary Description, XN-NF-80-19(P)(A), Volume 3, Revision 2, Exxon Nuclear Company, January 1987.

5.6.5.b.8 (8) Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis, XN-NF-80-19(P)(A), Volume 1 and Supplements 1 and 2, Exxon Nuclear Company, March 1983.

5.6.5.b.9 (9) Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel, XN-NF-85-67(P)(A) Revision 1, Exxon Nuclear Company, September 1986.

5.6.5.b.10 (10) Qualification of Exxon Nuclear Fuel for Extended Burnup Supplement 1: Extended Burnup Qualification of ENC 9x9 BWR Fuel, XN-NF-82-06(P)(A) Supplement 1, Revision 2, Advanced Nuclear Fuels Corporation, May 1988.

5.6.5.b.11 (11) Advanced Nuclear Fuels Corporation Generic Mechanical Design for Advanced Nuclear Fuels 9x9-IX and 9x9-9X BWR Reload Fuel, ANF-89-014(P)(A), Revision 1 and Supplements 1 and 2, Advanced Nuclear Fuels Corporation, October 1991.

5.6.5.b.12 (12) Generic Mechanical Design Criteria for BWR Fuel Designs, ANF-89-98(P)(A) Revision 1, and Revision 1 Supplement 1, Advanced Nuclear Fuels Corporation, May 1995.

5.6.5.b.13 (13) Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors, XN-NF-79-71(P)(A), Revision 2 Supplements 1, 2, and 3, Exxon Nuclear Company, March 1986.

5.6.5.b.14 (14) ANFB Critical Power Correlation, ANF-1125(P)(A) and Supplements 1 and 2, Advanced Nuclear Fuels Corporation, April 1990.

5.6.5.b.15 (15) Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors/Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors: Methodology for Analysis of Assembly Channel Bowing Effects/NRC Correspondence, ANF-524(P)(A), Revision 2, Supplement 1 Revision 2, Supplement 2, Advanced Nuclear Fuels Corporation, November 1990.

5.6.5.b.16 (16) COTRANSA 2: A Computer Program for Boiling Water Reactor Transient Analyses, ANF-913(P)(A) Volume 1 Revision 1 and Volume 1 Supplements 2, 3, and 4, Advanced Nuclear Fuels Corporation, August 1990.

5.6.5.b.17 (17) Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR Evaluation Model, ANF-91-048(P)(A), Advanced Nuclear Fuels Corporation, January 1993.

5.6.5.b.18 (18) Commonwealth Edison Topical Report NFSR-0091, "Benchmark of CASMO/MICROBURN BWR Nuclear Design Methods," Revision 0, Supplements 1 and 2, December 1991, March 1992, and May 1992, respectively; SER letter dated March 22, 1993.

5.6.5.b.19 (19) ANFB Critical Power Correlation Application for Coresident Fuel, EMF-1125(P)(A), Supplement 1, Appendix C, Siemens Power Corporation, August 1997.

5.6.5.b.20 (20) ANFB Critical Power Correlation Determination of ATRIUM-9B Additive Constant Uncertainties, ANF-1125(P)(A), Supplement 1, Appendix E, Siemens Power Corporation, September 1998.

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

5.6.5.d The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC ~~Document Control Desk with copies to the Regional Administrator and Resident Inspector~~. A.2

6.9.B ~~Special Reports~~ A.3

~~Special reports shall be submitted~~ ~~to the Regional Administrator of the NRC Regional Office~~ ~~within the time period specified for each report.~~ A.2 A.8

TABLE 3.2.F-1 (Continued)

ACCIDENT MONITORING INSTRUMENTATION

ACTION

ACTION 60 - a. With the number of OPERABLE accident monitoring instrumentation CHANNEL(s) less than the Required CHANNEL(s) shown in Table 3.2.F-1, restore the inoperable CHANNEL(s) to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.

b. With the number of OPERABLE accident monitoring instrumentation CHANNEL(s) less than the Minimum CHANNEL(s) shown in Table 3.2.F-1, restore the inoperable CHANNEL(s) to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

ACTION 61- With the number of OPERABLE accident monitoring instrumentation CHANNEL(s) less than the Minimum CHANNEL(s) shown in Table 3.2.F-1, initiate the preplanned alternate method of monitoring the appropriate parameter(s) within 72 hours, and:

a. Either restore the inoperable CHANNEL(s) to OPERABLE status within 7 days of the event, or

5.6.6 b. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.B ~~within 30 days following the event~~ outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status. M.1

ACTION 62- a. With the number of OPERABLE accident monitoring instrumentation CHANNEL(s) one less than the Required CHANNEL(s) shown in Table 3.2.F-1, restore the inoperable CHANNEL(s) to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.

b. With the number of OPERABLE accident monitoring instrumentation CHANNEL(s) less than the Minimum CHANNEL(s) shown in Table 3.2.F-1; restore at least one inoperable CHANNEL to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

<See ITS 3.3.3.1>

ADMINISTRATIVE

A.1 In the conversion of the Quad Cities 1 and 2 current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted that do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the BWR Standard Technical Specifications, NUREG 1433, Rev. 1 (i.e., the Improved Standard Technical Specifications (ISTS)).

A.2 Submittal details for reports required by CTS 6.9 (Reporting Requirements), CTS 6.9.A.5 (Monthly Operating Report), CTS 6.9.A.6.c (Core Operating Limits Report) and CTS 6.9.B (Special Reports) are being deleted. Proposed ITS 5.6 requires submittal of reports in accordance with 10 CFR 50.4, which identifies these requirements. This change is a presentation preference consistent with the BWR ISTS, NUREG-1433, Rev. 1, and with current NRC regulations (10 CFR 50.4) and is considered administrative.

A.3 ITS 5.6, "Reporting Requirements," does not use the current Technical Specification subtitles of "Routine Reports," "Annual Reports," or "Special Reports." The ITS names each individual report rather than grouping reports under subtitles. This change does not change reporting requirements and only affects the format of the Technical Specifications. Therefore, this change is considered to be administrative.

A.4 Proposed Notes for ITS 5.6.1, 5.6.2, and 5.6.3 allowing a single report submittal to satisfy the associated reporting requirement for both units is added to CTS 6.9.A.2.a, CTS 6.9.A.3, and CTS 6.9.A.4. This change provides clarification but does not change the regulatory reporting requirement; therefore, the change is considered administrative.

A.5 Another name for a new type of pocket dosimeter currently in use at Quad Cities 1 and 2 to estimate the whole body doses required to be reported in CTS 6.9.A.2.a, electronic dosimeter, has been added in ITS 5.6.1. This is considered administrative since the measurement tools described are accepted in the industry.

A.6 CTS 6.9.A.2.b requires reporting the results of specific activity analysis in which the primary coolant exceeded CTS 3.6.J limits. This reporting requirement is unnecessary since it is included in the LER requirements to report fuel cladding failures that exceed expected values or that are caused by unexpected factors, i.e., being seriously degraded. Since the criteria identified in 10 CFR 50.73 have been identified as the criteria in the area of degraded boundaries that

ADMINISTRATIVE

A.6 (cont'd) necessitates reporting, any minor differences are negligible with regard to safety. Therefore, the current reporting requirement of CTS 6.9.A.2.b is a duplication of the 10 CFR 50.73 reporting requirement and can be deleted.

A.7 CTS 6.9.A.4 requires submittal of the radioactive effluent release report "prior to April 1 of each year." Proposed ITS 5.6.3 also requires the submittal to be "in accordance with 10 CFR 50.36a." Compliance with 10 CFR 50 requirements is required by the Quad Cities 1 and 2 Operating Licenses. Therefore this change is considered to be administrative in nature.

A.8 The general statement in CTS 6.9.B to submit special reports within the time period specified for each report is not retained in the ITS. Each special report contains requirements for submittal. This change merely deletes duplicate requirements in the Technical Specifications or in regulations and is thus considered to be administrative in nature.

A.9 CTS 6.9.A.6, CORE OPERATING LIMITS REPORT, does not include reference to the LHGR limit and the transient linear heat generation rate limit of CTS 3.11.B, Transient Linear Heat Generation Rate. The requirements have been included in ITS 5.6.5.a.4. These changes are consistent with current practice (the limits are currently specified in the COLR), therefore this change is considered administrative.

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 CTS Table 3.2.F-1 Action 61.b requires a special report to be submitted within 30 days after a Drywell Radiation Monitor is inoperable, which is 23 days after the restoration time provided in CTS Table 3.2.F-1 Action 61.a has expired. ITS 5.6.6 will require the report within 14 days after the restoration time provided in ITS 3.3.3.1 ACTIONS has expired. This change is more restrictive on plant operations and is being made to be consistent with the BWR ISTS, NUREG-1433, Rev. 1.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 CTS 6.9.A.6.a (4) provides the detail associated with the MCPR Specification, which is addressed in the Core Operating Limits Report. This detail is to be relocated to the Bases of the individual Specification, i.e., B 3.2.2, MINIMUM

TECHNICAL CHANGES - LESS RESTRICTIVE

LA.1 (cont'd) CRITICAL POWER RATIO. The requirements of ITS 5.6.5 (Core Operating Limits Report) and LCO 3.2.2 are adequate to ensure the required limits are maintained. In addition, the requirements of ITS 5.6.5 provide regulatory controls over the detail to be relocated. As a result, the requirement proposed to be relocated is not required to be included in the ITS to provide adequate protection of the public health and safety. Additionally, changes to the Bases will be controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the ITS.

