SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. TO FACILITY OPERATING LICENSE NO. NPF-11

AND AMENDMENT NO. TO FACILITY OPERATING LICENSE NO. NPF-18

EXELON GENERATION COMPANY, LLC

LASALLE COUNTY STATION, UNITS 1 AND 2

DOCKET NOS. 50-373 AND 50-374

I. INTRODUCTION

LaSalle County Station, Units 1 and 2 (LaSalle), has been operating with Technical Specifications (TS), issued with the original operating licenses on April 17, 1982, for Unit 1 and December 16, 1983, for Unit 2, as amended from time to time.

By letter dated March 3, 2000, Exelon Generatoin Company, LLC (EGC, or the licensee, formerly Commonwealth Edison Company), proposed to amend the operating licenses for LaSalle to completely revise the TS with new TS based on the following:

- NUREG-1434, "Standard Technical Specifications - General Electric Plants, BWR/6" Revision 1, of April 1995.
- “NRC Final Policy Statement on Technical Specification Improvements for Nuclear Power Reactors" (Final Policy Statement), published on July 22, 1993 (58 FR 39132).
- The current LaSalle TS

The overall objective of EGC’s request, consistent with the Final Policy Statement, is to rewrite, reformat, and streamline TS consistent with 10 CFR 50.36.

Hereinafter, the proposed TS are referred to as the Improved TS (ITS), the existing LaSalle TS are referred to as the Current TS (CTS), and the TS in NUREG-1434 are referred to as the Standard TS (STS). The corresponding TS Bases are ITS Bases, CTS Bases, and STS Bases, respectively.

EGC retained portions of the CTS in the ITS in addition to basing the ITS on the STS and the Final Policy Statement. The NRC discussed plant-specific issues, including design features, requirements, and operating practices with EGC during a series of conference calls and meetings. In addition, EGC proposed generic changes that were not in the STS. The NRC staff asked EGC to submit such generic issues as proposed changes to the STS through the Nuclear Energy Institute's Technical Specifications Task Force (TSTF). These generic issues were considered for the LaSalle ITS before evaluating them generically. EGC proposed transferring some CTS requirements to EGC-controlled documents as this was consistent with the Final Policy Statement. In addition, EGC used human factors principles to clarify CTS

requirements being retained in the ITS and to define more clearly the appropriate scope of the ITS. Further, EGC proposed changes to the CTS Bases to make each ITS requirement clearer and easier to understand.

Since the licensee prepared the March 3, 2000, application, a number of amendments to the LaSalle operating license were approved, as follows:

Amendment No. (Unit 1, Unit 2)	Description of Change	Issue Date
- - 123	Exigent TS Change for Unit 2 Weld Examination	03/22/2000
139 124	UFSAR Change for High Energy Line Break	4/11/2001
140 125	Power Uprate - 5%	5/09/2000
- - 126	Increase Minimum Critical Power Ratio Limit	5/17/2000
141 127	Revise TS Requirements on Communications During Control Rod Movement	10/05/2000
142 128	Delete TS Requirements on Reactor Protection System Shorting Links	10/10/2000
143 129	Permit Functional Testing of Diesel Generators during Power Operation	10/16/2000
144 130	Revise Pressure/Temperature Limits	11/08/2000
145 131	Revise License Condition on Fuel Movement	11/09/2000
146 132	Transfer of Operating License to EGC	1/12/2001

These amendments have been incorporated, as appropriate, into the ITS.

The March 3, 2000, application was supplemented by letters dated March 24, June 5, July 18, July 31, September 1, September 22, October 5, October 9, November 20, November 30, December 18, **date (revision D)**, and **date (license conditions)**. The NRC staff issued requests for additional information (RAIs) by letters dated June 21, July 3, August 18, August 31, September 12 and November 3, 2000.

The NRC published its proposed actions on EGC's application for amendment of March 3, 2000, in the *Federal Register* on **date** (**citation**) and **date** (**citation**). This Safety Evaluation (SE) assesses EGC's application and supplemental information that resulted from NRC requests for information and discussions with EGC during the NRC staff's review. All ITS changes are within the scope of the actions described in the *Federal Register* notices.

The NRC staff relied on the Final Policy Statement and the STS as guidance for reviewing proposed deviations from the STS. This SE provides the basis for the NRC staff's conclusions that 1) EGC developed the ITS based on the STS as modified by plant-specific changes, and 2) using the LaSalle ITS is acceptable for continued plant operation. It is acceptable that the ITS

differs from STS, since the ITS reflects LaSalle's current licensing basis. The NRC staff approves EGC's changes to their CTS with modifications documented in their revised submittals.

For the reasons stated in this SE, the NRC staff finds that the TS issued with this license amendment comply with Section 182a of the Atomic Energy Act, 10 CFR 50.36, and the guidance in the Final Policy Statement and that the TS are in accord with the common defense and security and provide adequate protection of the health and safety of the public.

II. BACKGROUND

Section 182a of the Atomic Energy Act requires that applicants for nuclear power plant operating licenses will state:

> [S]uch technical specifications, including information of the amount, kind, and source of special nuclear material required, the place of the use, the specific characteristics of the facility, and such other information as the Commission may, by rule or regulation, deem necessary in order to enable it to find that the utilization . . . of special nuclear material will be in accord with the common defense and security and will provide adequate protection to the health and safety of the public. Such technical specifications shall be a part of any license issued.

In 10 CFR 50.36, the Commission established its regulatory requirements for TS content. In doing so, the Commission emphasized those matters related to preventing accidents and mitigating accident consequences. The Commission noted that applicants were expected to incorporate into their TS "those items that are directly related to maintaining the integrity of the physical barriers designed to contain radioactivity" (see Statement of Consideration, "Technical Specifications for Facility Licenses; Safety Analysis Reports," of December 17, 1968 (33 FR 18610)).

10 CFR 50.36 requires that TS include items in the following five specific categories:

(1) safety limits, limiting safety system settings and limiting control settings
(2) limiting conditions for operation (LCOs)
(3) surveillance requirements (SRs)
(4) design features
(5) administrative controls

However, the rule does not specify particular TS requirements.

For several years, NRC and industry representatives have tried to develop guidelines for improving nuclear power plant TS content and quality. On February 6, 1987, the Commission issued their "Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors" (52 FR 3788). During the period from 1989 to 1992, the utility Owners Groups and the NRC staff developed improved STS for each primary reactor type that would comply with the Commission's policy. In addition, the NRC staff, licensees, and Owners Groups developed a Writers Guide containing generic administrative and editorial guidelines for

preparing TS. The Guide emphasized human factors principles, and EGC used it to develop their ITS.

In September 1992, the Commission issued the General Electric STS as NUREG-1434, which was developed using the guidance and criteria contained in the Commission's Interim Policy Statement. The General Electric STS are a model for developing ITS for General Electric plants. The results from applying the Interim Policy Statement criteria to generic system functions were published in a "Split Report" issued to the Nuclear Steam System Supplier (NSSS) Owners Groups in May 1988. The Interim Policy Statement criteria along with the Writer's Guide ensured that the ITS would consistently reflect system configurations and operating characteristics for all NSSS designs. In addition, the generic Bases provide a lot of information about the basis for the STS requirements.

On July 22, 1993, the Commission issued its Final Policy Statement indicating that satisfying the guidance in the policy statement also satisfies Section 182a of the Act and 10 CFR 50.36 (58 FR 39132). The Final Policy Statement described the STS safety benefits and encouraged licensees to use the STS as the basis for plant-specific TS amendments and for complete conversions to the IST. Further, the Final Policy Statement gave guidance for evaluating the required scope of the ITS and defined the guidance criteria for determining which of the LCOs and associated surveillances should remain in the ITS. The Commission noted that, in allowing certain items to be relocated to licensee-controlled documents while requiring that other items be retained in the ITS, it was adopting the qualitative standard enunciated by the Atomic Safety and Licensing Appeal Board in Portland General Electric Company's hearing (Trojan Nuclear Plant), ALAB-531, 9 NRC 263, 273 (1979). There, the Appeal Board observed the following:

> [T]here is neither a statutory nor a regulatory requirement that every operational detail set forth in an applicant's safety analysis report (or equivalent) be subject to a technical specification, to be included in the license as an absolute condition of operation which is legally binding upon the licensee unless and until changed with specific Commission approval. Rather, as best we can discern it, the contemplation of both the Act and the regulations is that technical specifications are to be reserved for those matters as to which the imposition of rigid conditions or limitations upon reactor operation is deemed necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety.

Using this approach, licensees should keep in the ITS existing LCO requirements that fall within or satisfy any of the Final Policy Statement criteria. Those LCO requirements that do not fall within or satisfy these criteria may be relocated to licensee-controlled documents. The Commission codified the four criteria in 10 CFR 50.36 (60 FR 36593, July 19, 1995). The Final Policy Statement criteria are as follows:

Criterion 1 — Installed instrumentation that is used to detect and indicate in the control room a significant abnormal degradation of the reactor coolant pressure boundary.

Criterion 2 — A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to fission product barrier integrity.

Criterion 3 — A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to fission product barrier integrity.

Criterion 4 — A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

Part III of this SE explains the NRC staff's conclusion that converting LaSalle's CTS to those based on STS as modified by plant-specific changes is consistent with LaSalle's current licensing basis and the requirements and guidance of the Final Policy Statement and 10 CFR 50.36.

III. EVALUATION

The NRC staff's review evaluates changes to CTS that fall into categories, defined by EGC, and includes an evaluation of whether existing regulatory requirements are adequate for controlling future changes to requirements removed from the CTS and placed in EGC-controlled documents.

The NRC staff's review of the March 3, 2000, submittal, as supplemented, identified the need for clarifications and additions to the submittal in order to establish an appropriate regulatory basis for translation of CTS requirements into ITS. Each change to the CTS proposed in the amendment request is identified as a discussion of change (DOC) to the CTS. EGC also provided justifications for deviation from the STS, as appropriate. The NRC staff comments were documented as requests for additional information (RAIs) and forwarded to EGC. EGC provided written responses to the NRC staff requests in supplemental letters indicated above. The docketed letters clarified and revised EGC's basis for translating CTS requirements into ITS. The NRC staff finds that EGC's submittals provide sufficient detail to allow the staff to reach a conclusion regarding the adequacy of EGC's proposed changes.

