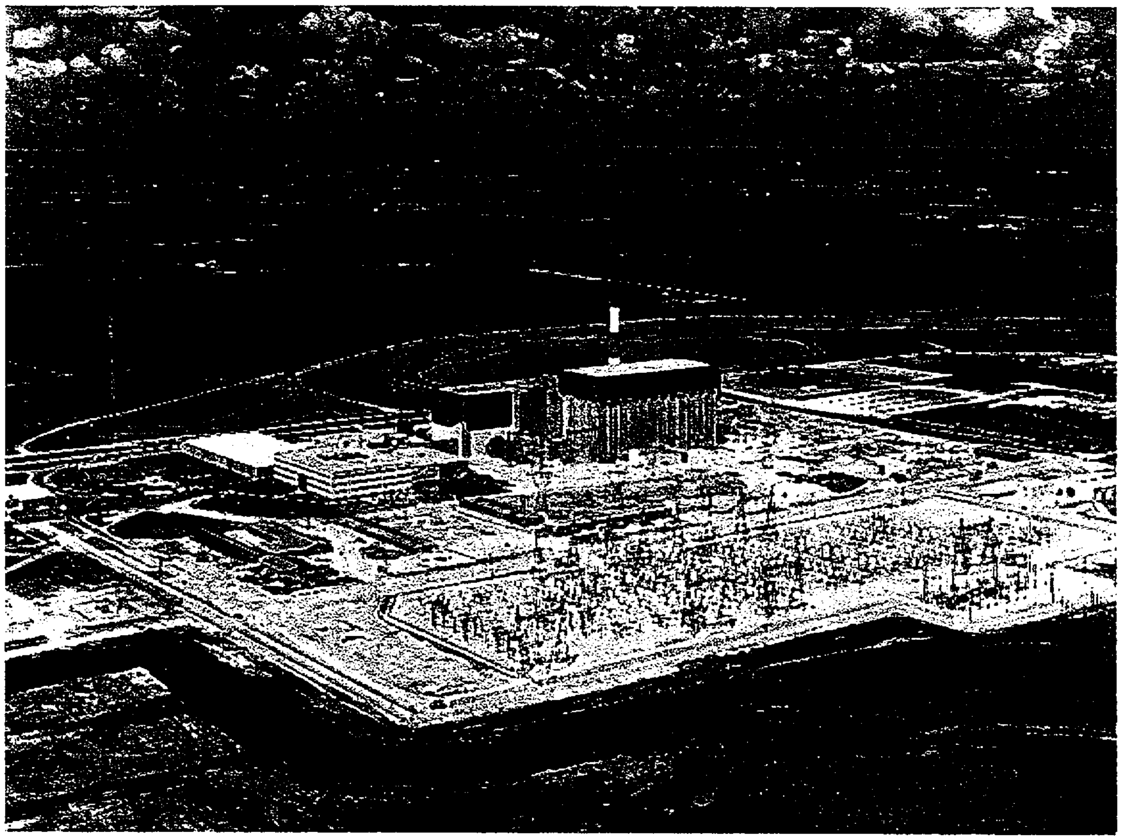
Improved Technical Specifications



LaSalle County 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 Boundaries

The site area and exclusion area boundaries are as shown in Figure 4.1-1.

4.1.2 Low Population Zone

The low population zone is all the land within a circle with its center at the vent stack and a radius of 3.98 miles.

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 764 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, and water rods or water boxes. Limited substitutions of Zircaloy, ZIRLO, 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 applicable NRC staff approved codes and methods and 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 185 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 6.26 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 819 ft.

4.3.3 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 3986 fuel assemblies for Unit 1 and 4078 fuel assemblies for Unit 2.

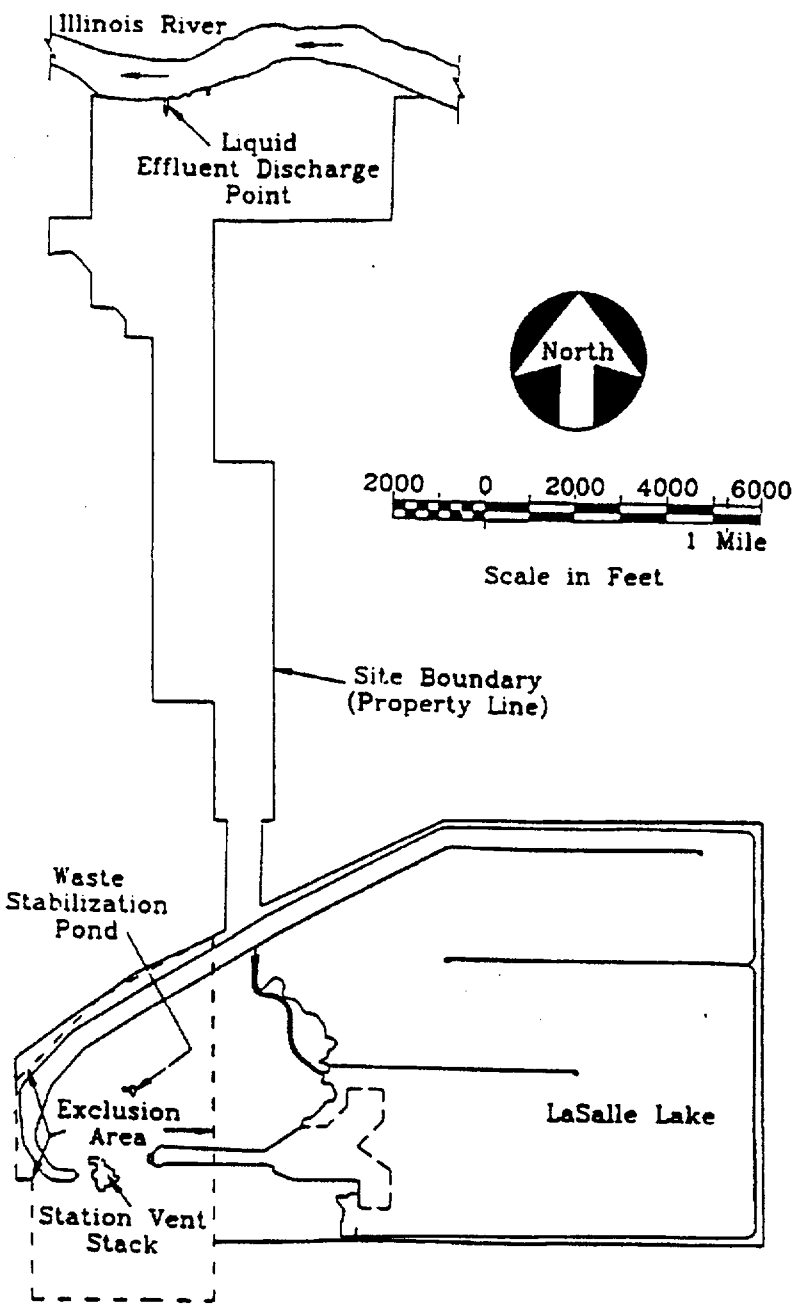


Figure 4.1-1 (Page 1 of 1)
Site and Exclusion Area Boundaries

A.1

ITS Chapter 4.0

4.0 5.0 DESIGN FEATURES

4.1 5.1 SITE

4.1.1 EXCLUSION AREA

A.2

5.1.1 The exclusion area shall be as shown in Figure 5.1.1-1.

4.1.2 LOW POPULATION ZONE

5.1.2 The low population zone shall be all the land within a circle with its center at the vent stack and a radius of 3.98 miles ~~as shown in Figure 5.1.2-1~~

4.1.1 SITE BOUNDARY FOR GASEOUS EFFLUENTS

5.1.3 The site boundary for gaseous effluents shall be as shown in Figure 5.1.1-1.

4.1.1 SITE BOUNDARY FOR LIQUID EFFLUENTS

5.1.4 The site boundary for liquid effluents shall be as shown in Figure 5.1.1-1.

~~5.2 CONTAINMENT~~

LA.1

~~CONFIGURATION~~

~~5.2.1 The primary containment is a steel lined post-tensioned concrete structure consisting of a drywell and suppression chamber. The drywell is a steel-lined post-stressed concrete vessel in the shape of a truncated cone closed by a steel dome. The drywell is above a cylindrical steel-lined post-stressed concrete suppression chamber and is attached to the suppression chamber through a series of downcomer vents. The drywell has a minimum free air volume of 229,538 cubic feet. The suppression chamber has an air region of 164,800 to 168,100 cubic feet and a water region of 128,800 to 131,900 cubic feet.~~

~~DESIGN TEMPERATURE AND PRESSURE~~

~~5.2.2 The primary containment is designed and shall be maintained for:~~

- ~~a. Maximum internal pressure 45 psig.~~
- ~~b. Maximum internal temperature: drywell 340°F. suppression chamber 275°F.~~
- ~~c. Maximum external pressure 5 psig.~~
- ~~d. Maximum floor differential pressure: 25 psid, downward. 5 psid, upward.~~

~~SECONDARY CONTAINMENT~~

~~5.2.3 The secondary containment consists of the Reactor Building, the equipment access structure and a portion of the main steam tunnel and has a minimum free volume of 2,875,000 cubic feet.~~

A.1

ITS Chapter 4.0

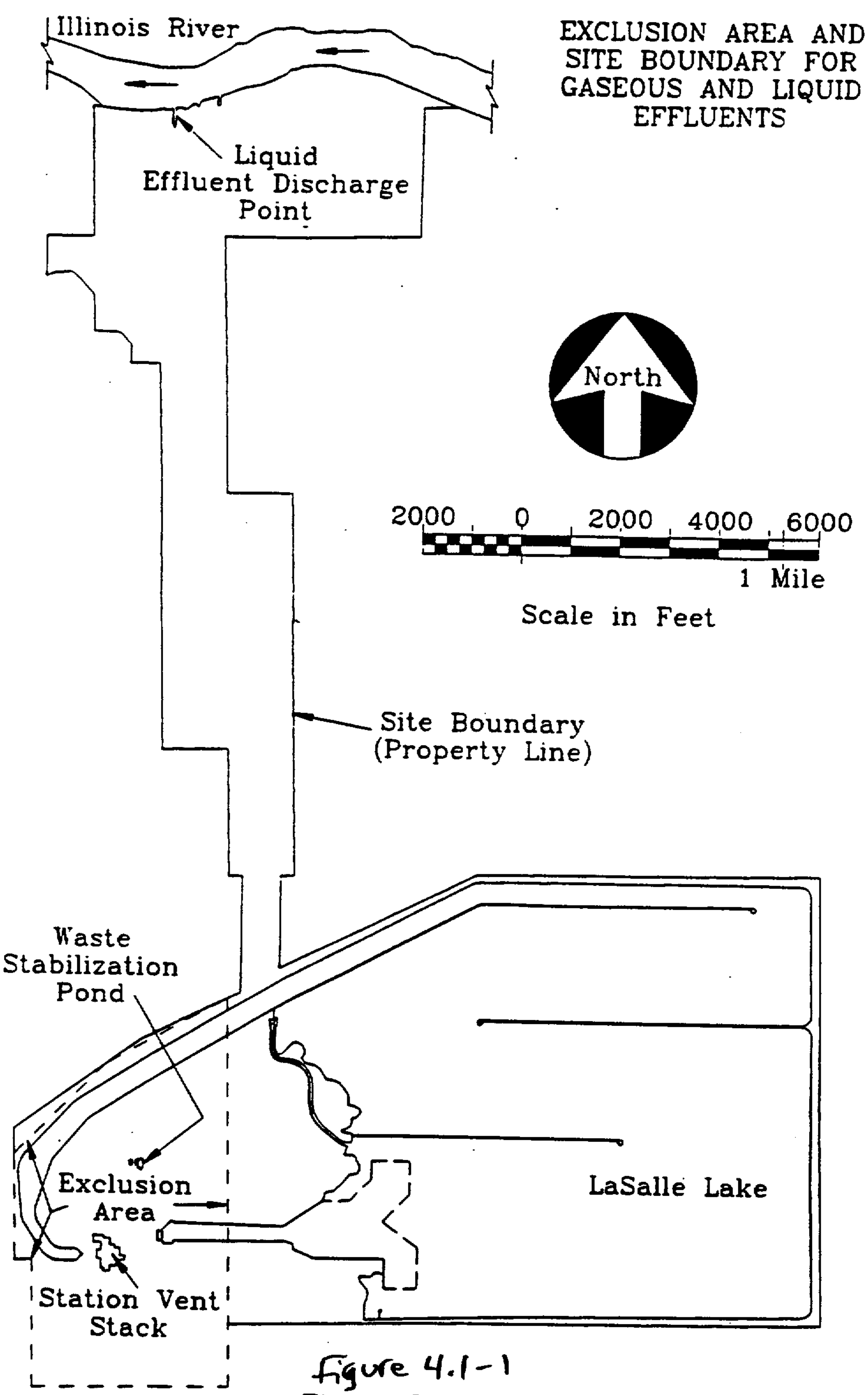


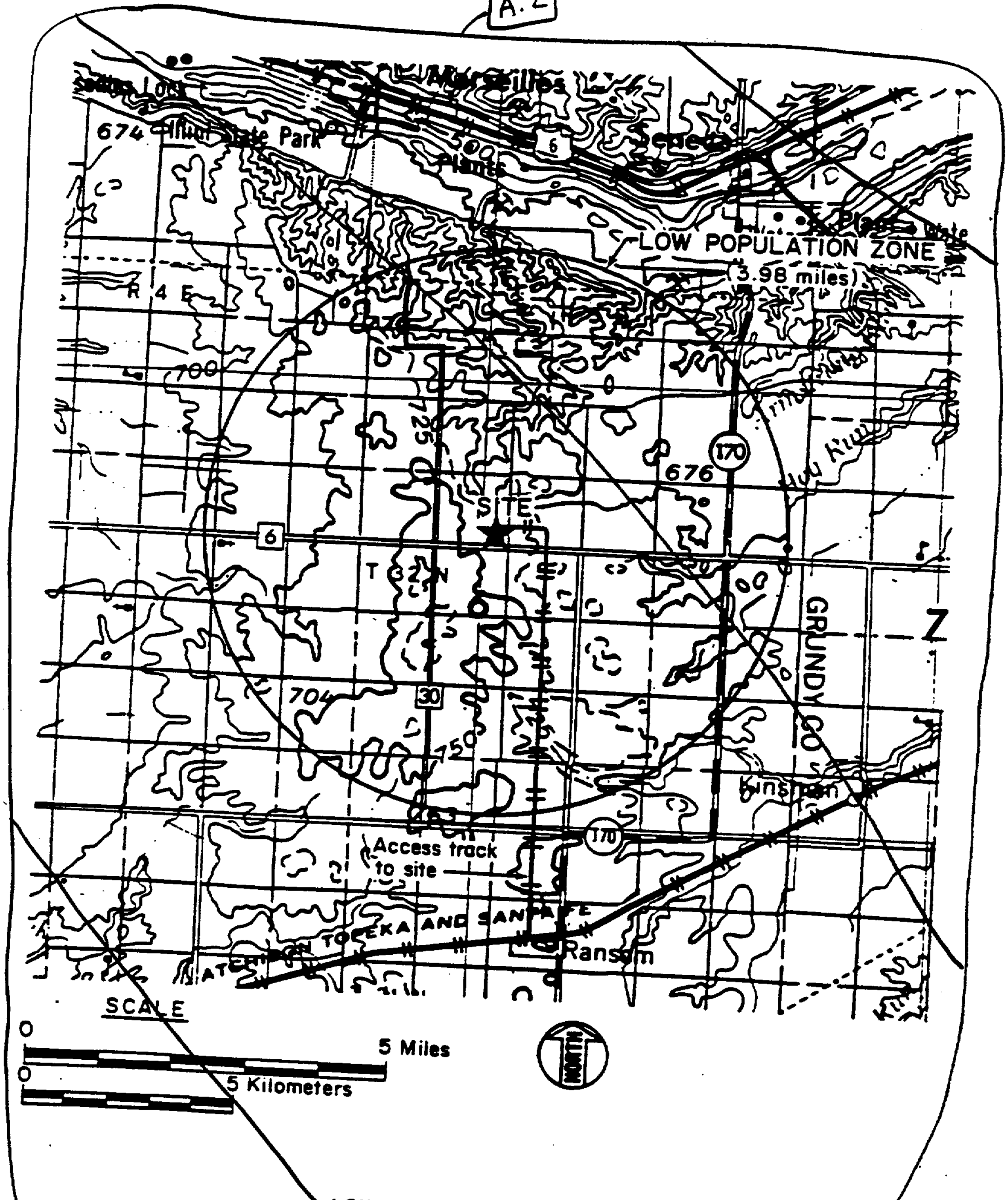
Figure 4.1-1

Figure 5.1.1-1

A.1

ITS Chapter 4.0

A.2



LOW POPULATION ZONE

Figure 5.1.2-1

ITS Chapter 4.0

4.2 5.3 REACTOR CORE A.1

FUEL ASSEMBLIES

4.2.1 5.3.1 The reactor shall contain 764 fuel assemblies. Each assembly shall consist of a matrix of Zircalloy clad A.1 fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. The bundles may contain water rods or water boxes. Limited substitutions of Zircalloy or ZIRLO 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 applicable NRC staff approved codes and methods and 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.

CONTROL ROD ASSEMBLIES

4.2.2 5.3.2 The reactor core shall contain 185 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.2

~~5.4 REACTOR COOLANT SYSTEM~~

~~DESIGN PRESSURE AND TEMPERATURE~~

LA.1

~~5.4.1 The reactor coolant system is designed and shall be maintained:~~

~~a. In accordance with the code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,~~

~~b. For a pressure of:~~

~~1. 1250 psig on the suction side of the recirculation pumps.~~

~~2. 1650 psig from the recirculation pump discharge to the outlet side of the discharge shutoff valve.~~

~~3. 1500 psig from the discharge shutoff valve to the jet pumps.~~

~~c. For a temperature of 575°F.~~

~~VOLUME~~

~~5.4.2 The total water and steam volume of the reactor vessel and recirculation system is ~ 21,000 cubic feet at a nominal T_{ave} of 533°F.~~

5.5 DELETED

Page 4 of 12

4.3 5.6 FUEL STORAGE

4.3.1 CRITICALITY

4.3.1.1 5.6.1 The spent fuel storage racks are designed and shall be maintained with:

a. A k_{eff} equivalent to ≤ 0.95 when flooded with unborated water, including all calculational uncertainties and biases, as described in Section 9.1 of the FSAR.

b. A nominal 6.26 inch center-to-center distance between fuel assemblies placed in the storage racks.

DRAINAGE

4.3.2 5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 819 feet.

CAPACITY

4.3.3 5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 3986 fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7.1-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7.1-1.

A.3
Moved to ITS Section 5.5

TABLE 5.7.1-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor	120 heatup and cooldown cycles	70°F to 560°F to 70°F
	10,000 power change cycles	75% to 100% to 75% of RATED THERMAL POWER
	80 step change cycles	Loss of Feedwater heaters
	190 reactor trip cycles	100% to 0% of RATED THERMAL POWER
	2000 power change cycles	50% to 100% to 50% of RATED THERMAL POWER
	400 control rod pattern exchanges	Not applicable
	130 hydrostatic pressure tests	Pressurized to ≥ 930 psig and ≤ 1250 psig.

A.1

A.3

moved to ITS Section 5.5

ITS Chapter 4.0

4.0 5.0 DESIGN FEATURES

4.1 5.1 SITE

4.1.1 EXCLUSION AREA

5.1.1 The exclusion area shall be as shown in Figure 5.1.1-1.

A.2

4.1.2 LOW POPULATION ZONE

5.1.2 The low population zone shall be ~~as shown in Figure 5.1.2-1~~ all the land within a circle with its center at the vent stack and a radius of 3.98 miles.

4.1.1 SITE BOUNDARY FOR GASEOUS EFFLUENTS

5.1.3 The site boundary for gaseous effluents shall be as shown in Figure 5.1.1-1.

4.1.1 SITE BOUNDARY FOR LIQUID EFFLUENTS

5.1.4 The site boundary for liquid effluents shall be as shown in Figure 5.1.1-1.

LA.1

5.2 CONTAINMENT

CONFIGURATION

5.2.1 The primary containment is a steel lined post-tensioned concrete structure consisting of a drywell and suppression chamber. The drywell is a steel-lined post-stressed concrete vessel in the shape of a truncated cone closed by a steel dome. The drywell is above a cylindrical steel-lined post-stressed concrete suppression chamber and is attached to the suppression chamber through a series of downcomer vents. The drywell has a minimum free air volume of 229,538 cubic feet. The suppression chamber has an air region of 164,800 to 168,100 cubic feet and a water region of 128,800 to 131,900 cubic feet.

DESIGN TEMPERATURE AND PRESSURE

5.2.2 The primary containment is designed and shall be maintained for:

a. Maximum internal pressure: 45 psig.

b. Maximum internal temperature: drywell 340°F.
suppression chamber 275°F.

c. Maximum external pressure: 5 psig.

d. Maximum floor differential pressure: 25 psid, downward.
5 psid, upward.

SECONDARY CONTAINMENT

5.2.3 The secondary containment consists of the Reactor Building, the equipment access structure and a portion of the main steam tunnel and has a minimum free volume of 2,875,000 cubic feet.

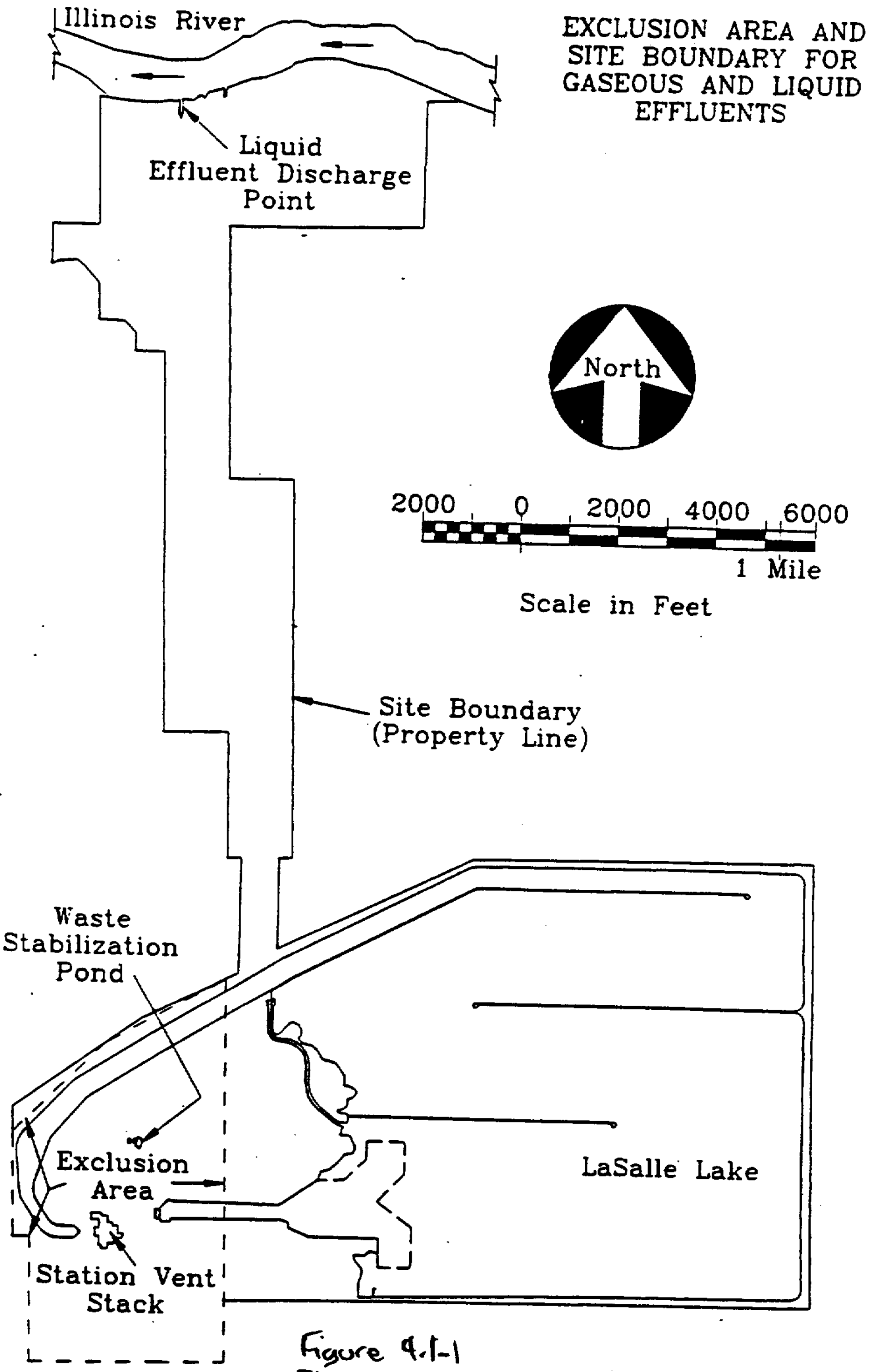
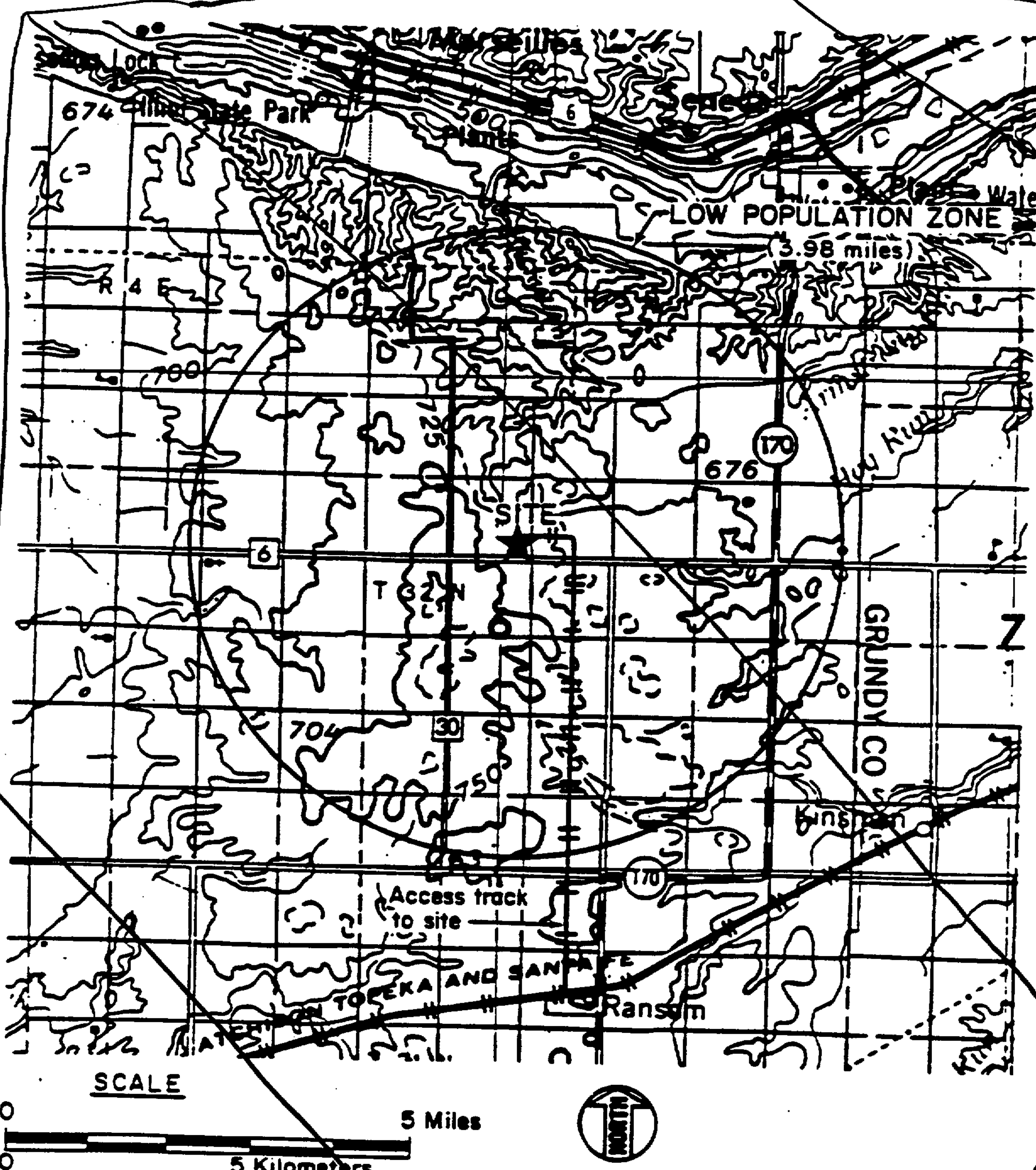


Figure 4.1-1
Figure 5.1.1-1



LOW POPULATION ZONE

Figure 5.1.2-1

DESIGN FEATURES

4.2 5.3 REACTOR CORE

4.2.1 FUEL ASSEMBLIES

5.3.1 The reactor shall contain 764 fuel assemblies. Each assembly shall consist of a matrix of Zircalloy clad A.1 fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. The bundles may contain water rods or water boxes. Limited substitutions of Zircalloy or ZIRLO 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 applicable NRC staff approved codes and methods and 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.