"Specific"

L.1 This change proposes to relax the CTS 6.9.A.3 and 6.9.A.4 requirements for submitting the Annual Radiological Environmental Operating Report and Radioactive Effluent Release Report. CTS 6.9.A.3 requires the Annual Radiological Environmental Operating Report to be submitted prior to May 1 of each year. This proposed change will allow the Annual Radiological Environmental Operating Report to be submitted by May 15 of each year. CTS 6.9.A.4 requires the Radioactive Effluent Release Report to be submitted prior to April 1 of each year. This proposed change will allow the Radioactive Effluent Release Report to be submitted prior to May 1 of each year. Given that the reports are still required to be provided to the NRC on or before May 15, for the Annual Radiological Environmental Operating Report, and May 1, for the Radioactive Effluent Release Report, and cover the previous calendar year, completion and submittal of these reports is clearly not necessary to assure operation in a safe manner. Additionally, there is no requirement for the NRC to approve the reports. Therefore, this change has no impact on the safe operation of the plant.

RELOCATED SPECIFICATIONS

None

$\boxed{A.1}$

5.7 6.12 HIGH RADIATION AREA

5.7.1 6.12.A Pursuant to 10 CFR 20.1601(c), in lieu of the requirements of paragraph 20.1601 of 10 CFR Part 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr at 30 cm (12 in.) shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP)[a] (or equivalent document). Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. 1. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. 2. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel have been made knowledgeable of them; or

c. 3. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified in the RWP (or equivalent document).

5.7.2 6.12.B In addition to the requirements of 6.12.A, above, areas accessible to personnel with radiation levels greater than 1000 mrem/hr at 30 cm (12 in.) from the radiation source or from any surface which the radiation penetrates shall require the following:

a. 1. Doors shall be locked to prevent unauthorized entry and shall not prevent individuals from leaving the area. In place of locking the door, direct or electronic surveillance that is capable of preventing unauthorized entry may be used. The keys shall be maintained under the administrative control of the Shift Engineer on duty ~~and/or health physics~~ supervision.

radiation protection $\boxed{A.2}$

b 2. Personnel access and exposure control requirements of activities being performed within these areas shall be specified by an approved RWP (or equivalent document).

Individuals qualified in radiation protection procedures $\boxed{A.2}$

such individuals

5.7.1 a Health Physics personnel or personnel escorted by health physics personnel shall be exempt from the RWP issuance requirements during the performance of their assigned radiation protection duties, provided they are otherwise following plant radiation protection procedures for entry into high radiation areas.

ADMINISTRATIVE CONTROLS A.1

c ~~3~~. Each person entering the area shall be provided with an alarming radiation monitoring device that continuously integrates the radiation dose rate (such as an electronic dosimeter.) Surveillance and radiation monitoring by a ~~R~~adiation ~~P~~rotection ~~T~~echnician A.2 may be substituted for an alarming dosimeter.

4. ~~Deleted.~~

5.7.3 5. For individual HIGH RADIATION AREAS accessible to personnel with radiation levels of greater than 1000 mrem/h at 30 cm (12 in.) that are located within large areas where no enclosure exists for purposes of locking, and where no enclosure can be reasonably constructed around the individual areas, then such individual areas shall be barricaded, conspicuously posted, and a flashing light shall be activated as a warning device.

DISCUSSION OF CHANGES
ITS: 5.7 - HIGH RADIATION AREA

ADMINISTRATIVE

A.1 In the conversion of the Quad Cities 1 and 2 current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted that do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the BWR Standard Technical Specifications, NUREG 1433, Rev. 1 (i.e., the Improved Standard Technical Specifications (ISTS)).

A.2 The title of the individual qualified to implement radiation protection procedures in CTS 6.12.B.3 has been changed from the specific title "Radiation Protection Technician" to just describe the generic function; radiation protection technician. Since the only individuals currently qualified are radiation protection technicians, this change is considered administrative. If other individuals are qualified in the future, they will meet the same qualifications. In addition, the term "health physics" in CTS 6.12.B.1 and CTS 6.12.A footnote a has been changed to radiation protection to be consistent. Therefore, these changes are considered administrative.

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

None

"Specific"

None

RELOCATED SPECIFICATIONS

None

CTS 6.4

Training 6.4

LA.1

6.4 TRAINING

A retraining and replacement program for the unit staff shall be maintained under the direction of the appropriate on site manager. Training shall be in accordance with ANSI N18.1-1971 and 10 CFR 55 for appropriate designated positions and shall include familiarization with relevant industry operational experience.

ADMINISTRATIVE

None

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The details contained in CTS 6.4 on training and replacement training for the unit staff are proposed to be relocated to the UFSAR. 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 unit staff requirements. ITS 5.2.2.a, 5.2.2.b, and 10 CFR 50.54 state minimum shift crew requirements. Training and requalification of licensed positions is contained in 10 CFR 50.55. Placement of training requirements in the UFSAR will ensure that training programs are properly maintained in accordance with Quad Cities 1 and 2 commitments and regulations. As such, the relocated details are not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the UFSAR are controlled by the provisions of 10 CFR 50.59 to ensure adequate reviews are performed.

"Specific"

None

RELOCATED SPECIFICATIONS

None

CTS 6.7

A.1

6.7 SAFETY LIMIT VIOLATION

6.7.A The following actions shall be taken in the event a Safety Limit is violated:

1. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within 1 hour. The Site Vice-President or his designated alternate shall be notified within 24 hours.

LA.1

2. Within 30 days, a Licensee Event Report (LER) shall be prepared documenting the event pursuant to 10 CFR 50.73. The LER shall be submitted to the NRC.

3. Critical operation of the Unit shall not be resumed until authorized by the Commission.

A.1

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DISCUSSION OF CHANGES
CTS: 6.7 - SAFETY LIMIT VIOLATION

ADMINISTRATIVE

A.1 The current Safety Limit Violation requirements of CTS 6.7, as they relate to NRC notification (portions of CTS 6.7.A.1 and 6.7A.2) and permission to restart the unit (CTS 6.7.A.3) are contained in and based upon the requirements located in 10 CFR 50.36(c)(1), 10 CFR 50.72, and 10 CFR 50.73. Since Quad Cities 1 and 2 are required by the Operating Licenses to comply with 10 CFR 50, the removal of these requirements from Technical Specifications is considered administrative.

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The CTS 6.7.A.1 requirement for notification of the Site-Vice President or his designated alternate in the event of a Safety Limit violation is proposed to be relocated to the Quality Assurance (QA) Manual. Given that the notification occurs following the Safety Limit violation, the proposed relocated requirement is clearly not necessary to assure operation of the unit in a safe manner. Additionally, in the event of a Safety Limit violation, 10 CFR 50.36(c)(1) does not allow operation of the unit to be resumed until authorization is received from the NRC. As such, the relocated requirement is not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the QA Manual are controlled by the provisions of 10 CFR 50.54.

"Specific"

None

RELOCATED SPECIFICATIONS

None

CTS 6.11

ADMINISTRATIVE CONTROLS

LA.1

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

Page 1 of 1

DISCUSSION OF CHANGES
CTS: 6.11 - RADIATION PROTECTION PROGRAM

ADMINISTRATIVE

None

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The details contained in CTS 6.11 are proposed to be relocated to the UFSAR. This relocated program requires procedures to be prepared for personnel radiation protection consistent with 10 CFR 20. These procedures are for nuclear plant personnel and have no impact on nuclear safety or the health and safety of the public. Requirements to have procedures to implement 10 CFR 20 are contained in 10 CFR 20.1101(b). Periodic review of these procedures is addressed in 10 CFR 20.1101(c). Since the CTS requirements are contained in the regulations and the Quad Cities 1 and 2 Operating Licenses require compliance with 10 CFR 20, there is no need to repeat the requirements in the ITS. As such, the relocated details are not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the UFSAR are controlled by the provisions of 10 CFR 50.59.

"Specific"

None

RELOCATED SPECIFICATIONS

None

6.13 PROCESS CONTROL PROGRAM (PCP)

6.13.A Changes to the PCP:

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

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

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

2. Shall become effective after review and acceptance, including approval by the Station Manager.

LA.1

CTS 6.13

1.0 DEFINITIONS

PRESSURE BOUNDARY LEAKAGE

PRESSURE BOUNDARY LEAKAGE shall be leakage through a non-isolable fault in a reactor coolant system component body, pipe wall or vessel wall.

PRIMARY CONTAINMENT INTEGRITY (PCI)

PRIMARY CONTAINMENT INTEGRITY (PCI) shall exist when:

a. All primary containment penetrations required to be closed during accident conditions are either:

1) Capable of being closed by an OPERABLE primary containment automatic isolation valve system, or

2) Closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position, except for valves that are open under administrative control as permitted by Specification 3.7.D.

b. All primary containment equipment hatches are closed and sealed.

c. Each primary containment air lock is in compliance with the requirements of Specification 3.7.C.

d. The primary containment leakage rates are maintained within the limits of Specification 3.7.A.

e. The suppression chamber is in compliance with the requirements of Specification 3.7.K.

f. The sealing mechanism associated with each primary containment penetration; e.g., welds, bellows or O-rings, is OPERABLE.