EGC's license amendment application categorized CTS changes as follows:

— Administrative Changes, (A), i.e., non-technical changes in existing CTS requirements.

— Technical Changes - More Restrictive, (M), i.e., new or additional CTS requirements.

— Technical Changes - Less Restrictive (specific), (L), i.e., deleting or relaxing CTS requirements.

— Technical Changes - Less Restrictive Relocated Requirements (generic), (LA), i.e., relocation of details out of the CTS and into licensee-controlled documents

— Technical Changes - Less Restrictive (generic), (LB), i.e., extending an instrument completion time or surveillance frequency according to approved vendor topical reports

— Technical Changes - Less Restrictive, (LC), i.e., eliminating instrumentation requirements for alarm and indication only functions

— Technical Changes - Less Restrictive, (LD), i.e., extending CTS surveillance intervals to 24 months from 18 months for items other than Channel Calibrations

— Technical Changes - Less Restrictive, (LE), i.e., extending CTS surveillance intervals to 24 months from 18 months for Channel Calibrations.

— Technical Changes - Less Restrictive, (LF), i.e., use of revised methodologies for determining Allowable Values and instrument setpoints, and analyzing channel/instrument performance to ensure that the design basis and associated safety limits will not be exceeded during plant operation.

— Relocated Specifications, (R), i.e., relaxations in which whole specifications are removed from the CTS and placed in EGC-controlled documents.

The changes that are in the ITS conversion for LaSalle are listed in the following tables attached to this SE:

- Table A of Administrative Changes to the CTS
- Table M of More-Restrictive Changes to the CTS
- Table L of Less-Restrictive Changes to the CTS (includes L, LD, LE, and LF categories)
- Table LA of Less-Restrictive, Relocated Requirements Changes to the CTS
- Table R of Relocated Specifications

The tables are only meant to summarize the changes being made to the CTS. The details, as to what the actual changes are and how they are being made to the CTS or ITS, are only provided in the licensee's application and supplemental letters.

The general categories of changes to the CTS requirements are described in more detail below.

A. Administrative Changes (A)

Administrative (non-technical) changes are intended to incorporate human factors principles into the form and structure of the ITS so that plant operations personnel can use them more easily. These changes are editorial in nature or involve the reorganization or reformatting of CTS requirements without affecting technical content or operational restrictions. Every section of the ITS reflects this type of change. In order to ensure consistency, the NRC staff and EGC have used STS as guidance to reformat and make other administrative changes. Among the changes proposed by EGC and found acceptable by the NRC staff are:

1 Providing the appropriate numbers, etc., for STS bracketed information (information that must be supplied on a plant-specific basis and that may change from plant to plant).

2 Identifying plant-specific wording for system names, etc.

3 Changing the wording of specification titles in the CTS to conform to STS.

4 Splitting up requirements currently grouped under a single current specification to more appropriate locations in two or more specifications of ITS.

5 Combining related requirements currently presented in separate specifications of the CTS into a single specification of ITS.

Table A lists the administrative changes proposed in ITS. Table A is organized by the corresponding ITS section DOC, and provides a summary description of the administrative change that was made, and CTS and ITS LCO references. The NRC staff reviewed all of the administrative and editorial changes proposed by EGC and finds them acceptable because they are compatible with the Writers Guide and STS, do not result in any substantive change in operating requirements, and are consistent with the Commission's regulations.

B. Technical Changes — More Restrictive (M)

EGC, in electing to implement the specifications of STS proposed a number of requirements more restrictive than those in the CTS. ITS requirements in this category include requirements that are either new, more conservative than corresponding requirements in the CTS, or have additional restrictions that are not in the CTS but are in the STS. Examples of more restrictive requirements are placing an LCO on plant equipment which is not required by the CTS to be operable, adopting more restrictive requirements to restore inoperable equipment, and adopting more restrictive SRs. Table M lists all the more restrictive changes proposed in ITS. Table M is organized by the corresponding ITS section DOC and provides a summary description of the more restrictive change that were adopted along with CTS and ITS LCO references. These changes are additional restrictions on plant operation that enhance safety. The staff reviewed these changes and found them to be acceptable.

C. Technical Changes — Less Restrictive (L, LB, LC, LD, LE and LF)

L, LB, LC, LD, LE and LF technical changes are grouped here to simplify discussion of the broad range of proposed less restrictive changes in technical requirements. L is used to designate a CTS change that requires a unique discussion. LB, LC, LD, LE and LF are used to identify a recurring change evaluated by a single discussion in the submittal. Less restrictive requirements include deletions and relaxations to portions of CTS requirements that are not being retained in ITS or relocated to an EGC-controlled document. When requirements have been shown to give little or no safety benefit, their relaxation or removal from the TS may be appropriate. In most cases, relaxations previously granted to individual plants on a plant-specific basis were the result of (1) generic NRC actions, (2) new staff positions that have evolved from technological advancements and operating experience, or (3) resolution of the Owners Groups' comments on STS. The NRC staff reviewed generic relaxations contained in the STS and found them acceptable because they are consistent with current licensing practices and the Commission's regulations. The LaSalle design was also reviewed to determine if the specific design basis and licensing basis are consistent with the technical basis for the model requirements in the STS and thus provide a basis for ITS.

A significant number of changes to the CTS involved deletions and relaxations to portions of CTS requirements evaluated as Categories 1 through 10 that follow:

Category 1 — Relaxation of LCO Requirements
Category 2 — Relaxation Applicability
Category 3 — Relaxation of Surveillance Requirement
Category 4 — Relaxation of Required Action Detail
Category 5 — Relaxation of Required Actions to Exit Applicability
Category 6 — Relaxation of Completion Time
Category 7 — Allow Mode Changes When LCO Not Met
Category 8 — Elimination of Requirement to Lock the Reactor Mode Switch in Shutdown or Refuel
Category 9 — Elimination of CTS Reporting Requirement
Category 10 — Relaxation of Surveillance Frequency from 18 months to 24 months

The following discussions address why the various categories of changes are acceptable.

Category 1 - Relaxation of the LCO Requirements

Certain CTS LCOs contain operational and system parameters beyond those necessary to meet safety analysis assumptions and therefore are considered overly restrictive. CTS also contain limits which have been shown to give little or no safety benefit to the safe operation of the plant. The ITS, consistent with the guidance in the STS, delete or revise operating limits in this category. CTS LCO changes included in this category are: (1) revising setpoints to be consistent with instrument setpoint methodologies; (2) deleting or revising operational limits to establish requirements consistent with applicable safety analyses; (3) deleting equipment or systems which establish redundant system capability beyond that assumed to function by the applicable safety analyses or which are implicit to the ITS requirement for systems, components and devices to be operable; and (4) adding allowances to use administrative controls on plant devices and equipments during times when automatic control is required or to establish temporary administrative limits, as appropriate, to allow time for systems to establish equilibrium operation;

TS changes represented by these categories of requirements allow operators to more clearly focus on issues important to safety. The resultant ITS LCOs maintain an adequate degree of protection consistent with the safety analysis. They also improve focus on issues important to safety and provide reasonable operational flexibility without adversely affecting the safe operation of the plant. These changes are consistent with STS and are acceptable.

Category 2 - Relaxation of Applicability

The CTS require compliance with the LCO during the Operational Mode(s) or other conditions specified in the LCO Applicability statement. Five Operating Modes are defined by TS according to average reactor coolant temperature and the position of the reactor mode switch located in the control room; Power Operation, Startup, Hot Shutdown, Cold Shutdown and Refueling. When CTS Applicability requirements are inconsistent with the applicable accident analyses assumptions for a system, subsystem or component specified in the LCO, the LCO is changed in the ITS to establish a consistent set of requirements. These modifications or deletions are acceptable because, during the conditions referenced in the ITS, the operability

requirements are consistent with the applicable safety analyses. These changes are consistent with STS and are acceptable.

Category 3 - Relaxation of Surveillance Requirement

CTS require maintaining the LCO equipment operable by meeting the SRs in accordance with the specified SR Frequency. This requires conducting tests to demonstrate equipment is operable, or that LCO parameters are within specified limits. When the test acceptance criteria and any specified conditions for the conduct of the test are met, the equipment is deemed operable. The changes in this category relate to relaxation of CTS SR acceptance criteria and/or the conditions for performing the SR.

Relaxing the SR acceptance criteria for these items provides operational flexibility consistent with the objective of the STS without reducing confidence that the equipment is operable. The ITS also permits the use of an actual, as well as a simulated, actuation signal to satisfy SRs for automatically actuated systems. TS required features cannot distinguish between an “actual” signal and a “test” signal. The changes to TS acceptance criteria are acceptable because appropriate testing standards are retained for determining that the LCO-required features are operable.

Relaxing conditions for performing SRs include, for example, not requiring testing of de-energized equipment (e.g., instrumentation Channel Checks) or equipment that is already performing its intended safety function (e.g., position verification of valves locked in their safety actuation position). The changes also include the allowance to verify the position of valves in high radiation areas by administrative means. ITS administrative controls (ITS 5.7) regarding access to high radiation areas make the likelihood of mispositioning valves small. These changes are acceptable because the changes do not affect the ability to determine whether equipment is capable of performing its intended safety function.

These relaxations of CTS SRs optimize test requirements for the affected safety systems and increase operational flexibility. These changes are consistent with STS and are acceptable.

Category 4 - Relaxation of Required Action Detail

LCOs are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When an LCO is not met, CTS specify actions to be taken until the equipment is restored to its required capability or performance level, or remedial measures are established. In revising the Required Actions, details are deleted or options are added such that resulting ITS actions continue to provide measures that conservatively compensate for the inoperable equipment. Furthermore, adopting STS action requirements results in simpler, more concise and more direct action requirements. This allows more effective use of operator resources for placing and maintaining the reactor in a safe condition when the LCO is not met. These changes are consistent with STS and are acceptable.