CONTROL ROD ASSEMBLIES

4.2.2 5.3.2 The reactor core shall contain 185 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.2

~~5.4 REACTOR COOLANT SYSTEM~~ LA.1

~~DESIGN PRESSURE AND TEMPERATURE~~

~~5.4.1 The reactor coolant system is designed and shall be maintained:~~

~~a. In accordance with the code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,~~

~~b. For a pressure of:~~

~~1. 1250 psig on the suction side of the recirculation pumps.~~

~~2. 1650 psig from the recirculation pump discharge to the outlet side of the discharge shutoff valve.~~

~~3. 1500 psig from the discharge shutoff valve to the jet pumps.~~

~~c. For a temperature of 575°F.~~

~~VOLUME~~

~~5.4.2 The total water and steam volume of the reactor vessel and recirculation system is ~ 21,000 cubic feet at a nominal T_{ave} of 533°F.~~

5.5 DELETED

A.1

ITS Chapter 4.0

4.3 5.6 FUEL STORAGE

4.3.1 CRITICALITY

4.3.1.1 5.6.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. A k_{eff} equivalent to $\leq$ 0.95 when flooded with unborated water, including all calculational uncertainties and biases, as described in Section 9.1 of the FSAR.

b. A nominal 6.26-inch center-to-center distance between fuel assemblies placed in the storage racks.

~~5.6.1.2 The k_{eff} for new fuel for the first core loading stored dry in the spent fuel storage racks shall not exceed 0.95 when flooded with water.~~ A.4

4.3.2 DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 819 feet.

4.3.3 CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 4078 fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7.1-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7.1-1. A.3

moved to ITS Section 5.5

TABLE 5.7.1-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor	120 heatup and cooldown cycles	70°F to 560°F to 70°F
	10,000 power change cycles	75% to 100% to 75% of RATED THERMAL POWER
	80 step change cycles	Loss of Feedwater heaters
	190 reactor trip cycles	100% to 0% of RATED THERMAL POWER
	2000 power change cycles	50% to 100% to 50% of RATED THERMAL POWER
	400 control rod pattern exchanges	Not applicable
	130 hydrostatic pressure tests	Pressurized to $\geq$ 930 psig and $\leq$ 1250 psig.

A.1

ITS Chapter 4.0

A.3

moved to ITS Section 5.5

DISCUSSION OF CHANGES
ITS: CHAPTER 4.0 - DESIGN FEATURES

ADMINISTRATIVE

A.1 In the conversion of the LaSalle 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-1434, Rev. 1 (i.e., the Improved Standard Technical Specifications (ISTS)).

A.2 CTS Figure 5.1.2-1, Low Population Zone, has been deleted since a description of the area has been provided. This figure and description continue to provide the information pertinent to 10 CFR 100 requirements. Since the requirements have not changed, this change is considered administrative.

A.3 The requirement in CTS 5.7 to maintain limits on component cyclic and transient stresses is being moved to ITS 5.5.5 in accordance with the format of the BWR ISTS, NUREG-1434, Revision 1. Any technical changes to this requirement will be addressed in the Discussion of Changes for ITS: Section 5.5.

A.4 (Unit 2 only) CTS 5.6.1.2 requires the k_{eff} for new fuel for the first core loading stored dry in the spent fuel storage racks to not exceed 0.95 when flooded with water. This requirement has been deleted, since LaSalle Unit 2 has completed the first core loading. Thus, this requirement is no longer applicable. This requirement has already been deleted from the LaSalle Unit 1 Technical Specifications for the same reason in Amendment 90 (NRC SER dated February 24, 1993). Therefore, since the requirement is no longer applicable, its removal from the Technical Specifications is administrative.

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 Primary containment configuration and design details in CTS 5.2.1, primary containment design temperatures and pressures in CTS 5.2.2, secondary containment design details in CTS 5.2.3, and the Reactor Coolant System design pressure and temperature and volume in CTS 5.4, 5.4.1, and 5.4.2 are proposed to be relocated to UFSAR, Sections 5.1, 5.2, 6.2.1, and 6.2.3. Any changes to these design parameters described in the UFSAR must conform to the

TECHNICAL CHANGES - LESS RESTRICTIVE

LA.1 (cont'd) 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 effect safety would require prior NRC review and approval. Since the features with a potential to effect safety are sufficiently addressed by LCOs, and other features, if altered in accordance with 10 CFR 50.59, would not result in a significant effect on safety, the criteria of 10 CFR 50.36(c)(4) for including as a Design 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.2 The nominal active control rod assembly absorber length described in CTS 5.3.2 is proposed to be relocated to the UFSAR, Section 4.2, 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> **4.1 Site Location** ~~[Text location of site location]~~ Insert SITE LOCATION 1

<5.3> **4.2 Reactor Core**

<5.3.1> **4.2.1 Fuel Assemblies**

764 clad 1

The reactor shall contain ~~[800]~~ fuel assemblies. Each assembly shall consist of a matrix of ~~[Zircalloy 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 applicable NRC staff approved codes and methods and 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.

2 Zircaloy, ZIRLO,

or water boxes

<5.3.2> **4.2.2 Control Rod Assemblies**

185 1

The reactor core shall contain ~~[193]~~ cruciform shaped control rod assemblies. The control material shall be ~~[~~boron carbide, and hafnium metal~~]~~ as approved by the NRC.

<5.6> **4.3 Fuel Storage**

4.3.1 Criticality

<5.6.1> 4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

3

~~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];~~

<5.6.1.a> a ~~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~~]~~; and

2 a 1 4

(continued)

<CTS>

Insert Site Location

4.1.1 Site and Exclusion Area Boundaries

<5.1.1>
<5.1.3>
<5.1.4>

The site and exclusion area boundaries are as shown in Figure 4.1-1.

4.1.2 Low Population Zone

<5.1.2>

The low population zone is all the land within a circle with its center at the vent stack and a radius of 3.98 miles.

<CTS>

4.0 DESIGN FEATURES

<5.6> **4.3 Fuel Storage (continued)**

1 <5.6.1.b>

4 ~~[c. A nominal fuel assembly center to center storage spacing of [7] inches within rows and [12.25] inches between rows in the [low density storage racks] in the upper containment pool; and]~~

2 3 6 [d. A nominal fuel assembly center to center storage spacing of [6.26] inches, within a neutron poison material between storage spaces, in the [high density storage racks] in the ~~spent fuel~~ storage pool ~~and in the upper containment pool~~.]

6.26 inch

distance between fuel assemblies placed

racks

5 ~~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 [6.26] inch center to center distance between fuel assemblies placed in storage racks.~~

<5.6.2> **4.3.2 Drainage**

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation [202] ft ~~5.25 inches~~.

819 1

<5.6.3> **4.3.3 Capacity**

6 ~~4.3.3.1~~ The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than [2324] fuel assemblies.

3986 1

for Unit 1 and 4078 fuel assemblies for Unit 2 1

(continued)

4.0 DESIGN FEATURES

~~4.3 Fuel Storage (continued)~~

~~4.3.3 Capacity~~

~~4.3.3.2 No more than [800] fuel assemblies may be stored in the upper containment pool.~~ 6

Insert Figure 4.1-1 1

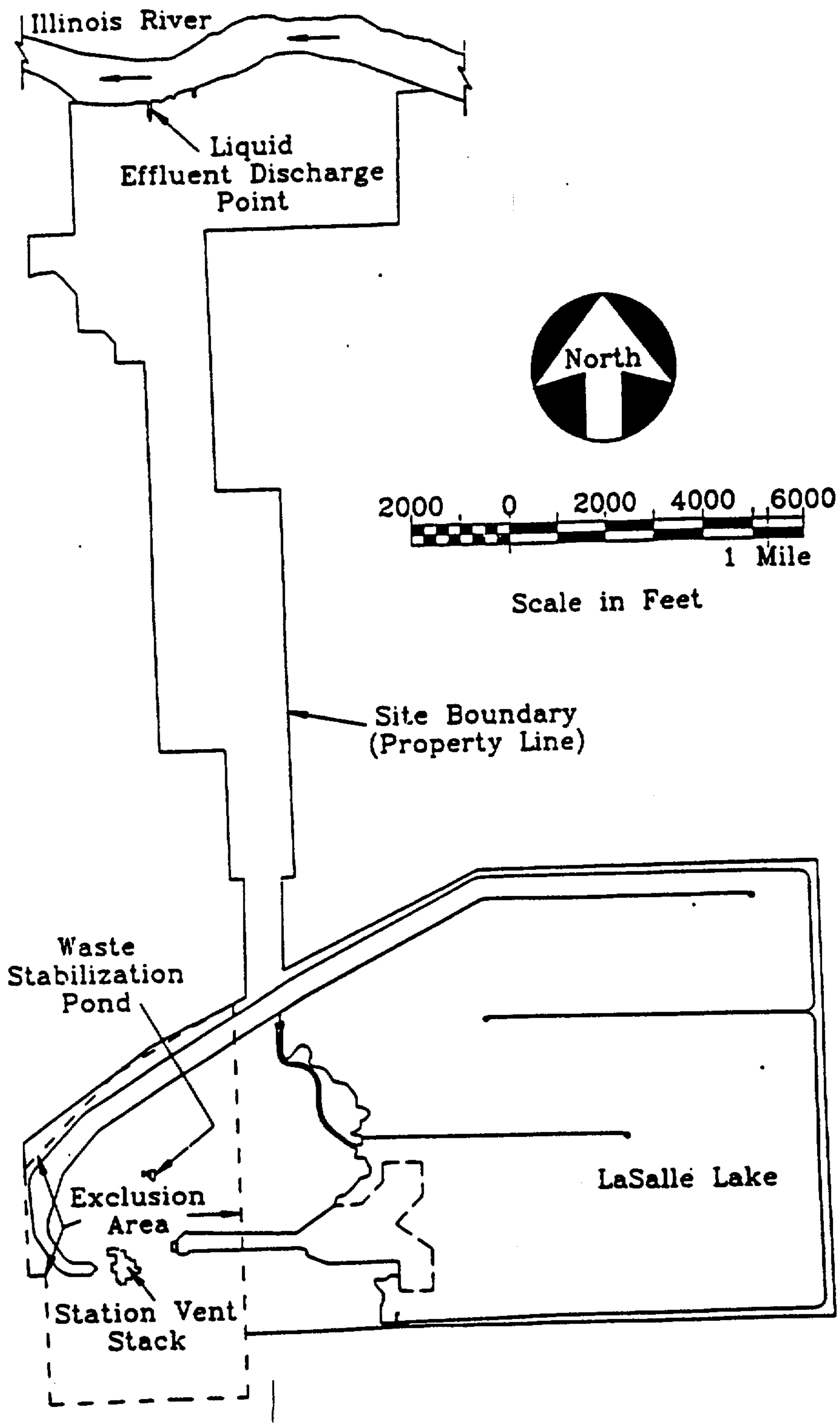


Figure 4.1-1 (Page 1 of 1)
Site and Exclusion Area Boundaries

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ITS: CHAPTER 4.0 - DESIGN FEATURES

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 requirement to specify the k_∞ or the average U-235 enrichment has been deleted. The current LaSalle 1 and 2 Technical Specifications, as well as NUREG-1434, Rev. 1 include a limit on k_{eff} for the spent fuel storage racks. In order to demonstrate compliance with this requirement, calculations have been performed, as described in the UFSAR, to determine the maximum k_{eff} of the racks. These calculations are dependent on the actual U-235 enrichment of the fuel stored in the racks. For ease of demonstrating compliance with the k_{eff} limit for the LaSalle 1 and 2 rack design, a bounding compliance criterion for each fuel type that can be stored in the new and spent fuel storage racks has been established such that the k_{eff} limit is still met. Because LaSalle 1 and 2 is required to maintain the $k_{eff} \leq 0.95$, each new fuel assembly loaded into the reactor must be compared to the storage racks bounding compliance criterion. This new limitation is a design feature of the fuel, not the racks. The limitations and requirements of the fuel is already provided in Specification 4.2.1. Design reviews for reloads will also verify continued compliance with the bounding requirements prior to storing the fuel in the new fuel storage racks and using the new fuel. This ensures continued compliance with the current k_{eff} limit for the new and spent fuel storage racks as required by the current LaSalle 1 and 2 Technical Specifications. In addition, this information is currently in the UFSAR. The following requirements have been renumbered, where applicable, to reflect this deletion.

4. This bracketed information has been deleted since it is not applicable to LaSalle 1 and 2.

5. The ISTS 4.3.1.2 new fuel storage requirements have been deleted. LaSalle 1 and 2 is consistent with the current licensing bases as provided in Amendment 90 (NRC SER dated February 24, 1993). This amendment deleted this requirement from the Unit 1 CTS, since the limit was only applicable to the first reload. In addition, this requirement has not been maintained in the Unit 2 ITS for the same reason. Subsequent requirements have been renumbered as applicable to reflect this change.

6. ISTS 4.3.3.2 has been deleted since it is not applicable to LaSalle 1 and 2. LaSalle 1 and 2 does not have an upper containment pool. The previous requirement (ISTS 4.3.3.1) has been renumbered to reflect this deletion.

GENERIC NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: CHAPTER 4.0 - DESIGN FEATURES

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-1434, 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 The 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:

a. A total of three non-licensed operators for the two units is required in all conditions. At least one of the required

(continued)

5.2 Organization

5.2.2 Unit Staff (continued)

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 the radiation protection manager who shall meet the requirements of "radiation protection manager" in Regulatory Guide 1.8, September 1975. Also, the ANSI N18.1-1971 qualification requirements for "radiation protection technician" may be met by either of the following alternatives:

a. Individuals who have completed the radiation protection technician training program and have accrued one year of working experience in the specialty; or

b. Individuals who have completed the radiation protection technician training program, but have not yet accrued one year of working experience in the specialty, who are supervised by on-shift radiation protection supervision who meet the requirements of ANSI N18.1-1971, Section 4.3.2 or Section 4.4.4.

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.

c. 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.

(continued)

5.5 Programs and Manuals

5.5.1 Offsite Dose Calculation Manual (ODCM) (continued)

Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

5.5.2 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 Low Pressure Core Spray, High Pressure Core Spray, Residual Heat Removal/Low Pressure Coolant Injection, Reactor Core Isolation Cooling, hydrogen recombiner, 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

(continued)

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

areas at or beyond the site boundary shall be in accordance with the following:

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;

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

k. Limitations on venting and purging of the primary containment through the Primary Containment Vent and Purge system or Standby Gas Treatment System to maintain releases as low as reasonably achievable.

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, Table 5.2-4, cyclic and transient occurrences to ensure that components are maintained within the design limits.

(continued)

5.5 Programs and Manuals

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, except that the Unit 1 and Unit 2 primary containments shall be treated as twin containments even though the initial structural integrity tests were not within 2 years of each other.

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

5.5.7 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:

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

(continued)

5.5 Programs and Manuals

5.5.7 Inservice Testing Program (continued)

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.

5.5.8 Ventilation Filter Testing Program (VFTP)

The VFTP shall establish the required testing of Engineered Safety Feature (ESF) filter ventilation systems. Tests described in Specification 5.5.8.a and 5.5.8.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 5.5.8.c shall be performed once per 24 months; after 720 hours of system operation; 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 5.5.8.d and 5.5.8.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 high efficiency particulate air (HEPA) filters shows a penetration and system bypass < 0.05% when tested in accordance with ANSI/ASME N510-1989 at the system flowrate specified below:

(continued)

5.5 Programs and Manuals

5.5.8 Ventilation Filter Testing Program (VFTP) (continued)

ESF Ventilation System	Flowrate(cfm)
Standby Gas Treatment (SGT) System	≥ 3600 and ≤ 4400
Control Room Area Filtration (CRAF) System Emergency Makeup Air Filter Units (EMUs)	≥ 3600 and ≤ 4400

b. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass less than the value specified below when tested in accordance with ANSI/ASME N510-1989 at the system flowrate specified below:

ESF Ventilation System	Penetration and System Bypass	Flowrate (cfm)
SGT System	0.05%	≥ 3600 and ≤ 4400
CRAF System		
EMUs	0.05%	≥ 3600 and ≤ 4400
Control Room Recirculation Filters(CRRFs)	2.0%	≥ 18000 and ≤ 28900
Auxiliary Electric Equipment Room Recirculation Filters (AEERRFs)	2.0%	≥ 14000 and ≤ 22800

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, a relative humidity of 70% and a face velocity as specified below:

(continued)

5.5 Programs and Manuals

5.5.8 Ventilation Filter Testing Program (VFTP) (continued)

ESF Ventilation System	Penetration	Face Velocity (fpm)
SGT System	0.5%	40
CRAF System		
EMUs	2.5%	40
CRRFs	15.0%	80
AEERRFs	15.0%	80

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

ESF Ventilation System	Delta P (inches WG)	Flowrate (cfm)
SGT System	8	$\geq$ 3600 and $\leq$ 4400
CRAF System		
EMUs	8	$\geq$ 3600 and $\leq$ 4400
CRRFs	3.0	$\geq$ 18000 and $\leq$ 28900
AEERRFs	3.0	$\geq$ 14000 and $\leq$ 22800

(continued)

5.5 Programs and Manuals

5.5.8 Ventilation Filter Testing Program (VFTP) (continued)

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

ESF Ventilation System	Wattage (kW)
SGT System	$\geq$ 21 and $\leq$ 25
CRAF System EMUs	$\geq$ 18 and $\leq$ 22

5.5.9 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the Condenser Offgas Treatment System and the quantity of radioactivity contained in any outside temporary tanks.

The program shall include:

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

b. A surveillance program to ensure that the quantity of radioactivity contained in all outside temporary 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 Waste Management Systems is less than the amount that would result in concentrations less than the limits specified in the ODCM, 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.

(continued)

5.5 Programs and Manuals

5.5.10 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,

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

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 ≤ 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.

5.5.11 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

(continued)

5.5 Programs and Manuals

5.5.11 Technical Specifications (TS) Bases Control Program (continued)

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 5.5.11.b.1 or 5.5.11.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 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.

(continued)

5.5 Programs and Manuals

5.5.12 Safety Function Determination Program (SFDP) (continued)

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

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.

5.5.13 Primary Containment Leakage Rate Testing Program

a. This program shall establish 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 exemptions. 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 39.6 psig.

(continued)

5.5 Programs and Manuals

5.5.13 Primary Containment Leakage Rate Testing Program (continued)

c. The maximum allowable primary containment leakage rage, L_a, at P_a, is 0.635% 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 are:

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

b) For each door, the seal leakage rate is $\leq$ 5 scf per hour when the gap between the door seals is pressurized to $\geq$ 10 psig.

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), electronic dosimeter, or film badge 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.

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/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.

3. The LHGR for Specification 3.2.3.

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

4. The Rod Block Monitor Upscale 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. ANFB Critical Power Correlation, ANF-1125(P)(A) and Supplements 1 and 2, Advanced Nuclear Fuels Corporation, April 1990.

2. Letter, Ashok C. Thadani (NRC) to R.A. Copeland (SPC), "Acceptance for Referencing of ULTRAFLOW™ Spacer on 9x9-IX/X BWR Fuel Design," July 28, 1993.

3. 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, XN-NF-524(P)(A) Revision 2 and Supplement 1 Revision 2, Supplement 2, Advanced Nuclear Fuels Corporation November 1990.

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

5. HUXY: A Generalized Multirod Heatup Code with 10 CFR 50, Appendix K Heatup Option, ANF-CC-33(P)(A), Supplement 1 Revision 1; and Supplement 2, Advanced Nuclear Fuels Corporation, August 1986 and January 1991, respectively.

6. Advanced Nuclear Fuel 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.

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

7. 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.

8. 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.

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

11. Volume 1 - STAIF - A Computer Program for BWR Stability Analysis in the Frequency Domain, Volume 2 - STAIF - A Computer Program for BWR Stability Analysis in the Frequency Domain, Code Qualification Report, EMF-CC-074(P)(A), Siemens Power Corporation, July 1994.

12. RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model, XN-NF-81-58(P)(A), Revision 2 Supplements 1 and 2, Exxon Nuclear Company, March 1984.

13. XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis, XN-NF-84-105(P)(A), Volume 1 and Volume 1 Supplements 1 and 2; Volume 1 Supplement 4, Advanced Nuclear Fuels Corporation, February 1987 and June 1988, respectively.

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

15. Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis, XN-NF-80-19(P)(A) Volume 1 and Supplements 1 ans 2, Exxon Nuclear Company, Richland, WA 99352, March 1983.

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

16. 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.

17. 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.

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

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

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

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

22. 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.

23. BWR Jet Pump Model Revision for RELAX, ANF-91-048(P)(A), Supplement 1 and Supplement 2, Siemens Power Corporation, October 1997.

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

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

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

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, 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 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 $>$ 100 mrem/hr and $\leq$ 1000 mrem/hr, provided they are otherwise following plant radiation protection procedures for entry into such 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 by the radiation protection manager in the RWP.

5.7.2 In addition to the requirements of Specification 5.7.1 for areas accessible to personnel with radiation levels such that a major portion of the body could receive in one hour a dose greater than 1000 mrem, the computer shall be programmed to permit entry through locked doors for any individual requiring access to any such high-high radiation areas for the time that access is required.

(continued)

5.7 High Radiation Area

5.7.3 Keys to manually open computer controlled high radiation area doors and high-high radiation area doors shall be maintained under the administrative control of the shift manager on duty or the radiation protection manager.

5.7.4 High-high radiation areas, as defined in Specification 5.7.2, not equipped with the computerized card readers shall be maintained in accordance with 10 CFR 20.1601(a)(3), locked except during periods when access to the area is required with positive control over each individual entry, or in the case of a high radiation area established for a period of 30 days or less, direct surveillance to prevent unauthorized entry may be substituted. Doors shall remain locked except during periods of access by personnel under an approved RWP which shall specify the dose rate levels in the immediate work area and the maximum allowable stay time for individuals in that area. For individual areas accessible to personnel with radiation levels such that a major portion of the body could receive in one hour a dose in excess of 1000 mrem that are located within large areas, such as the containment, where no enclosure exists for purposes of locking, and no enclosure can be reasonably constructed around the individual areas, then that area shall be roped off, conspicuously posted and a flashing light shall be activated as a warning device. In lieu of the stay time specification of the RWP, direct or remote, such as use of closed circuit TV cameras, continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities within the area.

5.0 6.0 ADMINISTRATIVE CONTROLS

5.1 6.1 RESPONSIBILITY ~~ORGANIZATION~~

A.1

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 the safety of the nuclear power plant.

<SEE ITS 5.2>

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.

M.1

5.1.1 2. The individual filling the ANSI N18.1-1971 Section 4.2.1 position of Plant Manager ("~~Plant~~ 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. and shall delegate in writing the succession to this responsibility during his absence

LA.1

<SEE ITS 5.2>

3. 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.

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

LA.2

~~B. The Shift Manager shall be responsible for directing and commanding the overall operation of the facility on his shift. The primary management responsibility of the Shift Manager shall be for safe operation of the nuclear facility on his shift under all conditions.~~

<SEE ITS 5.2>

C. The shift manning for the station shall be as shown in Figure 6.1-3.

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FIGURE 6.1-3
MINIMUM SHIFT CREW COMPOSITION(a)(c)

POSITION(b)	MINIMUM CREW NUMBER		
	EACH UNIT IN CONDITION 1, 2, OR 3	ONE UNIT IN CONDITION 1, 2, OR 3, AND ONE UNIT IN CONDITION 4 OR 5 OR DEFUELED	EACH UNIT IN CONDITION 4 OR 5 OR DEFUELED
SM	1	1	1
SRO	1	1	None
RO	3	3	2
AO	3	3	3
STA(d)	1	1	None

(a) This table reflects the total requirements for shift staffing of both units.

SEE ITS 5.2

With the exception of the Shift Manager, the shift crew composition may be one less than the minimum requirements of Figure 6.1-3 for not more than 2 hours to accommodate unexpected absence of on-duty shift crew members, provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Figure 6.1-3. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

(b) Table Notation:

SM Shift Manager with a Senior Reactor Operator license for each unit whose reactor contains fuel.

SRO Individual with a Senior Reactor Operator license for each unit whose reactor contains fuel.

During CORE ALTERATIONS on either unit a licensed SRO or licensed SRO limited to fuel handling, who has no other concurrent responsibilities, must be present to observe and directly supervise this operation.

RO An Individual with a Reactor Operator license or a Senior Reactor Operator license for unit assigned. At least one RO shall be assigned to each unit whose reactor contains fuel. Individuals acting as relief operators shall hold a license for both units. Otherwise, for each unit, provide a relief operator who holds a license for the unit assigned.

AO At least one auxiliary operator shall be assigned to each unit whose reactor contains fuel.

STA Shift Technical Advisor.

5.1.2 (c) While either unit is in CONDITION 1, 2, or 3, an individual with a valid SRO license shall be designated to assume the control room command function. With both Units in CONDITION 4 or 5 or defueled, an individual with a valid SRO or RO license shall be designated to assume the control room command function.

A.2

SEE ITS 5.2 (d) The STA position shall be filled by an individual who meets the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift.

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5.0 6.0 ADMINISTRATIVE CONTROLS

5.1 6.1 ~~ORGANIZATION~~ RESPONSIBILITY

A.1

ITS 5.1

SEE ITS 5.2

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 the safety of the nuclear power plant.

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.

M.1 5.1.1

and shall delegate in writing the succession to this responsibility during his absence

LA.1

2. The individual filling the ANSI N18.1-1971 Section 4.2.1 position of Plant Manager ("~~Plant~~ Manager"), shall be responsible for overall unit safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant. station

SEE ITS 5.2

3. 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.

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

~~B. The Shift Manager shall be responsible for directing and commanding the overall operation of the facility on his shift. The primary management responsibility of the Shift Manager shall be for safe operation of the nuclear facility on his shift under all conditions.~~

LA.2

SEE ITS 5.2

C. The shift manning for the station shall be as shown in Figure 6.1-3.

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FIGURE 6.1-3
MINIMUM SHIFT CREW COMPOSITION[(a)(c)]

POSITION[(b)]	MINIMUM CREW NUMBER		
	EACH UNIT IN CONDITION 1, 2, OR 3	ONE UNIT IN CONDITION 1, 2, OR 3, AND ONE UNIT IN CONDITION 4 OR 5 OR DEFUELED	EACH UNIT IN CONDITION 4 OR 5 OR DEFUELED
SM	1	1	1
SRO	1	1	None
RO	3	3	2
AO	3	3	3
STA[(d)]	1	1	None

<SEE ITS 5.2>

(a) This table reflects the total requirements for shift staffing of both units.