~~PROCESS CONTROL PROGRAM (PCP)~~

~~The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analysis, test, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71, State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.~~

LA.1

RATED THERMAL POWER (RTP)

RATED THERMAL POWER (RTP) shall be a total reactor core heat transfer rate to the reactor coolant of 2511 MWT.

<See ITS 1.0>

Page 2 of 2

ADMINISTRATIVE

None

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The details contained in CTS 6.13 and the definition of PROCESS CONTROL PROGRAM are proposed to be relocated to the UFSAR. The PCP implements the requirements of 10 CFR 20, 10 CFR 61, and 10 CFR 71. Compliance with these regulations is required by the Quad Cities 1 and 2 Operating Licenses, and as such, relocation of the description of the PCP from the ITS does not affect the safe operation of the facility. Therefore, the relocated details are not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the UFSAR are controlled by the provisions of 10 CFR 50.59.

"Specific"

None

RELOCATED SPECIFICATIONS

None

The following blank pages, have been deleted:

6-6, 6-7, and 6-17.

<CTS>

5.0 ADMINISTRATIVE CONTROLS 2

TSTF-65

5.1 Responsibility

station manager

Reviewer's Note not shown

<6.1.A> 5.1.1 The ~~[Plant Superintendent]~~ shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence. TSTF-65 1

manager

2 ~~The [Plant Superintendent] or his designee shall approve, prior to implementation, each proposed test, experiment or modification to systems or equipment that affect nuclear safety.~~ 3

A Senior Reactor Operator (SRO)

<6.1.B><LA.2> 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 or defueled, an individual with an active SRO license or Reactor Operator license shall be designated to assume the control room command function. 3

or defueled

both units are

while either unit is in MODE 1, 2, or 3

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 the TSTF-65 allowance. This is not meant to be retained in the final version of the plant specific submittal.

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

3. The second paragraph of ISTS 5.1.1, regarding review and approval of tests or experiments is deleted. CTS do not delineate this requirement. ISTS 5.1.2 is revised to reflect plant practice. The Shift Manager is responsible for directing the control room command function but is not necessarily in the control room. An SRO is in the control room and has the control room command function, when either unit is in MODE 1, 2, or 3.

<CTS>

5.0 ADMINISTRATIVE CONTROLS

5.2 Organization

5.2.1 Onsite and Offsite Organizations

<6.2.A>

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

<6.2.A.1> TSTF-65 2 Insert 5.2.1.a

a. Lines of authority, responsibility, and communication shall be defined and established throughout highest management levels, intermediate levels, and all operating organization positions. These relationships shall be documented and updated, as appropriate, in organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented in the ~~[FSAR]~~ 2 ~~QA Plan~~ Quality Assurance Manual 1

<6.2.A.2> 1 station manager TSTF-65

b. The ~~[Plant~~ ~~Superintendent]~~ shall be responsible for overall safe operation of the plant and shall have control over those onsite activities necessary for safe operation and maintenance of the plant~~;~~ . 2

<6.2.A.3> officer

c. ~~The [~~ ~~specified~~ corporate ~~executive position]~~ shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety~~; and~~ . 2

3 radiation protection <6.2.A.4> or perform 4

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.

5.2.2 Unit Staff

The unit staff organization shall include the following:

<6.2.B.1> 4 5

a. ~~A non-licensed operator shall be assigned to each reactor containing fuel and an additional non-licensed operator~~ Insert 5.2.2.a

(continued)

2 TSTF-65

INSERT 5.2.1.a

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

5 4

INSERT 5.2.2.a

5 A total of three non-licensed operators for the two units is required in all conditions. At least one of the required non-licensed operators shall be 4 assigned to each unit.

1

<CTS>

5.2 Organization

5.2.2 Unit Staff (continued)

5

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

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

8

<6.2.B.2>

TSTF -258 changes not adopted

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

6

<6.2.B.3>

c. Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i) and Specifications [5.2.2.a and 5.2.2.g] 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.

1

radiation protection

<6.2.B.4>

d. A ~~[Health Physics]~~ Technician[s] 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. TSTF-65

<6.2.B.4>

1

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

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

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

(continued)

<CTS>

5.2 Organization

5.2.2 Unit Staff (continued) 1

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

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

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

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

TSTF-65

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

OR

<6.2.B.4>

8

TSTF-258 changes not adopted

The amount of overtime worked by unit staff members performing safety related functions shall be limited and controlled in accordance with the NRC Policy Statement on working hours (Generic Letter 82-12). 1

<6.2.B.6> 8

f. The [O]perations [M]anager or [Assistant] shift [O]perations [Manager] supervisor shall hold an SRO license. 1 TSTF-65

TSTF-258 changes not adopted

<6.2.C>

g. The Shift Technical Advisor (STA) shall provide advisory technical support to the ~~Shift Supervisor (SS)~~ shift manager in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. In addition, the STA shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift. 7

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS: 5.2 - ORGANIZATION

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

2. Typographical/grammatical error corrected.

3. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, analysis description, or licensing basis description.

4. Editorial changes made for enhanced clarity.

5. Changes have been made to ISTS 5.2.2.a to be consistent with current licensing basis.

6. The referenced requirements are Specifications, not CFR requirements. Therefore, the word "Specifications" has been added to clearly state that "5.5.2.a and 5.2.2.g" are Specifications.

7. The proper plant specific description of the individual to whom the STA provides technical support has been provided.

8. ISTS 5.2 (Organization) is revised by TSTF-258, Rev. 4. In order to maintain consistency, to the maximum extent practicable, between the Administrative Controls Technical Specifications of the ComEd nuclear stations, the following changes of TSTF-258, Rev. 4, are not incorporated in ITS 5.2:

    a. ISTS 5.2.2.b contains shift manning requirements that duplicate requirements of 10 CFR 50.54(m)(2)(iii) and 10 CFR 50.54(k). As a result, ISTS 5.2.2.b was deleted by TSTF-258, Rev. 4.

    b. ISTS 5.2.2.e contains requirements for control of overtime of the plant staff. These requirements were revised by TSTF-258, Rev. 4.

    c. ISTS 5.2.2.g contains requirements for the Shift Technical Advisor. The title "Shift Technical Advisor (STA)" was deleted by TSTF-258, Rev. 4.

    Not incorporating these changes to ISTS 5.2 is consistent with the NRC approved ITS for the ComEd Byron and Braidwood Stations.

<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

<6.3> 5.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ~~[Regulatory Guide 1.8, Revision 2, 1987, or more recent revisions, or ANSI Standard acceptable to the NRC staff]. The staff not covered by [Regulatory Guide 1.8] shall meet or exceed the minimum qualifications of [Regulations, Regulatory Guides, or ANSI Standards acceptable to NRC staff].~~ 3 TSTF-258 changes not adopted

2 ANSI N18.1-1971, except for the radiation protection manager or lead radiation protection technician, who shall meet or exceed the qualifications for "Radiation Protection Manager" in Regulatory Guide 1.8, September 1975.

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

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

3. ISTS 5.3 (Unit Staff Qualifications) is revised by TSTF-258, Rev. 4. In order to maintain consistent, to the maximum extent practicable, between the Administrative Controls Technical Specifications of the ComEd nuclear stations, the following change of TSTF-258, Rev. 4, is not incorporated in ITS 5.3:

    ISTS 5.3.2 was added to define licensed Senior Reactor Operators and licensed Reactor Operators for the purpose of 10 CFR 55.4.

    Not incorporating this change to ISTS 5.3 is consistent with the NRC approved ITS for the ComEd Byron and Braidwood Stations.

<CTS>

5.0 ADMINISTRATIVE CONTROLS

5.4 Procedures

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

<6.8.A.1>

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

<6.8.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, Section 7.1 [1] [2]

~~c. Quality assurance for effluent and environmental monitoring;~~ [3]

<6.8.A.7>

c ~~d.~~ Fire Protection Program implementation; and

<DOC M.1>

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

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS: 5.4 - PROCEDURES

1. Typographical/grammatical error corrected.

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

3. ISTS 5.4.1.c is deleted and subsequent items renumbered. This change is consistent with the current licensing basis, which does not require these procedures to be controlled by Technical Specifications.

<CTS>

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

The following programs shall be established, implemented and maintained.