Category 5 - Relaxation of Required Actions

LCOs are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When an LCO is not met, CTS specify actions to be taken until the

equipment is restored to its required capability or performance level, or remedial measures are established. Compared to CTS required actions, the ITS actions result in extending the time period for taking the plant outside the applicability into shutdown conditions. For example, changes in this category include providing an option to: isolate a system, place equipment in the state assumed by the safety analysis, satisfy alternate criteria, take manual actions in place of automatic actions, “restore to operable status” within a specified time frame, place alternate equipment into service, or use more conservative TS setpoints. The resulting ITS actions continue to provide measures that conservatively compensate for the inoperable equipment. The ITS actions are commensurate with safety importance of the inoperable equipment, plant design and industry practice and do not compromise safe operation of the plant. These changes are consistent with STS and are acceptable.

Category 6 - Relaxation of Completion Time

Upon discovery of a failure to meet an LCO, TS specify times for completing Required Actions of the associated TS conditions. Required Actions establish remedial measures that must be taken within specified completion times (allowed outage times). These times define limits during which operation in a degraded condition is permitted.

Incorporating completion time extensions is acceptable because completion times take into account the operability status of the redundant systems of TS required features, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, vendor-developed standard repair times, and the low probability of a design basis accident (DBA) occurring during the repair period. These changes are consistent with STS, and allowed outage time extensions specified as Category 6 are acceptable.

Category 7 - Allow Mode Changes When LCO Not Met

CTS 3.0.D (ITS 3.0.4) precludes entry into the applicable Mode or specified conditions while relying on the Actions, even though the Actions are designed to provide for safe operation of the plant. Unless otherwise stated, LCO 3.0.4 is always applicable to ITS LCO Actions. However, ITS adds a Note to certain Actions stating “LCO 3.0.4 is not applicable.” The addition of this Note allows transition between Applicability Modes or other specified conditions with the LCO not met (i.e., relying on the Actions) even though the Actions may require plant shutdown. The addition of “LCO 3.0.4 is not applicable” notes does not impact normal operation of the plant for the specified LCO features and would not provide additional initiators for plant transients during the Mode or other specified conditions. This exception to ITS 3.0.4 is acceptable due to the passive function or the installed redundancy of the features, the plant conditions that apply to the Note, and the low probability of an event requiring the inoperable features. These changes are consistent with STS and are acceptable.

Category 8 - Elimination of the Requirement to Lock the Reactor Mode Switch in Shutdown or Refuel

Some CTS LCOs and Actions specify "lock" the mode switch in “Shutdown” (shutdown position) or “Refuel” (refueling position). Other CTS Action requirements also specify placing the reactor in the shutdown or refueling Mode without requiring the mode switch to be “locked.” The requirement to “lock” the mode switch in Shutdown or Refueling is not retained in the ITS. CTS Table 1-2, “Operational Modes” (ITS Table 1.1-1) defines reactor operational Modes based on

the reactor mode switch position and on average reactor coolant temperature. Moving a reactor mode switch from Shutdown into a position other than Shutdown causes a Mode change as defined by TS, and results in associated TS compliance requirements for the LCOs that become applicable in the new Mode. CTS 3.0.A (ITS 3.0.4) precludes changes in reactor Modes without all TS required equipment operable. Thus, ITS 3.0.4 is an administrative requirement put in place to prevent movement of the reactor mode switch between positions without first ensuring TS required equipment is operable, and changing the mode switch from the required position is adequately controlled by ITS Table 1.1-1 without adding a requirement to "lock" the mode switch. These changes are consistent with the STS and are acceptable.

Category 9 - Elimination of CTS Reporting Requirement

CTS include requirements to submit special reports to the NRC when specified limits or conditions are not met. Typically, the time period for the report to be issued is "within 30 days." However, the ITS eliminates the TS requirements for special reports and instead relies on the reporting requirements of 10 CFR 50.73. The changes to the reporting requirements are acceptable because 10 CFR 50.73 provides adequate reporting requirements, and the special reports do not affect continued plant operation.

CTS also include requirements for reports to be made to the NRC on data gathered as part of routine plant programs. These requirements are removed from the ITS. The requirement to report test frequency changes that occur due to consecutive SR failures has been deleted since the test schedule is already covered by the TS. In addition, a historical review has shown the SR has never failed.

Deleting TS reporting requirements reduces unnecessary regulatory burden on the plant and allows licensee efforts to be concentrated on maintaining TS required limits. These changes are consistent with the STS and are acceptable.

Category 10 - Relaxation of Surveillance Frequency from 18 months to 24 months (LD, LE and LF)

CTS require maintaining the LCO equipment operable by conducting SRs in accordance with the specified SR Frequency. The changes in this category relate to extending SR frequencies. Improved reactor fuels allow the licensee to consider an increase in the duration of the fuel cycle for their facility. TS that specify an 18-month surveillance interval are changed to specify a 24-month interval. The CTS 4.0.B (ITS SR 3.0.2) provision to extend surveillances by 25 percent of the specified interval would extend the time limit for completing these surveillances from the CTS limit of 22.5 months to a maximum of 30 months. The staff review of these items is covered in more detail in Section G of this SE. These changes are consistent with the STS and are acceptable.

Table L includes all L, LD, LE, and LF changes and is organized by ITS section. The table specifies: the section designation; a summary description of the change; CTS and ITS LCO references; a reference to the specific change category as discussed above; and a characterization of the DOC.

For the reasons presented above, these less restrictive requirements are acceptable because they will not affect the safe operation of the plant. The ITS requirements are consistent with

current licensing practices, operating experience, and plant accident and transient analyses, and provide reasonable assurance that public health and safety will be protected.

D. Technical Changes — Less Restrictive Relocated Requirements (Not Entire Specifications) (LA)

When requirements have been shown to give little or no safety benefit, their removal from the TS may be appropriate. These are grouped as LA changes. In most cases, relaxations previously granted to individual plants on a plant-specific basis were the result of (1) generic NRC actions, (2) new staff positions that have evolved from technological advancements and operating experience, or (3) resolution of the Owners Groups comments on STS. The NRC staff reviewed generic relaxations contained in the STS and found them acceptable because they are consistent with current licensing practices and the Commission's regulations. The LaSalle design was also reviewed to determine if the specific design basis and licensing basis are consistent with the technical basis for the model requirements in the STS and thus provide a basis for ITS. A significant number of changes to the CTS involved the removal of specific requirements and detailed information from individual specifications evaluated to be Types 1 through 3 that follow:

Type 1 Details of System Design and System Description including Design Limits

Type 2 Descriptions of Systems Operation

Type 3 Procedural Details for Meeting TS Requirements, Reporting Requirements, and Specification Requirements

The following discussions address why each of the three types of information or requirements is not required to be included in ITS .

<u>Type 1 Details of System Design and System Description Including Design Limits</u>

The design of the facility is required to be described in the UFSAR by 10 CFR 50.34. In addition, the quality assurance (QA) requirements of Appendix B to 10 CFR Part 50 require that plant design be documented in controlled procedures and drawings and maintained in accordance with an NRC-approved QA plan (UFSAR Chapter 17). In 10 CFR 50.59, controls are specified for changing the facility as described in the UFSAR, and in 10 CFR 50.54(a) criteria are specified for changing the QA plan. The ITS Bases also contain descriptions of system design. ITS 5.5.10 specifies controls for changing the Bases. Removing details of system design from the CTS is acceptable because this information will be adequately controlled in the UFSAR, controlled design documents and drawings, or the ITS Bases, as appropriate. Cycle-specific design limits are contained in the Core Operating Limits Report (COLR). ITS Administrative Controls include the programmatic requirements for the COLR.

<u>Type 2 Descriptions of Systems Operation</u>

The plans for the normal and emergency operation of the facility are required to be described in the UFSAR by 10 CFR 50.34. ITS 5.4.1.a requires written procedures to be established, implemented, and maintained for plant operating procedures including procedures

recommended in Regulatory Guide (RG) 1.33, Revision 2, Appendix A, February 1978. Controls specified in 10 CFR 50.59 apply to changes in procedures as described in the UFSAR. The ITS Bases also contain descriptions of system operation. It is acceptable to remove details of system operation from the TS because this type of information will be adequately controlled in the UFSAR, plant operating procedures, and the TS Bases, as appropriate.

Type 3 Procedural Details for Meeting TS Requirements, Reporting Requirements, and Specification Requirements

Details for performing TS Actions and SRs are more appropriately specified in the plant procedures required by ITS 5.4.1, the UFSAR, and ITS Bases. For example, control of the plant conditions appropriate to perform a surveillance test is an issue for procedures and scheduling and has previously been determined to be unnecessary as a TS restriction. As indicated in GL 91-04, allowing this procedural control is consistent with the vast majority of other SRs that do not dictate plant conditions for surveillances. Prescriptive procedural information in an Action requirement is unlikely to contain all procedural considerations necessary for the plant operators to complete the actions required, and referral to plant procedures is therefore required in any event. Other changes to procedural details include those associated with limits retained in the ITS. For example, the ITS requirement may refer to programmatic requirements such as COLR, included in ITS Section 5.5, which specifies the scope of the limits contained in the COLR and mandates NRC approval of the analytical methodology.

Relocating specification requirements, including LCO, required actions, and surveillance requirements, have been made in adopting the STS. For example, for certain power operated isolation valves that do not receive an automatic isolation signal and for which the closure time is not assumed in the safety analysis, requirements for periodic testing of these valves are moved to the procedures that implement the inservice testing program (10 CFR 50.55a). Support system specification requirements for other equipment with its own specifications are moved to the TRM. The definition of operability provides sufficient assurance that the supporting system can perform its required support function.

The removal of these kinds of procedural details from the CTS is acceptable because they will be adequately controlled in the UFSAR, plant procedures, Bases and COLR, as appropriate. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. Similarly, movement of reporting requirements from LCOs to licensee-controlled documents is appropriate because ITS 5.6, 10 CFR 50.36 and 10 CFR 50.73 adequately cover the reports deemed to be necessary.

Table LA consists of LA changes. Table LA lists CTS specifications and describes the information that is removed from individual specifications and deleted or relocated to EGC-controlled documents. Table LA is organized by ITS section and includes the following: a DOC identification number referenced to ITS Section; a CTS reference; a summary description of the requirement; the document that retains the CTS requirements; and the specific change type, as discussed above.