With the exception of the Shift Manager, the shift crew composition may be one less than the minimum requirements of Figure 6.1-3 for not more than 2 hours to accommodate unexpected absence of on-duty shift crew members, provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Figure 6.1-3. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

(b) Table Notation:

SM Shift Manager with a Senior Reactor Operator license for each unit whose reactor contains fuel.

SRO Individual with a Senior Reactor Operator license for each unit whose reactor contains fue

During CORE ALTERATIONS on either unit a licensed SRO or licensed SRO limited to fuel handling, who has no other concurrent responsibilities, must be present to observe and directly supervise this operation.

RO An Individual with a Reactor Operator license or a Senior Reactor Operator license for uni assigned. At least one RO shall be assigned to each unit whose reactor contains fuel. Individuals acting as relief operators shall hold a license for both units. Otherwise, for eac unit, provide a relief operator who holds a license for the unit assigned.

AO At least one auxiliary operator shall be assigned to each unit whose reactor contains fuel.

STA Shift Technical Advisor.

5.1.2 (c) While either unit is in CONDITION 1, 2, or 3, an individual with a valid SRO license shall be designated to assume the control room command function. With both Units in CONDITION 4 or 5, or defueled an individual with a valid SRO or RO license shall be designated to assume the control room command function. A.2

<SEE ITS 5.2> (d) The STA position shall be filled by an individual who meets the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift.

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ADMINISTRATIVE

A.1 In the conversion of the LaSalle 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 1434, Rev. 1 (i.e., the Improved Standard Technical Specifications (ISTS)).

A.2 Footnote (c) of CTS Figure 6.1-3 requires an individual with an SRO or RO license to be designated to assume the control room command function. The condition of defueled has been added in proposed TS 5.1.2. This requirement is consistent with current plant practice and ensures all possible conditions in which licensed personnel are required are covered. Since this omission is essentially an oversight in the CTS, the change is considered administrative.

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 Proposed ITS 5.1.1 requires the plant manager to delegate in writing the succession of the responsibility for overall plant operations during his absence. This change is in addition to the responsibility currently required by the CTS, and is consistent with the BWR ITS, NUREG-1434, Rev. 1. Therefore, this more restrictive change is acceptable.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 CTS 6.1.A.2 uses the title "Plant 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. 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.

TECHNICAL CHANGES - LESS RESTRICTIVE (continued)

LA.2 CTS 6.1.B delineates the responsibility of the Shift Manager for directing and commanding the overall operation of the facility on his shift. 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, inclusion of the detailed responsibilities of the Shift Manager 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.

"Specific"

None

RELOCATED SPECIFICATIONS

None

5.0 6.0 ADMINISTRATIVE CONTROLS

5.2 6.1 ORGANIZATION A.1 ITS 5.2

5.2.1 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 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, 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. LA-1

5.2.1.b 2. The individual filling the ANSI N18.1-1971 Section 4.2.1 position of Plant Manager ("~~Plant~~ Manager"), A corporate officer 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. station LA.1

5.2.1.c 3. ~~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. A.2

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.

SEE ITS 5.1 B. The Shift Manager shall be responsible for directing and commanding the overall operation of the facility on his shift. The primary management responsibility of the Shift Manager shall be for safe operation of the nuclear facility on his shift under all conditions.

5.2.2 C. The shift manning for the station shall be as shown in Figure 6.1-3.

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A.1

ITS 5.2

5.2.2.b 1. At least one licensed Reactor Operator shall be in the control room when fuel is in the reactor. In addition, while the reactor is in OPERATIONAL CONDITION 1, 2 or 3, at least one licensed Senior Reactor Operator who has been designated by the Shift Manager to assume the control room direction responsibility shall be in the Control Room.

5.2.2.d 2. A radiation protection technician* shall be on site when fuel is in the reactor.

LA.2

~~3. All CORE ALTERATIONS shall be observed and directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation.~~

4. DELETED

LA.3

~~5. The Independent Safety Engineering Group (ISEG) shall function to examine unit operating characteristics, NRC issuances, industry advisories, Licensee Event Reports and other sources of plant design and operating experience information, including plants of similar design, which may indicate areas for improving unit safety. The ISEG shall be composed of at least three, dedicated, full-time engineers of multi-disciplines located on site and shall be augmented on a part-time basis by personnel from other parts of the Commonwealth Edison Company organization to provide expertise not represented in the group. The ISEG shall be responsible for maintaining surveillance of unit activities to provide independent verification# that these activities are performed correctly and that human errors are reduced as much as practical. The ISEG shall make detailed recommendations for revised procedures, equipment modifications, maintenance activities, operations activities or other means of improving unit safety to the Manager of Quality and Safety Assessment and the Plant Manager.~~

shift manager

5.2.2.g 6. The Shift Technical Advisor 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.

LA.1

5.2.2.d * The radiation protection technician position may be less than the minimum requirement for a period of time not to exceed two hours in order to accommodate unexpected absence provided immediate action is taken to fill the required position.

~~# Not responsible for sign-off feature.~~

LA.3

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5.2.2.e 7. 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).

5.2.2.f 8. The operations manager or shift operations supervisor shall hold a Senior Reactor Operator License. LA.1

<SEE ITS 5.3>

D. Qualifications of the station management and operating staff shall meet minimum acceptable levels as described in ANSI N18.1, "Selection and Training of Nuclear Power Plant Personnel," dated March 8, 1971. The Health Physics Supervisor shall meet the requirements of radiation protection manager of Regulatory Guide 1.8, September 1975. The ANSI N18.1-1971 qualification requirements for Radiation Protection Technician may also be met by either of the following alternatives:

1. Individuals who have completed the Radiation Protection Technician training program and have accrued 1 year of working experience in the specialty, or

2. Individuals who have completed the Radiation Protection Technician training program, but have not yet accrued 1 year of working experience in the specialty, who are supervised by on-shift health physics supervision who meet the requirements of ANSI N18.1-1971 Section 4.3.2, "Supervisor Not Requiring AEC Licenses," or Section 4.4.4, "Radiation Protection."

<See CTS 6.1.E/F>

E. Retraining and replacement training of Station personnel shall be in accordance with ANSI N18.1, "Selection and Training of Nuclear Power Plant Personnel", dated March 8, 1971 and Appendix "A" of 10 CFR Part 55, and shall include familiarization with relevant industry operational experience.

F. Retraining shall be conducted at intervals not exceeding 2 years.

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G. ~~DELETED (The Review and Investigative Function and the Audit Function are described in the Quality Assurance Manual Topical Report CE-1-A).~~

FIGURE 6.1-3
MINIMUM SHIFT CREW COMPOSITION(a)(c)

A.1 ITS 5.2

POSITION(b)	MINIMUM CREW NUMBER		
	EACH UNIT IN CONDITION 1, 2, OR 3	ONE UNIT IN CONDITION 1, 2, OR 3, AND ONE UNIT IN CONDITION 4 OR 5 OR DEFUELED	EACH UNIT IN CONDITION 4 OR 5 OR DEFUELED
SM	1	1	1
SRO	1	1	None
RO	3	3	2
AO	3	3	3
STA(d)	1	1	None

LA.4

5.2.2.a

5.2.2.a (a) This table reflects the total requirements for shift staffing of both units.

5.2.2.c With the exception of the Shift Manager, the shift crew composition may be one less than the minimum requirements of Figure 6.1-3 for not more than 2 hours to accommodate unexpected absence of on-duty shift crew members, provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Figure 6.1-3. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent. A.3

(b) Table Notation: LA.4

SM Shift Manager with a Senior Reactor Operator license for each unit whose reactor contains fuel.

SRO Individual with a Senior Reactor Operator license for each unit whose reactor contains fuel.

During CORE ALTERATIONS on either unit a licensed SRO or licensed SRO limited to fuel handling, who has no other concurrent responsibilities, must be present to observe and directly supervise this operation. LA.2

LA.4

5.2.2.b RO An Individual with a Reactor Operator license or a Senior Reactor Operator license for unit assigned. At least one RO shall be assigned to each unit whose reactor contains fuel. Individuals acting as relief operators shall hold a license for both units. Otherwise, for each unit, provide a relief operator who holds a license for the unit assigned. LA.4

5.2.2.a AO At least one auxiliary operator shall be assigned to each unit whose reactor contains fuel. M.1

STA Shift Technical Advisor LA.4

SEE ITS 5.1 (c) While either unit is in CONDITION 1, 2, or 3, an individual with a valid SRO license shall be designated to assume the control room command function. With both Units in CONDITION 4 or 5 an individual with a valid SRO or RO license shall be designated to assume the control room command function.

5.2.2.g (d) The STA position shall be filled by an individual who meets the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift.

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5.0 6.0 ADMINISTRATIVE CONTROLS

5.2 6.1 ORGANIZATION A.1 ITS 5.2

5.2.1 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 the safety of the nuclear power plant. LA.1

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.

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

5.2.1.b 2. The individual filling the ANSI N18.1-1971 Section 4.2.1 position of Plant Manager ("~~Plant~~ Manager"), shall be responsible for overall unit safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

A corporate officer station LA.1

5.2.1.c 3. ~~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.

radiation protection A.2

5.2.1.d 4. The individuals who train the operating staff and those who carry out ~~health physics~~ and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

<SEE ITS 5.1> B. The Shift Manager shall be responsible for directing and commanding the overall operation of the facility on his shift. The primary management responsibility of the Shift Manager shall be for safe operation of the nuclear facility on his shift under all conditions.

5.2.2 C. The shift manning for the station shall be as shown in Figure 6.1-3.

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A.1

ITS 5.2

5.2.2.b 1. At least one licensed Reactor Operator shall be in the control room when fuel is in the reactor. In addition, while the reactor is in OPERATIONAL CONDITION 1, 2 or 3, at least one licensed Senior Reactor Operator who has been designated by the Shift Manager to assume the control room direction responsibility shall be in the Control Room.

5.2.2.d 2. A radiation protection technician* shall be on site when fuel is in the reactor.

~~3. All CORE ALTERATIONS shall be observed and directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation.~~ LA.2

4. DELETED

LA.3

~~5. The Independent Safety Engineering Group (ISEG) shall function to examine unit operating characteristics, NRC issuances, industry advisories, Licensee Event Reports and other sources of plant design and operating experience information, including plants of similar design, which may indicate areas for improving unit safety. The ISEG shall be composed of at least three, dedicated, full-time engineers of multi-disciplines located on site and shall be augmented on a part-time basis by personnel from other parts of the Commonwealth Edison Company organization to provide expertise not represented in the group. The ISEG shall be responsible for maintaining surveillance of unit activities to provide independent verification# that these activities are performed correctly and that human errors are reduced as much as practical. The ISEG shall make detailed recommendations for revised procedures, equipment modifications, maintenance activities, operations activities or other means of improving unit safety to the Manager of Quality and Safety Assessment and the Plant Manager.~~

5.2.2.g 6. The Shift Technical Advisor 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. LA.1

shift manager

5.2.2.d * The radiation protection technician position may be less than the minimum requirement for a period of time not to exceed two hours in order to accommodate unexpected absence provided immediate action is taken to fill the required position.

~~# Not responsible for sign-off feature.~~ LA.3

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5.2.2.e 7. 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).

5.2.2.f 8. The ~~O~~perations ~~M~~anager or ~~S~~hift ~~O~~perations ~~S~~upervisor shall hold a Senior Reactor Operator License. } LA.1

<SEE ITS 5.3>

D. Qualifications of the station management and operating staff shall meet minimum acceptable levels as described in ANSI N18.1, "Selection and Training of Nuclear Power Plant Personnel," dated March 8, 1971. The Health Physics Supervisor shall meet the requirements of radiation protection manager of Regulatory Guide 1.8, September 1975. The ANSI N18.1-1971 qualification requirements for Radiation Protection Technician may also be met by either of the following alternatives:

1. Individuals who have completed the Radiation Protection Technician training program and have accrued 1 year of working experience in the specialty, or

2. Individuals who have completed the Radiation Protection Technician training program, but have not yet accrued 1 year of working experience in the specialty, who are supervised by on-shift health physics supervision who meet the requirements of ANSI N18.1-1971 Section 4.3.2, "Supervisor Not Requiring AEC Licenses," or Section 4.4.4, "Radiation Protection."

<See CTS 6.1.E/F>

E. Retraining and replacement training of Station personnel shall be in accordance with ANSI N18.1, "Selection and Training of Nuclear Power Plant Personnel", dated March 8, 1971 and Appendix "A" of 10 CFR Part 55, and shall include familiarization with relevant industry operational experience.

F. Retraining shall be conducted at intervals not exceeding 2 years.

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~~6. DELETED (The Review and Investigative Function and the Audit Function are described in the Quality Assurance Manual Topical Report CE-1-A).~~

FIGURE 6.1-3 A.1 ITS 5.2

MINIMUM SHIFT CREW COMPOSITION(a)(c)

POSITION(b)	MINIMUM CREW NUMBER		
	EACH UNIT IN CONDITION 1, 2, OR 3	ONE UNIT IN CONDITION 1, 2, OR 3, AND ONE UNIT IN CONDITION 4 OR 5 OR DEFUELED	EACH UNIT IN CONDITION 4 OR 5 OR DEFUELED
SM	1	1	1
SRO	1	1	None
RO	3	3	2
AO	3	3	3
STA(d)	1	1	None

LA.4

5.2.2.a {

LA.4

5.2.2.a (a) This table reflects the total requirements for shift staffing of both units.

5.2.2.c With the exception of the Shift Manager, the shift crew composition may be one less than the minimum requirements of Figure 6.1-3 for not more than 2 hours to accommodate unexpected absence of on-duty shift crew members, provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Figure 6.1-3. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent. A.3

(b) Table Notation: LA.4

SM Shift Manager with a Senior Reactor Operator license for each unit whose reactor contains fuel.

SRO Individual with a Senior Reactor Operator license for each unit whose reactor contains fuel.

During CORE ALTERATIONS on either unit a licensed SRO or licensed SRO limited to fuel handling, who has no other concurrent responsibilities, must be present to observe and directly supervise this operation. LA.2

5.2.2.b { RO An Individual with a Reactor Operator license or a Senior Reactor Operator license for unit assigned. At least one RO shall be assigned to each unit whose reactor contains fuel. Individuals acting as relief operators shall hold a license for both units. Otherwise, for each unit, provide a relief operator who holds a license for the unit assigned. LA.4

5.2.2.a AO At least one auxiliary operator shall be assigned to each unit whose reactor contains fuel. M.1

STA Shift Technical Advisor. LA.4

<SEE ITS 5.1> (c) While either unit is in CONDITION 1, 2, or 3, an individual with a valid SRO license shall be designated to assume the control room command function. With both Units in CONDITION 4 or 5 an individual with a valid SRO or RO license shall be designated to assume the control room command function.

5.2.2.g (d) The STA position shall be filled by an individual who meets the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift.

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ADMINISTRATIVE

A.1 In the conversion of the LaSalle 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 1434, Rev. 1 (i.e., the Improved Standard Technical Specifications (ISTS)).

A.2 The term "health physics" in CTS 6.1.A.4 has been changed to radiation protection. This terminology is equivalent. Thus, the change is administrative.

A.3 Footnote (a) of CTS Table 6.1-3 does not allow any shift crew position to be unmanned upon shift change because an oncoming shift crewman scheduled to come on duty is late or absent. ITS 5.2.2.c allows a period of time not to exceed two hours in order to accommodate unexpected absence of "on-duty" shift crew members or personnel. The wording "on-duty," implies that the absence refers to on-duty shift crew members or personnel and not the oncoming crew or personnel. If anyone in the oncoming crew or personnel is not present, the "on-duty" person may not leave. Therefore, the requirement of this footnote is covered in ITS 5.2.2.c. The minimum shift crew requirements continue to be maintained in ITS 5.2.2.c. Therefore, the deletion of this portion of the footnote is administrative.

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 The wording in CTS Table 6.1-3 footnote (b) has 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 at all times, in lieu of the CTS requirement that the non-licensed operator be assigned only when fuel is in the reactor vessel. 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-1434, Rev. 1, and is considered more restrictive on plant operations.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 CTS 6.1.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.A.1.2 uses the title "Plant Manager." In ITS 5.2.1.b, this specific title is replaced with the generic title "station manager." CTS 6.1.C.6 uses the title "Shift Manager." In ITS 5.2.2.g, this specific title is replaced with the generic term "shift manager." CTS 6.1.C.8 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. 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 Details contained in CTS 6.1.C.3 and Figure 6.1-3, footnote (b) that require all Core Alterations to be supervised by either a licensed Senior Operator or Senior Reactor Operator Limited to Fuel Handling are proposed to be relocated to the UFSAR. These current TS requirements are contained in 10 CFR 50.54 (m)(2)(iv) and do not need to be repeated in the ITS to provide adequate protection of the public health and safety. Once in the UFSAR, these requirements will be under the change control provisions of 10 CFR 50.59. 10 CFR 50.54 (m)(2)(iv) specifies the minimum requirements for moving reactor fuel. It does not require a non-licensed member of the reactor analyst group (or any other type of engineer) to monitor the fuel movement. This is an additional administrative requirement that is not needed to be in the ITS for protection of the public health and safety. Once in the UFSAR, this requirement will also be under the change control provisions of 10 CFR 50.59.

LA.3 The Independent Safety Engineering Group (ISEG) requirements in CTS 6.1.C.5 are proposed to be relocated to the Quality Assurance (QA) Manual since they can be adequately addressed elsewhere and there is adequate regulatory authority to do so. The ISEG performs independent safety reviews. Since the ISEG provides after-the-fact recommendations to improve safety, this organization is

TECHNICAL CHANGES - LESS RESTRICTIVE

LA.3 (cont'd) not necessary to ensure safe operation of the facility. Therefore, inclusion of the requirements for the ISEG in ITS is not necessary to provide adequate protection of the public health and safety. Changes to the QA Manual will be controlled by the provisions of 10 CFR 50.54.

LA.4 Details of the minimum shift crew requirements located in CTS Table 6.1-3, including portions of footnotes (a) and (b), are proposed to be relocated to the UFSAR. The minimum shift crew requirements for licensed operators and senior operators are also contained in 10 CFR 50.54 (k), (l), and (m) and do not need to be repeated in the ITS. The minimum shift crew requirements for non-licensed plant equipment operators are transferred from CTS Table 6.1-3 to ITS 5.2.2.a. In addition, ITS 5.1.2 contains requirements for the control room command function, ITS 5.2.2.c contains minimum requirements for licensed Reactor Operators and Senior Operators to be present in the control room, and ITS 5.2.2.g contains STA requirements. The relocation of the details of the minimum shift crew requirements to the UFSAR is acceptable considering the controls provided by regulations, the remaining requirements in the ITS, and the UFSAR change control process (10 CFR 50.59). Therefore, the relocated requirements are not required to be in the ITS to provide adequate protection of the public health and safety.

"Specific"

None

RELOCATED SPECIFICATIONS

None

SEE ITS 5.2

7. 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).

8. The Operations Manager or Shift Operations Supervisor shall hold a Senior Reactor Operator License.

LA.2

5.3.1 D. Qualifications of the ~~station management and~~ ~~operating~~ unit staff shall meet minimum acceptable levels as described in ANSI N18.1, "Selection and Training of Nuclear Power Plant Personnel," dated March 8, 1971. The Health Physics radiation protection Supervisor shall meet the requirements of radiation protection manager of Regulatory Guide 1.8, September 1975. The ANSI N18.1-1971 qualification requirements for Radiation Protection Technician may also be met by either of the following alternatives: A.1 LA.1

5.3.1.a 1. Individuals who have completed the Radiation Protection Technician training program and have accrued 1 year of working experience in the specialty, or LA.1

5.3.1.b 2. Individuals who have completed the Radiation Protection Technician training program, but have not yet accrued 1 year of working experience in the specialty, who are supervised by on-shift health physics supervision who meet the requirements of ANSI N18.1-1971 Section 4.3.2, "Supervisor Not Requiring AEC Licenses," or Section 4.4.4, "Radiation Protection." LA.1

E. Retraining and replacement training of Station personnel shall be in accordance with ANSI N18.1, "Selection and Training of Nuclear Power Plant Personnel", dated March 8, 1971 and Appendix "A" of 10 CFR Part 55, and shall include familiarization with relevant industry operational experience.

F. Retraining shall be conducted at intervals not exceeding 2 years.

See CTS 6.1.E/F

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See ITS 5.2

7. 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).

8. The Operations Manager or Shift Operations Supervisor shall hold a Senior Reactor Operator License. LA.2

5.3.1 D. Qualifications of the ~~station management and operating~~ unit staff shall meet minimum acceptable levels as described in ANSI N18.1, "Selection and Training of Nuclear Power Plant Personnel," dated March 8, 1971. A.1 The ~~Health Physics~~ radiation protection Supervisor shall meet the requirements of radiation protection manager of Regulatory Guide 1.8, September 1975. The ANSI N18.1-1971 qualification requirements for Radiation Protection Technician may also be met by either of the following alternatives: LA.1

5.3.1.a 1. Individuals who have completed the Radiation Protection Technician training program and have accrued 1 year of working experience in the specialty, or LA.1

5.3.1.b 2. Individuals who have completed the Radiation Protection Technician training program, but have not yet accrued 1 year of working experience in the specialty, who are supervised by on-shift health physics supervision who meet the requirements of ANSI N18.1-1971 Section 4.3.2, "Supervisor Not Requiring AEC Licenses," or Section 4.4.4, "Radiation Protection." LA.1

E. Retraining and replacement training of Station personnel shall be in accordance with ANSI N18.1, "Selection and Training of Nuclear Power Plant Personnel", dated March 8, 1971 and Appendix "A" of 10 CFR Part 55, and shall include familiarization with relevant industry operational experience.

F. Retraining shall be conducted at intervals not exceeding 2 years.

See CTS 6.1.E/F

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ADMINISTRATIVE

A.1 In the conversion of the LaSalle 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 1434, 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.D uses the plant specific titles "Health Physics Supervisor" and "Radiation Protection Technician." In ITS 5.3.1, these specific titles are replaced with generic titles "radiation protection manager" and "radiation protection technician." The specific title is proposed to be relocated to the Quality Assurance (QA) Manual. 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.

LA.2 CTS 6.1.D states that the qualifications of the station management shall meet those described in ANSI N18.1-1971. This requirement is proposed to be relocated to the QA Manual. ITS 5.3.1 continues to require each member of the unit staff to meet the qualifications of ANSI N18.1-1971. Since plant management is not directly involved with the operation of the facility, their qualifications are not required to be in the ITS to ensure adequate protection of the health and safety of the public. 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

<SEE ITS 5.7>

HIGH RADIATION AREAS (Continued)

individual areas accessible to personnel with radiation levels such that a major portion of the body could receive in one hour a dose in excess of 1000 mrem* that are located within large areas, such as the containment, where no enclosure exists for purposes of locking, and no enclosure can be reasonably constructed around the individual areas, then that area shall be roped off, conspicuously posted and a flashing light shall be activated as a warning device. In lieu of the stay time specification of the RWP, direct or remote, such as use of closed circuit TV cameras, continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities within the area.

5.4 6.2 PLANT OPERATING PROCEDURES AND PROGRAMS <See ITS 5.5>

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

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

5.4.1.b b. The emergency operating procedures required to implement the requirements of NUREG-0737 and Supplement 1 to NUREG-0737 as stated in Section 7.1 of Generic Letter No. 82-33,

~~c. Station Security Plan implementation,~~

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

~~e. PROCESS CONTROL PROGRAM implementation,~~ LA.1

~~f. OFFSITE DOSE CALCULATION MANUAL implementation, and~~ A.3

5.4.1.c g. Fire Protection Program implementation.

Add proposed TS 5.4.1.d M.1

<SEE ITS 5.7> *Measurement made at 18" from source of radioactivity.

ITS 5.4

5.4 PLANT OPERATING PROCEDURES AND PROGRAMS (Continued) <See ITS 5.5>

B. Radiation control procedures shall be maintained, made available to all station personnel, and adhered to. These procedures shall show permissible radiation exposure and shall be consistent with the requirements of 10 CFR 20. This radiation protection program shall be organized to meet the requirements of 10 CFR 20. <See CTS 6.2.B>

LA.2

C. TECHNICAL REVIEW AND CONTROL

Procedures required by Specification 6.2.A and 6.2.B and other procedures which affect nuclear safety, as determined by the Plant Manager, and changes thereto, other than editorial or typographical changes, shall be reviewed as follows prior to implementation except as noted in Specification 6.2.D:

1. Each procedure or procedure change shall be independently reviewed by a qualified individual knowledgeable in the area affected other than the individual who prepared the procedure or procedure change. This review shall include a determination of whether or not additional cross-disciplinary reviews are necessary. If deemed necessary, the reviews shall be performed by the qualified review personnel of the appropriate discipline(s).

2. Individuals performing these reviews shall meet the applicable experience requirements of ANSI N18.1-1971, Sections 4.2, 4.3, 4.4, 4.5.1, or 4.6, and be approved by the Plant Manager.

3. Applicable Administrative Procedures recommended by Regulatory Guide 1.33, Plant Emergency Operating Procedures, and changes thereto shall be submitted to the Onsite Review and Investigative Function for review and approval prior to implementation.

4. Review of the procedure or procedure change will include a determination of whether or not an unreviewed safety question is involved. This determination will be based on the review of a written safety evaluation prepared by a qualified individual or documentation that a safety evaluation is not required. Onsite Review, Offsite Review and Commission approval of items involving unreviewed safety questions shall be obtained prior to Station approval for implementation.

5. The Department Head approval authority shall be specified in station procedures.

6. Written records of reviews performed in accordance with this specification shall be prepared and maintained in accordance with Specification 6.5.

7. Editorial and Typographical changes shall be made in accordance with station procedures.

~~D. Temporary changes to procedures 6.2.A and 6.2.B above may be made provided:~~

~~1. The intent of the original procedure is not altered.~~

~~2. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.~~

~~3. The change is documented, reviewed and approved in accordance with Specification 6.2.C, within 14 days of implementation.~~

LA.2

~~E. Drills of the emergency procedures described in Specification 6.2.A.d shall be conducted at frequencies as specified in the Generating Stations Emergency Plan (GSEP). These drills will be planned so that during the course of the year, communication links are tested and outside agencies are contacted.~~

A.4

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

1. Primary Coolant Sources Outside Primary Containment

A program to reduce 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 LPCS, HPCS, RHR/LPCI, RCIC, hydrogen recombiner, process sampling, containment monitoring, and standby gas treatment systems. The program shall include the following:

a. Preventive maintenance and periodic visual inspection requirements, and

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

SEE ITS 5.5

2. In-Plant Radiation Monitoring

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

a. Training of personnel,

b. Procedures for monitoring, and

c. Provisions for maintenance of sampling and analysis equipment.

3. Post-accident Sampling

A program which will 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,

c. Provisions for maintenance of sampling and analysis equipment.

HIGH RADIATION AREAS (Continued)

<SEE ITS 5.7> individual areas accessible to personnel with radiation levels such that a major portion of the body could receive in one hour a dose in excess of 1000 mrem* that are located within large areas, such as the containment, where no enclosure exists for purposes of locking, and no enclosure can be reasonably constructed around the individual areas, then that area shall be roped off, conspicuously posted and a flashing light shall be activated as a warning device. In lieu of the stay time specification of the RWP, direct or remote, such as use of closed circuit TV cameras, continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities within the area.

5.4 6.2 PLANT OPERATING PROCEDURES AND PROGRAMS <See ITS 5.5>

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

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

5.4.1.b b. The emergency operating procedures required to implement the requirements of NUREG-0737 and Supplement 1 to NUREG-0737 as stated in Section 7.1 of Generic Letter No. 82-33,

~~c. Station Security Plan implementation,~~

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

~~e. PROCESS CONTROL PROGRAM implementation,~~ LA.1

~~f. OFFSITE DOSE CALCULATION MANUAL implementation, and~~ A.3

5.4.1.c g. Fire Protection Program implementation.