5.5.1 Offsite Dose Calculation Manual (ODCM)

<Def-ODCM>

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

<Def-ODCM>

b. The ODCM shall also contain the radioactive effluent controls and radiological environmental monitoring activities (1) 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]. 2

<6.14.A>

c. Licensee initiated changes to the ODCM: 3

<6.14.A.1>

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

<6.14.A.1.a>

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

<6.14.A.1.b>

(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 (do) not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations; 1

TSTF-76

<6.14.A.2>

2 ~~b.~~ Shall become effective after ~~review and acceptance by the [onsite review function] and~~ the approval of the ~~[Plant Superintendent]~~ station manager; and 2

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<6.14.A.3>

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

station 2

(continued)

L <CTS>

5.5 Programs and Manuals

5.5.1 Offsite Dose Calculation Manual (ODCM) (continued)

<6.14.A.3>

that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

5.5.2 Primary Coolant Sources Outside Containment

<6.8.D.1>

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, containment monitoring, and Standby Gas Treatment, Reactor Water Cleanup. The program shall include the following: 2

a. Preventive maintenance and periodic visual inspection requirements; and

b. Integrated leak test requirements for each system at 24 month ~~refueling cycle~~ intervals ~~or less~~. The provisions of SR 3.0.2 are applicable to the 24 month Frequency for performing integrated system leak test activities. 4

5.5.3 Post Accident Sampling

<6.8.D.3>

This program provides controls that ensure the capability to obtain and analyze reactor coolant, radioactive gases, iodines, and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the following: 5 1

a. Training of personnel;

b. Procedures for sampling and analysis; and

c. Provisions for maintenance of sampling and analysis equipment.

5.5.4 Radioactive Effluent Controls Program

<6.8.D.4>

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably

(continued)

<CTS>

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

<6.8.D.4> 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.4.a 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;

6.8.D.4.b b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to ten times the concentration values in ~~10 CFR 20~~ Appendix B, Table 2, Column 2 to 10 CFR 20.1001 - 20.2402;

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6

<6.8.D.4.c> c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM;

<6.8.D.4.d> 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;

<6.8.D.4.e> e. Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days;

<6.8.D.4.f> 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;

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<6.8.D.4.g> 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 ~~conforming to the dose associated with 10 CFR 20, Appendix B, Table 2, Column 1~~; shall be in accordance with the following:

6

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

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

(continued)

<CTS>

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

<6.8.D.4.h>

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;

<6.8.D.4.i>

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

6 TSTF-258

, beyond the site boundary,

<6.8.D.4.j>

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

7

~~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).~~

5.5.5 Component Cyclic or Transient Limit

<DOC M.2>

This program provides controls to track the ~~FSAR Section [ ], [ ],~~ U 3.9 cyclic and transient occurrences to ensure that components are maintained within the design limits.

8 2 9

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

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

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

(continued)

<DOC A.2> The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluents Control Program Surveillance Frequencies.

6 TSTF-256

ε <CTS>

5.5 Programs and Manuals (continued)

5.5.~~7~~ 6 9

<6.8.E>

Inservice Testing Program

pumps and valves 20

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 ~~components~~ ~~including applicable supports~~. ~~The program shall include the following:~~ 20 TSTF-279

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

<6.8.E.2>

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice testing activities	Required Frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days
Every 48 months	At least once per 1461 days

22

<6.8.E.3> b. The provisions of SR 3.0.2 are applicable to the above required Frequencies for performing inservice testing activities;

<DOC A.2> c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

<6.8.E.5> d. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any TS.

<4.7.P.2> 5.5.~~8~~ 7 9

Ventilation Filter Testing Program (VFTP)

<4.7.P.3>
<4.7.P.4>
<4.7.P.5>
<4.7.P.6>
<4.8.D.3>
<4.8.D.4>
<4.8.D.5>
<4.8.D.6>
<4.8.D.7>

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

Insert 5.5.7.f from page 5.0-13

(continued)

21 INSERT 5.5.7

Tests described in Specification 5.5.7.a and 5.5.7.b shall be performed once per 24 months; after each complete or partial replacement of the HEPA filter bank or charcoal adsorber bank; after any structural maintenance on the HEPA filter bank or charcoal adsorber bank housing; and, following significant painting, fire, or chemical release in any ventilation zone communicating with the subsystem while it is in operation.

Tests described in Specification 5.5.7.c shall be performed once per 24 months; after 1440 hours of adsorber operation for the Standby Gas Treatment System; after 720 hours of adsorber operation for the Control Room Emergency Ventilation System; after any structural maintenance on the charcoal adsorber bank housing; and, following significant painting, fire, or chemical release in any ventilation zone communicating with the subsystem while it is in operation.

Tests described in Specification 5.5.7.d and 5.5.7.e shall be performed once per 24 months.

<CTS>

5.5 Programs and Manuals

5.5.~~8~~ 7 9 Ventilation Filter Testing Program (VFTP) (continued)

<4.7.P.2.a>
<4.7.P.5>
<4.8.D.3.a>
<4.8.D.3.c>
<4.8.D.6>

a. Demonstrate for each of the ESF systems that an inplace test of the HEPA filters shows a penetration and system bypass ~~< [0.05]%~~ specified below when tested in accordance with {Regulatory Guide 1.52, Revision 2, ~~and~~ ANSI/ ASME N510-1989} at the system flowrate specified below ~~[± 10%]~~. 2 1 0

Insert 5.5.7.a

ESF Ventilation System	Flowrate
[ ]	[ ]

2

<4.7.P.2.a>
<4.7.P.6>
<4.8.D.3.a>
<4.8.D.3.c>
<4.8.D.7>

b. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass ~~< [0.05]%~~ specified below when tested in accordance with {Regulatory Guide 1.52, Revision 2, ~~and~~ ANSI/ ASME N510-1989} at the system flowrate specified below ~~[± 10%]~~. 2 1 0

Insert 5.5.7.b

ESF Ventilation System	Flowrate
[ ]	[ ]

2

<4.7.P.2.b>
<4.7.P.3>
<4.8.D.3.b>
<4.8.D.4>

c. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in {Regulatory Guide 1.52, Revision 2}, shows the methyl iodide penetration less than the value specified below when tested in accordance with {ASTM D3803-1989} at a temperature of ≤ {30°C} and ~~greater than or equal to the~~ relative humidity (RH) specified below. 2 11 1

Insert 5.5.7.c

ESF Ventilation System	Penetration	RH
[ ]	[ ]	[ ]

2

(continued)

2 Insert 5.5.7.a

ESF Ventilation System	Penetration	Flowrate
Standby Gas Treatment (SGT) System	< 1.0%	≥ 3600 cfm and ≤ 4400 cfm
Control Room Emergency Ventilation (CREV) System	< 0.05%	≥ 1800 scfm and ≤ 2200 scfm

2 Insert 5.5.7.b

ESF Ventilation System	Penetration	Flowrate
Standby Gas Treatment (SGT) System	< 1.0%	≥ 3600 cfm and ≤ 4400 cfm
Control Room Emergency Ventilation (CREV) System	< 0.05%	≥ 1800 scfm and ≤ 2200 scfm

2 Insert 5.5.7.c

ESF Ventilation System	Penetration	RH
Standby Gas Treatment (SGT) System	2.5%	70%
Control Room Emergency Ventilation (CREV) System	0.5%	70%

<CTS>

5.5 Programs and Manuals

5.5.~~8~~ 7 [9] Ventilation Filter Testing Program (VFTP) (continued)

[12]

~~[Reviewer's Note: Allowable penetration = [100% - methyl iodide efficiency for charcoal credited in staff safety evaluation]/(safety factor).~~

~~Safety factor = [5] for systems with heaters.~~
~~= [7] for systems without heaters.]~~

<4.7.P.4.a>
<4.8.D.5.a>

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

Insert 5.5.7.d

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

[13] , corrected for voltage variations at the 480V bus,

<4.7.P.4.c>
<4.8.D.5.d>

e. Demonstrate that the heaters for each of the ESF systems dissipate the value specified below ~~[± 10%]~~ when tested in accordance with ~~[~~ ANSI/ ASME N510-1989~~]~~: [1] [2]

Insert 5.5.7.e

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

<DOC A.6>

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

[21] move this 5.5.7.f to page 5.0-11 as indicated

5.5.~~9~~ 8 [9] Explosive Gas and Storage Tank Radioactivity Monitoring Program

<DOC M.2>

This program provides controls for potentially explosive gas mixtures contained in the ~~[Waste~~ Off- 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 [2] [14]

(continued)

2 Insert 5.5.7.d

ESF Ventilation System	Delta P	Flowrate
Standby Gas Treatment (SGT) System	< 6 inches water guage	≥ 3600 cfm and ≤ 4400 cfm
Control Room Emergency Ventilation (CREV) System	< 6 inches water guage	≥ 1800 scfm and ≤ 2200 scfm

2 Insert 5.5.7.e

ESF Ventilation System	Wattage
Standby Gas Treatment (SGT) System	≥ 27 kW and ≤ 33 kW
Control Room Emergency Ventilation (CREV) System	≥ 10.8 kW and ≤ 13.2 kW

L <CTS>

5.5 Programs and Manuals

5.5.8 9

Explosive Gas and Storage Tank Radioactivity Monitoring Program (continued)

~~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"].~~ 14

The program shall include:

<3.8.H>
<4.8.H.a>

a. The limits for concentrations of hydrogen and oxygen in the Off- Waste Gas Holdup System and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion); 15 2