The NRC staff has concluded that these types of detailed information and specific requirements are not necessary in the ITS to ensure the effectiveness of ITS to adequately protect the health

and safety of the public. Accordingly, these requirements may be deleted or moved to one of the following EGC-controlled documents for which changes are adequately governed by a regulatory or TS requirement:

(1) TS Bases controlled by ITS 5.5.14, "Technical Specifications Bases Control Program."
(2) UFSAR (includes the Technical Requirements Manual (TRM) by reference) controlled by 10 CFR 50.59.
(3) ODCM controlled by ITS 5.5.1, “Offsite Dose Calculation Manual.”
(4) QA Manual controlled by 10 CFR 50.54.
(5) Inservice Testing Program controlled by ITS 5.5.6, “Inservice Testing Program.”

(6) Inservice Inspection program controlled by 10 CFR 50.55a.
(7) Core Operating Limits Report controlled by ITS 5.6.5, “Core Operating Limits Report (COLR).”

To the extent that requirements and information have been relocated to EGC-controlled documents, such information and requirements are not required to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety. Further, where such information and requirements are contained in LCOs and associated requirements in the CTS, the NRC staff has concluded that they do not fall within any of the four criteria in the Final Policy Statement (discussed in Part II of this SE). Accordingly, existing detailed information and specific requirements, such as generally described above, may be deleted from the CTS.

E. Relocated Specifications (R)

The Final Policy Statement states that LCOs and associated requirements that do not satisfy or fall within any of the four specified criteria may be relocated from CTS (an NRC-controlled document) to appropriate licensee-controlled documents. These requirements include the LCOs, Action Statements (Actions), and associated SRs. EGC proposed, in accordance with the criteria in the Final Policy Statement, to entirely remove certain TS from the CTS and place them in EGC-controlled documents. The staff has reviewed EGC's submittals, and finds that relocation of these requirements to licensee-controlled documents (described above) is acceptable in that changes to these documents will be adequately controlled by 10 CFR 50.59 and other regulations (described above). These provisions will continue to be implemented by appropriate plant procedures (i.e., operating procedures, maintenance procedures, surveillance and testing procedures, and work control procedures).

Table R lists all specifications that are relocated, based on the Final Policy Statement, to EGC-controlled documents. Table R provides: a DOC identification number referenced to ITS Section; a CTS reference; a summary description of the requirement; the name of the document that retains the CTS requirements; and the method for controlling future changes to relocated requirements. The NRC staff evaluation of each relocated specification and specific CTS detail presented in Table R is provided below.

3/4.1.6 Economic Generation Control System

The Economic Generation Control System limits are relocated to the TRM. CTS 3/4.1.6 specify that the economic generation control system (EGCS) may be in operation with automatic flow control provided that core flow is ≥ 65% of rated core flow, and thermal power is greater than or equal to 20% of rated thermal power. The system was designed to allow the load dispatcher to control power output of the station within appropriate limits based on reactor operating conditions. These EGCS limiting conditions for operation were chosen to be well within the analyzed system setpoints utilized in design basis accident (DBA) and transient analyses; however, the EGCS limits do not rely on any assumptions used in DBA or transient analyses. The requirements of the EGCS LCO do not meet the requirements for TS and have been relocated to the TRM.

3/4.3.1 Reactor Protection System Instrumentation

The Control Rod Drive Charging Water Header Pressure - Low Function and associated timer function of the Reactor Protection System are relocated to the TRM. CTS 3/4.3.1 specifies that as a minimum, the reactor protection system instrumentation channels shown in Table 3.3.1-1 shall be operable with the Reactor Protection System (RPS) Response Time as shown in Table 3.3.1-2. The Control Rod Drive Charging Water Header Pressure - Low Function and associated time delay Function provide a reactor scram signal when a low control rod drive (CRD) charging water header condition is detected. CRD charging water pressure is normally maintained by a CRD pump with a backup source of pressure supplied by an accumulator. If the CRD pump is tripped, pressure to the control rod drives is maintained by the accumulator and a check valve in the charging line.

In the CTS this scram is only required to be operable in Modes 2 and 5 when reactor pressure is low and control rods are permitted to be withdrawn because during normal operation, reactor pressure is continuously applied to the control rod drive piston and this pressure is sufficient to insert the rod without the accumulator pressure. However, loss of charging water header pressure in Modes 2 and 5 will only inhibit a control rod scram if the CRD accumulators are concurrently inoperable and incapable of providing the pressure needed to insert the control rods. In addition, the ITS requires that the accumulators be operable in Mode 2 , and if they are not, the affected control rods would be declared inoperable or slow, depending upon the most recent scram times. Also, upon loss of two or more accumulators when reactor vessel pressure is ≥ 900 psig or one accumulator when reactor vessel pressure is less than 900 psig, the charging water header must be at normal pressure or a scram is required (within 20 minutes when reactor vessel pressure is ≥ 900 psig and immediately when reactor vessel pressure is < 900 psig). In Mode 5 the ITS requires that the accumulators be operable and if they are not, the inoperable rods are required to be inserted. These requirements will ensure that the motive force required to scram the control rods will be available when needed. The RPS limits for Control Rod Drive Charging Water Header Pressure - Low (CTS 3/4.3.1.13.a) and for associated the Control Rod Drive Delay Timer (CTS 3/4.3.1.13.b) functions are not assumed in any design basis or transient analyses and are therefore relocated to the TRM.

3/4.3.3 Emergency Core Cooling System (ECCS) Actuation Instrumentation

The ADS Manual Inhibit Functions for Trip Systems A and B of the ECCS Actuation Instrumentation are relocated to the TRM. CTS 3/4.3.3 specifies that the ECCS actuation

instrumentation channels shown in Table 3.3.3-1 shall be operable with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.3-2 and with Emergency Core Cooling System Response Time as shown in Table 3.3.3-3. ECCS instrumentation functions for ADS 'A' - Manual Inhibit (3/4.3.3.A.2.i) and ADS 'B' - Manual Inhibit (3/4.3.3.B.2.h) are relocated to the TRM. The ADS Manual Inhibit switch allows the operator to defeat automatic ADS actuation, as directed by the emergency operating procedures, under conditions for which ADS would not be desirable. However, such manual operator action is not credited in a design basis accident or transient analysis. For example, during an ATWS event low pressure ECCS system activation would dilute sodium pentaborate injected by the Standby Liquid Control (SLC) System thereby reducing the effectiveness of the SLC System ability to shutdown. The assumptions used in the DBA or transient analyses do not require ADS manual inhibit functions. The requirements for ADS manual inhibit to be operable do not meet the requirements for TS and have been relocated to the TRM. Since the screening criteria have not been satisfied, the portions of the LCO and surveillance applicable to the ADS Manual Inhibit switch may be relocated to other plant controlled documents outside the Technical Specifications.

3/4.3.6 Control Rod Withdrawal Block Instrumentation

The CTS requires the control rod withdrawal block instrumentation channels shown in Table 3.3.6-1 to be operable with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.6-2. Several control rod withdrawal block instrumentation functions are relocated to the TRM.

3/4.3.6.2 Average Power Range Monitors (APRM)

The APRM control rod block instrumentation is installed to prevent conditions that would otherwise require actuation of the RPS if plant conditions were allowed to persist, such as during a "control rod withdrawal error at power." The APRMs use LPRM signals to provide information about the average core power and to create the APRM rod block signal. However, the rod block function of the APRMs is not used to mitigate a DBA or transient.

3/4.3.6.3 Source Range Monitor (SRM)

The SRM control rod block instrumentation is installed to monitor neutron flux during refueling, shutdown, and startup conditions. When IRMs are not above Range 2, the SRM control rod block prevents a control rod withdrawal if the count rate exceeds a preset value or falls below a preset limit. However, the rod block signals initiated by the SRMs are not used to mitigate a DBA or transient.

3/4.3.6.4 Intermediate Range Monitors (IRM)

The IRM control rod block instrumentation is installed to monitor the neutron flux levels during refueling, shutdown, and startup conditions. The IRM control rod block prevents a control rod withdrawal if the IRM reading exceeds a preset value, or if the IRM is inoperable. However, the rod block signals initiated by the IRMs are not used to mitigate a DBA or transient.

3/4.3.6.5 Scram Discharge Volume (SDV)

The Scram Discharge Volume (SDV) control rod block instrumentation uses signals derived from SDV level monitors to prevent control rod withdrawals when accumulated water reaches a pre-set level in the SDV. This instrumentation ensures there is sufficient volume remaining in the SDV to contain the water discharged by the control rod drives during a scram, thus ensuring that the control rods will be able to insert fully. This rod block signal also provides an indication to the operator that water is accumulating in the SDV and prevents further rod withdrawals. With continued water accumulation, a reactor protection system initiated scram signal will occur. Thus, the SDV water level rod block signal provides an opportunity for the operator to take action to avoid a reactor scram. However, the rod block signals initiated by the SDV instrumentation is not used to mitigate a DBA or transient.

3/4.3.6.6 Recirculation Flow Unit

Reactor recirculation flow is monitored as an early indication of an increase in neutron flux and reactor power. The recirculation flow converter upscale or flow converter inoperative initiate a control rod withdrawal block to prevent a continued increase in power without operable monitoring instrumentation. The recirculation flow comparator prevents control rod withdrawal unless the outputs are within limits and the comparator is operable. However, flow increases are detected by neutron flux monitors which provide input to the reactor protection system. The control rod block signals initiated by the recirculation flow unit are not used to mitigate a DBA or transient.

3/4.3.7.3 Meteorological Monitoring Instrumentation

Meteorological monitoring instrumentation are relocated to the TRM. The CTS requires the meteorological monitoring instrumentation channels shown in Table 3.3.7.3-1 to be OPERABLE. Meteorological instrumentation measures environmental parameters that may affect distribution of fission products and gases following a design basis accident (DBA). Meteorological information is required to evaluate the need for initiating protective measures to protect the health and safety of the public; however, the information collected from the meteorological monitoring instrumentation is not used to monitor or mitigate a DBA or transient.