Add proposed TS 5.4.1.d M.1

<SEE ITS 5.7> *Measurement made at 18" from source of radioactivity.

 (See ITS 5.5)

5.4 PLANT OPERATING PROCEDURES AND PROGRAMS (Continued) A.1 ITS 5.4

B. Radiation control procedures shall be maintained, made available to all station personnel, and adhered to. These procedures shall show permissible radiation exposure and shall be consistent with the requirements of 10 CFR 20. This radiation protection program shall be organized to meet the requirements of 10 CFR 20. (see CTS 6.2.B)

C. ~~TECHNICAL REVIEW AND CONTROL~~ LA.2

~~Procedures required by Specification 6.2.A and 6.2.B and other procedures which affect nuclear safety, as determined by the Plant Manager, and changes thereto, other than editorial or typographical changes, shall be reviewed as follows prior to implementation except as noted in Specification 6.2.D:~~

1. ~~Each procedure or procedure change shall be independently reviewed by a qualified individual knowledgeable in the area affected other than the individual who prepared the procedure or procedure change. This review shall include a determination of whether or not additional cross-disciplinary reviews are necessary. If deemed necessary, the reviews shall be performed by the qualified review personnel of the appropriate discipline(s).~~

2. ~~Individuals performing these reviews shall meet the applicable experience requirements of ANSI N18.1-1971, Sections 4.2, 4.3, 4.4, 4.5.1, or 4.6, and be approved by the Plant Manager.~~

3. ~~Applicable Administrative Procedures recommended by Regulatory Guide 1.33, Plant Emergency Operating Procedures, and changes thereto shall be submitted to the Onsite Review and Investigative Function for review and approval prior to implementation.~~

4. ~~Review of the procedure or procedure change will include a determination of whether or not an unreviewed safety question is involved. This determination will be based on the review of a written safety evaluation prepared by a qualified individual or documentation that a safety evaluation is not required. Onsite Review, Offsite Review and Commission approval of items involving unreviewed safety questions shall be obtained prior to Station approval for implementation.~~

5. ~~The Department Head approval authority shall be specified in station procedures.~~

6. ~~Written records of reviews performed in accordance with this specification shall be prepared and maintained in accordance with Specification 6.5.~~

7. ~~Editorial and Typographical changes shall be made in accordance with station procedures.~~

ITS 5.4

LA.2

D. ~~Temporary changes to procedures 6.2.A and 6.2.B above may be made provided:~~

1. ~~The intent of the original procedure is not altered.~~
2. ~~The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.~~
3. ~~The change is documented, reviewed and approved in accordance with Specification 6.2.C within 14 days of implementation.~~

A.4

E. ~~Drills of the emergency procedures described in Specification 6.2.A.d shall be conducted at frequencies as specified in the Generating Stations Emergency Plan (GSEP). These drills will be planned so that during the course of the year, communication links are tested and outside agencies are contacted.~~

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

SEE ITS 5.5

1. Primary Coolant Sources Outside Primary Containment

A program to reduce 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 LPCS, HPCS, RHR/LPCI, RCIC, hydrogen recombiner, process sampling, containment monitoring, and standby gas treatment systems. The program shall include the following:

a. Preventive maintenance and periodic visual inspection requirements, and

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

2. In-Plant Radiation Monitoring

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

a. Training of personnel,

b. Procedures for monitoring, and

c. Provisions for maintenance of sampling and analysis equipment.

3. Post-accident Sampling

A program which will 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,

c. Provisions for maintenance of sampling and analysis equipment.

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ADMINISTRATIVE

A.1 In the conversion of the LaSalle 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 1434, Rev. 1 (i.e., the Improved Standard Technical Specifications (ISTS)).

A.2 Procedures required by CTS 6.2.A.c and d 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 6.2.A.f, 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.

A.4 CTS 6.2.E requires that drills of the emergency procedures be conducted at frequencies as specified in the Generating Station Emergency Response Plan, and that certain communications link are tested in the course of a year. These requirements are already required by 10 CFR 50, Appendix E. Therefore, this duplicative requirement has not been retained in the ITS. This is a change in the presentation of the requirements only, and therefore, is considered an administrative change. Therefore, there is no need to retain the requirement in the ITS.

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 thirteen 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 6.2.A.e 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 LaSalle 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.

LA.2 The details of procedure reviews and approvals including temporary changes contained in CTS 6.2.C and 6.2.D are proposed to be relocated to the Quality Assurance (QA) Manual. The ability to relocate these requirements is based on regulations and standards that contain these provisions such that duplication in the ITS is not necessary. The requirements for the establishment, maintenance, and implementation of procedures related to activities affecting quality are contained in 10 CFR 50, Appendix B, Criterion II and Criterion V; ANSI/ANS 3.2 - 1982; and ANSI/ASME NQAI - 1983, including 1983 addenda. In accordance with these requirements, the QA Manual will include adequate detail with respect to the administrative control of procedures related to activities affecting quality and nuclear safety. 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 QA Manual will be controlled by the provisions of 10 CFR 50.54.

"Specific"

None

RELOCATED SPECIFICATIONS

None

A.1 ITS 5.5

D. Temporary changes to procedures 6.2.A and 6.2.B above may be made provided:

SEE ITS 5.4

1. The intent of the original procedure is not altered.

2. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.

3. The change is documented, reviewed and approved in accordance with Specification 6.2.C. within 14 days of implementation.

E. Drills of the emergency procedures described in Specification 6.2.A.d shall be conducted at frequencies as specified in the Generating Stations Emergency Plan (GSEP). These drills will be planned so that during the course of the year, communication links are tested and outside agencies are contacted.

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

5.5.2 1. Primary Coolant Sources Outside Primary Containment

A program to reduce 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 LPCS, HPCS, RHR/LPCI, RCIC, hydrogen recombiner, process sampling, containment monitoring, and standby gas treatment systems. The program shall include the following:

a. Preventive maintenance and periodic visual inspection requirements, and

24 month LD.1

b. Integrated leak test requirements for each system at ~~refueling~~ ~~cycle~~ intervals ~~or less~~. A.2 LD.1

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

~~2. In-Plant Radiation Monitoring~~ LA.1

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

~~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

A program which will 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,

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 operating procedures, and (3) shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

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

5.5.4.b b. Limitations on the concentrations of radioactive material released in liquid effluents to UNRESTRICTED AREAS conforming to 10 times the concentration values in Appendix B, Table 2, Column 2 to 10 CFR 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,

5.5.4.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 Appendix I to 10 CFR Part 50,

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,

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 1. 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 2. 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,

<See ITS 5.4> PLANT OPERATING PROCEDURES AND PROGRAMS (Continued) A.1 ITS 5.5

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 half-lives greater than 8 days in gaseous effluents released from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50,

5.5.4.k j. Limitations on venting and purging of the containment through the Primary Containment Vent and Purge System or Standby Gas Treatment System to maintain releases as low as reasonably achievable,

5.5.4.j k. 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.

LA.2

~~5. Radiological Environmental Monitoring Program~~

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

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

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

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

5.5.6 6. Inservice Inspection Program for Post Tensioning Tendons

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, except that the unit 1 and 2 primary containments shall be treated as twin containments even though the Initial Structural Integrity Tests were not within 2 years of each other.

LA.3

~~The Onsite Review and Investigative Function shall be responsible for reviewing and approving changes to the Inservice Inspection Program for Post Tensioning Tendons.~~

The provisions of 4.0.2 and 4.0.3 are applicable to the Tendon Surveillance Program inspection frequencies.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluent Control Program surveillance frequencies. A.2

<See ITS 5.4> PLANT OPERATING PROCEDURES AND PROGRAMS (Continued)

5.5.13 7. Primary Containment Leakage Rate Testing Program

5.5.13.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 exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Testing Program," dated September 1995.

5.5.13.b The peak calculated primary containment internal pressure for the design basis loss of coolant accident, P_a, is 39.6 psig.

5.5.13.c The maximum allowable primary containment leakage rate, L_a, at P_a, is 0.635% of primary containment air weight per day.

5.5.13.d Leakage rate acceptance criteria are:

5.5.13.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.

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

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

5.5.13.d.2.b 2) For each door, the seal leakage rate is $\leq$ 5 scf per hour when the gap between the door seals is pressurized to $\geq$ 10 psig.

~~The provisions of specification 4.0.2 do not apply to the test frequencies specified in the Primary Containment Leakage Rate Testing Program.~~ A.3

5.5.13.e The provisions of specification 4.0.3 are applicable to the Primary Containment Leakage Rate Testing Program.

5.5.8 8. Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems ~~at the frequencies specified in Regulatory Guide 1.52, Revision 2, dated March 1978,~~ and in accordance with ASME N510-1989. A.13 A.4 LD.2 LD.3

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

5.5.8.a a. Demonstrate for each of the ESF systems that an inplace test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass <0.05% when tested in accordance with ASME N510-1989, at the system flowrate specified below:

A.12 EMUs

ESF Ventilation System	Flowrate (cfm)
SBGT System	$\geq$ 3600 and $\leq$ 4400
~~CREF~~ CRAF System	$\geq$ 3600 and $\leq$ 4400

5.5.8.b b. Demonstrate for each of the ESF system filter units that an inplace test of the charcoal adsorber shows a penetration and system bypass less than the value specified below, when tested in accordance with ASME N510-1989, at the system flowrate specified below:

A.12 EMUs

ESF Ventilation System	Penetration and System Bypass	Flowrate (cfm)
CRAF		
SBGT System	0.05 %	≥ 3600 and ≤ 4400
~~CREF~~ System	0.05 %	≥ 3600 and ≤ 4400
CRRF ~~System~~	2.0 %	≥18000 and ≤ 28900
AEERRF ~~System~~	2.0 %	≥14000 and ≤ 22800

A.12

5.5.8.c 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, a relative humidity of 70 % and a face velocity as specified below.

ESF Ventilation System	Penetration	Face Velocity (fpm)
CRAF		
SBGT System	0.5 %	40
~~CREF~~ System	2.5 %	40
CRRF ~~System~~	15.0 %	80
AEERRF ~~System~~	15.0 %	80

A.12

5.5.8.d d. Demonstrate for each of the ESF systems that the pressure drop across the combined moisture separator, heater, prefilter, HEPA filters and the charcoal adsorbers is less than the value specified below when tested at the system flowrate specified below:

ESF Ventilation System	Delta P (inches wg)	Flowrate (cfm)
CRAF		
SBGT System	8	≥ 3600 and ≤ 4400
~~CREF~~ System	8	≥ 3600 and ≤ 4400
CRRF ~~System~~	3.0	≥ 18000 and ≤ 28900
AEERRF ~~System~~	3.0	≥ 14000 and ≤ 22800

A.12

5.5.8.e e. Demonstrate that the heaters for each of the ESF systems dissipate the electrical power specified below when tested in accordance with ASME N510-1989. These readings shall include appropriate corrections for variations from 480 Volts at the bus.

ESF Ventilation System	Wattage (kw)
CRAF	
SBGT System	≥ 21 and ≤ 25
~~CREF~~ System	≥ 18 and ≤ 22

A.12

<SEE CTS 6.3>

6.3 ACTION TO BE TAKEN IN THE EVENT OF A REPORTABLE EVENT IN PLANT OPERATION

The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified and a Licensee Event Report submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50, and

b. Each REPORTABLE EVENT shall be reviewed by the Onsite Review and Investigative Function.

Add proposed ITS 5.5.11 — M.1

Add proposed ITS 5.5.12 — M.1

A.1

ITS 5.5

APPLICABILITY

SURVEILLANCE REQUIREMENTS

See ITS Section 3.0

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

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

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

4.0.4 Entry into an OPERATIONAL CONDITION or other specified applicable CONDITION shall not be made unless the Surveillance Requirements 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 CONDITIONS as required to comply with ACTION requirements.

5.5.7 4.0.5 Surveillance Requirements for inservice inspection and testing of ASME Code Class 1, 2, & 3 components shall be applicable as follows: LA.4

Pumps and Valves

LA.4

a. ~~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 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(g)(6)(i). LA.5 LA.4

5.5.7.a b. 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:

LA.4

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

A.5

A.1

APPLICABILITY

SURVEILLANCE REQUIREMENTS (Continued)

5.5.7.b c. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice ~~inspection and~~ testing activities. LA.4

d. ~~Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements.~~ A.6

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

f. ~~The inservice inspection program for piping identified in NRC Generic Letter 88-01 shall be performed in accordance with the NRC staff positions on schedule, methods, personnel, and sample expansion included in Generic Letter 88-01 or in accordance with alternate measures approved by the NRC staff.~~ LA.4

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

A.1

ITS 5.5

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS <See ITS 3.8.1>

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

7. Verifying the pressure in required diesel generator air start receivers to be greater than or equal to 200 psig.

A.7

add proposed ITS 5.5.10

b. At least once per 31 days and after each operation of the diesel where the period of operation was greater than or equal to 1 hour by checking for and removing accumulated water from the day fuel tanks.

5.5.10.a,b,c c. By sampling and analyzing stored and new fuel oil in accordance with the following:

L.1

5.5.10.a 1. ~~At least once per 92 days, and~~ for new fuel oil prior to addition to the storage tanks, that a sample obtained and tested in accordance with the applicable ASTM Standards has:

add proposed ITS 5.5.10.a.1

M.2

a) A water and sediment content within applicable ASTM limits, or a clear and bright appearance with proper color

b) A kinematic viscosity at 40°C and flash point M.2 within applicable ASTM limits.

add proposed ITS 5.5.10.b

5.5.10.c 2. At least every 31 days, ~~and for new fuel oil prior to addition to the storage tanks,~~ that a sample obtained in accordance with the applicable ASTM Standard has a total particulate contamination of ~~less than~~ ≤ 10 mg/l when tested in accordance with the applicable ASTM Standard.

L.1

d. At least once per 18 months during shutdown by:

1. (Not used).

2. Verifying the diesel generator capability* to reject a load of greater than or equal to 1190 kW for diesel generator 0, greater than or equal to 638 kW for diesel generators 1A and 2A, and greater than or equal to 2421 kW for diesel generator 1B while maintaining engine speed less than or equal to 75% of the difference between nominal speed and the overspeed trip setpoint or 15% above nominal, whichever is less.

3. Verifying the diesel generator capability* to reject a load of 2600 kW without tripping. The generator voltage shall not exceed 5000 volts during and following the load rejection.

4. Simulating a loss of offsite power* by itself, and:

*All planned diesel generator starts performed for the purpose of meeting these surveillance requirements may be preceded by an engine prelube period, as recommended by the manufacturer.

<See ITS 3.8.1>

A.7

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

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

A.1

ITS 5.5

LIQUID HOLDUP TANKS

LIMITING CONDITION FOR OPERATION

5.5.9.b 3.11.1.1 The quantity of radioactive material contained in any outside temporary tanks shall be limited to less than or equal to the limits calculated in the ODCM.

add proposed ITS 5.5.9 A.8

~~APPLICABILITY: At all times.~~

~~ACTION:~~

~~a. With the quantity of radioactive material in any of the above listed tanks exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.~~

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

LA.6

SURVEILLANCE REQUIREMENTS

5.5.9.b 4.11.1.1 The quantity of radioactive material contained in each of the above listed tanks shall be determined to be within the above limit by analyzing a representative sample of the tank's contents ~~at least once per 7 days when radioactive materials are being added to the tank.~~

LA.6

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.

A.1

ITS 5.5

3/4 11.2 GASEOUS EFFLUENTS

EXPLOSIVE GAS MIXTURE

add Proposed ITS 5.5.9 — A.8

LIMITING CONDITION FOR OPERATION

5.5.9.a 3.11.2.1 The concentration of hydrogen in the main condenser offgas treatment system shall be limited ~~to less than or equal to 4% by volume.~~

~~APPLICABILITY: Whenever the main condenser air ejector system is in operation.~~

~~ACTION:~~

~~a. With the concentration of hydrogen in the main condenser offgas treatment system exceeding the limit, restore the concentration to within the limit within 48 hours.~~

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

LA.6

SURVEILLANCE REQUIREMENTS

5.5.9.a 4.11.2.1 The concentration of hydrogen in the main condenser offgas treatment system shall be determined to be within the above limits ~~as required by Table 3.3.7.11-1 of Specification 3.3.7.11.~~ — LA.6

(See ITS chapter 4.0)

5.6 FUEL STORAGE

CRITICALITY

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

a. A k_{eff} equivalent to ≤ 0.95 when flooded with unborated water, including all calculational uncertainties and biases, as described in Section 9.1 of the FSAR.

b. A nominal 6.26 inch center-to-center distance between fuel assemblies placed in the storage racks.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 819 feet.

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 3986 fuel assemblies.

5.5.5

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in UFSAR Table ~~5.7.1-1~~ 5.2-4 are designed and shall be maintained within the cyclic or transient limits ~~of Table 5.7.1-1~~. LA.7

TABLE 5.7.1-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor	120 heatup and cooldown cycles	70°F to 560°F to 70°F
	10,000 power change cycles	75% to 100% to 75% of RATED THERMAL POWER
	80 step change cycles	Loss of Feedwater heaters
	190 reactor trip cycles	100% to 0% of RATED THERMAL POWER
	2000 power change cycles	50% to 100% to 50% of RATED THERMAL POWER
	400 control rod pattern exchanges	Not applicable
	130 hydrostatic pressure tests	Pressurized to ≥ 930 psig and ≤ 1250 psig.

LA.7

ITS 5.5

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A.1

ITS 5.5

1.20 Deleted

See ITS Chapter 1.0

LIMITING CONTROL ROD PATTERN

1.21 A LIMITING CONTROL ROD PATTERN shall be a pattern which results in the core being on a thermal hydraulic limit, i.e., operating on a limiting value for APLHGR, LHGR, or MCPR.

LINEAR HEAT GENERATION RATE

1.22 LINEAR HEAT GENERATION RATE (LHGR) shall be the heat generation per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length.

LOGIC SYSTEM FUNCTIONAL TEST

1.23 A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all logic components, i.e., all relays and contacts, all trip units, solid state logic elements, etc. of a logic circuit, from sensor through and including the actuated device to verify OPERABILITY. THE LOGIC SYSTEM FUNCTIONAL TEST may be performed by any series of sequential, overlapping or total system steps such that the entire logic system is tested.

MAXIMUM FRACTION OF LIMITING POWER DENSITY

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

MEMBER(S) OF THE PUBLIC

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

MINIMUM CRITICAL POWER RATIO

1.26 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be the smallest CPR which exists in the core.

5.5.1 OFFSITE DOSE CALCULATION MANUAL

5.5.1.a

1.27 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Technical Specification Section 6.2.F.4 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Annual Radioactive Effluent Release Reports required by Technical Specification Sections 6.6.A.3 and 6.6.A.4.

5.5.1.b

5.5.1 6.8 OFFSITE DOSE CALCULATION MANUAL (ODCM)* A.9

~~6.8.1 The ODCM shall be approved by the Commission prior to implementation.~~

5.5.1.c 6.8.2 Licensee initiated changes to the ODCM:

5.5.1.c.1 a. Shall be documented and records of reviews performed shall be retained ~~as required by Specification 6.5.B.18~~. This documentation shall contain: A.10

5.5.1.c.1(a) 1) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s), and A.11

5.5.1.c.1(b) 2) A determination that the change will maintain the level of radioactive effluent control required by 10 CFR 20.~~106~~ 1302, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50 and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

5.5.1.c.2 b. Shall become effective after ~~review and acceptance~~ by the On-Site Review ~~and Investigative Function~~ and ~~the~~ approval of the ~~Plant~~ station Manager ~~on the date specified by the On-Site Review and Investigative Function~~. LA.3 LA.8 LA.3

5.5.1.c.3 c. Shall be submitted to the Commission in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Annual Radioactive Effluent Release Report for the period of the report in which any change to the ODCM was made 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.

<SEE CTS 6.9>

6.9 MAJOR CHANGES TO RADIOACTIVE WASTE TREATMENT SYSTEMS

6.9.1 Licensee initiated major changes to the radioactive waste treatment systems (liquid, gaseous and solid):

a. Shall be reported to the Commission in the Monthly Operating Report for the period in which the evaluation was reviewed by the Onsite Review and Investigative Function. The discussion of each change shall contain:

1. A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR 50.59;

2. Sufficient detailed information to totally support the reason for the change without benefit or additional or supplemental information;

3. A detailed description of the equipment, components and processes involved and the interfaces with other plant systems;

5.5.1 *The OFFSITE DOSE CALCULATION MANUAL (ODCM) is common to La Salle Unit 1 and La Salle Unit 2.

D. Temporary changes to procedures 6.2.A and 6.2.B above may be made provided:

1. The intent of the original procedure is not altered.

2. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.

3. The change is documented, reviewed and approved in accordance with Specification 6.2.C. within 14 days of implementation.

E. Drills of the emergency procedures described in Specification 6.2.A.d shall be conducted at frequencies as specified in the Generating Stations Emergency Plan (GSEP). These drills will be planned so that during the course of the year, communication links are tested and outside agencies are contacted.

< SEE ITS 5.4 >

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

5.5.2 1. Primary Coolant Sources Outside Primary Containment

A program to reduce 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 LPCS, HPCS, RHR/LPCI, RCIC, hydrogen recombiner, process sampling, containment monitoring, and standby gas treatment systems. The program shall include the following:

a. Preventive maintenance and periodic visual inspection requirements, and

b. Integrated leak test requirements for each system at ~~refueling cycle~~ 24 month intervals ~~or less~~ LD.1

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

~~2. In-Plant Radiation Monitoring~~

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

~~a. Training of personnel,~~

~~b. Procedures for monitoring, and~~

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

LA.1

5.5.3 3. Post-accident Sampling

A program which will 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,

c. Provisions for maintenance of sampling and analysis equipment.

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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 operating procedures, and (3) shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

5.5.4.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 concentrations of radioactive material released in liquid effluents to UNRESTRICTED AREAS conforming to 10 times the concentration values in Appendix B, Table 2, Column 2 to 10 CFR 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,

5.5.4.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 Appendix I to 10 CFR Part 50,

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,

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 1. 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 2. 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,

<See ITS 5.4> PLANT OPERATING PROCEDURES AND PROGRAMS (Continued) A.1 ITS 5.5

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 half-lives greater than 8 days in gaseous effluents released from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50,

5.5.4.k j. Limitations on venting and purging of the containment through the Primary Containment Vent and Purge System or Standby Gas Treatment System to maintain releases as low as reasonably achievable,

5.5.4.j k. 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. LA.2

~~5. Radiological Environmental Monitoring Program~~

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

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

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

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

5.5.6 6. Inservice Inspection Program for Post Tensioning Tendons

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, except that the unit 1 and 2 primary containments shall be treated as twin containments even though

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluent Control Program surveillance frequencies. A.2

5.5.6 the Initial Structural Integrity Tests were not within 2 years of each other.

~~The Onsite Review and Investigative Function shall be responsible for reviewing and approving changes to the Inservice Inspection Program for Post Tensioning Tendons.~~ LA.3

The provisions of 4.0.2 and 4.0.3 are applicable to the Tendon Surveillance Program inspection frequencies.

5.5.13 7. Primary Containment Leakage Rate Testing Program

5.5.13.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 exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Testing Program," dated September 1995.

5.5.13.b The peak calculated primary containment internal pressure for the design basis loss of coolant accident, P_a, is 39.6 psig.

5.5.13.c The maximum allowable primary containment leakage rate, L_a, at P_a, is 0.635% of primary containment air weight per day.

5.5.13.d Leakage rate acceptance criteria are:

5.5.13.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.

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

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

5.5.13.d.2(b) 2) For each door, the seal leakage rate is $\leq$ 5 scf per hour when the gap between the door seals is pressurized to $\geq$ 10 psig.

~~The provisions of specification 4.0.2 do not apply to the test frequencies specified in the Primary Containment Leakage Rate Testing Program.~~ A.3

5.5.13.e The provisions of specification 4.0.3 are applicable to the Primary Containment Leakage Rate Testing Program.

5.5.8 8. Ventilation Filter Testing Program (VFTP) A.13

A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems at the frequencies specified in Regulatory Guide 1.52, Revision 2, dated March 1978, and in accordance with ASME N510-1989. A.4 LD.2 LD.3

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

<See ITS 5.4> PLANT OPERATING PROCEDURES AND PROGRAMS (Continued)

5.5.8.a a. Demonstrate for each of the ESF systems that an inplace test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass < 0.05 % when tested in accordance with ASME N510-1989, at the system flowrate specified below:

A.12 CRAF System EMUs

ESF Ventilation System	Flowrate (cfm)
SBGT System	≥ 3600 and ≤ 4400
~~CREF~~ System	≥ 3600 and ≤ 4400

5.5.8.b b. Demonstrate for each of the ESF system filter units that an inplace test of the charcoal adsorber shows a penetration and system bypass less than the value specified below, when tested in accordance with ASME N510-1989, at the system flowrate specified below:

ESF Ventilation System	Penetration and System Bypass	Flowrate (cfm)
SBGT System	0.05 %	≥ 3600 and ≤ 4400
~~CREF~~ System	0.05 %	≥ 3600 and ≤ 4400
CRRF ~~System~~ A.12	2.0 %	≥ 18000 and ≤ 28900
AEERRF ~~System~~	2.0 %	≥ 14000 and ≤ 22800

5.5.8.c 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, a relative humidity of 70 % and a face velocity as specified below.

ESF Ventilation System	Penetration	Face Velocity (fpm)
SBGT System	0.5 %	40
~~CREF~~ System	2.5 %	40
CRRF ~~System~~ A.12	15.0 %	80
AEERRF ~~System~~	15.0 %	80

5.5.8.d d. Demonstrate for each of the ESF systems that the pressure drop across the combined moisture separator, heater, prefilter, HEPA filters and the charcoal adsorbers is less than the value specified below when tested at the system flowrate specified below:

ESF Ventilation System	Delta P (inches wg)	Flowrate (cfm)
SBGT System	8	≥ 3600 and ≤ 4400
~~CREF~~ System	8	≥ 3600 and ≤ 4400
CRRF ~~System~~ A.12	3.0	≥ 18000 and ≤ 28900
AEERRF ~~System~~	3.0	≥ 14000 and ≤ 22800

See ITS 5.4

PLANT OPERATING PROCEDURES AND PROGRAMS (Continued)

5.5.8.e e. Demonstrate that the heaters for each of the ESF systems dissipate the electrical power specified below when tested in accordance with ASME N510-1989. These readings shall include appropriate corrections for variations from 480 Volts at the bus.

A.12 CRAF EMUs

ESF Ventilation System	Wattage (kw)
SBGT System	≥ 21 and ≤ 25
CREF System	≥ 18 and ≤ 22

6.3 ACTION TO BE TAKEN IN THE EVENT OF A REPORTABLE EVENT IN PLANT OPERATION

The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified and a Licensee Event Report submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50, and

b. Each REPORTABLE EVENT shall be reviewed by the Onsite Review and Investigative Function.

SEE CTS 6.3

Add proposed ITS 5.5.11 — M.1

Add proposed ITS 5.5.12 — M.1

A.1

ITS 5.5

APPLICABILITY

SURVEILLANCE REQUIREMENTS

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

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

See ITS Section 3.0

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

4.0.4 Entry into an OPERATIONAL CONDITION or other specified applicable CONDITION shall not be made unless the Surveillance Requirements 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 CONDITIONS as required to comply with ACTION requirements.