~~b. A surveillance program to ensure that the quantity of radioactivity contained in [each gas storage tank and fed into the offgas treatment system] is less than the amount that would result in a whole body exposure of ≥ 0.5 rem to any individual in an unrestricted area, in the event of [an uncontrolled release of the tanks' contents]; and~~ 16

<DOC M.2>

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

<DOC A.8>

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies. 1

L <CTS>

5.5 Programs and Manuals (continued)

5.5.~~10~~ 9 [9] **Diesel Fuel Oil Testing Program** shall establish [11]

<4.9.A.5.a> A diesel fuel oil testing program ~~to implement~~ required testing of both new fuel oil and stored fuel oil ~~shall be established~~. The [10] 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.A.5.c> a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:

<4.9.A.5.b>

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

2. a flash point and kinematic viscosity within limits ~~for ASTM 2D fuel oil~~, and [17] or water and sediment within limits

3. a clear and bright appearance with proper color;

<4.9.A.5.c> [16] Insert 5.5.9.b
<4.9.A.6.a> b. ~~Other properties for ASTM 2D fuel oil are within limits within 31 days following sampling and addition to storage tanks;~~ and [TSTF-106] of the new fuel oil

<4.9.A.6.b> [17] Standard c. Total particulate concentration of the fuel oil [18] in the storage tanks is ≤ 10 mg/l applicable when tested every 31 days in accordance with ASTM ~~D-2276, Method A-2 or A-3~~. [TSTF-118]

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

5.5.~~11~~ 10 [9] **Technical Specifications (TS) Bases Control Program**

<DOC M.2> This program provides a means for processing changes to the Bases of these Technical Specifications.

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

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

[1]

1. ~~a~~ A change in the TS incorporated in the license; or

2. ~~a~~ A change to the ~~updated~~ U FSAR or Bases that involves an unreviewed safety question as defined in 10 CFR 50.59. [8]

(continued)

TSTF-106

INSERT 5.5.9.b

verify that the properties of the new fuel oil, other than those addressed in a., above, are within limits for ASTM 2D fuel oil.

17

<CTS>

5.5 Programs and Manuals

<DOC M.2> 5.5.~~11~~ 10 9 **Technical Specifications (TS) Bases Control Program** (continued)

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the ~~FSAR~~ U. 8

d. Proposed changes that meet the ~~criteria~~ (ion) of Specification ~~5.5.11b~~ 5.5.10.b.1 or 5.5.10.b.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).

5.5.~~12~~ 11 9 **Safety Function Determination Program (SFDP)**

<DOC M.2>

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

3

a.

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

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

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

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

b. A loss of safety function exists when, assuming no concurrent single failure, and assuming no concurrent loss of offsite power or loss of onsite diesel generator(s), a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and: TSTF-273

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

(continued)

<CTS>

5.5 Programs and Manuals

11 9

5.5.12 Safety Function Determination Program (SFDP) (continued)

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

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

18 3

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.

<6.8.D.5> Insert 5.5.12 19

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.

TSTF-273

<CTS> 19 INSERT 5.5.12

5.5.12 Primary Containment Leakage Rate Testing Program

<6.8.D.5>

a. This program shall establish the leakage testing of the primary containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemption. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Testing Program," dated September 1995.

b. The peak calculated primary containment internal pressure for the design basis loss of coolant accident, P_a, is 48 psig.

c. The maximum allowable primary containment leakage rate, L_a, at P_a, is 1% of primary containment air weight per day.

d. Leakage rate acceptance criteria are:

1. Primary containment overall 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 $\leq 0.60\ L_a$ for the combined Type B and Type C tests, and $\leq 0.75\ L_a$ for Type A tests.

2. Air lock testing acceptance criteria is the overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

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

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS: 5.5 - PROGRAMS AND MANUALS

1. Typographical/grammatical error corrected.

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

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

4. The Surveillance Frequency has been extended to 24 months to be consistent with the proposed "refueling cycle interval" Surveillance Frequency in the Quad Cities 1 and 2 ITS LCO Sections. The normal "refueling cycle intervals" (i.e., 18 months) have been extended to 24 months in the Quad Cities 1 and 2 ITS, thus this requirement, which is essentially a Surveillance Requirement, has also been extended. In addition, since normal Surveillance Requirements in the LCO Sections allow a 25% extension of the Frequency per proposed SR 3.0.2 (CTS 4.0.B), this allowance has also been added for this Surveillance Requirement (since SR 3.0.2 only applies to the LCO Sections (i.e., LCO Sections 3.1 through 3.10). Also, the term "or less" is unnecessary and has been deleted for consistency.

5. The term "radioactive gases" has been changed to "radioactive iodines" consistent with current licensing basis.

6. This change has been made to comply with the new 10 CFR 20 requirements or have been added for clarity. In addition, these requirements in ITS 5.5.4 at one time were located in individual Specifications in the CTS. Thus, CTS 4.0.B (ITS SR 3.0.2) and CTS 4.0.C (ITS SR 3.0.3) applied to the CTS surveillance frequencies. To maintain this, an allowance that SR 3.0.2 and SR 3.0.3 are applicable to the surveillance frequencies has been added to ITS 5.5.4. This change is consistent with TSTF-258, Rev. 4, except that in the Quad Cities 1 and 2 submittal, the words are “surveillance frequencies” in lieu of “surveillance frequency” since the surveillance tests required by ITS 5.5.4 are not all performed at the same frequency.

7. This requirement has been deleted since Quad Cities 1 and 2 have Mark I containments. This change is consistent with current licensing basis.

8. The proper plant specific information/nomenclature has been provided.

9. This bracketed requirement has been deleted because it is not applicable to Quad Cities 1 and 2 (Quad Cities 1 and 2 do not have prestressed concrete containments). The following Specifications were also renumbered to reflect the deletion.

10. The words of the Ventilation Filter Testing Program and the Diesel Fuel Oil Testing Program have been modified to be consistent with the purpose statements of the other programs in this Section. The current words require a program to be established. These current words imply that a program does not exist and this statement is directing

10. (continued)

the utility to establish the program. However, when ITS is implemented, a program will already have been established. The purpose statement needs to say that the applicable program establishes certain requirements (e.g., testing of ESF filter ventilation systems). The other ITS programs (e.g., IST Program, Specification 5.5.6) provide the proper words, assuming that the program is already established. Therefore, these changes are bringing the VFTP and the Diesel Fuel Oil Testing Program in line with the words of the other programs.

11. Editorial change for enhanced clarity.

12. The bracketed "Reviewer's Note" in ISTS 5.5.8 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.

13. ISTS 5.5.8.d demonstrates that the pressure drop across the HEPA filters and charcoal adsorbers is less than the specified pressure drop when tested at the specified system flow rate. The referenced methods for performing the test, Regulatory Guide 1.52 and ASME N510-1989 do not provide the methods for performing this test. As a result, these test method references have been deleted in ITS 5.5.7.d. In addition, the requirement in ISTS 5.5.8.d to test across the prefilter has been deleted and the words ", corrected for voltage variations at the 480 V bus," have been added to ISTS 5.5.8.e to be consistent with the current licensing basis.

14. The provisions in ISTS 5.5.9 for Waste Gas Systems are for PWRs and not applicable to Quad Cities 1 and 2. Quantities of radioactivity contained in all outdoor liquid radwaste tanks meeting the conditions of ITS 5.5.8 are determined in accordance with the specified Surveillance Program (ITS 5.5.8.b). Therefore, the sentence in the introductory paragraph is not necessary to specify a method to determine liquid radwaste quantities.

15. The requirement to limit oxygen in the Off-gas System has been deleted consistent with current licensing basis.

16. The provisions in ISTS 5.5.9.b are only for the PWRs and are not applicable for Quad Cities 1 and 2. Due to this deletion, the following Specification has been renumbered.

17. The following changes have been made to ISTS 5.5.10:

    a. An allowance to perform a water and sediment test instead of the clear and bright test has been provided.

17. (continued)

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

    c. The words in ISTS 5.5.10.c "ASTM D-2276 Method A-2 or A-3" have been changed to "the applicable ASTM Standard" in ITS 5.5.9.c to be consistent with current licensing basis.

18. These words have been added for clarity.

19. The Primary Containment Leakage Rate Testing Program has been added to be consistent with the current licensing basis and TSTF-52.

20. The Inservice Testing (IST) Program 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 the IST Program applies to (i.e., pumps and valves) have been added for clarity. In addition, the statement "The program shall include the following:" has been deleted since not all the statements that follow are really part of the program requirements.

21. The current licensing basis Surveillance Frequencies have been provided. 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.

22. An additional testing frequency of 48 months has been added to the Inservice Testing Program requirements in ITS 5.5.6 consistent with the ASME Boiler and Pressure Vessel Code. The 48 month frequency is the frequency recommended for Class 2 and 3 pressure relief devices.

<CTS>

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

<6.9> The following reports shall be submitted in accordance with 10 CFR 50.4.