3/4.3.7.5 Accident Monitoring Instrumentation

All Regulatory Guide 1.97 non-Type A instruments and all Regulatory Guide 1.97 non-Category 1 instruments specified in the plant's Safety Evaluation Report (SER) on Regulatory Guide 1.97 are relocated to the TRM. The CTS require the accident monitoring instrumentation channels shown in Table 3.3.7.5-1 to be operable. Accident monitoring instrumentation is provided to monitor variables and systems over their anticipated ranges for accident conditions as appropriate to ensure adequate safety during and following accidents. These variables are used by the control room operating personnel to perform their role in the emergency plan in the evaluation and assessment, monitoring and execution of control room functions when other emergency response facilities are not effectively manned.

The NRC staff documented deterministic screening criteria for post-accident monitoring instrumentation in letter dated May 7, 1988 from T.E. Murley (NRC) to R.F. Janecek (BWROG). The staff requires all plant-specific Regulatory Guide 1.97 Type A instruments specified in the

plant's Safety Evaluation Report (SER) on Regulatory Guide 1.97, and all Regulatory Guide 1.97 Category 1 instruments to be included in ITS. Accordingly, this position has been applied to the LaSalle 1 and 2 Regulatory Guide 1.97 instruments.

The CTS accident monitoring instruments that do not meet the RG 1.97 deterministic criteria and which are relocated include: suppression chamber air temperature, drywell air temperature, safety/relief valve position indicators, noble gas monitor-main stack and noble gas monitor-standby gas treatment system stack. Those instruments meeting the criteria, including neutron flux (wide range) monitors are retained by the ITS criteria. However, category 1 requirements as they relate to the neutron flux (wide range monitor) are revised. The BWR Owners Group submitted a Licensing Topical Report, NEDO-31558, that provided alternative neutron monitoring functional design criteria to that of RG 1.97. By letter dated January 13, 1993 the staff found the BWR Owners Group the alternate design criteria acceptable. Based on the acceptance letter LaSalle 1 and 2 reclassified the neutron flux (wide range) monitor as neither a Type A or a Category 1 variable. Therefore, the neutron flux (wide range monitor) is not added to the ITS.

3/4.3.7.11 Explosive Gas Monitoring Instrumentation

Explosive gas monitoring instrumentation are relocated to the TRM. The CTS require explosive gas monitoring instrumentation channels shown in Table 3.3.7.11-1 to be operable with their Alarm/Trip setpoints set to ensure that the limits of specification 3.11.2.1 are not exceeded. The explosive gas monitoring instrumentation monitors the gaseous radwaste treatment system process for potentially explosive gas mixtures to ensure that hydrogen concentration is maintained below the flammability limit. However, the offgas system is designed to contain detonations without affecting safety related equipment functions. Neither the concentration of hydrogen in the offgas stream, nor the instrumentation used to monitor the hydrogen concentration are an initial assumption of any design basis accident (DBA) or transient analysis.

3/4.3.7.12 Loose-part Detection System

Loose-part detection system requirements are relocated to the TRM. The CTS require the loose-part detection system to be operable. The loose-part detection system is used to detect the presence of loose parts in the reactor vessel. The presence of a loose part indicates there is a potential for damaging components; however, loose-part detection instrumentation is not used for quantifying degradation of the primary coolant pressure boundary. Other component failures related to loose parts, such as fuel failure due to fuel bundle flow blockage from a lost part will be detected by the radiation monitors in the offgas stream. The instrumentation for monitoring loose parts are not an initial assumption of any design basis accident (DBA) or transient analysis.

3/4.4.8 Structural Integrity

The CTS require the structural integrity of ASME Code Class 1, 2 and 3 components (pumps and valves) to be maintained operable in accordance with Specification 4.4.8 are relocated to the TRM. Specification 4.4.8 establishes the programmatic elements for conducting ASME Code Class 1, 2, and 3 component inspections by reference to Section XI of the ASME Boiler and Pressure Vessel Code. The safety basis for establishing programmatic requirements on structural integrity in CTS relate to prevention of component degradation and continued long

term maintenance of acceptable structural conditions. Therefore, structural integrity of safety systems are not operational limits that are an initial assumption of any design basis accident (DBA) or transient analysis.

3/4.7.4 Sealed Source Contamination

Sealed Source Contamination limits are relocated to the TRM. CTS specifies removable contamination limits for sealed sources. Each sealed source containing radioactive material in excess of 100 microcuries of either beta or gamma emitting material or 5 microcuries of alpha emitting material shall be free of greater than or equal to 0.005 microcuries of removable contamination. These limits ensure that the total body or individual organ irradiation doses do not exceed ingestion or inhalation limits. This TS requirement and the associated Surveillance Requirements do not relate to the operational conditions or limitations that are necessary to ensure safe reactor operation. Sealed source contamination limits are not an initial assumption of any design basis accident (DBA) or transient analysis.

3/4.7.7 Area Temperature Monitoring

Area temperature monitoring requirements are relocated to the TRM. CTS require maintaining the temperatures of the areas of Unit 1 and Unit 2 specified in Table 3.7.7-1 within the limits indicated in the table. Area temperature monitors ensure the environmental conditions for safety-related equipment do not exceed environmental qualification envelope assumed for the equipment. Area temperature monitoring instrumentation is separate from leak detection and system isolation instrumentation used to detect or mitigate a DBA such as break detection and leak isolation. Area temperature monitoring instrumentation is not an initial assumption of any design basis accident (DBA) or transient analysis.

3/4.7.8 Structural Integrity of Class 1 Structures

Requirements for maintaining structural integrity of Class 1 structures are relocated to the TRM. CTS 4.7.8.1 and 4.7.8.2 require periodic verification of the structural integrity of Class 1 structures. These TS establish surveillance to monitor Class 1 structures subject to settlement. By ensuring that excessive differential and total settlement is detected, the safety analysis assumptions of Class 1 structures housed in these structures on the LaSalle site are maintained. However, monitoring structural settlement is not related to operational limits that are an initial assumption of any design basis accident (DBA) or transient analysis.

3/4.8.3.1 A.C. Circuits Inside Primary Containment

CTS requirements that specify A.C. circuits inside primary containment shall be de-energized, except during entry into the drywell, are relocated to the TRM. A.C. circuits included in the CTS are the following: (a) installed welding grid systems 1A and 1B for Unit 1, and 2A and 2B for Unit 2, (b) all drywell lighting circuits, (c) all drywell hoists and cranes circuits. These circuits are installed to supply power primarily for lighting, utility outlets, and convenient power plugs; used to conduct plant walk downs, maintenance, and in-situ tests and/or observations. These circuits are non-Class 1E circuits which are de-energized except during drywell entries and are not assumed to be energized in response to plant accidents or transients. The circuits are physically separated from Class 1E circuits such that their operation or failure will not affect operability of Class 1E circuits. A.C. circuits inside primary containment do not establish

operational requirements for plant safety systems and the circuits listed are not an initial assumption of any design basis accident (DBA) or transient analysis.

3/4.8.3.2 Primary Containment Penetration Conductor Overcurrent Protective Devices

Primary containment penetration conductor overcurrent protective devices are relocated to the TRM. CTS require that primary containment penetration conductor overcurrent protective devices for medium and high voltage (6.9 kV, 4.16 kV and 480 volts) electrical penetration circuits shall be operable. This LCO excludes devices on circuits for which credible fault currents would not exceed the electrical penetration design rating. These protective devices will interrupt control and power circuits by opening the circuit whenever the load conditions exceed the preset current demands to protect the circuit conductors against damage or failure due to overcurrent heating effects. In the event a protective device fails to trip the circuit, an alternate protective device is installed to isolate the faulted circuit. Thus, this protection design ensures the worst case fault condition is the loss of a single (redundant) division of protective functions as required by the single failure design criterion.

The overcurrent protection devices also ensure the pressure integrity of the containment penetration. With failure of a device, wire insulation is postulated to degrade resulting in a containment leak path during a LOCA. However, penetration conductor integrity is not directly monitored, rather it is assumed that containment penetration degradation will be identified during the normal containment leak rate tests required by 10 CFR Part 50, Appendix J. Overcurrent protective devices are not operational limits that are an initial assumption of any design basis accident (DBA) or transient analysis.

3/4.9.4 Decay Time

The minimum required decay time (24 hours) prior to fuel movement ensures sufficient time has elapsed to allow the radioactive decay of the short lived fission products. This decay time is consistent with assumptions used in the accident analyses. However, preparing to move fuel requires operations (e.g., containment entry, removal of drywell head, removal of vessel head, removal of vessel internals) involving the reactor vessel that require more than 24 hours to complete. Thus, although CTS 3/4.9.4, Decay Time, may satisfy Criterion 2 of the Technical Specifications Selection Criteria in 10 CFR 50.36 (c)(2)(ii), the 24 hour decay time limit following subcriticality is a time limit that will always be met for a refueling outage because the plant cannot be placed in a condition that would violate decay time TS requirements. Therefore, the decay time requirement does not result in a limiting condition for reactor operation.

3/4.9.5 Communications

Communication requirements are relocated to the TRM. CTS specify that direct communications are to be maintained between the control room and refueling platform personnel to ensure that refueling personnel can be promptly informed of significant changes in the plant status or core reactivity condition during refueling operations. Communications between control room and refuel platform personnel are necessary for coordinating activities such as the insertion of control rods prior to loading fuel. However, operable control room communications with refueling platform personnel is not an assumption for response to refueling system failures, or design accident or transient response.

3/4.9.6 Crane and Hoist

Crane and hoist requirements are relocated to the TRM. CTS specify that all cranes and hoists (fuel hoist and auxiliary hoist) used for handling fuel assemblies or control rods within the reactor pressure vessel are to be operable. These TS ensure that hoists have sufficient load capacity for handling fuel assemblies and/or control rods and the core internals and pressure vessel are protected from excessive lifting force if they are inadvertently engaged during lifting operations. The interlocks designed to provide the above protection capabilities can prevent damage to the refueling platform equipment and core internals they are; however, not assumed to function to prevent or mitigate the consequences of a design basis accident.