5.5.7 4.0.5 Surveillance Requirements for inservice inspection and testing of ASME Code Class 1, 2, & 3 components shall be applicable as follows: LA.4

pumps and valves LA.4

a. ~~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 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).~~ LA.5 LA.4

5.5.7.a b. 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
Bienially or every 2 years	At least once per 731 days
Every 48 months	At least once per 1461 days

A.5

APPLICABILITY

SURVEILLANCE REQUIREMENTS (Continued)

5.5.7.b c. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice ~~inspection and~~ testing activities. LA.4

~~d. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements.~~ A.6

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

~~f. The inservice inspection program for piping identified in NRC Generic Letter 88-01 shall be performed in accordance with the NRC staff positions on schedule, methods, personnel, and sample expansion included in Generic Letter 88-01 or in accordance with alternate measures approved by the NRC staff.~~ LA.4

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

ITS 5.5

A.1

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS

A.7

add proposed ITS 5.5.10

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

7. Verifying the pressure in required diesel generator air start receivers to be greater than or equal to 200 psig.

b. At least once per 31 days and after each operation of the diesel where the period of operation was greater than or equal to 1 hour by checking for and removing accumulated water from the day fuel tanks.

See ITS 3.8.1

5.5.10.a,b,c c. By sampling and analyzing stored and new fuel oil in accordance with the following: L.1

5.5.10.a 1. ~~At least once per 92 days, and~~ for new fuel oil prior to addition to the storage tanks, that a sample obtained and tested in accordance with the applicable ASTM Standards has: M.2

add proposed ITS 5.5.10.a.1

a) A water and sediment content within applicable ASTM limits, or a clear and bright appearance with proper color

b) A kinematic viscosity at 40°C and flash point M.2 within applicable ASTM limits.

add proposed ITS 5.5.10.b

2. At least every 31 days, ~~and for new fuel oil prior to addition to the storage tanks~~, that a sample obtained in accordance with the applicable ASTM Standard has a total particulate contamination of ~~less than~~ ≤ 10 mg/l when tested in accordance with the applicable ASTM Standard. L.1

d. At least once per 18 months during shutdown by:

1. (Not Used).

2. Verifying the diesel generator capability* to reject a load of greater than or equal to 1190 kW for diesel generator 0, greater than or equal to 638 kW for diesel generators 1A and 2A, and greater than or equal to 2421 kW for diesel generator 2B while maintaining engine speed less than or equal to 75% of the difference between nominal speed and the overspeed trip setpoint or 15% above nominal, whichever is less.

3. Verifying the diesel generator capability* to reject a load of 2600 kW without tripping. The generator voltage shall not exceed 5000 volts during and following the load rejection.

4. Simulating a loss of offsite power* by itself, and:

See ITS 3.8.1

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program testing frequencies. A.7

*All planned diesel generator starts performed for the purpose of meeting these surveillance requirements may be preceded by an engine prelube period, as recommended by the manufacturer.

See ITS 3.8.1

3/4.11.1 LIQUID EFFLUENTS A.1 ITS 5.5

LIQUID HOLDUP TANKS

LIMITING CONDITION FOR OPERATION

5.5.9.b 3.11.1.1 The quantity of radioactive material contained in any outside temporary tanks shall be limited to less than or equal to the limits calculated in the ODCM.

Add proposed ITS 5.5.9 A.8

~~APPLICABILITY: At all times.~~

~~ACTION:~~

~~a. With the quantity of radioactive material in any of the above listed tanks exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.~~

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

LA.6

SURVEILLANCE REQUIREMENTS

5.5.9.b 4.11.1.1 The quantity of radioactive material contained in each of the above listed tanks shall be determined to be within the above limit by analyzing a representative sample of the tank's contents ~~at least once per 7 days when radioactive materials are being added to the tank.~~

LA.6

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. A.8

3/4 11.2 GASEOUS EFFLUENTS

A.1

ITS 5.5

EXPLOSIVE GAS MIXTURE

LIMITING CONDITION FOR OPERATION

add proposed ITS 5.5.9 A.8

5.5.9.a 3.11.2.1 The concentration of hydrogen in the main condenser offgas treatment system shall be limited ~~to less than or equal to 4% by volume.~~

~~APPLICABILITY: Whenever the main condenser air ejector system is in operation.~~

~~ACTION:~~

~~a. With the concentration of hydrogen in the main condenser offgas treatment system exceeding the limit, restore the concentration to within the limit within 48 hours.~~

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

LA.6

SURVEILLANCE REQUIREMENTS

5.5.9.a 4.11.2.1 The concentration of hydrogen in the main condenser offgas treatment system shall be determined to be within the above limits ~~as required by Table 3.3.7.11-1 of Specification 3.3.7.11.~~

LA.6

A.1

ITS 5.5

DESIGN FEATURES <See ITS chapter 4.0>

5.6 FUEL STORAGE

CRITICALITY

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

a. A k_{eff} equivalent to $\leq$ 0.95 when flooded with unborated water, including all calculational uncertainties and biases, as described in Section 9.1 of the FSAR.

b. A nominal 6.26-inch center-to-center distance between fuel assemblies placed in the storage racks.

5.6.1.2 The k_{eff} for new fuel for the first core loading stored dry in the spent fuel storage racks shall not exceed 0.95 when flooded with water.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 819 feet.

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 4078 fuel assemblies.

5.5.5 **5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT**

5.7.1 The components identified in UFSAR Table ~~5.7.1-1~~ 5.2-4 are designed and shall be maintained within the cyclic or transient limits ~~of Table 5.7.1-1~~. LA.7

TABLE 5.7.1-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor	120 heatup and cooldown cycles	70°F to 560°F to 70°F
	10,000 power change cycles	75% to 100% to 75% of RATED THERMAL POWER
	80 step change cycles	Loss of Feedwater heaters
	190 reactor trip cycles	100% to 0% of RATED THERMAL POWER
	2000 power change cycles	50% to 100% to 50% of RATED THERMAL POWER
	400 control rod pattern exchanges	Not applicable
	130 hydrostatic pressure tests	Pressurized to ≥ 930 psig and ≤ 1250 psig.

LA.7

ITS 5.5

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A.1

ITS 5.5

LIMITING CONTROL ROD PATTERN

1.21 A LIMITING CONTROL ROD PATTERN shall be a pattern which results in the core being on a thermal hydraulic limit, i.e., operating on a limiting value for APLHGR, LHGR, or MCPR.

LINEAR HEAT GENERATION RATE

1.22 LINEAR HEAT GENERATION RATE (LHGR) shall be the heat generation per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length. LHGR is monitored by the ratio of LHGR to its fuel specific limit, as specified in the CORE OPERATING LIMITS REPORT.

See ITS Chapter 1.0

LOGIC SYSTEM FUNCTIONAL TEST

1.23 A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all logic components, i.e, all relays and contacts, all trip units, solid state logic elements, etc. of a logic circuit, from sensor through and including the actuated device to verify OPERABILITY. THE LOGIC SYSTEM FUNCTIONAL TEST may be performed by any series of sequential, overlapping or total system steps such that the entire logic system is tested.

1.24 Deleted

MEMBERS(S) OF THE PUBLIC

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

MINIMUM CRITICAL POWER RATIO

1.26 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be the smallest CPR which exists in the core.

5.5.1 OFFSITE DOSE CALCULATION MANUAL

5.5.1.a 1.27 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 Technical Specification Section 6.2.F.4 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Annual Radioactive Effluent Release Reports required by Technical Specification Sections 6.6.A.3 and 6.6.A.4.

5.5.1 6.8 OFFSITE DOSE CALCULATION MANUAL (ODCM)*

~~6.8.1 The ODCM shall be approved by the Commission prior to implementation.~~ A.9

5.5.1.c 6.8.2 Licensee initiated changes to the ODCM:

5.5.1.c.1 a. Shall be documented and records of reviews performed shall be retained ~~as required by Specification 6.5.B.10~~. This documentation shall contain: A.10

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

5.5.1.c.1(b) 2) A determination that the change will maintain the level of radioactive effluent control required by 10 CFR 20.~~106~~ 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. A.9

5.5.1.c.2 b. Shall become effective after review and acceptance by the On-Site Review and Investigative Function and the approval of the ~~Plant~~ Station Manager on the date specified by the On-Site Review and Investigative Function. LA.3 LA.8 LA.8 LA.3

5.5.1.c.3 c. Shall be submitted to the Commission in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Annual Radioactive Effluent Release Report for the period of the report in which any change to the ODCM was made 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.

<SEE CTS 6.9>

6.9 MAJOR CHANGES TO RADIOACTIVE WASTE TREATMENT SYSTEMS

6.9.1 Licensee initiated major changes to the radioactive waste treatment systems (liquid, gaseous and solid):

a. Shall be reported to the Commission in the Monthly Operating Report for the period in which the evaluation was reviewed by the Onsite Review and Investigative Function. The discussion of each change shall contain:

1. A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR 50.59;

5.5.1 *The OFFSITE DOSE CALCULATION MANUAL (ODCM) is common to La Salle Unit 1 and La Salle Unit 2.

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ADMINISTRATIVE

A.1 In the conversion of the LaSalle 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 1434, 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 6.2.F.1 (ITS 5.5.2), a statement of applicability of SR 3.0.3 has been added to CTS 4.0.5 (ITS 5.5.7.c), and a statement of applicability of SR 3.0.2 and SR 3.0.3 has been added to CTS 6.2.F.4 (ITS 5.5.4) and CTS 4.8.1.1.2.c (ITS 5.5.10). 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 CTS 6.2.F.7 exempts the requirements of CTS 4.0.2 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 4.0.1 and 4.0.4 also do not apply to the Primary Containment Leakage Rate Testing Program, but this is not stated in CTS 6.2.F.7. Therefore, the specific statement to exempt this requirement is redundant and has been deleted. Also, this has been previously approved for the most recent BWR/5 ITS submittal (NMP2).

A.4 CTS 6.2.F.8 states that the test frequencies for the Ventilation Filter Testing Program shall be in accordance with Regulatory Guide 1.52, Rev. 2, dated March 1978. The Regulatory Guide requires certain tests to be performed every 18 months. However, this Frequency is being changed to 24 months, as described in Discussion of Changes LD.2 and LD.3 below. Therefore, the actual test frequencies are being added into ITS 5.5.8. Since the Frequencies are not changed (except as discussed in Discussion of Changes LD.2 and LD.3 below), this change is considered administrative.

ADMINISTRATIVE (continued)

A.5 Additional definitions of frequencies "Biennially or every two years" and "Every 48 months" are identified for the Inservice Testing Program of CTS 4.0.5.b. This change includes no new requirements, but only provides clarification of terms. Therefore, this change is considered administrative.

A.6 CTS 4.0.5.d restates that all applicable requirements must be met. Repeating this overall requirement as a specific detail is redundant and unnecessary. Therefore, this detail can be omitted without any technical change in the requirements and is considered administrative in nature.

A.7 The diesel fuel oil testing requirements in CTS 4.8.1.1.2.c 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.

A.8 The liquid holdup tank requirements in CTS 3/4.11.1.1 and the explosive gas mixture requirements in CTS 3/4.11.2.1 have 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 CTS 6.8.1, which requires the Offsite Dose Calculation Manual (ODCM) to be approved by the Commission prior to implementation, is deleted. The ODCM has already been approved by the NRC. As a result, this change is considered administrative since the activity has already been completed.

A.10 CTS 6.8.2 contains a reference to Specification 6.5.B.18. CTS Section 6.5 provides the requirements regarding plant records. This change simply deletes a reference to this section. The Discussion of Change for relocating CTS 6.5 is DOC LA.1 to CTS 6.5. It provides the justification for relocating the requirements regarding plant records to the Quality Assurance Manual. As such, this change is considered a presentation change only.

ADMINISTRATIVE (continued)

A.11 CTS 6.8.2.a.2 contains a reference to 10 CFR 20.106. In proposed TS 5.5.1.c.1(b), this reference has been changed to 10 CFR 20.1302. This change reflects the recent revision to 10 CFR 20, and as such, is considered administrative.

A.12 The Ventilation Filter Testing Program (VFTP) requirements of CTS 6.2.F.8 includes testing requirements for the plant's Control Room outside air intake filters. CTS designates these filter units as the CREF System. ITS 5.5.8 contains the VFTP requirements but designates these filters as emergency makeup filter units (EMUs). Furthermore, EMUs, Control Room Recirculation Filters (CRRFs), and Auxiliary Electric Equipment Room Recirculation Filters (AEERRFs) are considered subsystems of the Control Room Area Filtration (CRAF) System. This change includes no new requirements, but only provides consistency with other ITS Specifications and plant specific nomenclature. Therefore, this change is administrative.

A.13 CTS 6.2.F.8 states that the testing frequencies for the Ventilation Filter Testing Program shall be in accordance with Regulatory Guide 1.52, Rev. 2. As a result, certain SGT and CRAF System filter testing is required 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 LaSalle 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, NMP2. 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 Two new programs are included in the proposed Technical Specifications. These programs are:

ITS 5.5.11	Technical Specification (TS) Bases Control Program
ITS 5.5.12	Safety Function Determination Program (SFDP)

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 two programs may be found in ITS 5.5.11 and 5.5.12.

M.2 CTS 4.8.1.1.2.c requirements for fuel oil testing do not address flash point, gravity, or other properties not addressed in the Specification. ITS 5.5.10.a.1 includes a requirement to verify either the API gravity or the absolute specific gravity of new fuel is within limits and ITS 5.5.10.a.2 includes a requirement to verify the new fuel oil flash point is within the requirements of the applicable ASTM standard. These properties are verified to ensure the new fuel oil is adequate for operation prior to addition to the storage tanks. In addition, ITS 5.5.10.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 limits for ASTM fuel. These changes are consistent with BWR ISTS, NUREG-1434, Rev. 1, impose additional operational requirements, and are considered more restrictive.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The details contained in CTS 6.2.F.2, "In-Plant Radiation Monitoring," are proposed to be relocated to the UFSAR. This program is required by the LaSalle 1 and 2 commitment to NUREG-0737, Item III.D.3.3 as stated in the NRC Safety Evaluation Report dated March, 1981 (NUREG-0519). 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

TECHNICAL CHANGES - LESS RESTRICTIVE

LA.1 (cont'd) 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 The details contained in CTS 6.2.F.5, "Radiological Environmental Monitoring Program," are proposed to be relocated to the Offsite Dose Calculation Manual (ODCM). This program is a redundant verification of the effectiveness of the effluent monitoring program contained in the ODCM and specified in the administrative controls section of the ITS. The relocated program has no impact or effect on nuclear safety of the plant. Proposed ITS 5.5.1 for the ODCM requires the ODCM to contain these activities. 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 ODCM will be controlled by the provisions of proposed ITS 5.5.1.c.

LA.3 The CTS 6.2.F.6 requirement that the changes to the Inservice Inspection Program for Post Tensioning Tendons must be reviewed and approved by the Onsite Review and Investigative Function and the CTS 6.8.2.b requirement that the ODCM must be reviewed and accepted by the Onsite Review and Investigative Function, prior to implementation and to document this review and acceptance are proposed to be relocated to the Quality Assurance (QA) Manual. The review activities performed by the Onsite Review and Investigative Function are required by ANSI N18.7-1976. Thus, the provisions are not necessary to be included in the ITS to provide adequate protection of the public health and safety, given the existence of these redundant requirements. Changes to the QA Manual are controlled by the provisions of 10 CFR 50.54.

LA.4 Details of the Inservice Inspection (ISI) Program in CTS 4.0.5 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 LaSalle 1 and 2 Operating Licenses. The LaSalle 1 and 2 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. LaSalle 1 and 2 commitments to Generic Letter 88-01 are documented in an NRC Safety Evaluation dated August 22, 1990, and do not need to be repeated in the ITS. Regulations and LaSalle 1 and 2 commitments to the NRC contain the necessary programmatic requirements for

TECHNICAL CHANGES - LESS RESTRICTIVE

LA.4 (cont'd) 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 4.0.5.a).

LA.5 Details of the Inservice Testing Program (IST) in the CTS 4.0.5 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 LaSalle 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.6 The details for implementing the requirements contained in CTS 3/4.11.1.1 and CTS 3/4.11.2.1 are proposed to be relocated to the Technical Requirements Manual (TRM). The requirements of ITS 5.5.9 are adequate to ensure the quantity of radioactivity in outside temporary tanks is maintained within limits and explosive gas mixtures in the main condenser offgas treatment system are maintained within limits. ITS 5.5.9 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 quantity of radioactivity in outside temporary tanks is maintained within limits and explosive gas mixtures in the main condenser offgas treatment 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 LaSalle 1 and 2 UFSAR at ITS implementation. Changes to the TRM will be controlled by the provisions of 10 CFR 50.59.

LA.7 Details of the components governed by CTS 5.7 (Component Cyclic or Transient Limit) are proposed to be relocated to the UFSAR. The requirement to monitor the cyclic and transient occurrences is maintained as a program in ITS 5.5.5 (Component Cyclic or Transient Limit). ITS 5.5.5 provides adequate regulatory

TECHNICAL CHANGES - LESS RESTRICTIVE

LA.7 (cont'd) control over the details to be relocated. As a result, 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.8 CTS 6.8.2.b uses the title "Plant 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. 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.

LD.1 The Frequency for performing CTS 6.2.F.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 4.0.2 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 4.0.2 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 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

TECHNICAL CHANGES - LESS RESTRICTIVE

LD.1 (cont'd) 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 6.2.F.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 6.2.F.8 has been extended from 18 months to 24 months in ITS 5.5.8. 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 4.0.2 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 4.0.2 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. 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 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 5.5.8 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

TECHNICAL CHANGES - LESS RESTRICTIVE

LD.2 (cont'd) 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 6.2.F.8 as implemented in ITS 5.5.8. 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 6.2.F.8 has been extended from 18 months to 24 months in ITS 5.5.8. These requirements ensure that in-place Control Room Area Filtration 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 4.0.2 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 4.0.2 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.

The Control Room Area Filtration (CRAF) System provides filtration for control room area air intake and recirculated air during a high radiation accident and maintains a positive pressure in the control room area during control room isolation. Regulatory positions C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, require CRAF System filters and charcoal adsorbers be in-place tested (1) initially, (2) at least once per 18 months thereafter, and (3) following

TECHNICAL CHANGES - LESS RESTRICTIVE

LD.3 (cont'd) 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 5.5.8 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 CRAF 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 CRAF System is normally in standby. In addition, the CRAF 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 CRAF 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 6.2.F.8 as implemented in ITS 5.5.8. 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.

"Specific"

L.1 CTS 4.8.1.1.2.c.2 requires sampling and verification that new fuel oil meets ASTM standards for "particulate contamination" prior to addition to the fuel oil storage tanks. Proposed ITS 5.5.10.b relaxes this requirement for new fuel by allowing "particulate contaminant" analyses of the stored fuel be performed every 31 days after the addition of any new fuel oil to the storage tanks.

CTS 4.8.1.1.2.c.1 requires sampling of stored fuel oil every 92 days to verify "water and sediment" and "kinematic viscosity" within ASTM limits. Proposed ITS 5.5.10.c relaxes the requirements for bulk stored fuel oil by not including the 92 day requirement to verify "water and sediment" and "kinematic viscosity." "Water and sediment" and "kinematic viscosity" testing for new fuel oil prior to addition to the storage tanks is retained. However, a clear and bright "appearance" test is added as an alternate to performing the "water and

TECHNICAL CHANGES - LESS RESTRICTIVE

L.1 (cont'd) sediment" test, consistent with BWR ISTS, NUREG-1434, Rev. 1. The clean and bright test is a visual check for evidence of water and particulate contamination. The clear and bright test will only be used for fuel oil meeting the ASTM D4176 color requirements (as described in Bases B 3.8.3). For dyed fuel oil not meeting the color requirements of ASTM D4176, use of the clear and bright test is not appropriate since the presence of free water and particulates may be obscured by the dye. However, for fuel oils meeting the ASTM D4176 color requirements, the clear and bright test provides a test with the required sensitivity for detection of water and particulates in the fuel oil CTS 4.8.1.1.2.c.2 also provides a particulate contamination limit of less than 10 mg/liter. ITS 5.5.10.c changes the limit to less than or equal to 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 5.5.10.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 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 5.5.10.b (31 days after new fuel oil addition) and the normal 31 day sampling frequency of ITS 5.5.10.c evaluate properties that would not have an immediate effect on 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 reflect 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

 ITS 5.6

A.11

PLANT OPERATING RECORDS (Continued)

5. Records of plant radiation and contamination surveys;

6. Records of offsite environmental monitoring surveys;

7. Records of radiation exposure for all plant personnel, including all contractors and visitors to the plant, in accordance with 10 CFR Part 20;

<SEE CTS 6.5>

8. Records of radioactivity in liquid and gaseous wastes released to the environment;

9. Records of transient or operational cycling for those components that have been designed to operate safety for a limited number of transient or operational cycles (identified in Table 5.7.1-1);

10. Records of individual staff members indicating qualifications, experience, training, and retraining;

11. Inservice inspections of the reactor coolant system;

12. Minutes of meetings and results of reviews and audits performed by the offsite and onsite review and audit functions;

13. Records of reactor tests and experiments;

14. Records of Quality Assurance activities required by the QA Manual, except for those items specified in Section 6.5.A;

15. Records of reviews performed for changes made to procedures on equipment or reviews of tests and experiments pursuant to 10 CFR 50.59;

16. Records of the service lives of all hydraulic and mechanical snubbers required by specification 3.7.9 including the date at which the service life commences and associated installation and maintenance records;

17. Records of analyses required by the radiological environmental monitoring program;

18. Records of reviews performed for changes made to the OFFSITE DOSE CALCULATION MANUAL and the PROCESS CONTROL PROGRAM; and

19. Records of pre-stressed concrete containment tendon surveillances.

6.6 REPORTING REQUIREMENTS

5.6 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following identified reports shall be submitted

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6.6 REPORTING REQUIREMENTS (Continued)

~~to the director of the appropriate Regional Office of Inspection and Enforcement unless otherwise noted.~~ A.2 In accordance with 10 CFR 50.4

A. Routine Reports A.3

LA.1

~~1. Startup Report~~

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

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

A.3 2. Annual Report Add proposed ITS 5.6.1 Note A.4

L.1

5.6.1 A tabulation shall be submitted on an annual basis ~~prior to March 1~~ by April 30 of each year of the number of station, utility, and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man rem exposure according to work and job functions (Note: this tabulation supplements the requirements of Section 20.407 of 10 CFR 20), 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 pocket dosimeter, TLD, or film badge measurements. 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 shall be assigned to specific major work functions.

A.5 2206

A.5 electronic or

A.6

~~The results of specific activity analysis in which the primary coolant exceeded the limits of Specification 3.4.5 shall be included in the Annual Report along with the following information: (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~~

A.1

~~performed prior to exceeding the limit, results of analysis while limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up system flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.~~ A.6

5.6.2 3. Annual Radiological Environmental Operating Report* L.1

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted before May 1 of each year. The report shall include summaries, interpretations, and analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in (1) the ODCM and (2) Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50. A.7

64 15

5.6.3 4. Annual Radioactive Effluent Release Report**

in accordance with 10CFR 50.36a

The Annual Radioactive Effluents Release Report covering the operation of the unit during the previous calendar year shall be submitted prior to May 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 unit. The material provided shall be (1) consistent with the objectives outlined in the ODCM and PCP and (2) in conformance with 10 CFR 50.36a and Section IV.B.1 of Appendix I to 10 CFR Part 50.

5.6.4 5. Monthly Operating Report

Routine reports of operating statistics and shutdown experience, including documentation of all challenges to safety/relief valves, shall be submitted on a monthly basis to the addresses specified in 10 CFR 50.4 no later than the 15th of each month following the calendar month covered by the report.

A.4

Add 2nd sentence of proposed ITS 5.6.2 Note

5.6.2 NOTE * A single submittal may be made for a multi-unit station.

5.6.3 NOTE ** A single submittal may be made for a multi-unit station. The submittal should combine those sections that are common to all units at the station; ~~however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.~~ A.9

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Monthly Operating Report (Continued)

See CTS 6.9 A report of any major changes to the radioactive waste treatment systems shall be submitted with the Monthly Operating Report for the period in which the evaluation was reviewed and accepted by Onsite Review and Investigative Function.

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.1 (1) The Average Planar Linear Heat Generation Rate (APLHGR) for Technical Specification 3.2.1.

5.6.5.a.2 (2) The minimum Critical Power Ratio (MCPR) scram time, dependent MCPR limits, and power and flow dependent MCPR limits for Technical Specification 3.2.3. Effects of analyzed equipment out of service are included. LA.2

5.6.5.a.3 (3) The Linear Heat Generation Rate (LHGR) for Technical Specification 3.2.4.

5.6.5.a.4 (4) The Rod Block Monitor Upscale Instrumentation Setpoints for Technical Specification Table 3.3.6-2.

5.6.5.b b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC. For LaSalle County Station Unit 1, the topical reports are:

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

5.6.5.b.2 (2) Letter, Ashok C. Thadani (NRC) to R.A. Copeland (SPC), "Acceptance for Referencing of ULTRAFLOW™ Spacer on 9x9-IX/X BWR Fuel Design," July 28, 1993.

5.6.5.b.3 (3) 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, XN-NF-524(P)(A) Revision 2, and Supplement 1 Revision 2, Supplement 2, Advanced Nuclear Fuels Corporation, November 1990.

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

Core Operating Limits Report (Continued)

5.6.5.b.5 (5) HUXY: A Generalized Multirod Heatup Code with 10 CFR 50, Appendix K Heatup Option, ANF-CC-33(P)(A), Supplement 1 Revision 1; and Supplement 2, Advanced Nuclear Fuels Corporation, August 1986 and January 1991, respectively.

5.6.5.b.6 (6) Advanced Nuclear Fuel 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.7 (7) 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.8 (8) 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.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) Advanced Nuclear Fuels Corporation Generic Mechanical Design for Advanced Nuclear Fuels Corporation 9x9-IX and 9x9-9X BWR Reload Fuel, ANF-89-014(P)(A), Revision 1 and Supplements 1 and 2, October 1991.

5.6.5.b.11 (11) Volume 1 - STAIF - A Computer Program for BWR Stability Analysis in the Frequency Domain, Volume 2 - STAIF - A Computer Program for BWR Stability Analysis in the Frequency Domain, Code Qualification Report, EMF-CC-074(P)(A), Siemens Power Corporation, July 1994.

5.6.5.b.12 (12) RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model, XN-NF-81-58(P)(A), Revision 2 Supplements 1 and 2, Exxon Nuclear Company, March 1984.

5.6.5.b.13 (13) XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis, XN-NF-84-105(P)(A), Volume 1 and Volume 1 Supplements 1 and 2; Volume 1 Supplement 4, Advanced Nuclear Fuels Corporation, February 1987 and June 1988, respectively.

5.6.5.b.14 (14) 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.15 (15) 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, Richland, WA 99352, March 1983.

Core Operating Limits Report (Continued)

5.6.5.b.16 (16) 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.17 (17) 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.18 (18) NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," (latest approved revision).

5.6.5.b.19 (19) Commonwealth Edison Topical Report NFSR-0085, "Benchmark of BWR Nuclear Design Methods," (latest approved revision).

5.6.5.b.20 (20) 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.21 (21) Commonwealth Edison Topical Report NFSR-0085, Supplement 2, "Benchmark of BWR Nuclear Design Methods - Neutronic Licensing Analyses," (latest approved revision).

5.6.5.b.22 (22) 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.23 (23) BWR Jet Pump Model Revision for RELAX, ANF-91-048(P)(A), Supplement 1 and Supplement 2, Siemens Power Corporation, October 1997.

5.6.5.b.24 (24) 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.25 (25) 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 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 U.S. Nuclear Regulatory Commission ~~Document Control Desk with copies to the Regional Administrator and Resident Inspector.~~ A.2

B. Deleted

C. Unique Reporting Requirements A.3

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

6.7 PROCESS CONTROL PROGRAM (PCP)*

6.7.1 The PCP shall be approved by the Commission prior to implementation.

<SEE CTS 6.7>

6.7.2 Licensee initiated changes to the PCP:

a. Shall be documented and records of reviews performed shall be retained as required by Specification 6.5.B.18. This documentation shall contain:

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

2) 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.

b. Shall become effective upon review and acceptance by the Onsite Review and Investigative Function.