5.6.1 Occupational Radiation Exposure Report

TSTF-152

----------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

Insert 5.6.1 1 2

<6.9.A.2>
<6.9.A.2.c>

~~A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures > 100 mrem/yr and their associated man rem exposure 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 dosimeter, thermoluminescent dosimeter (TLD), or film badge measurements. Small exposures totalling < 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources should be assigned to specific major work functions. The report shall be submitted by April 30 of each year. [The initial report shall be submitted by April 30 of the year following initial criticality.]~~

5.6.2 Annual Radiological Environmental Operating Report

<6.9.A.3>

----------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

(continued)

TSTF-152 INSERT 5.6.1 1 2 3

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 initial criticality.]~~

man

(

)

%

<CTS>

5.6 Reporting Requirements

5.6.2 Annual Radiological Environmental Operating Report (continued)

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

~~The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements [in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979]. [The report shall identify the TLD results that represent collocated dosimeters in relation to the NRC TLD program and the exposure period associated with each result.] In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.~~ 4 TSTF-348

5.6.3 Radioactive Effluent Release Report

TSTF-152 changes not adopted 5

<6.9.A.4>

-----NOTE-----
A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit. 1

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

prior to May 1 of each year 11

the 6

5.6.4 Monthly Operating Reports

<6.9.A.5>

Routine reports of operating statistics and shutdown experience~~s~~, including documentation of all challenges to the ~~safety/~~relief 1

safety and 12

TSTF-258 change not adopted

(continued)

<CTS>

5.6 Reporting Requirements

12 TSTF-258 change not adopted

5.6.4 Monthly Operating Reports (continued)

[valves,] shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report. 1

<6.9.A.6> 5.6.5 CORE OPERATING LIMITS REPORT (COLR)

<6.9.A.6.a> <A.9>

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: Insert 5.6.5.a

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

<6.9.A.6.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 5.6.5.b

~~[Identify the Topical Report(s) by number, title, date, and NRC staff approval document, or identify the staff Safety Evaluation Report for a plant specific methodology by NRC letter and date.]~~ 1

<6.9.A.6.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.

<6.9.A.6.c>

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)~~ 7

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

(continued)

<CTS>

1 **INSERT 5.6.5.a**

1. The APLHGR for Specification 3.2.1.
2. The MCPR for Specification 3.2.2. <6.9.A.6.a>
3. The LHGR for Specification 3.2.3.
4. The LHGR and transient linear heat generation rate limit for Specification 3.2.4. <DOC A.9>
5. Control Rod Block Instrumentation Setpoint for the Rod Block Monitor – Upscale Function Allowable Value for Specification 3.3.2.1.

1 **INSERT 5.6.5.b**

1. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," (latest approved revision).
2. Commonwealth Edison Topical Report NFSR-0085, "Benchmark of BWR Nuclear Design Methods," (latest approved revision). <6.9.A.6.b>
3. Commonwealth Edison Topical Report NFSR-0085, Supplement 1, "Benchmark of BWR Nuclear Design Methods - Quad Cities Gamma Scan Comparisons," (latest approved revision).
4. Commonwealth Edison Topical Report NFSR-0085, Supplement 2, "Benchmark of BWR Nuclear Design Methods - Neutronic Licensing Analyses," (latest approved revision).
5. Advanced Nuclear Fuels Methodology for Boiling Water Reactors, XN-NF-80-19(P)(A), Volume 1, Supplement 3, Supplement 3 Appendix F, and Supplement 4, Advanced Nuclear Fuels Corporation, November 1990.
6. Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads, XN-NF-80-19(P)(A), Volume 4, Revision 1, Exxon Nuclear Company, June 1986.
7. Exxon Nuclear Methodology for Boiling Water Reactors THERMEX: Thermal Limits Methodology Summary Description, XN-NF-80-19(P)(A), Volume 3, Revision 2, Exxon Nuclear Company, January 1987.
8. Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis, XN-NF-80-19(P)(A), Volume 1 and Supplements 1 and 2, Exxon Nuclear Company, March 1983.
9. Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel, XN-NF-85-67(P)(A) Revision 1, Exxon Nuclear Company, September 1986.
10. Qualification of Exxon Nuclear Fuel for Extended Burnup Supplement 1: Extended Burnup Qualification of ENC 9x9 BWR Fuel, XN-NF-82-06(P)(A) Supplement 1, Revision 2, Advanced Nuclear Fuels Corporation, May 1988.

<CTS>

$\boxed{1}$ INSERT 5.6.5.b (continued)

11. Advanced Nuclear Fuels Corporation Generic Mechanical Design for Advanced Nuclear Fuels 9x9-IX and 9x9-9X BWR Reload Fuel, ANF-89-014(P)(A), Revision 1 and Supplements 1 and 2, Advanced Nuclear Fuels Corporation, October 1991.

12. Generic Mechanical Design Criteria for BWR Fuel Designs, ANF-89-98(P)(A) Revision 1, and Revision 1 Supplement 1, Advanced Nuclear Fuels Corporation, May 1995.

13. Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors, XN-NF-79-71(P)(A), Revision 2 Supplements 1, 2, and 3, Exxon Nuclear Company, March 1986.

14. ANFB Critical Power Correlation, ANF-1125(P)(A) and Supplements 1 and 2, Advanced Nuclear Fuels Corporation, April 1990.

<6.9.A.6.b>

15. Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors/Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors: Methodology for Analysis of Assembly Channel Bowing Effects/NRC Correspondence, ANF-524(P)(A), Revision 2, Supplement 1 Revision 2, Supplement 2, Advanced Nuclear Fuels Corporation, November 1990.

16. COTRANSA 2: A Computer Program for Boiling Water Reactor Transient Analyses, ANF-913(P)(A) Volume 1 Revision 1 and Volume 1 Supplements 2, 3, and 4, Advanced Nuclear Fuels Corporation, August 1990.

17. Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR Evaluation Model, ANF-91-048(P)(A), Advanced Nuclear Fuels Corporation, January 1993.

18. Commonwealth Edison Topical Report NFSR-0091, "Benchmark of CASMO/MICROBURN BWR Nuclear Design Methods," Revision 0, Supplements 1 and 2, December 1991, March 1992, and May 1992, respectively; SER letter dated March 22, 1993.

19. ANFB Critical Power Correlation Application for Coresident Fuel, EMF-1125(P)(A), Supplement 1, Appendix C, Siemens Power Corporation, August 1997.

20. ANFB Critical Power Correlation Determination of ATRIUM-9B Additive Constant Uncertainties, ANF-1125(P)(A), Supplement 1, Appendix E, Siemens Power Corporation, September 1998.

5.6 Reporting Requirements

7

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

[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. [Identify the NRC staff approval document by date.]

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

[Reviewers' Notes: The methodology for the calculation of the P-T limits for NRC approval should include the following provisions:

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

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

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

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

5. The limiting ART shall be incorporated into the calculation of the pressure and temperature limit curves in accordance with NUREG-0800 Standard Review Plant 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.]

(continued)

5.6 Reporting Requirements

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) (continued) [7]

[7. Licensees who have removed two or more capsules should compare for each surveillance material the measured increase in reference temperature (RT_{NDT}) to the predicted increase in RT_{NDT}; where the predicted increase in RT_{NDT} is based on the mean shift in RT_{NDT} plus the two standard deviation value ($2\sigma_\Delta$) specified in Regulatory Guide 1/99, Revision 2. If the measured value exceeds the predicted value (increase in $RT_{NDT} + 2\sigma_\Delta$), the licensee should provide a supplement to the PTLR to demonstrate how the results affect the approved methodology.]

5.6.7 EDG Failures Report [8]

[If an individual emergency diesel generator (EDG) experiences four or more valid failures in the last 25 demands, these failures and any nonvalid failures experienced by that EDG in that time period shall be reported within 30 days. Reports on EDG failures shall include the information recommended in Regulatory Guide 1/9, Revision 3, Regulatory Position C.5, or existing Regulatory Guide 1.108 reporting requirement.]

[8] 5.6.~~8~~ 6 (PAM) Report Post Accident Monitoring Instrumentation [9]

〈Table 3.2.F-1〉

When a report is required by Condition B or ~~G~~ F of LCO 3.3.~~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. [1]

[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.] [10]

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

2. Certain changes to ISTS 5.6.1 per TSTF-152 have not been incorporated in ITS 5.6.1. The symbol "%" is used in lieu of "percent" for consistency with other specifications. The term "man-rem" has been retained since "person-rem" is not the unit defined in the regulations or guides. The term "film badge" has not been used since film badges are not used at Quad Cities 1 and 2 to comply with this requirement.

3. The initial report requirement for ISTS 5.6.1 is being deleted since this initial report has been submitted on a one-time basis.

4. ISTS 5.6.2 was revised to delete specific details of the annual radiological environmental operating report. This change is in accordance with changes approved in an SER dated April 2, 1996.