The relocated CTS discussed above are not required to be in the TS under 10 CFR 50.36 and do not meet any of the four criteria in the Final Policy Statement. They are not needed to obviate the possibility that an abnormal situation or event will give rise to an immediate threat to the public health and safety. In addition, the NRC staff finds that sufficient regulatory controls exist under the regulations cited above to maintain the effect of the provisions in these specifications. The NRC staff has concluded that appropriate controls have been established for all of the current specifications, information, and requirements that are being moved to EGC-controlled documents. This is the subject of a license condition established herewith. Until incorporated in the UFSAR and procedures, changes to these specifications, information, and requirements will be controlled in accordance with the applicable current procedures that control these documents. Following implementation, the NRC will audit the removed provisions to ensure that an appropriate level of control has been achieved. The NRC staff has concluded that, in accordance with the Final Policy Statement, sufficient regulatory controls exist under the regulations, particularly 10 CFR 50.59. Accordingly, these specifications, information, and requirements, as described in detail in this SE, may be relocated from CTS and placed in the UFSAR or other EGC-controlled documents as specified in EGC's letter of **date.**

F. Control of Specifications, Requirements, and Information Removed from the CTS

The facility and procedures described in the UFSAR and TRM, incorporated into the UFSAR by reference, can only be revised in accordance with the provisions of 10 CFR 50.59, which ensures records are maintained and establishes appropriate control over requirements removed from CTS and over future changes to the requirements. Other licensee-controlled documents contain provisions for making changes consistent with other applicable regulatory requirements: for example, the ODCM can be changed in accordance with ITS 5.5.1; the emergency plan implementing procedures (EPIPs) can be changed in accordance with 10 CFR 50.54(q); and the administrative instructions that implement the QA Plan can be changed in accordance with 10 CFR 50.54(a) and 10 CFR Part 50, Appendix B. Temporary procedure changes are also controlled by 10 CFR 50.54(a). The documentation of these changes will be maintained by EGC in accordance with the record retention requirements specified in EGC's QA plan for LaSalle and such applicable regulations as 10 CFR 50.59.

The license condition for the relocation of requirements from the CTS addresses the implementation of the ITS conversion and when the relocation of the CTS requirements into licensee-controlled documents will be completed. The submittal of the updated licensee-controlled documents (e.g., UFSAR) to the Commission will be as required by, and in

accordance with, the regulations (e.g., 10 CFR 50.71(e) for the updated UFSAR), and not be as part of the implementation of the ITS.

G. Other TS Changes Included in the Application

This section evaluates other TS changes included in EGC's ITS conversion application. These include items which deviate from both the CTS and the STS, do not fall clearly into a category, or are in addition to those changes that are needed to meet the overall purpose of the conversion.

Conversion to ITS Section 3.6.1.3

CTS 4.6.1.1.b verifies that all penetrations not capable of being closed by automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges or deactivated automatic valves secured in their position, except as provided in CTS 3.6.3. In the ITS, this surveillance is relocated from the CTS Primary Containment Integrity specification (CTS 3/4.6.1.1) to the ITS Primary Containment Isolation Valve Specification (ITS 3.6.1.3) and broken up into two specifications - one for valves and blind flanges outside containment and one for valves and blind flanges inside containment. During the review of the licensee's submittal, a difference of opinion arose between the staff and the licensee as to what would constitute a failure of this CTS surveillance and what appropriate actions should be taken. The staff concedes that the wording and structure of the LaSalle CTS would allow several interpretations of how CTS 4.6.1.1.b is to be met, what actions to take if the surveillance is not met, and which ITS Action Notes are implied by the CTS wording in CTS 3/4.7.A. Depending on the interpretation, the change from the CTS to the ITS could be characterized as Administrative, More Restrictive, Less Restrictive, or a combination thereof.

In addition, the staff concedes that there are several interpretations of how CTS 3.6.M Action and 3.7.D Action 1 can be applied to penetrations with one primary containment isolation valve. One interpretation would require an immediate shutdown since there is no other OPERABLE isolation valve. Another interpretation considers the closed system boundary as the other OPERABLE isolation valve. Depending on which interpretation is used, the change from the CTS to ITS 3.6.1.3 Action C could be characterized as Administrative, Less Restrictive, or a combination of the two.

One objective of the conversion to the ITS is to correct these types of problem areas. The LaSalle ITS provide the appropriate SRs and Actions, if the surveillances are not met, to correct the ambiguity of the CTS while not degrading the safe operation of the plant. Thus, the staff finds that ITS 3.6.1.3 is acceptable.

Conversion to 24 Month Surveillance Interval (LD, LE, LF)

Improved reactor fuels allow licensees to consider increasing the duration of the fuel cycle for their facilities. The staff has reviewed and approved a number of requests to extend surveillance requirements to accommodate a 24-month fuel cycle. The staff has found that the effect on plant safety is small because safety systems use redundant electrical and mechanical components and because licensees perform other surveillances during plant operation that confirm that these systems and components can perform their safety functions.

Generic Letter (GL) 91-04, “Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle,” issued on April 2, 1991, provides staff guidance that identifies the types of information that must be addressed when proposing extensions of the fuel cycle to 24 months. The GL addressed steam generator inspections (which are not applicable to LaSalle), leak rate testing pursuant to Appendix J to 10 CFR Part 50 (which is not applicable to LaSalle because individual leak testing requirements have been replaced by the Primary Containment Leakage Rate Testing Program), instrument drift, and other 18-month surveillances that are extended to 24 months.

The GL requires that licensees address instrument drift when proposing an increase in the surveillance interval for calibrating instruments that perform safety functions including providing the capability for safe shutdown. The effect of the increased calibration interval on instrument errors must be addressed because instrument errors caused by drift were considered when determining safety system setpoints and when performing safety analyses.

For the remaining 18-month surveillances, the GL requires the following information to support conversion to a 24-month operating cycle:

(1) Licensees should evaluate the effect on safety of an increase in 18-month surveillance intervals to accommodate a 24-month fuel cycle. This evaluation should support a conclusion that the effect on safety is small.

(2) Licensees should confirm that historical plant maintenance and surveillance data support this conclusion.

(3) Licensees should confirm that assumptions in the plant licensing basis would not be invalidated on the basis of performing any surveillance at the bounding surveillance interval limit provided to accommodate a 24-month fuel cycle.

In consideration of these confirmations, the staff concluded that licensees need not quantify the effect of the change in surveillance intervals on the availability of individual systems or components.

INSTRUMENT DRIFT

The staff’s review grouped the instrumentation changes together. This primarily includes extensions of channel calibrations and logic system functional tests from 18 to 24 months.

By letter dated March 3, 2000, the licensee submitted a request to amend the Facility Operating Licenses for Dresden, LaSalle, and Quad Cities nuclear power plants. The amendment proposes changes to the technical specifications (TS) to extend the surveillance intervals for selected TS items from 18 months to 24 months. By letter dated March 24, 2000, the licensee submitted the methodology used for the determination of instrument setpoints and allowable values. On April 27, 2000, a meeting was held with the licensee to discuss the staff request for additional information and by letter dated June 5, 2000, the licensee provided the information requested by the staff. On August 22 and 23, a meeting was held with the licensee to review their sample calculations. During that meeting, the staff identified some concerns with the

licensee's response of June 5, 2000, and by letter dated November 30, 2000, the licensee provided the response to resolve the staff's concerns.

GL 91-04 required that information in seven specific areas be addressed in order to provide an acceptable basis for increasing the calibration interval for instruments that are used to perform safety functions. The following discussion identifies these seven areas and includes a summary of the licensee's response along with the staff's conclusions.

(1) Confirm that instrument drift as determined by as-found and as-left calibration data from surveillance and maintenance records have not, except on rare occasions, exceeded acceptable limits for a calibration interval.

(2) Confirm that the values of drift for each instrument type (make, model, and range) and application have been determined with a high probability and a high degree of confidence. Provide a summary of the methodology and assumptions used to determine the rate of instrument drift with time based upon historical plant calibration data.

(3) Confirm that the magnitude of instrument drift has been determined with a high probability and a high degree of confidence for a bounding calibration interval of 30 months for each instrument type (make, model number, and range) and application that performs a safety function. Provide a list of the channels by TS section that identifies these instrument applications.

(4) Confirm that a comparison of the projected instrument drift errors has been made with the values of drift used in the setpoint analysis. If this results in revised setpoints to accommodate large drift errors, provide proposed TS changes to update trip setpoints. If the drift errors result in a revised safety analysis to support existing setpoints, provide a summary of the updated analysis conclusions to confirm that safety limits and safety analysis assumptions are not exceeded.

(5) Confirm that the projected instrument errors caused by drift are acceptable for control of plant parameters to effect a safe shutdown with the associated instrumentation.

(6) Confirm that all conditions and assumptions of the setpoint and safety analyses have been checked and are appropriately reflected in the acceptance criteria of plant surveillance procedures for channel checks, channel functional tests, and channel calibrations.

(7) Provide a summary description of the program for monitoring and assessing the effects of increased calibration surveillance intervals of instrument drift and its effect on safety.

The licensee performed a safety assessment for the proposed changes to the surveillance test intervals in accordance with the GL 91-04 guidance stated above. This assessment entailed reviewing the historical maintenance and surveillance test data at the bounding surveillance test interval limit, performing an evaluation to ensure that a 24-month surveillance test interval would not invalidate any assumption in the plant licensing bases, and the determination that the effect of the surveillance interval extension is small.

In their submittals of March 3, and 24, 2000, the licensee identified Nuclear Engineering Standard NES-EIC-20.04, Rev. 1, "Analysis of Instrument Channel Setpoint Error and

Instrument Loop Accuracy," which included Appendix J, "Guidelines For the Analysis and Use of As-Found/As-Left Data," as the basis for performing analyses of drift for all affected instrument loops in order to establish the effect of a 30-month (24 months + 25% allowable tolerance) calibration frequency on instrument performance. This appendix is based on Electric Power Research Institute (EPRI) TR-103335, "Guidelines for Instrument Calibration Extension/Reduction Programs," Rev. 1, October 1998. The licensee has used Microsoft Excel spreadsheets to document information for performing additional analyses to be consistent with the analyses recommended by NRC in its safety evaluation report (SER) for the Peach Bottom Atomic Power Station, Units 2 and 3.