*The Process Control Program (PCP) is common to La Salle Unit 1 and La Salle Unit 2.

<SEE ITS 3.3.3.1>

Table 3.3.7.5-1 (Continued)

ACCIDENT MONITORING INSTRUMENTATION
ACTION STATEMENTS

ACTION 80 -

a. With the number of OPERABLE accident monitoring instrumentation channels less than the Required Number of Channels shown in Table 3.3.7.5-1, restore the inoperable channel(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

b. With the number of OPERABLE accident monitoring instrumentation channels less than the Minimum Channels OPERABLE requirements of Table 3.3.7.5-1, restore the inoperable channel(s) to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

ACTION 81 -

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

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

5.6.6 2) prepare and submit a Special Report to the Commission pursuant to Specification 6.6.C within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

ACTION 82 -

a. With the number of OPERABLE channels one less than the required number of channels shown in Table 3.3.7.5-1, restore the inoperable channel to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.

b. With the number of OPERABLE channels less than the minimum channels OPERABLE requirements of Table 3.3.7.5-1, restore at least one 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>

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A.1

ITS 5.6

PLANT OPERATING RECORDS (Continued)

7. Records of radiation exposure for all plant personnel, including all contractors and visitors to the plant, in accordance with 10 CFR Part 20;

8. Records of radioactivity in liquid and gaseous wastes released to the environment;

9. Records of transient or operational cycling for those components that have been designed to operate safety for a limited number of transient or operational cycles (identified in Table 5.7.1-1);

10. Records of individual staff members indicating qualifications, experience, training, and retraining;

SEE CTS 6.5

11. Inservice inspections of the reactor coolant system;

12. Minutes of meetings and results of reviews and audits performed by the offsite and onsite review and audit functions;

13. Records of reactor tests and experiments;

14. Records of Quality Assurance activities required by the QA Manual, except for those items specified in Section 6.5.A;

15. Records of reviews performed for changes made to procedures on equipment or reviews of tests and experiments pursuant to 10 CFR 50.59;

16. Records of the service lives of all hydraulic and mechanical snubbers required by specification 3.7.9 including the date at which the service life commences and associated installation and maintenance records;

17. Records of analyses required by the radiological environmental monitoring program;

18. Records of reviews performed for changes made to the OFFSITE DOSE CALCULATION MANUAL and the PROCESS CONTROL PROGRAM; and

19. Records of pre-stressed concrete containment tendon surveillances.

6.6 REPORTING REQUIREMENTS

5.6

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

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6.6 REPORTING REQUIREMENTS (Continued) A.1 ITS 5.6

~~to the director of the appropriate Regional Office of Inspection and Enforcement unless otherwise noted.~~ in accordance with 10CFR 50.4 A.2

A. Routine Reports A.3

LA.1

~~1. Startup Report~~

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

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

A3

2. Annual Report Add proposed ITS 5.6.1 Note A.4

5.6.1 A tabulation shall be submitted on an annual basis ~~prior to March 1~~ by April 30 of each year of the number of station, utility, and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man rem exposure according to work and job functions (Note: this tabulation supplements the requirements of Section 20.407 2206 of 10 CFR 20), 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, TLD, or film badge measurements. 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 shall be assigned to specific major work functions.

L.1 A.5 A.5

A.6

~~The results of specific activity analysis in which the primary coolant exceeded the limits of Specification 3.4.5 shall be included in the Annual Report along with the following information: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded; (2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up system flow history starting 48 hours~~

 A.1

ITS 5.6

~~prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.~~ A.6

5.6.2 3. Annual Radiological Environmental Operating Report* L.1

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

5.6.3 4. Annual Radioactive Effluent Release Report**

The Annual Radioactive Effluent Release Report covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a A.7. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be (1) consistent with the objectives outlined in the ODCM and PCP and (2) in conformance with 10 CFR 50.36a and Section IV.B.1 of Appendix I to 10 CFR Part 50.

5.6.4 5. Monthly Operating Report

Routine reports of operating statistics and shutdown experience, including documentation of all challenges to safety/relief valves, shall be submitted on a monthly basis to the addressees specified in 10 CFR 50.4 no later than the 15th of each month following the calendar month covered by the report.

A report of any major changes to the radioactive waste treatment systems shall be submitted with the Monthly Operating Report for the period in which the evaluation was reviewed and accepted by Onsite Review and Investigative Function. (See CTS 6.9)

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: A.4

5.6.2 Note * A single submittal may be made for a multi-unit station. Add 2nd sentence of Proposed ITS 5.6.2 Note

5.6.3 Note ** A single submittal may be made for a multi-unit station. The submittal should combine those sections that are common to all units at the station; ~~however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.~~ A.9

Core Operating Limits Report (Continued)

5.6.5.a.1 (1) The Average Planar Linear Heat Generation Rate (APLHGR) for Technical Specification 3.2.1.

5.6.5.a.2 (2) The minimum Critical Power Ratio (MCPR) ~~scram time dependent MCPR limits, and power and flow dependent MCPR limits~~ for Technical Specification 3.2.3. ~~Effects of analyzed equipment out of service are included.~~ LA2

5.6.5.a.3 (3) The Linear Heat Generation Rate (LHGR) for Technical Specification 3.2.4.

5.6.5.a.4 (4) The Rod Block Monitor Upscale Instrumentation Setpoints for Technical Specification Table 3.3.6-2.

5.6.5.b b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC. For LaSalle County Station Unit 2, the topical reports are:

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

5.6.5.b.2 (2) Letter, Ashok C. Thadani (NRC) to R.A. Copeland (SPC), "Acceptance for Referencing of ULTRAFLOW™ Spacer on 9x9-IX/X BWR Fuel Design," July 28, 1993.

5.6.5.b.3 (3) 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, XN-NF-524(P)(A) Revision 2 and Supplement 1 Revision 2, Supplement 2, Advanced Nuclear Fuels Corporation November 1990.

5.6.5.b.4 (4) COTRANSA 2: A Computer Program for Boiling Water Reactor Transient Analysis, 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.5 (5) HUXY: A Generalized Multirod Heatup Code with 10 CFR 50, Appendix K Heatup Option, ANF-CC-33(P)(A), Supplement 1 Revision 1; and Supplement 2, Advanced Nuclear Fuels Corporation, August 1986 and January 1991, respectively.

5.6.5.b.6 (6) Advanced Nuclear Fuel 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.7 (7) 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.8 (8) 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.

Core Operating Limits Report (Continued)

5.6.5.6.9 (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.

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

5.6.5.6.11 (11) Volume 1 - STAIF - A Computer Program for BWR Stability Analysis in the Frequency Domain, Volume 2 - STAIF - A Computer Program for BWR Stability Analysis in the Frequency Domain, Code Qualification Report, EMF-CC-074(P)(A), Siemens Power Corporation, July 1994.

5.6.5.6.12 (12) RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model, XN-NF-81-58(P)(A), Revision 2 Supplements 1 and 2, Exxon Nuclear Company, March 1984.

5.6.5.6.13 (13) XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis, XN-NF-84-105(P)(A), Volume 1 and Volume 1 Supplements 1 and 2; Volume 1 Supplement 4, Advanced Nuclear Fuels Corporation, February 1987 and June 1988, respectively.

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

5.6.5.6.15 (15) 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, Richland, WA 99352, March 1983.

5.6.5.6.16 (16) 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.

5.6.5.6.17 (17) 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.

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

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

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

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

Core Operating Limits Report (Continued)

5.6.5.b.22 (22) 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.23 (23) BWR Jet Pump Model Revision for RELAX, ANF-91-048(P)(A), Supplement 1 and Supplement 2, Siemens Power Corporation, October 1997.

5.6.5.b.24 (24) 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.25 (25) ANFB Critical Power Correlation Determination of ATRIUM-9B Additive Constant Uncertainties, ANF-1125(P)(A), Supplement 1, Appendix E, Siemens Power Corporation, September 1998.

Core Operating Limits Report (Continued)

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 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 U.S. Nuclear Regulatory Commission ~~Document Control Desk with copies to the Regional Administrator and Resident Inspector.~~ A.2

B. Deleted A.3

C. Unique Reporting Requirements A.8

1. 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

6.7 PROCESS CONTROL PROGRAM (PCP)*

6.7.1 The PCP shall be approved by the Commission prior to implementation.

6.7.2 Licensee initiated changes to the PCP:

a. Shall be documented and records of reviews performed shall be retained as required by Specification 6.5.B.18. This documentation shall contain:

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

2) 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.

b. Shall become effective upon review and acceptance by the Onsite Review and Investigative Function.

SEE <CTS 6.7>

*The Process Control Program (PCP) is common to La Salle Unit 1 and La Salle Unit 2.

Table 3.3.7.5-1 (Continued)

ACCIDENT MONITORING INSTRUMENTATION
ACTION STATEMENTS

ACTION 80 -

a. With the number of OPERABLE accident monitoring instrumentation channels less than the Required Number of Channels shown in Table 3.3.7.5-1, restore the inoperable channel(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

b. With the number of OPERABLE accident monitoring instrumentation channels less than the Minimum Channels OPERABLE requirements of Table 3.3.7.5-1, restore the inoperable channel(s) to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

ACTION 81 -

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

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

5.6.6 2) prepare and submit a Special Report to the Commission pursuant to Specification 6.6.c within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

ACTION 82 -

a. With the number of OPERABLE channels one less than the required number of channels shown in Table 3.3.7.5-1, restore the inoperable channel to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.

b. With the number of OPERABLE channels less then the minimum channels OPERABLE requirements of Table 3.3.7.5-1, restore at least one 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 LaSalle 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 1434, Rev. 1 (i.e., the Improved Standard Technical Specifications (ISTS)).

A.2 Submittal details for reports required by CTS 6.6 (Reporting Requirements), CTS 6.6.A.6 (Core Operating Limits Report) and CTS 6.6.C (Unique Reporting Requirements) 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-1434, 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 "Unique Reporting Requirements." 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 A proposed Note for ITS 5.6.1 allowing a single report submittal to satisfy the associated reporting requirement for both units is added to CTS 6.6.A.2. In addition, the CTS 6.6.A.3 footnote * has been clarified in ITS 5.6.2 Note to state that the submittal should combine only those sections common to both units. 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 LaSalle 1 and 2 to estimate the whole body doses required to be reported in CTS 6.6.A.2, electronic dosimeter, has been added in ITS 5.6.1. This is considered administrative since the measurement tools described are accepted in the industry. In addition, the CTS 6.6.A.2 reference to 10 CFR 20 has been modified in ITS 5.6.1 to reflect the proper reference to 10 CFR 20, based on the recent revision to 10 CFR 20.

A.6 CTS 6.6.A.2 requires reporting the results of specific activity analysis in which the primary coolant exceeded CTS 3.4.5 limits. This reporting requirement is unnecessary since it is included in the LER requirements to report fuel cladding

ADMINISTRATIVE

A.6 (cont'd) 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 necessitates reporting, any minor differences are negligible with regard to safety. Therefore, the current reporting requirement of CTS 6.6.A.2 is a duplication of the 10 CFR 50.73 reporting requirement and can be deleted.

A.7 CTS 6.6.A.4 requires submittal of the radioactive effluent release report "prior to May 1 of each year." Proposed ITS 5.6.3 requires the submittal to be "in accordance with 10 CFR 50.36a." Compliance with 10 CFR 50 requirements is required by the LaSalle 1 and 2 Operating Licenses. Therefore this change is considered to be administrative in nature.

A.8 The general statement in CTS 6.6.C 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 6.6.A.4 (ITS 5.6.3), Annual Radioactive Effluent Release Report, is modified by footnote **. This footnote allows a single submittal to be made for a multi-unit station and requires the submittal to combine those sections that are common to all units at the station. However, the footnote requires, that for units with separate radwaste systems, the submittal specify the releases of radioactive material from each unit. At LaSalle 1 and 2, the radwaste systems are common to both units. Therefore, the Note to ITS 5.6.3 has been revised to delete reference to requirements for units with separate radwaste systems. Since the actual reporting requirements are unchanged, the change is considered to be administrative.

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The details associated with CTS 6.6.A.1, "Startup Report," are proposed to be relocated to the Technical Requirements Manual (TRM). The Startup Report is a summary of plant startup and power escalation testing following receipt of the Operating License, increase in licensed power level, installation of nuclear fuel with a different design or manufacturer than the current fuel, and modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the unit. The report provides the NRC a mechanism to review the appropriateness of licensee activities after-the-fact, but provides no regulatory authority once the report is submitted (i.e., no requirement for NRC approval). The Quality Assurance requirements of 10 CFR 50, Appendix B, and the Startup Test Program provisions contained in the UFSAR provide assurance the listed activities will be adequately performed and that appropriate corrective actions, if required, are taken. Given that the report was required to be provided to the Commission no sooner than 90 days following completion of the respective milestone, report completion and submittal was clearly not necessary to assure operation of the facility in a safe manner for the interval between completion of the startup testing and submittal of the report. Additionally, given there is no requirement for the Commission to approve the report, the Startup Report is 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 LaSalle 1 and 2 UFSAR at ITS implementation. Changes to the TRM will be controlled by the provisions of 10 CFR 50.59.

LA.2 CTS 6.6.A.6.a(2) 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 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.

TECHNICAL CHANGES - LESS RESTRICTIVE (continued)

"Specific"

L.1 This change proposes to relax the CTS 6.6.A.2 requirement for submitting the Occupational Radiation Exposure Report and the CTS 6.6.A.3 requirement for submitting the Annual Radiological Environmental Operating Report. The CTS require the reports to be submitted prior to March 1 and May 1 of each year, respectively. This proposed change will allow the reports to be submitted by April 30 and May 15 of each year, respectively. Given that the reports are still required to be provided to the NRC on or before April 30 or May 15, as applicable, and covers the previous calendar year, report completion and submittal is clearly not necessary to assure operation in a safe manner for the interval between March 1 and April 30 and May 1 and May 15. Additionally, there is no requirement for the NRC to approve the report. Therefore, this change has no impact on the safe operation of the plant.

RELOCATED SPECIFICATIONS

None

A.1 ITS 5.7

5.7 6.1.1 HIGH RADIATION AREAS

20.1601(c)

20.1601

A.2

5.7.1 6.1.1.1 Pursuant to Paragraph ~~20.203(c)(5)~~ of 10 CFR 20, in lieu of the "control device" or "alarm signal" required by paragraph ~~20.203(c)(2)~~ of 10 CFR 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/hr* but less than 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, 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 in which the intensity of radiation is greater than 100 mrem/hr* but less than 1000 mrem/hr*, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

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

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

c. A health physics qualified individual, i.e., qualified in radiation protection procedures, with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the ~~Health Physicist~~ in the Radiation Work Permit (RWP). LA.1

radiation protection manager

5.7.2 6.1.1.2 In addition to the requirements of 6.1.1.1, above, for areas accessible to personnel with radiation levels such that a major portion of the body could receive in one hour a dose greater than 1000 mrem*, the computer shall be programmed to permit entry through locked doors for any individual requiring access to any such High-High Radiation Areas for the time that access is required.

5.7.3 6.1.1.3 Keys to manually open computer controlled High Radiation Area doors and High-High Radiation Area doors shall be maintained under the Administration control of the Shift Manager on duty ~~and/~~or the ~~Health Physicist~~.

5.7.4 6.1.1.4 High-High Radiation areas, as defined in 6.1.1.2 above, not equipped with the computerized card readers shall be maintained in accordance with 10 CFR ~~20.203 c.2 (iii)~~, locked except during periods when access to the area is required with positive control over each individual entry, or ~~10 CFR 20.203 c.4~~ in the case of a High Radiation Area established for a period of 30 days or less, direct surveillance to prevent unauthorized entry may be substituted. Doors shall remain locked except during periods of access by personnel under an approved RWP which shall specify the dose rate levels in the immediate work area and the maximum allowable stay time for individuals in that area. For A.2

20.1601(a)(3) A.2

5.7 HIGH RADIATION AREAS (Continued)

5.7.4 individual areas accessible to personnel with radiation levels such that a major portion of the body could receive in one hour a dose in excess of 1000 mrem* that are located within large areas, such as the containment, where no enclosure exists for purposes of locking, and no enclosure can be reasonably constructed around the individual areas, then that area shall be roped off, conspicuously posted and a flashing light shall be activated as a warning device. In lieu of the stay time specification of the RWP, direct or remote, such as use of closed circuit TV cameras, continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities within the area.

6.2 PLANT OPERATING PROCEDURES AND PROGRAMS

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

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

b. 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,

c. Station Security Plan implementation,

d. Generating Station Emergency Response Plan implementation,

e. PROCESS CONTROL PROGRAM implementation,

f. OFFSITE DOSE CALCULATION MANUAL implementation, and

g. Fire Protection Program implementation.

<See ITS 5.4>

~~*Measurement made at 18" from source of radioactivity.~~ A.2

A.1 ITS 5.7

5.7 6.1.1 HIGH RADIATION AREAS

5.7.1 6.1.1.1 Pursuant to Paragraph ~~20.203(c)(5)~~ 20.1601(c) of 10 CFR 20, in lieu of the "control device" or "alarm signal" required by paragraph ~~20.203(c)(2)~~ 20.1601 of 10 CFR 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/hr* but less than 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, 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 in which the intensity of radiation is greater than 100 mrem/hr* but less than 1000 mrem/hr*, provided they are otherwise following plant radiation protection procedures for entry into such 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.2

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

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

c. A health physics qualified individual, i.e., qualified in radiation protection procedures, with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the ~~Health Physicist~~ radiation protection manager in the Radiation Work Permit (RWP). LA.1

5.7.2 6.1.1.2 In addition to the requirements of 6.1.1.1, above, for areas accessible to personnel with radiation levels such that a major portion of the body could receive in one hour a dose greater than 1000 mrem*, the computer shall be programmed to permit entry through locked doors for any individual requiring access to any such High-High Radiation Areas for the time that access is required.

5.7.3 6.1.1.3 Keys to manually open computer controlled High Radiation Area doors and High-High Radiation Area doors shall be maintained under the Administration control of the shift manager on duty ~~and/~~or the ~~Health Physicist~~.

5.7.4 6.1.1.4 High-High Radiation areas, as defined in 6.1.1.2 above, not equipped with the computerized card readers shall be maintained in accordance with 10 CFR ~~20.203 c.2 (iii)~~, locked except during periods when access to the area is required with positive control over each individual entry, or ~~10 CFR 20.203.c.4.~~ in the case of a High Radiation Area established for a period of 30 days or less, direct surveillance to prevent unauthorized entry may be substituted. Doors shall remain locked except during periods of access by personnel under an approved RWP which shall specify the dose rate levels in the immediate work area and the maximum allowable stay time for individuals in that area. For A.2

20.1601(a)(3) A.2

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5.7 HIGH RADIATION AREAS (Continued)

5.7.4 individual areas accessible to personnel with radiation levels such that a major portion of the body could receive in one hour a dose in excess of 1000 mrem* that are located within large areas, such as the containment, where no enclosure exists for purposes of locking, and no enclosure can be reasonably constructed around the individual areas, then that area shall be roped off, conspicuously posted and a flashing light shall be activated as a warning device. In lieu of the stay time specification of the RWP, direct or remote, such as use of closed circuit TV cameras, continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities within the area.

6.2 PLANT OPERATING PROCEDURES AND PROGRAMS

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

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

b. 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,

c. Station Security Plan implementation,

d. Generating Station Emergency Response Plan implementation,

e. PROCESS CONTROL PROGRAM implementation,

f. OFFSITE DOSE CALCULATION MANUAL implementation, and

g. Fire Protection Program implementation.

~~*Measurement made at 18" from source of radioactivity.~~

<See ITS 5.4>

A.2

ADMINISTRATIVE

A.1 In the conversion of the LaSalle 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 1434, Rev. 1 (i.e., the Improved Standard Technical Specifications (ISTS)).

A.2 CTS 6.1.1.1, 6.1.1.2, and 6.1.1.4 have been revised, as appropriate, to incorporate the latest revision of 10 CFR 20. Since the requirements remain the same, i.e., the station is required to meet 10 CFR 20, the change is considered a presentation preference only. Therefore, the change is considered administrative.

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 CTS 6.1.1.1.c and 6.1.1.3 use the title "Health Physicist." In ITS 5.7.1 and 5.7.3, this specific title is replaced with the generic title "radiation protection manager." CTS 6.1.1.3 uses the title "Shift Manager." In ITS 5.7.3, this specific title is replaced with the generic title "shift manager." The specific titles are proposed to be relocated to the Quality Assurance (QA) Manual. 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 radiation protection manager and shift 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.

TECHNICAL CHANGES - LESS RESTRICTIVE (continued)

"Specific"

None

RELOCATED SPECIFICATIONS

None

CTS 6.1.E/F

7. 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).

8. The Operations Manager or Shift Operations Supervisor shall hold a Senior Reactor Operator License.

See ITS 5.2

D. Qualifications of the station management and operating staff shall meet minimum acceptable levels as described in ANSI N18.1, "Selection and Training of Nuclear Power Plant Personnel," dated March 8, 1971. The Health Physics Supervisor shall meet the requirements of radiation protection manager of Regulatory Guide 1.8, September 1975. The ANSI N18.1-1971 qualification requirements for Radiation Protection Technician may also be met by either of the following alternatives:

1. Individuals who have completed the Radiation Protection Technician training program and have accrued 1 year of working experience in the specialty, or

2. Individuals who have completed the Radiation Protection Technician training program, but have not yet accrued 1 year of working experience in the specialty, who are supervised by on-shift health physics supervision who meet the requirements of ANSI N18.1-1971 Section 4.3.2, "Supervisor Not Requiring AEC Licenses," or Section 4.4.4, "Radiation Protection."

See ITS 5.3

~~E. Retraining and replacement training of Station personnel shall be in accordance with ANSI N18.1, "Selection and Training of Nuclear Power Plant Personnel", dated March 8, 1971 and Appendix "A" of 10 CFR Part 55 and shall include familiarization with relevant industry operational experience.~~

~~F. Retraining shall be conducted at intervals not exceeding 2 years.~~

LA.1

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CTS 6.1.E/F

7. 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).

8. The Operations Manager or Shift Operations Supervisor shall hold a Senior Reactor Operator License.

See ITS 5.2

D. Qualifications of the station management and operating staff shall meet minimum acceptable levels as described in ANSI N18.1, "Selection and Training of Nuclear Power Plant Personnel," dated March 8, 1971. The Health Physics Supervisor shall meet the requirements of radiation protection manager of Regulatory Guide 1.8, September 1975. The ANSI N18.1-1971 qualification requirements for Radiation Protection Technician may also be met by either of the following alternatives:

1. Individuals who have completed the Radiation Protection Technician training program and have accrued 1 year of working experience in the specialty, or

2. Individuals who have completed the Radiation Protection Technician training program, but have not yet accrued 1 year of working experience in the specialty, who are supervised by on-shift health physics supervision who meet the requirements of ANSI N18.1-1971 Section 4.3.2, "Supervisor Not Requiring AEC Licenses," or Section 4.4.4, "Radiation Protection."

See ITS 5.3

E. ~~Retraining and replacement training of Station personnel shall be in accordance with ANSI N18.1, "Selection and Training of Nuclear Power Plant Personnel", dated March 8, 1971 and Appendix "A" of 10 CFR Part 55, and shall include familiarization with relevant industry operational experience.~~

F. ~~Retraining shall be conducted at intervals not exceeding 2 years.~~

LA.1

Page 2 of 2

ADMINISTRATIVE

None

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The details contained in CTS 6.1.E and 6.1.F on retraining 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 N 18.1-1971 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 LaSalle 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

PLANT OPERATING PROCEDURES AND PROGRAMS (Continued)

B. ~~Radiation control procedures shall be maintained, made available to all station personnel, and adhered to. These procedures shall show permissible radiation exposure and shall be consistent with the requirements of 10 CFR 20. This radiation protection program shall be organized to meet the requirements of 10 CFR 20.~~ LA.1

C. TECHNICAL REVIEW AND CONTROL

Procedures required by Specification 6.2.A and 6.2.B and other procedures which affect nuclear safety, as determined by the Plant Manager, and changes thereto, other than editorial or typographical changes, shall be reviewed as follows prior to implementation except as noted in Specification 6.2.D:

1. Each procedure or procedure change shall be independently reviewed by a qualified individual knowledgeable in the area affected other than the individual who prepared the procedure or procedure change. This review shall include a determination of whether or not additional cross-disciplinary reviews are necessary. If deemed necessary, the reviews shall be performed by the qualified review personnel of the appropriate discipline(s).

2. Individuals performing these reviews shall meet the applicable experience requirements of ANSI N18.1-1971, Sections 4.2, 4.3, 4.4, 4.5.1, or 4.6, and be approved by the Plant Manager.

3. Applicable Administrative Procedures recommended by Regulatory Guide 1.33, Plant Emergency Operating Procedures, and changes thereto shall be submitted to the Onsite Review and Investigative Function for review and approval prior to implementation.

4. Review of the procedure or procedure change will include a determination of whether or not an unreviewed safety question is involved. This determination will be based on the review of a written safety evaluation prepared by a qualified individual or documentation that a safety evaluation is not required. Onsite Review, Offsite Review and Commission approval of items involving unreviewed safety questions shall be obtained prior to Station approval for implementation.

5. The Department Head approval authority shall be specified in station procedures.

6. Written records of reviews performed in accordance with this specification shall be prepared and maintained in accordance with Specification 6.5.

7. Editorial and Typographical changes shall be made in accordance with station procedures.

<See ITS 5.4>

CTS 6.2.B

PLANT OPERATING PROCEDURES AND PROGRAMS (Continued)

LA.1

B. ~~Radiation control procedures shall be maintained, made available to all station personnel, and adhered to. These procedures shall show permissible radiation exposure and shall be consistent with the requirements of 10 CFR 20. This radiation protection program shall be organized to meet the requirements of 10 CFR 20.~~

C. TECHNICAL REVIEW AND CONTROL

Procedures required by Specification 6.2.A and 6.2.B and other procedures which affect nuclear safety, as determined by the Plant Manager, and changes thereto, other than editorial or typographical changes, shall be reviewed as follows prior to implementation except as noted in Specification 6.2.D:

1. Each procedure or procedure change shall be independently reviewed by a qualified individual knowledgeable in the area affected other than the individual who prepared the procedure or procedure change. This review shall include a determination of whether or not additional cross-disciplinary reviews are necessary. If deemed necessary, the reviews shall be performed by the qualified review personnel of the appropriate discipline(s).

2. Individuals performing these reviews shall meet the applicable experience requirements of ANSI N18.1-1971, Sections 4.2, 4.3, 4.4, 4.5.1, or 4.6, and be approved by the Plant Manager.

3. Applicable Administrative Procedures recommended by Regulatory Guide 1.33, Plant Emergency Operating Procedures, and changes thereto shall be submitted to the Onsite Review and Investigative Function for review and approval prior to implementation.

4. Review of the procedure or procedure change will include a determination of whether or not an unreviewed safety question is involved. This determination will be based on the review of a written safety evaluation prepared by a qualified individual or documentation that a safety evaluation is not required. Onsite Review, Offsite Review and Commission approval of items involving unreviewed safety questions shall be obtained prior to Station approval for implementation.

5. The Department Head approval authority shall be specified in station procedures.

6. Written records of reviews performed in accordance with this specification shall be prepared and maintained in accordance with Specification 6.5.