5. ISTS 5.6.3 (Radioactive Effluent Release Report) is revised by TSTF-152. Certain changes of TSTF-152 are not incorporated in ITS 5.6.3 for the following reasons:

   a. The Note allowing a single submittal to be made for a multiple unit station is revised by TSTF-152 to state that the submittal "shall" combine sections common to all units of the station. This change is inconsistent with similar Notes that are provided in ISTS 5.6.1 and 5.6.2. In addition, the NRC guidance provided in the proposed Generic Letter on Technical Specification changes for 10 CFR 20 implementation (referenced as the justification for these changes in TSTF-152) did not include this change.

   b. TSTF-152 revises the first sentence of ISTS 5.6.3 to state that the Radioactive Effluent Release Report covering operation of the unit "during the previous year" shall be submitted "prior to May 1 of each year" in accordance with 10 CFR 50.36a. The first portion of this change is duplicative of the requirements in 10 CFR 50.36a and is therefore not required to be in the Technical Specifications. 10 CFR 50.36a states that the report must be submitted within one year of the previous report. Since Technical Specifications cannot supersede the requirements of 10 CFR 50, implementation of this change would require NRC approval of an exemption request in accordance with 10 CFR 50.12. This is considered to be outside the scope of the ITS conversion.

   c. TSTF-152 revises the last sentence of ISTS 5.6.3 to state "10 CFR Part 50," in lieu of "10 CFR 50". This change is inconsistent with similar words in ISTS 5.6.2, as well as other places in the ISTS (notably the Bases). Therefore, the ITS leaves the words "10 CFR 50."

6. Typographical/grammatical error corrected.

7. The utilization of a Pressure and Temperature Limits Report (PTLR) requires the development and NRC approval of detailed methodologies for future revisions to P/T limits. At this time, ComEd does not have the necessary methodologies submitted to the NRC for review and approval. Therefore, the proposed presentation removes references to the PTLR and proposes that the specific limits and curves be included in the P/T limits Specification (ITS 3.4.9).

8. ISTS 5.6.7 has been deleted in accordance with the guidance of Generic Letter 94-01. Quad Cities 1 and 2 have implemented a maintenance program for monitoring and maintaining diesel generator performance in accordance with the provisions of the maintenance rule and consistent with the guidance of Regulatory Guide 1.160. This change is also consistent with TSTF-37. In addition, the following Specification was renumbered to reflect this deletion.

9. The acronym "PAM" has been defined, consistent with the format of the ITS, since it is the first use of this term in this Specification. The term "Instrumentation" has also been added for clarity. Also, the proper Condition has been referenced.

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

11. ISTS 5.6.3 has been revised to identify the required submittal date, "prior to May 1 of each year," for the Radioactive Effluent Release Report. This change is consistent with the NRC approved ITS requirements for the Byron and Braidwood Stations.

12. ISTS 5.6 (Reporting Requirements) is revised by TSTF-258, Rev. 4. In order to maintain consistency, to the maximum extent practicable, between the Administrative Controls Technical Specifications of the ComEd nuclear stations, the following change of TSTF-258, Rev. 4, is not incorporated in ITS 5.6:

    ISTS 5.6.4 contains a requirement for the Monthly Operating Report to document challenges to safety/relief valves. This requirement is deleted by TSTF-258, Rev. 4.

    Not incorporating this change to ISTS 5.6.4 is consistent with the NRC approved ITS for the ComEd Byron and Braidwood Stations.

1

TSTF-258 changes not adopted

 1

5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area 1

at 30 cm (12 in.) radiation protection

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

(or equivalent document)

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

1

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

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

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

(or equivalent document)

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

7

Insert 5.7.2

(continued)

1 INSERT 5.7.2

at 30 cm (12 in.) from the radiation source or from any surface which the radiation penetrates shall require the following:

a. Doors shall be locked to prevent unauthorized entry and shall not prevent individuals from leaving the area. In place of locking the door, direct or electronic surveillance that is capable of preventing unauthorized entry may be used. The keys shall be maintained under the administrative control of the Shift Engineer on duty or radiation protection supervision.

b. Personnel access and exposure control requirements of activities being performed within these areas shall be specified by an approved RWP (or equivalent document).

c. Each person entering the area shall be provided with an alarming radiation monitoring device that continuously integrates the radiation dose rate (such as an electronic dosimeter). Surveillance and radiation monitoring by a radiation protection technician may be substituted for an alarming dosimeter.

TSTF-258 changes not adopted. 1

5.7 High Radiation Area 1

5.7.2 (continued)

areas and the maximum allowable stay times for individuals in those areas. ~~In lieu of the stay time specification of the RWP, direct or remote (such as closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area.~~ 1

5.7.3 For individual high radiation areas with radiation levels of > 1000 mrem/hr at 30 cm (12 in.) accessible to personnel, that are located within large areas ~~such as reactor containment~~, where no enclosure exists for purposes of locking, ~~or that cannot be continuously guarded~~, and where no enclosure can be reasonably constructed around the individual area, that individual area shall be barricaded and conspicuously posted, and a flashing light shall be activated as a warning device.

1. The brackets have been removed and the proper plant-specific information has been provided. In addition, the changes to ISTS 5.7 from TSTF-258, Rev. 4, are not adopted since Quad Cities 1 and 2 choose to maintain their CTS requirements for High Radiation Area controls.

ADMINISTRATIVE CHANGES
("A.x" Labeled Comments/Discussions)

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   The proposed change involves reformatting, renumbering, and rewording the existing Technical Specifications. The reformatting, renumbering, and rewording process involves no technical changes to the existing Technical Specifications. As such, this change is administrative in nature and does not impact initiators of analyzed events or assumed mitigation of accident or transient events. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The proposed change will not impose any new or eliminate any old requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change will not reduce a margin of safety because it has no impact on any safety analyses assumptions. This change is administrative in nature. Therefore, the change does not involve a significant reduction in a margin of safety.

TECHNICAL CHANGES - MORE RESTRICTIVE
("M.x" Labeled Comments/Discussions)

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   The proposed change provides more stringent requirements for operation of the facility. These more stringent requirements do not result in operation that will increase the probability of initiating an analyzed event and do not alter assumptions relative to mitigation of an accident or transient event. The more restrictive requirements continue to ensure process variables, structures, systems, and components are maintained consistent with the safety analyses and licensing basis. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in the methods governing normal plant operation. The proposed change does impose different requirements. However, these changes are consistent with the assumptions in the safety analyses and licensing basis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The imposition of more restrictive requirements either has no impact on or increases the margin of plant safety. As provided in the discussion of the change, each change in this category is by definition, providing additional restrictions to enhance plant safety. The change maintains requirements within the safety analyses and licensing basis. Therefore, this change does not involve a significant reduction in a margin of safety.

GENERIC NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: CHAPTER 5.0 - ADMINISTRATIVE CONTROLS

"GENERIC" LESS RESTRICTIVE CHANGES:
RELOCATING DETAILS TO TECHNICAL SPECIFICATION BASES, UFSAR, TRM, OR OTHER PLANT CONTROLLED DOCUMENTS
("LA.x" Labeled Comments/Discussions)

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   The proposed change relocates certain details from the Technical Specifications to the Bases, UFSAR, TRM, or other plant controlled documents. The Bases, UFSAR, TRM, and other plant controlled documents containing the relocated information will be maintained in accordance with 10 CFR 50.59. In addition to 10 CFR 50.59 provisions, the Technical Specification Bases are subject to the change control provisions in the Administrative Controls Chapter of the ITS. The UFSAR is subject to the change control provisions of 10 CFR 50.71(e), and the plant procedures and other plant controlled documents are subject to controls imposed by plant administrative procedures, which endorse applicable regulations and standards. Since any changes to the Bases, UFSAR, TRM, or other plant controlled documents will be evaluated per the requirements of the Bases Control Program in Chapter 5.0 of the ITS or 10 CFR 50.59, no increase (significant or insignificant) in the probability or consequences of an accident previously evaluated will be allowed. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. The proposed change will not impose or eliminate any requirements, and adequate control of the information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the details to be transposed from the Technical Specifications to the Bases, UFSAR, TRM, or other plant controlled

"GENERIC" LESS RESTRICTIVE CHANGES:
RELOCATING DETAILS TO TECHNICAL SPECIFICATION BASES, UFSAR, TRM, OR OTHER PLANT CONTROLLED DOCUMENTS
("LA.x" Labeled Comments/Discussions)

3. (continued)

documents are the same as the existing Technical Specifications. Since any future changes to these details in the Bases, UFSAR, TRM, or other plant controlled documents will be evaluated per the requirements of 10 CFR 50.59, no reduction (significant or insignificant) in a margin of safety will be allowed. Based on 10 CFR 50.92, the existing requirement for NRC review and approval of revisions, to these details proposed for relocation, does not have a specific margin of safety upon which to evaluate. However, since the proposed change is consistent with the BWR ISTS, NUREG-1433, Rev. 1, approved by the NRC Staff, revising the Technical Specifications to reflect the approved level of detail ensures no significant reduction in the margin of safety.

"GENERIC" LESS RESTRICTIVE CHANGES: EXTENDING SURVEILLANCE FREQUENCIES FROM 18 MONTHS TO 24 MONTHS FOR SURVEILLANCES OTHER THAN CHANNEL CALIBRATIONS ("LD.x" Labeled Comments/Discussions)

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   The proposed change involves a change in the surveillance testing intervals from 18 months to 24 months. The proposed change does not physically impact the plant nor does it 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 analysis. The proposed change does not impact the Surveillance Requirements themselves nor the way in which the Surveillances are performed. Additionally, the proposed change does not introduce any new accident initiators since 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 indicated 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 increase the probability or consequences of an accident previously evaluated.