During the meeting of April 27, 2000, the staff identified concerns with the licensee's sample data, outlier determination, time dependency, the graded approach to instrument setpoint determination (Appendix D to the Nuclear Engineering Standard), and miscellaneous other items. Based on the staff's comments, the licensee, by letter dated June 5, 2000, submitted the revised Nuclear Engineering Standard and their justification for surveillance extensions. The staff reviewed the revised documents and was still concerned with the outlier determination, time dependency, and the graded approach to instrument setpoint determination. However, during a conference call the licensee was able to satisfy the staff's concerns and it was decided to have a meeting to review some sample calculations to better understand the licensee's methodology. The staff reviewed the sample calculations and determined the licensee's approach acceptable but wanted the licensee to revise the Nuclear Engineering Standard to clearly describe their methodology. Based on this, the licensee provided Rev. 3 of the Nuclear Engineering Standard and submitted a letter dated November 30, 2000, to state that graded approach to setpoint determination has not been used by the licensee.

The staff has reviewed the licensee's submittals, including the responses to additional information, and has verified that the licensee has addressed the issues identified in GL 91-04 and provided an acceptable basis for increasing the calibration interval and for determining the instrument setpoint and allowable values for instruments that are used to perform safety functions. On the basis of the evaluation, the staff concludes that the licensee has confirmed that safety limits and safety analysis assumptions will not be exceeded after the worst-case drift is considered for the instruments whose surveillance intervals will be extended to 24 months.

On the basis of its review, the staff concludes that the proposed methodology for extending surveillance intervals for certain safety-related instrumentation components is consistent with the guidance in GL 91-04 in that the licensee has demonstrated that the effect of extending the surveillance intervals to 24 months is negligible and the system will continue to perform within assumed limits during the longer surveillance interval. The staff also finds that the instrument setpoint methodology used by the licensee to determine the allowable values is acceptable.

NON-INSTRUMENTATION CHANGES

Regarding non-instrumentation changes, GL 91-04 requires licensees to evaluate the effect on safety of the change in surveillance intervals to accommodate a 24-month fuel cycle. This evaluation should support a conclusion that the effect on safety is small. In addition, licensees should confirm that the performance of surveillances at the bounding surveillance interval limit provided to accommodate a 24-month fuel cycle would not invalidate any assumption in the plant licensing basis. In consideration of these confirmations, the licensees need not quantify

the effect of the change in surveillance intervals on the availability of individual systems or components.

To address the requirements of the GL 91-04, the licensee has referenced the NRC SER (dated August 2, 1993) relating to the extension of the Peach Bottom Units 2 and 3 surveillance intervals from 18 months to 24 months. In this SER, the staff stated the following:

> *Industry reliability studies for boiling water reactors (BWRs), prepared by the BWR Owners Group (NEDC-30936P) show that the overall safety systems' reliabilities are not dominated by the reliabilities of the logic system, but by that of the mechanical components, (e.g., pumps and valves), which are consequently tested on a more frequent basis. Since the probability of a relay, or contact failure is small relative to the probability of mechanical component failure, increasing the Logic System Functional Test interval represents no significant change in the overall safety system unavailability.*

The licensee has reviewed the surveillance test history at LaSalle and has validated this conclusion. The licensee's review has demonstrated that there are no failures that would invalidate the conclusion that the impact, if any, on system availability is minimal from a change to a 24-month operating cycle.

The following discussion describes how the staff determined that the effect of extending surveillance intervals on plant safety is small. The staff's review focused on redundant electrical and mechanical components as well as other surveillances conducted during plant operation that confirm that these systems and components can perform their safety functions.

<u>TS 3.1.7</u> <u>Standby Liquid Control System</u>

<u>SR 3.1.7.8 and 3.1.7.9</u>

> These SRs ensure that the Standby Liquid Control System is capable of injecting into the reactor pressure vessel by verifying a flow path and by firing one of the explosive valves.

The licensee provided the following justification for concluding that the effect on safety due to the extended surveillance frequency will be small:

- The licensee reviewed historical maintenance and surveillance data which shows that these tests normally pass at the current frequency and that no failures have occurred that would invalidate the conclusion that the impact, if any, on system availability is small due to this change.
- The following tests ensure that the system is operable during the operating cycle:
  - SR 3.1.7.7 which verifies system capacity
  - SR 3.1.7.2 and 3.1.7.3 which ensure that the temperature in the SLC system tank and SLC pump suction piping is maintained to prevent precipitation of sodium pentaborate
  - SR 3.1.7.4 which verifies the continuity of the charge in the explosive valves.

Based on the information above, the staff concludes that the impact on plant safety due to the proposed change is small and, therefore, the change is acceptable.

TS 3.1.8 Scram Discharge Volume Vent and Drain Valves

SR 3.1.8.3

This SR ensures that the scram discharge volume vent and drain valves close in ≤ 30 seconds after receipt of an actual or simulated scram signal and open when the actual or simulated scram signal is reset.

The licensee provided the following justification for concluding that the effect on safety due to the extended surveillance frequency will be small:

- The licensee reviewed historical maintenance and surveillance data which shows that these tests normally pass at the current frequency and that no failures have occurred that would invalidate the conclusion that the impact, if any, on system availability is small due to this change.
- The following test ensures that the system is operable during the operating cycle. SR 3.1.8.2 requires that the scram discharge volume vent and drain valves be cycled fully closed and fully open every 92 days during the operating cycle. Although this test does not ensure that the logic of the SDV vent and drain valves is operable, logic systems are inherently more reliable and, therefore, the impact of the increased surveillance interval is small.

Based on the information above, the staff concludes that the impact on plant safety due to the proposed change is small and, therefore, the change is acceptable.

TS 3.4.2 Flow Control Valves

SR 3.4.2.1 and 3.4.2.2

These SRs ensure that flow control valves fail “as is” on loss of hydraulic pressure at the hydraulic control unit and that the average rate of flow control valve (FCV) movement is within the specific limit of 11% of stroke per second.

The licensee provided the following justification for concluding that the effect on safety due to the extended surveillance frequency will be small:

- The licensee reviewed historical surveillance data which shows that no failures have occurred that would invalidate the conclusion that the impact, if any, on system availability is small due to this change.
- The FCVs are utilized during normal plant operation and major deviations will be identified.
- The licensee’s analysis has shown that, given a LOCA event, no single failure in the electronic/hydraulic controls can cause the FCV to close while in the normal manual control mode.
- Backup electronic velocity limiters are included in the recirculation control system to limit FCV velocity to 11%. Additional multiple specific component failures in these limiters must occur to cause the full closure of the FCV at velocities in excess of this value.

Based on the information above, the staff concludes that the impact on plant safety due to the proposed change is small and, therefore, the change is acceptable.

TS 3.4.7 Reactor Coolant System Leakage Detection Instrumentation

SR 3.4.7.3

These SRs ensure that the required primary containment atmosphere particulate, atmospheric gaseous, floor drain sump flow, and air cooler condensate flow rate monitoring systems are operable and within the established calibration requirements.

The licensee provided the following justification for concluding that the effect on safety due to the extended surveillance frequency will be small:

- The licensee reviewed historical surveillance data which shows that no failures have occurred that would invalidate the conclusion that the impact, if any, on system availability is small due to this change.
- These systems do not provide for actuation of any safety devices and provide a monitoring function only. In addition, the setpoint of these devices is not an assumption in any event analysis.
- These systems provide redundant detection methods.

Based on the information above, the staff concludes that the impact on plant safety due to the proposed change is small and, therefore, the change is acceptable.

ITS 3.5 ECCS

SR 3.5.1.6

This SR ensures that a system initiation signal (actual or simulated) to the automatic initiation logic of HPCS, LPCS, and LPCI will cause the subsystems to operate as designed, including actuation of the system throughout its emergency operation sequence, automatic pump startup and actuation of all automatic valves to their required positions.

SR 3.5.1.9

This SR ensures that each ECCS injection/spray subsystem responds in a manner consistent with the values assumed in the accident analysis. (This test applies to Modes 1, 2, and 3)

SR 3.5.2.7

This SR ensures that each ECCS injection/spray subsystem responds in a manner consistent with the values assumed in the accident analysis. (This test applies to Modes 4 and 5)

SR 3.5.1.7

This SR ensures that the mechanical portions of the Automatic Depressurization System (ADS) function as designed when initiated either by an actual or simulated initiation signal.

SR 3.5.1.8

This SR ensures that the ADS valves and solenoids operate properly.

The licensee provided the following justification for concluding that the effect on safety due to the extended surveillance frequency for each of the above surveillances will be small:

- The licensee reviewed historical surveillance data which shows that no failures have occurred that would invalidate the conclusion that the impact, if any, on system availability is small due to this change.
- The ECCS network has built-in redundancy so that no single failure will prevent the starting of the ECCS system.
- ADS is equipped with two redundant trip systems
- Other, more frequent tests will detect significant failures in the ECCS subsystems to perform their safety function. For example: each of the ECCS injection/spray systems are tested every three months according to the ASME Section XI inservice testing program, and surveillances are performed every 31 days to ensure that the subsystems are available to perform their safety function.

Based on the information above, the staff concludes that the impact on plant safety due to the proposed changes is small and, therefore, the changes are acceptable.

TS 3.5.3 RCIC System

SR 3.5.3.4

This SR ensures that the RCIC system is capable of performing its design function before reactor pressure is increased above the system minimum operating pressure.

SR 3.5.3.5

This SR ensures that a system initiation signal (actual or simulated) to the automatic initiation logic of RCIC will cause the systems or subsystems to operate as designed, including actuation of the system throughout its emergency operating sequence, automatic pump startup and actuation of all automatic valves to their required positions.

The licensee provided the following justification for concluding that the effect on safety due to the extended surveillance frequency for each of the above surveillances will be small:

- The licensee reviewed historical surveillance data which shows that no failures have occurred that would invalidate the conclusion that the impact, if any, on system availability is small due to this change.
- The safety analysis does not take credit for the RCIC system
- The functions performed by RCIC are redundant to those performed by HPCS.

- RCIC will continue to be tested every three months to ensure required flow at normal operating pressure. This test would detect significant failures of the RCIC turbine or pump that could lead to the failure of the system to perform its safety function at low pressures.

Based on the information above, the staff concludes that the impact on plant safety due to the proposed changes is small and, therefore, the changes are acceptable.

TS 3.6.1.1 Primary Containment

SR 3.6.1.1.3

This SR ensures that the drywell-to-suppression chamber bypass leakage is less than or equal to the bypass leakage limit.