7. Editorial and Typographical changes shall be made in accordance with station procedures.

<See ITS 5.4>

ADMINISTRATIVE

None

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The details contained in CTS 6.2.B 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 LaSalle 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

CTS 63

b. Demonstrate for each of the ESF system filter units that an inplace test of the charcoal adsorber shows a penetration and system bypass less than the value specified below, when tested in accordance with ASME N510-1989, at the system flowrate specified below:

ESF Ventilation System	Penetration and System Bypass	Flowrate (cfm)
SBGT System	0.05 %	≥ 3600 and ≤ 4400
CREF System	0.05 %	≥ 3600 and ≤ 4400
CRRF System	2.0 %	≥18000 and ≤ 28900
AEERRF System	2.0 %	≥14000 and ≤ 22800

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, a relative humidity of 70 % and a face velocity as specified below.

ESF Ventilation System	Penetration	Face Velocity (fpm)
SBGT System	0.5 %	40
CREF System	2.5 %	40
CRRF System	15.0 %	80
AEERRF System	15.0 %	80

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

<SEE ITS 5.5>

ESF Ventilation System	Delta P (inches wg)	Flowrate (cfm)
SBGT System	8	≥ 3600 and ≤ 4400
CREF System	8	≥ 3600 and ≤ 4400
CRRF System	3.0	≥ 18000 and ≤ 28900
AEERRF System	3.0	≥ 14000 and ≤ 22800

e. Demonstrate that the heaters for each of the ESF systems dissipate the electrical power specified below when tested in accordance with ASME N510-1989. These readings shall include appropriate corrections for variations from 480 Volts at the bus.

ESF Ventilation System	Wattage (kw)
SBGT System	≥ 21 and ≤ 25
CREF System	≥ 18 and ≤ 22

~~6.3 ACTION TO BE TAKEN IN THE EVENT OF A REPORTABLE EVENT IN PLANT OPERATION~~ A.1

~~The following actions shall be taken for REPORTABLE EVENTS:~~

~~a. The Commission shall be notified and a Licensee Event Report submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50, and~~

~~b. Each REPORTABLE EVENT shall be reviewed by the Onsite Review and Investigative Function.~~ LA.1

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PLANT OPERATING PROCEDURES AND PROGRAMS (Continued)

<SEE ITS 5.5>

e. Demonstrate that the heaters for each of the ESF systems dissipate the electrical power specified below when tested in accordance with ASME N510-1989. These readings shall include appropriate corrections for variations from 480 Volts at the bus.

ESF Ventilation System	Wattage (kw)
SBGT System	≥ 21 and ≤ 25
CREF System	≥ 18 and ≤ 22

6.3 ACTION TO BE TAKEN IN THE EVENT OF A REPORTABLE EVENT IN PLANT OPERATION — A.1

The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified and a Licensee Event Report submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50, and

b. Each REPORTABLE EVENT shall be reviewed by the Onsite Review and Investigative Function. — LA.1

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DISCUSSION OF CHANGES
CTS: 6.3 - REPORTABLE EVENT ACTION

ADMINISTRATIVE

A.1 The requirements of CTS 6.3.a (Reportable Event notification requirements) are to be removed from the Technical Specifications. CTS 6.3.a requires, in the case of a Reportable Event, that the Commission be notified and a report submitted pursuant to the requirements of 10 CFR 50.72 and 10 CFR 50.73. The requirements of CTS 6.3.a of Reportable Event Action are contained in 10 CFR 50.72 and 10 CFR 50.73. Therefore, there is no need to repeat these requirements in the Technical Specifications. Since these requirements are contained in the regulations and since the LaSalle 1 and 2 Operating Licenses require compliance with 10 CFR 50, the change is considered to be administrative in nature.

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The requirements of CTS 6.3.b (Reportable Events reviews by the Onsite Review and Investigative Function are proposed to be relocated to the Quality Assurance (QA) Manual. Given that these reviews and submittal of results are required following the event without a specified completion time, the proposed relocated requirements are not necessary to assure operation of the facility in a safe manner. As such, the relocated requirements 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.

"Specific"

None

RELOCATED SPECIFICATIONS

None

CTS 6.4

~~6.4 ACTION TO BE TAKEN IN THE EVENT A SAFETY LIMIT IS EXCEEDED~~

~~If a safety limit is exceeded, the reactor shall be shut down immediately pursuant to Specification 2.1.1, 2.1.2 and 2.1.3, and critical reactor operation shall not be resumed until authorized by the NRC.~~ A.1 ~~The conditions of shutdown shall be promptly reported to the Site Vice President or his designated alternate. The incident shall be reviewed by the Onsite and Offsite Review and Investigative Functions~~ LA.1 ~~and a separate Licensee Event Report for each occurrence shall be prepared in accordance with Section 50.73 to 10 CFR Part 50. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within one hour.~~ A.1 ~~The Site Vice President and the Director of Safety Review shall be notified within 24 hours.~~ LA.1

6.5 PLANT OPERATING RECORDS

SEE CTS 6.5

A. Records and/or logs relative to the following items shall be kept in a manner convenient for review and shall be retained for at least 5 years:

1. Records of normal plant operation, including power levels and periods of operation at each power level;
2. Records of principal maintenance and activities, including inspection and repair, regarding principal items of equipment pertaining to nuclear safety;
3. Records and reports of reportable events;
4. Records and periodic checks, inspection and/or calibrations performed to verify that the surveillance requirements (see Section 4 of these specifications) are being met. All equipment failing to meet surveillance requirements and the corrective action taken shall be recorded;
5. Records of changes to operating procedures;
6. Shift Manager logs; and
7. Byproduct material inventory records and source leak test results.

B. Records and/or logs relative to the following items shall be recorded in a manner convenient for review and shall be retained for the life of the plant:

1. Substitution or replacement of principal items of equipment pertaining to nuclear safety;
2. Changes made to the plant as it is described in the SAR;
3. Records of new and spent fuel inventory and assembly histories;
4. Updated, corrected, and as-built drawings of the plant;

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~~6.4 ACTION TO BE TAKEN IN THE EVENT A SAFETY LIMIT IS EXCEEDED~~ A.1

~~If a safety limit is exceeded, the reactor shall be shut down immediately pursuant to Specification 2.1.1, 2.1.2 and 2.1.3, and critical reactor operation shall not be resumed until authorized by the NRC.~~ A.1 ~~The conditions of shutdown shall be promptly reported to the Site Vice President or his designated alternate. The incident shall be reviewed by the Onsite and Offsite Review and Investigative Functions~~ LA.1 ~~and a separate Licensee Event Report for each occurrence shall be prepared in accordance with Section 50.73 to 10 CFR Part 50. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within one hour.~~ A.1 ~~The Site Vice President and the Director of Safety Review shall be notified within 24 hours.~~ LA.1

6.5 PLANT OPERATING RECORDS

<SEE CTS 6.5>

A. Records and/or logs relative to the following items shall be kept in a manner convenient for review and shall be retained for at least 5 years:

1. Records of normal plant operation, including power levels and periods of operation at each power level;
2. Records of principal maintenance and activities, including inspection and repair, regarding principal items of equipment pertaining to nuclear safety;
3. Records and reports of reportable events;
4. Records and periodic checks, inspection and/or calibrations performed to verify that the surveillance requirements (see Section 4 of these specifications) are being met. All equipment failing to meet surveillance requirements and the corrective action taken shall be recorded;
5. Records of changes to operating procedures;
6. Shift Manager logs; and
7. Byproduct material inventory records and source leak test results.

B. Records and/or logs relative to the following items shall be recorded in a manner convenient for review and shall be retained for the life of the plant:

1. Substitution or replacement of principal items of equipment pertaining to nuclear safety;
2. Changes made to the plant as it is described in the SAR;
3. Records of new and spent fuel inventory and assembly histories;
4. Updated, corrected, and as-built drawings of the plant;
5. Records of plant radiation and contamination surveys;
6. Records of offsite environmental monitoring surveys;

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DISCUSSION OF CHANGES
CTS: 6.4 - SAFETY LIMIT VIOLATION

ADMINISTRATIVE

A.1 The current Safety Limit Violation requirements of Specification 6.4, as they relate to NRC notification and permission to restart the unit 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 LaSalle 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 6.4 requirements for: 1) notification of the Site Vice President and Director of Safety Review in the event of a Safety Limit violation; and 2) the Onsite and Offsite Review Investigative Functions to review the Safety Limit Violation Report, are proposed to be relocated to the Quality Assurance (QA) Manual. Given that the notification occurs following the Safety Limit violation and that the Safety Limit Violation Report is an after-the-fact report, the proposed relocated requirements are 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 requirements 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.

"Specific"

None

RELOCATED SPECIFICATIONS

None

6.4 ACTION TO BE TAKEN IN THE EVENT A SAFETY LIMIT IS EXCEEDED

<SEE CTS 6.4>

If a safety limit is exceeded, the reactor shall be shut down immediately pursuant to Specification 2.1.1, 2.1.2 and 2.1.3, and critical reactor operation shall not be resumed until authorized by the NRC. The conditions of shutdown shall be promptly reported to the Site Vice President or his designated alternate. The incident shall be reviewed by the Onsite and Offsite Review and Investigative Functions and a separate Licensee Event Report for each occurrence shall be prepared in accordance with Section 50.73 to 10 CFR Part 50. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within one hour. The Site Vice President and the Director of Safety Review shall be notified within 24 hours.

LA.1

~~6.5 PLANT OPERATING RECORDS~~

~~A. Records and/or logs relative to the following items shall be kept in a manner convenient for review and shall be retained for at least 5 years:~~

~~1. Records of normal plant operation, including power levels and periods of operation at each power level;~~

~~2. Records of principal maintenance and activities, including inspection and repair, regarding principal items of equipment pertaining to nuclear safety;~~

~~3. Records and reports of reportable events;~~

~~4. Records and periodic checks, inspection and/or calibrations performed to verify that the surveillance requirements (see Section 4 of these specifications) are being met. All equipment failing to meet surveillance requirements and the corrective action taken shall be recorded;~~

~~5. Records of changes to operating procedures;~~

~~6. Shift Manager logs; and~~

~~7. Byproduct material inventory records and source leak test results.~~

~~B. Records and/or logs relative to the following items shall be recorded in a manner convenient for review and shall be retained for the life of the plant:~~

~~1. Substitution or replacement of principal items of equipment pertaining to nuclear safety;~~

~~2. Changes made to the plant as it is described in the SAR;~~

~~3. Records of new and spent fuel inventory and assembly histories;~~

~~4. Updated, corrected, and as-built drawings of the plant;~~

Page 1 of 4

CTS 6.5

PLANT OPERATING RECORDS (Continued)

LA.1

5. Records of plant radiation and contamination surveys;
6. Records of offsite environmental monitoring surveys;
7. Records of radiation exposure for all plant personnel, including all contractors and visitors to the plant, in accordance with 10 CFR Part 20;
8. Records of radioactivity in liquid and gaseous wastes released to the environment;
9. Records of transient or operational cycling for those components that have been designed to operate safety for a limited number of transient or operational cycles (identified in Table 5.7.1-1);
10. Records of individual staff members indicating qualifications, experience, training, and retraining;
11. Inservice inspections of the reactor coolant system;
12. Minutes of meetings and results of reviews and audits performed by the offsite and onsite review and audit functions;
13. Records of reactor tests and experiments;
14. Records of Quality Assurance activities required by the QA Manual, except for those items specified in Section 6.5.A;
15. Records of reviews performed for changes made to procedures on equipment or reviews of tests and experiments pursuant to 10 CFR 50.59;
16. Records of the service lives of all hydraulic and mechanical snubbers required by specification 3.7.9 including the date at which the service life commences and associated installation and maintenance records;
17. Records of analyses required by the radiological environmental monitoring program;
18. Records of reviews performed for changes made to the OFFSITE DOSE CALCULATION MANUAL and the PROCESS CONTROL PROGRAM; and
19. Records of pre-stressed concrete containment tendon surveillances.

6.6 REPORTING REQUIREMENTS

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

<SEE ITS 5.6>

Page 2 of 4

SEE CTS 6.4

6.4 ACTION TO BE TAKEN IN THE EVENT A SAFETY LIMIT IS EXCEEDED

If a safety limit is exceeded, the reactor shall be shut down immediately pursuant to Specification 2.1.1, 2.1.2 and 2.1.3, and critical reactor operation shall not be resumed until authorized by the NRC. The conditions of shutdown shall be promptly reported to the Site Vice President or his designated alternate. The incident shall be reviewed by the Onsite and Offsite Review and Investigative Functions and a separate Licensee Event Report for each occurrence shall be prepared in accordance with Section 50.73 to 10 CFR Part 50. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within one hour. The Site Vice President and the Director of Safety Review shall be notified within 24 hours.

6.5 PLANT OPERATING RECORDS

A. Records and/or logs relative to the following items shall be kept in a manner convenient for review and shall be retained for at least 5 years: LA.1

1. Records of normal plant operation, including power levels and periods of operation at each power level;
2. Records of principal maintenance and activities, including inspection and repair, regarding principal items of equipment pertaining to nuclear safety;
3. Records and reports of reportable events;
4. Records and periodic checks, inspection and/or calibrations performed to verify that the surveillance requirements (see Section 4 of these specifications) are being met. All equipment failing to meet surveillance requirements and the corrective action taken shall be recorded;
5. Records of changes to operating procedures;
6. Shift Manager logs; and
7. Byproduct material inventory records and source leak test results.

B. Records and/or logs relative to the following items shall be recorded in a manner convenient for review and shall be retained for the life of the plant:

1. Substitution or replacement of principal items of equipment pertaining to nuclear safety;
2. Changes made to the plant as it is described in the SAR;
3. Records of new and spent fuel inventory and assembly histories;
4. Updated, corrected, and as-built drawings of the plant;
5. Records of plant radiation and contamination surveys;
6. Records of offsite environmental monitoring surveys;

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CTS 6.5

PLANT OPERATING RECORDS (Continued)

7. Records of radiation exposure for all plant personnel, including all contractors and visitors to the plant, in accordance with 10 CFR Part 20;

8. Records of radioactivity in liquid and gaseous wastes released to the environment;

LA.1

9. Records of transient or operational cycling for those components that have been designed to operate safety for a limited number of transient or operational cycles (identified in Table 5.7.1-1);

10. Records of individual staff members indicating qualifications, experience, training, and retraining;

11. Inservice inspections of the reactor coolant system;

12. Minutes of meetings and results of reviews and audits performed by the offsite and onsite review and audit functions;

13. Records of reactor tests and experiments;

14. Records of Quality Assurance activities required by the QA Manual, except for those items specified in Section 6.5.A;

15. Records of reviews performed for changes made to procedures on equipment or reviews of tests and experiments pursuant to 10 CFR 50.59;

16. Records of the service lives of all hydraulic and mechanical snubbers required by specification 3.7.9 including the date at which the service life commences and associated installation and maintenance records;

17. Records of analyses required by the radiological environmental monitoring program;

18. Records of reviews performed for changes made to the OFFSITE DOSE CALCULATION MANUAL and the PROCESS CONTROL PROGRAM; and

19. Records of pre-stressed concrete containment tendon surveillances.

6.6 REPORTING REQUIREMENTS

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

SEE ITS 5.6

page 4 of 4

DISCUSSION OF CHANGES
CTS: 6.5 - PLANT OPERATING RECORDS

ADMINISTRATIVE

None

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The details contained in CTS 6.5 are proposed to be relocated to the Quality Assurance (QA) Manual. The requirement for retention of records related to activities affecting quality is contained in 10 CFR 50, Appendix B, Criterion XVII and other sections of 10 CFR 50 that are applicable to LaSalle 1 and 2 (i.e., 10 CFR 50.71, 10 CFR 73, etc.). These record retention requirements provide a record of certain activities important to plant safety, but the records themselves do not assure safe operation of the facility since review of these records is a post-compliance review. Relocation of these CTS provisions to the QA Manual will provide adequate controls over record retention requirements for LaSalle 1 and 2. The QA Manual will be revised to contain adequate detail with respect to these requirements to ensure recordkeeping is implemented in an appropriate manner. 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 QA Manual will be controlled by the provisions of 10 CFR 50.54.

"Specific"

None

RELOCATED SPECIFICATIONS

None

CTS 6.7

<See ITS chapter 1.0>

e. The suppression chamber is OPERABLE pursuant to Specification 3.6.2.1.

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

g. Primary containment structural integrity has been verified in accordance with Surveillance Requirement 4.6.1.1.e.

~~PROCESS CONTROL PROGRAM~~

~~1.33 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, 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 20, 61, and 71, State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.~~ LA.1

PURGE - PURGING

1.34 PURGE or PURGING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

RATED THERMAL POWER

1.35 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3323 MWT.

REACTOR PROTECTION SYSTEM RESPONSE TIME

1.36 REACTOR PROTECTION SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

REPORTABLE EVENT

1.37 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

ROD DENSITY

1.38 ROD DENSITY shall be the number of control rod notches inserted as a fraction of the total number of control rod notches. All rods fully inserted is equivalent to 100% ROD DENSITY.

<See ITS Chapter 1.0>

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.

<SEE ITS 5.6>

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 U.S. Nuclear Regulatory Commission Document Control Desk with copies to the Regional Administrator and Resident Inspector.

B. Deleted

C. Unique Reporting Requirements

1. Special Reports shall be submitted to the Regional Administrator of the NRC Regional Office within the time period specified for each report.

LA.1

~~6.7 PROCESS CONTROL PROGRAM (PCP)*~~

~~6.7.1 The PCP shall be approved by the Commission prior to implementation.~~

~~6.7.2 Licensee initiated changes to the PCP:~~

~~a. Shall be documented and records of reviews performed shall be retained as required by Specification 6.5.B.18. This documentation shall contain:~~

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

~~2) 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.~~

~~b. Shall become effective upon review and acceptance by the Onsite Review and Investigative Function.~~

~~*The Process Control Program (PCP) is common to La Salle Unit 1 and La Salle Unit 2.~~

<See ITS Chapter 1.0>

DEFINITIONS

PRIMARY CONTAINMENT INTEGRITY (Continued)

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

g. Primary containment structural integrity has been verified in accordance with Surveillance Requirement 4.6.1.1.e.

~~PROCESS CONTROL PROGRAM~~

~~1.33 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, 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 20, 61, and 71, State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.~~

LA.1

PURGE - PURGING

1.34 PURGE or PURGING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

RATED THERMAL POWER

1.35 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3323 MWT.

REACTOR PROTECTION SYSTEM RESPONSE TIME

1.36 REACTOR PROTECTION SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

REPORTABLE EVENT

1.37 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

ROD DENSITY

1.38 ROD DENSITY shall be the number of control rod notches inserted as a fraction of the total number of control rod notches. All rods fully inserted is equivalent to 100% ROD DENSITY.

<See ITS Chapter 1.0>

Core Operating Limits Report (Continued)

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.

SEE ITS 5.6

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 U.S. Nuclear Regulatory Commission Document Control Desk with copies to the Regional Administrator and Resident Inspector.

B. Deleted

C. Unique Reporting Requirements

1. Special Reports shall be submitted to the Regional Administrator of the NRC Regional Office within the time period specified for each report.

LA.1

~~6.7 PROCESS CONTROL PROGRAM (PCP)*~~

~~6.7.1 The PCP shall be approved by the Commission prior to implementation.~~

~~6.7.2 Licensee initiated changes to the PCP:~~

~~a. Shall be documented and records of reviews performed shall be retained as required by Specification 6.5.B.18. This documentation shall contain:~~

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

~~2) 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.~~

~~b. Shall become effective upon review and acceptance by the Onsite Review and Investigative Function.~~

~~*The Process Control Program (PCP) is common to La Salle Unit 1 and La Salle Unit 2.~~

ADMINISTRATIVE

None

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The details contained in CTS 6.7 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 LaSalle 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

6.8 OFFSITE DOSE CALCULATION MANUAL (ODCM)*

6.8.1 The ODCM shall be approved by the Commission prior to implementation.

6.8.2 Licensee initiated changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained as required by Specification 6.5.B.18. This documentation shall contain:

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

2) A determination that the change will maintain the level of radioactive effluent control required by 10 CFR 20.106, 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.

b. Shall become effective after review and acceptance by the On-Site Review and Investigative Function and the approval of the Plant Manager on the date specified by the On-Site Review and Investigative Function.

c. 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 Annual Radioactive Effluent Release 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.

<SEE ITS 5.5>

6.9 MAJOR CHANGES TO RADIOACTIVE WASTE TREATMENT SYSTEMS

6.9.1 Licensee initiated major changes to the radioactive waste treatment systems (liquid, gaseous and solid): L.1

a. Shall be reported to the Commission in the Monthly Operating Report for the period in which the evaluation was reviewed by the OnSite Review and Investigative Function. The discussion of each change shall contain: LA.1

1. A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR 50.59;

2. Sufficient detailed information to totally support the reason for the change without benefit or additional or supplemental information;

3. A detailed description of the equipment, components and processes involved and the interfaces with other plant systems;

*The OFFSITE DOSE CALCULATION MANUAL (ODCM) is common to La Salle Unit 1 and La Salle Unit 2.

<SEE ITS 5.5>

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 CTS 6.9

MAJOR CHANGES TO RADIOACTIVE WASTE TREATMENT SYSTEMS (Continued)

LA.1

4. An evaluation of the change which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the license application and amendments thereto;

5. An evaluation of the change which shows the expected maximum exposures to individual in the unrestricted area and to the general population that differ from those previously estimated in the license application and amendments thereto;

6. A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period to when the changes are to be made;

7. An estimate of the exposure to plant operating personnel as a result of the change; and

8. Documentation of the fact that the change was reviewed and found acceptable by the Onsite Review and Investigative Function.

b. Shall become effective upon review and acceptance by the Onsite Review and Investigative Function.

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Monthly Operating Report (Continued)

~~A report of any major changes to the radioactive waste treatment systems shall be submitted with the Monthly Operating Report for the period in which the evaluation was reviewed and accepted by Onsite Review and Investigative Function.~~ L.1

6. Core Operating Limits Report

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:

(1) The Average Planar Linear Heat Generation Rate (APLHGR) for Technical Specification 3.2.1.

(2) The minimum Critical Power Ratio (MCPR) scram time, dependent MCPR limits, and power and flow dependent MCPR limits for Technical Specification 3.2.3. Effects of analyzed equipment out of service are included.

(3) The Linear Heat Generation Rate (LHGR) for Technical Specification 3.2.4.

(4) The Rod Block Monitor Upscale Instrumentation Setpoints for Technical Specification Table 3.3.6-2.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC. For LaSalle County Station Unit 1, the topical reports are:

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

(2) Letter, Ashok C. Thadani (NRC) to R.A. Copeland (SPC), "Acceptance for Referencing of ULTRAFLOW™ Spacer on 9x9-IX/X BWR Fuel Design," July 28, 1993.

(3) 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, XN-NF-524(P)(A) Revision 2, and Supplement 1 Revision 2, Supplement 2, Advanced Nuclear Fuels Corporation, November 1990.

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

<See ITS 5.6>

6.8 OFFSITE DOSE CALCULATION MANUAL (ODCM)*

6.8.1 The ODCM shall be approved by the Commission prior to implementation.

6.8.2 Licensee initiated changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained as required by Specification 6.5.B.18. This documentation shall contain:

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

2) A determination that the change will maintain the level of radioactive effluent control required by 10 CFR 20.106, 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.

b. Shall become effective after review and acceptance by the On-Site Review and Investigative Function and the approval of the Plant Manager on the date specified by the On-Site Review and Investigative Function.

c. 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 Annual Radioactive Effluent Release 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.

<SEE ITS 5.5>

6.9 MAJOR CHANGES TO RADIOACTIVE WASTE TREATMENT SYSTEMS

LA.1

6.9.1 Licensee initiated major changes to the radioactive waste treatment systems (liquid, gaseous and solid):

L.1

a. Shall be reported to the Commission in the Monthly Operating Report for the period in which the evaluation was reviewed by the Onsite Review and Investigative Function. The discussion of each change shall contain:

1. A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR 50.59;

*The OFFSITE DOSE CALCULATION MANUAL (ODCM) is common to La Salle Unit 1 and La Salle Unit 2.

<SEE ITS 5.5>

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CTS 6.9

MAJOR CHANGES TO RADIOACTIVE WASTE TREATMENT SYSTEMS (Continued)

2. Sufficient detailed information to totally support the reason for the change without benefit or additional or supplemental information;

3. A detailed description of the equipment, components and processes involved and the interfaces with other plant systems;

4. An evaluation of the change which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the license application and amendments thereto;

5. An evaluation of the change which shows the expected maximum exposures to individual in the unrestricted area and to the general population that differ from those previously estimated in the license application and amendments thereto;

6. A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period to when the changes are to be made;

7. An estimate of the exposure to plant operating personnel as a result of the change; and

8. Documentation of the fact that the change was reviewed and found acceptable by the Onsite Review and Investigative Function.

b. Shall become effective upon review and acceptance by the Onsite Review and Investigative Function.

LA.1

Page 5 of 6

CTS 6.9

prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.

3. Annual Radiological Environmental Operating Report*

<See ITS 5.6>

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

4. Annual Radioactive Effluent Release Report**

The Annual Radioactive Effluent Release Report covering the operation of the unit during the previous calendar year shall be submitted prior to May 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 unit. The material provided shall be (1) consistent with the objectives outlined in the ODCM and PCP and (2) in conformance with 10 CFR 50.36a and Section IV.B.1 of Appendix I to 10 CFR Part 50.

5. Monthly Operating Report

Routine reports of operating statistics and shutdown experience, including documentation of all challenges to safety/relief valves, shall be submitted on a monthly basis to the addressees specified in 10 CFR 50.4 no later than the 15th of each month following the calendar month covered by the report.

~~A report of any major changes to the radioactive waste treatment systems shall be submitted with the Monthly Operating Report for the period in which the evaluation was reviewed and accepted by Onsite Review and Investigative Function.~~ L.1

6. Core Operating Limits Report

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:

---

* A single submittal may be made for a multi-unit station.

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

<See ITS 5.6>

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DISCUSSION OF CHANGES
CTS: 6.9 - MAJOR CHANGES TO RADIOACTIVE WASTE
TREATMENT SYSTEMS

ADMINISTRATIVE

None

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 CTS 6.9 provides requirements regarding major changes to the radioactive waste treatment systems. The requirements are: 1) a description of the content of the report to be submitted to the NRC regarding the major changes; and 2) a requirement that the major changes become effective upon review and acceptance by the Onsite Review and Investigative Function. These requirements will be relocated to the Offsite Dose Calculation Manual (ODCM), since the ODCM deals with systems that handle radioactive wastes. The requirements do not deal with any systems that are required to mitigate a design basis accident or transient. Thus, the Technical Specifications do not need to include these requirements to assure operation of the facility in a safe manner. Given the above, the requirements of CTS 6.9 can be adequately maintained within the ODCM without impacting public health and safety. Changes to the ODCM will be controlled by the provisions of the ODCM change process described in Chapter 5 of the ITS.

"Specific"

L.1 CTS 6.6.A.5 and CTS 6.9.1.a require submitting a report of any major changes to the radioactive waste treatment system with the Monthly Operating Report. This reporting requirement is being changed to only require submittal of major changes to the radioactive waste treatment system in the Radioactive Release Report. As described in Discussion of Change LA.1 above, this requirement will be located in the ODCM. Changes to the radioactive waste systems are controlled in accordance with the plant modification process and the provisions of 10 CFR 50.59. Additionally, changes to the UFSAR (which describes the radioactive waste system) are submitted to the NRC every two years. Consequently, changes to the radwaste system are controlled in accordance with

TECHNICAL CHANGES - LESS RESTRICTIVE

L.1 (cont'd) plant programs and any modifications are communicated to the NRC. There is no requirement for the NRC to approve the Monthly Operating Report that might contain information on changes to the radwaste systems. Therefore, this change has no impact on the safe operation of the plant. This change is consistent with the BWR ISTS, NUREG-1434, Rev. 1.