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

   The proposed change involves a change in the surveillance testing intervals 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 Requirements themselves and the way Surveillances are performed will remain unchanged. Furthermore, an historical review of surveillance test results indicated 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.

"GENERIC" LESS RESTRICTIVE CHANGES:
EXTENDING SURVEILLANCE FREQUENCIES FROM 18 MONTHS TO 24 MONTHS FOR SURVEILLANCES OTHER THAN CHANNEL CALIBRATIONS
("LD.x" Labeled Comments/Discussions) (continued)

3. Does this change involve a significant reduction in a margin of safety?

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, 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 assumptions in the licensing basis are not impacted, and the proposed change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 5.1 - RESPONSIBILITY

There are no plant specific less restrictive changes identified for this Specification.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 5.2 - ORGANIZATION

L.1 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   The proposed change will remove the requirement for a licensed Senior Reactor Operator (SRO) to be present in the control room while the unit is in MODE 4. As a result, an SRO will not be required to be present in the control room in MODE 4 or 5. The proposed change conforms to 10 CFR 50.54(m)(2)(iii) and is consistent with the BWR ISTS, NUREG-1433, Revision 1 (ISTS 5.2.2.b). In MODE 4, all control rods are normally fully inserted and the probability and consequences of a Design Basis Accident (DBA) are significantly reduced due to the limitations on pressure and temperature. In addition, pursuant to 10 CFR 50.54(m)(2), a Reactor Operator (RO) will still be required to be present at the controls (in the control room) at all times and, in MODE 4, at least one SRO, who is assigned supervisory responsibility, will be required to be on-site and readily available to the RO for consultation. The proposed change does not involve any physical alteration of plant systems, structures, or components, changes in parameters governing normal plant operation, or methods of operation. Thus, the proposed change will not impact the plant's response to a DBA and the probability and consequences of such an accident will be reduced. Therefore, the proposed change will not significantly increase the probability or consequences of an accident previously evaluated.

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

   The proposed change does not involve any design change or plant modifications, nor will the change alter any technical requirements or system parameters. The proposed change does not introduce any new modes or alter any existing modes of plant operation in a manner that could create a new precursor of an accident. As such, plant structures, systems, and components will continue to function as previously analyzed. Therefore, the proposed change will not create the possibility of a new or different kind of an accident from any accident previously evaluated.

L.1 CHANGE (continued)

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

The proposed change results in an SRO not being required to be present in the control room in MODE 4. The proposed change conforms to 10 CFR 50.54(m)(2)(iii) and is consistent with the BWR ISTS, NUREG-1433, Revision 1 (ISTS 5.2.2.b). The proposed change does not involve any physical alteration of plant systems, structures, or components, changes in parameters governing normal plant operation, or methods of operation. Furthermore, in MODE 4, all control rods are normally fully inserted and the probability and consequences of a DBA are significantly reduced due to the limitations on pressure and temperature. In addition, pursuant to 10 CFR 50.54(m)(2), an RO will still be required to be present at the controls (in the control room) at all times and, in MODE 4, at least one SRO, who is assigned supervisory responsibility, will be required to be on-site and readily available to the RO for consultation. Thus, the proposed change will not impact the plant's response to a DBA and the limitations on pressure and temperature in MODE 4 provide increased safety margins. Therefore, this proposed change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 5.3 - UNIT STAFF QUALIFICATIONS

There are no plant specific less restrictive changes identified for this Specification.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 5.4 - PROCEDURES

There are no plant specific less restrictive changes identified for this Specification.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 5.5 - PROGRAMS AND MANUALS

L.1 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   The proposed changes relax current technical specification monitoring requirements for specific emergency diesel generator fuel oil analyses. These proposed changes continue to ensure that diesel fuel oil acquired and stored for emergency diesel generators meets established ASTM standards and the quality of the fuel oil is sufficiently maintained to support diesel generator operation. The proposed changes do not affect the probability of an accident and are not considered initiators of any previously evaluated accident. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not introduce a new mode of plant operation and does not involve physical modification to the plant. Therefore, it does not create the possibility of a new or different type of accident from any accident previously evaluated.

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

   The proposed changes to the emergency diesel generator fuel oil monitoring requirements are consistent with ASTM standards for emergency diesel generator fuel oil. The margin of safety is not reduced due to these proposed changes. The proposed changes have no impact on the safe operation of the plant and the safety analysis assumptions will still be maintained, thus no question of safety exists. Therefore, these changes do not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 5.6 - REPORTING REQUIREMENTS

L.1 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   This change proposes to relax the requirements for submitting the Annual Radiological Environmental Operating Report and Radioactive Effluent Release Report. The CTS require the reports to be submitted prior to May 1 of each year and prior to April 1 of each year, respectively. This proposed change will allow the reports to be submitted by May 15 of each year and May 1 of each year, respectively. The proposed change does not affect the probability of an accident. The submittal dates of the Annual Radiological Environmental Operating Report and Radioactive Effluent Release Report are not assumed to be initiators of any analyzed event. Also, the consequences of an accident are not affected by the submittal dates of the Annual Radiological Environmental Operating Report and Radioactive Effluent Release Report. This proposed change does not impact the assumptions of any design basis accident. This change will not alter assumptions relative to the mitigation of an accident or transient event. This change has no impact on the safe operation of the plant. The reports will still be required to be submitted each year and do not affect any plant equipment or requirements for maintaining plant equipment. The submittal dates of these report are not required for the mitigation of any accident. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   This change proposes to relax the requirement for submitting the Annual Radiological Environmental Operating Report and the Radioactive Effluent Release Report. The current TS require the reports to be submitted prior to May 1 of each year and prior to April 1 of each year, respectively. This proposed change will allow the reports to be submitted by May 15 of each year and May 1 of each year, respectively. The proposed change will not create the possibility of an accident. This change will not physically alter the plant (no new or different type of equipment will be installed). The changes in methods governing normal plant operation are consistent with the current safety analysis assumptions. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

L.1 CHANGE (continued)

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

This change proposes to relax the requirement for submitting the Annual Radiological Environmental Operating Report and Radioactive Effluent Release Report. The current TS require the reports to be submitted prior to May 1 of each year and prior to April 1 of each year, respectively. This proposed change will allow the reports to be submitted by May 15 of each year and May 1 of each year, respectively. The margin of safety is not reduced by this change. This proposed change has no effect on the assumptions of the design basis accident. This change has no impact on the safe operation of the plant. The reports will still be required to be submitted each year and do not affect any plant equipment or requirements for maintaining plant equipment. The safety analysis assumptions will still be maintained, thus no question of safety exists. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 5.7 - HIGH RADIATION AREA

There are no plant specific less restrictive changes identified for this Specification.

NO SIGNIFICANT HAZARDS CONSIDERATION
CTS: 6.4 - TRAINING

There are no plant specific less restrictive changes identified for this Specification.

NO SIGNIFICANT HAZARDS CONSIDERATION
CTS: 6.7 - SAFETY LIMIT VIOLATION

There are no plant specific less restrictive changes identified for this Specification.

NO SIGNIFICANT HAZARDS CONSIDERATION
CTS: 6.11 - RADIATION PROTECTION PROGRAM

There are no plant specific less restrictive changes identified for this Specification.

NO SIGNIFICANT HAZARDS CONSIDERATION
CTS: 6.13 - PROCESS CONTROL PROGRAM (PCP)

There are no plant specific less restrictive changes identified for this Specification.

ENVIRONMENTAL ASSESSMENT
ITS: CHAPTER 5.0 - ADMINISTRATIVE CONTROLS

In accordance with the criteria set forth in 10 CFR 50.21, ComEd has evaluated this proposed Technical Specification change for identification of licensing and regulatory actions requiring environmental assessment, determined it meets the criteria for a categorical exclusion set forth in 10 CFR 51.22(c)(9) and as such, has determined that no irreversible consequences exist in accordance with 10 CFR 50.92(b). This determination is based on the fact that this change is being proposed as an amendment to a license issued pursuant to 10 CFR which changes a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or which changes an inspection or a surveillance requirement, and the amendment meets the following specific criteria:

1. The amendment involves no significant hazards consideration.

   As demonstrated in the No Significant Hazards Consideration, this proposed amendment does not involve any significant hazards consideration.

2. There is no significant change in the type or significant increase in the amounts of any effluents that may be released offsite.

   The proposed change will not result in changes in the operation or configuration of the facility. There will be no change in the level of controls or methodology used for processing of radioactive effluents or handling of solid radioactive waste, nor will the proposal result in any change in the normal radiation levels within the plant. Therefore, there will be no change in the types or significant increase in the amounts of any effluents released offsite resulting from this change.

3. There is no significant increase in individual or cumulative occupational radiation exposure.

   The proposed change will not result in changes in the operation or configuration of the facility which impact radiation exposure. There will be no change in the level of controls or methodology used for processing of radioactive effluents or handling of solid radioactive waste, nor will the proposal result in any change in the normal radiation levels within the plant. Therefore, there will be no increase in individual or cumulative occupational radiation exposure resulting from this change.

Therefore, based upon the above evaluation, ComEd has concluded that no irreversible consequences exist with the proposed change.