RELOCATED SPECIFICATIONS

None

The following blank pages, have been deleted:

6-5 through 6-12 and 6-14.

TSTF-65 Reviewer's Note Not Shown [1]

TSTF-65

<CTS>

5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

<6.1.A.2>

5.1.1 The ~~[Plant Superintendent]~~ Station manager [2] shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence. [2] TSTF-65

~~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]

Fig 6.1^e-3 footnote (c)

5.1.2 (A) The ~~[Shift Supervisor (SS)]~~ Senior Reactor Operator (SRO) [2] 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.~~ While either unit is in MODE 1, 2, or 3 [3] During any absence of the [SS] from the control room while ~~the unit is~~ 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.

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

<6.1.A> 5.2.1 Onsite and Offsite Organizations

TSTF-65

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

<6.1.A.1> a. Lines of authority, responsibility, and communication shall be defined and established throughout highest management levels, intermediate levels, and all operating organization positions. These relationships shall be documented and updated, as appropriate, in organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements, including the plant-specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications 2, shall be documented in the ~~FSAR~~ Quality Assurance Manual. 2 1 ~~QA Plan~~ TSTF-65

<6.1.A.2> b. The ~~Plant Superintendent~~ station manager 2 TSTF-65 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

<6.1.A.3> c. The ~~A specified~~ corporate ~~executive position~~ officer TSTF-65 TSTF-65 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 1

<6.1.A.4> d. The individuals who train the operating staff, ~~carry out~~ ~~health physics~~ radiation protection 3, or ~~perform~~ or perform 4 quality assurance functions, 4 may report to the appropriate onsite manager; however, these individuals shall have sufficient organizational freedom to ensure their independence from operating pressures. 4

<6.1.C> 5.2.2 Unit Staff

The unit staff organization shall include the following:

<Fig. 6.1-3 footnotes (a) & (b)> a. ~~A non-licensed operator shall be assigned to each reactor containing fuel and an additional non-licensed operator~~ 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. 5 4

(continued)

<CTS>

5.2 Organization

5.2.2 Unit Staff (continued)

shall be assigned for each control room from which a reactor is operating in MODES 1, 2, or 3. 5

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

<6.1.C.1> 8

<Fig. 6.1-3 footnote (b)> TSTF-65 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.

<Fig 6.1-3 footnote(a)> 2 to

c. Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i) and Specification 5.2.2.a and 5.2.2.g for a period of time not be exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements. 6

<6.1.C.2> 1

<6.1.C.2 footnote *> radiation protection technician

d. A Health Physics 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. TSTF-65

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). 1

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:

(continued)

<CTS>

5.2 Organization

5.2.2 Unit Staff (continued)

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

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 [~~Plant Superintendent~~] station 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.~~ 1 TSTF-65

~~Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the [~~Plant Superintendent~~] station manager or his designee to ensure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized.~~ TSTF-65

OR

<6.1.C.7>

~~[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

<6.1.C.8>

TSTF-258 changes not adopted 8

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

<6.1.C.6>

<Fig. 6.1-3 footnote (d)>

g. The Shift Technical Advisor (STA) shall provide advisory technical support to the Shift ~~Supervisor (SS)~~ 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-1434, 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

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

Insert 5.3.1-A

TSTF-258 change not adopted 3

<CTS>

Insert 5.3.1-A

<6.1.D> ANSI N18.1-1971, except the radiation protection manager who shall meet the requirements of "radiation protection manager" in Regulatory Guide 1.8, September 1975. Also, the ANSI N18.1-1971 qualification requirements for "radiation protection technician" may be met by either of the following alternatives:

a. Individuals who have completed the radiation protection technician training program and have accrued one year of working experience in the specialty; or <6.1.D.1>

b. Individuals who have completed the radiation protection technician training program, but have not yet accrued one year of working experience in the specialty, who are supervised by on-shift radiation protection supervision who meet the requirements of ANSI N18.1-1971, Section 4.3.2 or Section 4.4.4. <6.1.D.2>

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ITS: 5.3 - UNIT STAFF QUALIFICATIONS

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 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 5.3:

    ISTS 5.3.2 was added to define the 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

<6.2.A> 5.4.1 Written procedures shall be established, implemented, and maintained covering the following activities:

<6.2.A.a> a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978; [1]

<6.2.A.b> b. The emergency operating procedures required to implement the requirements of NUREG-0737 and ~~to~~ NUREG-0737, Supplement 1, as stated in ~~(~~Generic Letter 82-33~~)~~, Section 7.1; [2]

~~c. Quality assurance for effluent and environmental monitoring;~~

<6.2.A.g> c ~~d~~. Fire Protection Program implementation; and

<DOC M.1> d ~~e~~. All programs specified in Specification 5.5.

[3]

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, 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 the remaining 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.

<1,27> 5.5.1 Offsite Dose Calculation Manual (ODCM)

<6,8>

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]. 2

3 (c.) Licensee initiated changes to the ODCM:

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

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

(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; do 1

3 (2) ~~b.~~ Shall become effective after ~~review and acceptance by the [onsite review function] and~~ the approval of the ~~[Plant Superintendent]~~; and station manager TSTF-65 TSTF-76 2

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.

(continued)

<CTS>

5.5 Programs and Manuals

<1.27> <6.8> 5.5.1 Offsite Dose Calculation Manual (ODCM) (continued)

Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

<6.2.F.1> 5.5.2 Primary Coolant Sources Outside Containment

2 /Low Pressure Coolant Injection

2 Containment monitoring

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 Core Spray, Residual Heat Removal, Reactor Core Isolation Cooling, hydrogen recombiner, process sampling, 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 ~~refueling cycle~~ intervals ~~or less~~. 4

4 24-month

The provisions of SR 3.0.2 are applicable to the 24 month Frequency for performing integrated System leak test activities. 4

<6.2.F.3> 5.5.3 Post Accident Sampling

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

a. Training of personnel;

b. Procedures for sampling and analysis; and

c. Provisions for maintenance of sampling and analysis equipment.

<6.2.F.4> 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

(continued)

<CTS>

5.5 Programs and Manuals

5.5.4 **Radioactive Effluent Controls Program** (continued)

<6.2.F.4>

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 10 CFR 20, Appendix B, Table 2, Column 2 to 10 CFR 20.1001 – 20.2402;

TSTF-258

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 the site from radioactive material released in gaseous effluents to areas at or ~~beyond~~ the site boundary ~~conforming to the dose associated with 10 CFR 20, Appendix B, Table 2, Column 1~~;

Insert 5.5.4.a

TSTF-258

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each

(continued)

<CTS>

Insert 5.5.4a

<6.2.F.4>
shall be in accordance with the following:

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.

<CTS>

5.5 Programs and Manuals

5.5.4 <6.2.F.4> Radioactive Effluent Controls Program (continued)

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 7

7 Insert 5.5.4.j TSTF -258

j. Limitations on the annual dose or dose commitment to any member of the public, beyond the site boundary, due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190; and 7 TSTF -258

5.5.5 <5.7> Component Cyclic or Transient Limit

This program provides controls to track the U 8 FSAR, ~~Section~~ Table 5.2-4 2, cyclic and transient occurrences to ensure that components are maintained within the design limits.

5.5.6 <6.2.F.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~~]~~ Insert 5.5.6.a 2

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Tendon Surveillance Program inspection frequencies. 2

5.5.7 <4.0.5> Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 ~~components~~ pumps and valves 9 ~~including applicable supports~~ TSTF-279. ~~The program shall include the following:~~ 9 9

(continued)

<CTS>

Insert 5.5.4.b

<6.2.F.4>
k. Limitations on venting and purging of the primary containment through the Primary Containment Vent and Purge System or Standby Gas Treatment System to maintain releases as low as reasonably achievable.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluents Control Program Surveillance Frequencies. TSTF-258

Insert 5.5.6.a

<6.2.F.6>
, except that the Unit 1 and Unit 2 primary containments shall be treated as twin containments even though the initial structural integrity tests were not within 2 years of each other.

<CTS>

5.5 Programs and Manuals

<4.0.5> 5.5.7 Inservice Testing Program (continued)

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

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

21

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.

<6.2.F.8> 5.5.8 Ventilation Filter Testing Program (VFTP)

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

Insert 5.5.8

Insert 5.5.8.f from page 5.0-13

a. Demonstrate for each of the ESF systems that an inplace test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass < ~~[~~0.05~~]~~% when tested in 2

(continued)

19

<CTS>

<6.2.F.8>

Tests described in Specification 5.5.8.a and 5.5.8.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 5.5.8.c shall be performed once per 24 months; after 720 hours of system operation; 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 5.5.8.d and 5.5.8.e shall be performed once per 24 months.

⟨CTS⟩

5.5 Programs and Manuals

⟨6.2.F.8⟩

5.5.8 Ventilation Filter Testing Program (VFTP) (continued)

accordance with Regulatory Guide 1.52, Revision 2, and ASME N510-1989 at the system flowrate specified below (± 10%). [2] ANSI/ [1]

[2] Standby Gas Treatment (SGT) System
Control Room Area Filtration (CRAF) System
Emergency Makeup Air Filter Units (EMUs)

ESF Ventilation System	Flowrate (cfm) [2]
[ ] [2]	[ ]
	≥ 3600 and ≤ 4400
	≥ 3600 and ≤ 4400

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

[2] less than the value specified below

[2] INSERT 5.5.8.b

ESF Ventilation System	Flowrate
[ ]	[ ]

[2]

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 ~~specified below~~: [2] } [1]

[2] a relative humidity of 70% and a face velocity as

[2] Insert 5.5.8.c

ESF Ventilation System	Penetration	RH
[ ]	[ ]	[ ]

[2]

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.

[11]

(continued)

<CTS>

Insert 5.5.8.b

<6.2.F.8>

ESF Ventilation System	Penetration and System Bypass	Flowrate (cfm)
SGT System	0.05%	≥ 3600 and ≤ 4400
CRAF System		
EMUs	0.05%	≥ 3600 and ≤ 4400
Control Room Recirculation Filters (CRRFs)	2.0%	≥ 18000 and ≤ 28900
Auxiliary Electric Equipment Room Recirculation Filters (AEERRFs)	2.0%	≥ 14000 and ≤ 22800

INSERT 5.5.8.c

<6.2.F.8>

ESF Ventilation System	Penetration	Face Velocity (fpm)
SGT System	0.5%	40
CRAF System		
EMUs	2.5%	40
CRRFs	15.0%	80
AEERRFs	15.0%	80

5.5 Programs and Manuals

5.5.8 Ventilation Filter Testing Program (VFTP) (continued)

⟨6.2.F.8⟩

d. Demonstrate for each of the ESF systems that the pressure drop across the combined HEPA filters, the prefilters, and the charcoal adsorbers is less than the value specified below when tested in accordance with [Regulatory Guide 1.52, Revision 2, and ASME N510-1989] at the system flowrate specified below (± 10%): 1

12 moisture separator, heater,

12

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

2 Insert 5.5.8.d

2

, corrected for voltage variations at the 480V bus,

12

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

2 ANSI/

ESF Ventilation System	Wattage (kW)
SGT System	≥ 21 and ≤ 25
CRAF System	≥ 18 and ≤ 22
EMUs	

2

2

19

Move this Insert 5.5.8 to to page 5.0-13

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

5.5.9 Explosive Gas and Storage Tank Radioactivity Monitoring Program

⟨3/4.11.1.1⟩

2 Condenser offgas treatment system

This program provides controls for potentially explosive gas mixtures contained in the ~~Waste Gas Holdup System~~, [the quantity of radioactivity contained in gas storage tanks or fed into the offgas treatment system,] and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks. ~~The gaseous radioactivity quantities shall be determined following the methodology in [Branch Technical Position (BTP) ETSB 11-5, "Postulated Radioactive Release due to Waste Gas System Leak or Failure"]. The liquid radwaste quantities shall be determined in accordance with [Standard Review Plan, Section 15.7.3, "Postulated Radioactive Release due to Tank Failures"].~~

2

any outside temporary tanks

13

⟨DOC A.8⟩

⟨3/4.11.2.1⟩

(continued)

<CTS>

Insert 5.5.8.d

<6.2.F.8>

ESF Ventilation System	Delta P (inches WG)	Flowrate (cfm)
SGT System	8	$\geq$ 3600 and $\leq$ 4400
CRAF System		
EMUs	8	$\geq$ 3600 and $\leq$ 4400
CRRFs	3.0	$\geq$ 18000 and $\leq$ 28900
AEERRFs	3.0	$\geq$ 14000 and $\leq$ 22800

<CTS>

5.5 Programs and Manuals

5.5.9 **Explosive Gas and Storage Tank Radioactivity Monitoring Program** (continued)

<3/4.11.1.1>
<DOC A.8>
<3/4.11.2.1>

The program shall include:

a. The limits for concentrations of hydrogen ~~and oxygen~~ in the ~~Waste Gas Holdup~~ Condenser Offgas Treatment System~~s~~ 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 15 — 14 — 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~~ 15

~~c.~~ b. 15 A surveillance program to ensure that the quantity of radioactivity contained in all ~~outdoor liquid radwaste tanks~~ outside temporary tanks (8) 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 Systems~~ Liquid Waste Management Systems (2) is less than the amount that would result in concentrations less than the limits ~~of 10 CFR 20, Appendix B, Table 2, Column 2~~ specified in the ODCM (20), 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.

1 { 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.

5.5.10 **Diesel Fuel Oil Testing Program**

<DOC A.7>
<4.8.1.1.2.c>

A diesel fuel oil testing program shall establish the ~~to implement~~ required testing of both new fuel oil and stored fuel oil ~~shall be established~~. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following: } 10

(continued)

5.5.10 Diesel Fuel Oil Testing Program (continued)

<DOC A.7>
<4.8.1.1.2.c>

TSTF-106

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

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

2. ~~a flash point and~~ kinematic viscosity within limits for ASTM 2D fuel oil, or water and sediment within limits 16

3. a clear and bright appearance with proper color;

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

b. ~~Other properties for ASTM 2D fuel oil are within limits~~ within 31 days following sampling and addition of the new fuel oil to storage tanks; and TSTF-106

c. Total particulate concentration of the fuel oil in the storage tanks is ≤ 10 mg/l ~~when tested every 31 days~~ in accordance with the applicable ASTM ~~D-2276, Method A-2 or A-3~~ standard. 16 17

5.5.11 Technical Specifications (TS) Bases Control Program

<DOC M.1>

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

TSTF-118

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

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

d. Proposed changes that meet the criteria ion 17 of ~~5.5.11b~~ 5.5.11.b.1 or 5.5.11.b.2 above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without 17

(continued)

<CTS>

5.5 Programs and Manuals

5.5.11 Technical Specifications (TS) Bases Control Program (continued)

<DOC M.1>

prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

5.5.12 Safety Function Determination Program (SFDP)

<DOC M.1>

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. [3] The SFDP shall contain the following:

[3]

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

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

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

(continued)

<CTS>

5.5 Programs and Manuals

<DOC M.1>

5.5.12 Safety Function Determination Program (SFDP) (continued)

17

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

INSERT 5.5.13 18

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>

INSERT 5.5.13

5.5.13 Primary Containment Leakage Rate Testing Program

<6.2.F.7>

a. This program shall establish 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 exemptions. 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 39.6 psig.

c. The maximum allowable primary containment leakage rage, L_a, at P_a, is 0.635% 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 are:

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

b) For each door, the seal leakage rate is $\leq$ 5 scf per hour when the gap between the door seals is pressurized to $\geq$ 10 psig.

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

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, 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 LaSalle Units 1 and 2 ITS LCO Sections. The normal "refueling cycle intervals" (i.e., 18 months) have been extended to 24 months in the LaSalle Units 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 4.0.2), 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. In addition, these requirements in ITS 5.5.4 at one time were located in individual Specifications in the CTS. Thus, CTS 4.0.2 (ITS SR 3.0.2) and CTS 4.0.3 (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 LaSalle 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 added since LaSalle 1 and 2 have Mark II containments. This change is consistent with current licensing basis.

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

9. 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

9. (continued)

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.

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 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 5.5.7) 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. 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.

12. ISTS 5.5.8.d demonstrates that the pressure drop across the HEPA filters, prefilters, 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 5.5.8.d. In addition, the requirement to test across the moisture separator and heater has been added in ITS 5.5.8.d and the words ", corrected for voltage variations at the 480 V bus," have been added in ITS 5.5.8.e to be consistent with the current licensing basis.

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

14. The requirement to limit oxygen in the Condenser Offgas Treatment System has been deleted consistent with current licensing basis.

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

16. 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 consistent with the current licensing basis.

    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 5.5.10.c to be consistent with current licensing basis.

17. These words have been added for clarity.

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

19. 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.

20. Changes have been made to be consistent with the LaSalle 1 and 2 current licensing basis.

21. An additional testing frequency of 48 months has been added to the Inservice Testing Program requirements of ITS 5.5.7 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

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

5.6.1 Occupational Radiation Exposure Report

⟨6.6.A.2⟩ 1

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

1

~~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.]~~

1 2 3 Insert 5.6.1

TSTF-152

5.6.2 Annual Radiological Environmental Operating Report

⟨6.6.A.3⟩

⟨6.6.A.3 footnote *⟩ 1

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

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)

<CTS>

<6.6.A.2>

INSERT 5.6.1

TSTF-152

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 6
collective deep dose equivalent (reported in person-rem) according to work and 2
job functions (e.g., reactor operations and surveillance, inservice man
inspection, routine maintenance, special maintenance (describe maintenance), 1
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 2
total deep dose equivalent received from external sources should be assigned
to specific major work functions. The report covering the previous calendar
year shall be submitted by April 30 of each year. ~~[The initial report shall
be submitted by April 30 of the year following the initial criticality.]~~ 3

<CTS>

5.6 Reporting Requirements

<6.6.A.3>

<6.6.A.3 footnote *>

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

<6.6.A.4>

<6.6.A.4 footnote *>

5.6.3 Radioactive Effluent Release Report TSTF-152 Changes not adopted. 5

1 -----------------------------NOTE-----------------------------
A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station~~; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit~~. 1 11

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. 6

<6.6.A.5>

5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience~~,~~ 1
~~including documentation of all challenges to the safety/relief~~ TSTF-258 Changes not adopted

(continued) 12

<CTS>

5.6 Reporting Requirements

12 TSTF-258 changes not adopted

5.6.4 Monthly Operating Reports (continued)

<6.6.A.5>

1 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)

<6.6.A.6>

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 INSERT 5.6.5.a

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

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 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.~~

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

7

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

~~(continued)~~

<CTS>

<6.6.A.6>

INSERT 5.6.5.a

1. The APLHGR for Specification 3.2.1.

2. The MCPR for Specification 3.2.2.

3. The LHGR for Specification 3.2.3.

4. The Rod Block Monitor Upscale Instrumentation Setpoint for the Rod Block Monitor - Upscale Function Allowable Value for Specification 3.3.2.1.

INSERT 5.6.5.b
page 1 of 3

<6.6.A.6>

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

2. Letter, Ashok C. Thadani (NRC) to R.A. Copeland (SPC), "Acceptance for Referencing of ULTRAFLOW™ Spacer on 9x9-IX/X BWR Fuel Design," July 28, 1993.

3. 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, XN-NF-524(P)(A) Revision 2 and Supplement 1 Revision 2, Supplement 2, Advanced Nuclear Fuels Corporation November 1990.

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

5. HUXY: A Generalized Multirod Heatup Code with 10 CFR 50, Appendix K Heatup Option, ANF-CC-33(P)(A), Supplement 1 Revision 1; and Supplement 2, Advanced Nuclear Fuels Corporation, August 1986 and January 1991, respectively.

6. Advanced Nuclear Fuel 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.

7. 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.

8. 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.

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.

<CTS>
<6.6.A.6>

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

11. Volume 1 - STAIF - A Computer Program for BWR Stability Analysis in the Frequency Domain, Volume 2 - STAIF - A Computer Program for BWR Stability Analysis in the Frequency Domain, Code Qualification Report, EMF-CC-074(P)(A), Siemens Power Corporation, July 1994.

12. RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model, XN-NF-81-58(P)(A), Revision 2 Supplements 1 and 2, Exxon Nuclear Company, March 1984.

13. XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis, XN-NF-84-105(P)(A), Volume 1 and Volume 1 Supplements 1 and 2; Volume 1 Supplement 4, Advanced Nuclear Fuels Corporation, February 1987 and June 1988, respectively.

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

15. Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis, XN-NF-80-19(P)(A) Volume 1 and Supplements 1 ans 2, Exxon Nuclear Company, Richland, WA 99352, March 1983.

16. 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.

17. 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.

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

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

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

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

<CTS>
<6.6.A.6>

22. 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.

23. BWR Jet Pump Model Revision for RELAX, ANF-91-048(P)(A), Supplement 1 and Supplement 2, Siemens Power Corporation, October 1997.

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

25. 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

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.] 7

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.

Reviewer's 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 Plan 5.3.2, Pressure-Temperature Limits.

(continued)

5.6 Reporting Requirements

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

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

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

5.6.7 EDG Failure Reports 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.

5.6.8 6 8 9 1 Post Accident Monitoring 9 (PAM) Report Instrumentation 9

Table 3.3.7.5-1 ACTION 81 part 2

When a Special Report is required by Condition B or ~~G~~ F 9 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.

5.6.9 Tendon Surveillance Report 10

Any abnormal degradation of the containment structure detected during the tests required by the Pre-Stressed Concrete

(continued)

5.6 Reporting Requirements

5.6.9 Tendon Surveillance Report (continued)

[Containment Tendon Surveillance Program shall be reported to the NRC within 30 days. The report shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, and the corrective action taken.]

[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

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ITS: 5.6 - REPORTING REQUIREMENTS

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.

3. The initial report requirement for the 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 the LaSalle 1 and 2 SER for Operating License Amendment numbers 85 and 69, respectively, dated September 1, 1992, and October 7, 1992.

5. ISTS 5.6.3 (Radioactive Effluent Release Report) is revised by TSTF-152. The 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 3.4.11).

8. ISTS 5.6.7 has been deleted in accordance with the guidance of Generic Letter 94-01. LaSalle 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. In addition, the term "Special Report" has been replaced by "report" since LCO 3.3.3.1 does not refer to this as a Special Report, and this report is not under the old (revision 0) header of "Special Reports." Also, the proper Condition has been referenced.

10. ISTS 5.6.9 has not been added to the ITS. This Technical Specification report is not currently required by the LaSalle 1 and 2 Technical Specifications. Reports concerning the degradation of tendons in pre-stressed concrete containments will be made in accordance with 10 CFR 50.73 and Regulatory Guide 1.35.

11. 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.

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 to 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.

<CTS>

 1

TSTF-258 changes not adopted 1

5.0 ADMINISTRATIVE CONTROLS

<6.1.1> {5.7 High Radiation Area}

<6.1.1.1> 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 > 100 mrem/hr and ≤ 1000 mrem/hr, provided they are otherwise following plant radiation protection procedures for entry into such 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. 1

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

TSTF-45 {Radiation Protection Manager} in the RWP.

~~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~~

(continued)

[1]

5.7 High Radiation Area

Insert 5.7

5.7.2 (continued)

under an approved RWP that shall specify the dose rate levels in the immediate work 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.

5.7.3 For individual high radiation areas with radiation levels of > 1000 mrem/hr, 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]

TSTF-258 changes not adopted [1]

<CTS>

Insert 5.7

5.7.2 In addition to the requirements of Specification 5.7.1 for areas accessible to personnel with radiation levels such that a major portion <6.1.1.2> of the body could receive in one hour a dose greater than 1000 mrem, the computer shall be programmed to permit entry through locked doors for any individual requiring access to any such high-high radiation areas for the time that access is required.

5.7.3 Keys to manually open computer controlled high radiation area doors and high-high radiation area doors shall be maintained under the <6.1.1.3> administrative control of the shift manager on duty or the radiation protection manager.

5.7.4 High-high radiation areas, as defined in Specification 5.7.2, not equipped with the computerized card readers shall be maintained in accordance with 10 CFR 20.1601(a)(3), locked except during periods when <6.1.1.4> access to the area is required with positive control over each individual entry, or in the case of a high radiation area established for a period of 30 days or less, direct surveillance to prevent unauthorized entry may be substituted. Doors shall remain locked except during periods of access by personnel under an approved RWP which shall specify the dose rate levels in the immediate work area and the maximum allowable stay time for individuals in that area. For individual areas accessible to personnel with radiation levels such that a major portion of the body could receive in one hour a dose in excess of 1000 mrem that are located within large areas, such as the containment, where no enclosure exists for purposes of locking, and no enclosure can be reasonably constructed around the individual areas, then that area shall be roped off, conspicuously posted and a flashing light shall be activated as a warning device. In lieu of the stay time specification of the RWP, direct or remote, such as use of closed circuit TV cameras, continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities within the area.

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 LaSalle 1 and 2 choose to maintain their CTS requirements for High Radiation Area controls.

GENERIC NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: CHAPTER 5.0 - ADMINISTRATIVE 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-1434, 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

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

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 requirement for submitting the Occupational Radiation Exposure Report and the Annual Radiological Environmental Operating Report. The CTS require the reports to be submitted by March 1 and May 1 of each year, respectively. This proposed change will allow the reports to be submitted by April 30 and May 15 of each year, respectively. The proposed change does not affect the probability of an accident. The submittal dates of the Occupational Radiation Exposure Report and the Annual Radiological Environmental Operating 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 Occupational Radiation Exposure Report and the Annual Radiological Environmental Operating 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 reports 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 Occupational Radiation Exposure Report and the Annual Radiological Environmental Operating Report. The current TS require the reports to be submitted by March 1 and May 1 of each year, respectively. This proposed change will allow the report to be submitted by April 30 and May 15 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 Occupational Radiation Exposure Report and the Annual Radiological Environmental Operating Report. The current TS require the reports to be submitted by March 1 and May 1 of each year, respectively. This proposed change will allow the reports to be submitted by April 30 and May 15 of each year, respectively. The margin of safety is not reduced by allowing the reports to be submitted 60 days and 15 days later, respectively. 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: 6.1.E/F - TRAINING

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

NO SIGNIFICANT HAZARDS CONSIDERATION
CTS: 6.2.B - RADIATION PROTECTION PROGRAM

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

NO SIGNIFICANT HAZARDS CONSIDERATION
CTS: 6.3 - REPORTABLE EVENT ACTION

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

NO SIGNIFICANT HAZARDS CONSIDERATION
CTS: 6.4 - SAFETY LIMIT VIOLATION

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

NO SIGNIFICANT HAZARDS CONSIDERATION
CTS: 6.5 - PLANT OPERATING RECORDS

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

NO SIGNIFICANT HAZARDS CONSIDERATION
CTS: 6.7 - PROCESS CONTROL PROGRAM (PCP)

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

NO SIGNIFICANT HAZARDS CONSIDERATION
CTS: 6.9 - MAJOR CHANGES TO RADIOACTIVE WASTE
TREATMENT SYSTEMS

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 proposed relaxation of the schedule requirement for submitting a report of any major changes to the radioactive waste treatment system as part of the Monthly Operating Report. The proposed change does not affect the probability of an accident. The submittal of the Monthly Operating Report containing or not containing information related to changes to the radioactive waste treatment system is not assumed to be an initiator of any analyzed event. Also, the consequences of an accident are not affected by the submittal of these reports. 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 information will still be required to be submitted and does not affect any plant equipment or requirements for maintaining plant equipment. The submittal of this information is 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 proposed to relax the requirement for submitting a report of any major changes to the radioactive waste treatment system as part of the Monthly Operating Report. This proposed change does not affect the probability of an accident. 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 relaxation of the schedule requirement for submitting a report of any major changes to the radioactive waste treatment system as part of the Monthly Operating Report. The margin of safety is not reduced by allowing the information to be submitted every year as part of the Radioactive Effluent Release Report. 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 information will still be required to be submitted and does 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.

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.