The licensee provided the following justification for concluding that the effect on safety due to the extended surveillance frequency for the above surveillance will be small:

- The licensee reviewed historical surveillance data which shows that no failures have occurred that would invalidate the conclusion that the impact, if any, on system availability is small due to this change.
- The risk of high radiation exposure requires that this surveillance be performed during shutdown.
- Tests are performed in accordance with the Primary Containment Leakage Rate Testing Program that would identify most component failures.

Based on the information above, the staff concludes that the impact on plant safety due to the proposed change is small and, therefore, the change is acceptable.

TS 3.6.1.3 Primary Containment Isolation Valves (PCIV)

SR 3.6.1.7

This SR ensures that each automatic PCIV will actuate to its isolation position on a primary containment isolation signal.

SR 3.6.1.3.8

This SR ensures that each excess flow check valve (EFCV) actuates to the isolation position on an actual or simulated instrument line break condition.

SR 3.6.1.3.9

This SR requires that the explosive squib from be removed and tested for the shear isolation valve of the Traversing Incore Probe System.

The licensee provided the following justification for concluding that the effect on safety due to the extended surveillance frequency for each of the above surveillances will be small:

- The licensee reviewed historical surveillance data which shows that no failures have occurred that would invalidate the conclusion that the impact, if any, on system availability is small due to this change.
- During the operating cycle, the PCIVs are either exercised or partially stroked. These exercises test a significant portion of the PCIV's circuitry and will detect failures of the circuitry or failures with valve movement.
- The PCIVs, including the actuating logic, are designed to be single failure proof and are, therefore, highly reliable.
- The EFCVs are required to be tested under the conditions that apply during a plant outage. In addition, the potential for an unplanned transient increases if the surveillance were performed with the reactor at power.
- The instrument lines associated with the EFCVs are provided with flow-restricting orifices which are sized to ensure that in the event of a postulated failure of the piping or component, the potential offsite exposure would be substantially below the guidelines of 10 CFR 100.
- The TIP shear isolation valve explosive charge is also verified on a monthly basis by the requirement of TS 3.6.1.3.4. In addition, administrative controls on the explosive charges, such as limits on shelf life and operating life, provide further assurance that the explosive squibs will operate as designed.

Based on the information above, the staff concludes that the impact on plant safety due to the proposed changes is small and, therefore, the changes are acceptable.

<u>TS 3.6.1.6</u> <u>Suppression Chamber to Drywell Vacuum Breakers</u>

<u>SR 3.6.1.6.3</u>

This SR verifies that the opening setpoint of each suppression chamber-to-drywell vacuum breaker is less than or equal to the specified differential pressure.

The licensee provided the following justification for concluding that the effect on safety due to the extended surveillance frequency for each of the above surveillances will be small:

- The licensee reviewed historical surveillance data which shows that no failures have occurred that would invalidate the conclusion that the impact, if any, on system availability is small due to this change.
- This surveillance must be performed under the conditions that apply during a plant outage.
- The potential for an unplanned transient is increased if the surveillance were performed with the reactor at power.
- Other surveillances, such as a functional test of each vacuum breaker every 92 days and verification that each breaker is closed every 14 days, provide additional assurance that the breakers would function as designed.

Based on the information above, the staff concludes that the impact on plant safety due to the proposed change is small and, therefore, the change is acceptable.

TS 3.6.3.1 Primary Containment Hydrogen Recombiners

SR 3.6.3.1.1

This SR verifies the ability of the recombiner system to actuate and prevent the hydrogen-oxygen level within the primary containment from reaching the flammability limit.

SR 3.6.3.1.2

This SR ensures that there are not detectable grounds in any heater phase by verifying that the resistance to ground fro any heater phase is greater than the required resistance value when this SR is performed following performance of the system functional test.

The licensee provided the following justification for concluding that the effect on safety due to the extended surveillance frequency for each of the above surveillances will be small:

- The licensee reviewed historical surveillance data which shows that no failures have occurred that would invalidate the conclusion that the impact, if any, on system availability is small due to this change.
- Redundancy of the recombiner system and the availability of alternate hydrogen control system.
- The backup Hydrogen Purge System also functions in conjunction with the hydrogen recombiner and can filter purged air from the primary containment, post-LOCA, after the containment pressure has dropped below a predetermined value.

Based on the information above, the staff concludes that the impact on plant safety due to the proposed change is small and, therefore, the change is acceptable.

TS 3.6.4.1 Secondary Containment

These SRs ensure secondary containment boundary integrity by demonstrating that secondary containment vacuum assumed in the safety analysis can be established and maintained under design basis conditions.

SR 3.6.4.1.3

This SR verifies the secondary containment can be drawn down to the specified vacuum in the time required using one standby gas treatment subsystem.

SR 3.6.4.1.3

This SR verifies the secondary containment can be maintained at the specified vacuum for the required time using one SGT subsystem at the specified flow rate.

The licensee provided the following justification for concluding that the effect on safety due to the extended surveillance frequency for each of the above surveillances will be small:

- The licensee reviewed historical surveillance data which shows that no failures have occurred that would invalidate the conclusion that the impact, if any, on system availability is small due to this change.
- Secondary containment is maintained at a negative pressure during normal operations.
- Secondary containment structural integrity is maintained through administrative controls which ensure that no significant changes will be made to the secondary containment structure without proper evaluation. Any event which would cause significant structural degradation, such as a seismic event, would require a plant evaluation.

Based on the information above, the staff concludes that the impact on plant safety due to the proposed change is small and, therefore, the change is acceptable.

TS 3.6.4.2 Secondary Containment Isolation Valves (SCIVs)

SR 3.6.4.2.3

This SR verifies each automatic secondary containment isolation valve actuates to the isolation position on an actual or simulated automatic isolation signal.

The licensee provided the following justification for concluding that the effect on safety due to the extended surveillance frequency for each of the above surveillances will be small:

- The licensee reviewed historical surveillance data which shows that no failures have occurred that would invalidate the conclusion that the impact, if any, on system availability is small due to this change.
- These valves are also tested every 92 days to satisfy the requirements of SR 3.6.4.2.2 which verifies isolation times are within limits. These tests would detect significant failures affecting valve operation.
- The SCIV system active components and power supplies are designed with redundancy to meet the single active failure criteria.
- Industry reliability studies show that the overall safety systems' reliabilities are not dominated by the reliabilities of the logic system, but by that of the mechanical components which are tested on a more frequent basis.

Based on the information above, the staff concludes that the impact on plant safety due to the proposed change is small and, therefore, the change is acceptable.

TS 3.6.4.3 Standby Gas Treatment (SGT) System

SR 3.6.4.3.3

This SR verifies that each SGT subsystem actuates on an actual or simulated initiation signal.

The licensee provided the following justification for concluding that the effect on safety due to the extended surveillance frequency for each of the above surveillances will be small:

- The licensee reviewed historical surveillance data which shows that no failures have occurred that would invalidate the conclusion that the impact, if any, on system availability is small due to this change.
- The system is operated every 31 days to satisfy the requirements of SR 3.6.4.3.1, which operates each SGT subsystem for a specified period of time, ensures that both subsystems are operable and that all associated controls are functioning properly. This test will detect significant failures affecting system operation.
- The SGT system is designed with redundancy to meet the single active failure criteria.
- Industry reliability studies show that the overall safety systems' reliabilities are not dominated by the reliabilities of the logic system, but by that of the mechanical components which are tested on a more frequent basis.

Based on the information above, the staff concludes that the impact on plant safety due to the proposed change is small and, therefore, the change is acceptable.

TS 3.7.2 Diesel Generator Cooling Water (DGSW) System

SR 3.7.2.2

This SR verifies that each DGCW pump starts automatically on each required actual or simulated initiation signal.

The licensee provided the following justification for concluding that the effect on safety due to the extended surveillance frequency for each of the above surveillances will be small:

- The licensee reviewed historical surveillance data which shows that no failures have occurred that would invalidate the conclusion that the impact, if any, on system availability is small due to this change.
- This requirement is also verified on a more frequent basis by the following tests: (1) diesel generator start testing every 31 days per TS 3.8.1.2, and (2) Low Pressure Coolant Spray pump start testing every 92 days for the Inservice Testing Program.

Based on the information above, the staff concludes that the impact on plant safety due to the proposed change is small and, therefore, the change is acceptable.

TS 3.7.3 Ultimate Heat Sink

SR 3.7.3.2 and 3.7.3.3

These SRs verify the sediment deposition and bottom elevation of the cooling pond. They ensure that the volume of water in the CSCS pond will be adequate to support long term cooling for a 30 day period after a design basis accident.

The licensee provided the following justification for concluding that the effect on safety due to the extended surveillance frequency for each of the above surveillances will be small:

- The licensee reviewed historical surveillance data which shows that no failures have occurred that would invalidate the conclusion that the impact, if any, on system availability is small due to this change.

- A hydrographic survey of the pond was performed in 1999 which showed that the amount of sediment that accumulated from the time of original licensing to the survey date (over 17 years) was negligible.

Based on the information above, the staff concludes that the impact on plant safety due to the proposed change is small and, therefore, the change is acceptable.

TS 3.7.4 Control Room Area Filtration System

SR 3.7.4.4 and 3.7.4.5

These SRS ensure that each CRAF subsystem is capable of automatic initiation and that the mechanical components operate as designed on system actuation and that the control room area boundary leakage is within the capacity of the CRAF system by demonstrating that control room area can be maintained at a positive pressure with respect to adjacent areas when in the pressurization mode of operation.

The licensee provided the following justification for concluding that the effect on safety due to the extended surveillance frequency for each of the above surveillances will be small:

- The licensee reviewed historical surveillance data which shows that no failures have occurred that would invalidate the conclusion that the impact, if any, on system availability is small due to this change.
- The control room boundary is maintained at a positive pressure during normal operation. Therefore, any substantial degradation of the boundary that would prevent maintaining the control room area at the required pressure will be evident.
- The CRAF system will be tested every 31 days by SR 3.7.4.1 and SR 3.7.4.2 which will detect any significant mechanical component failures and verify the operability of the majority of the CRAF system circuitry.

Based on the information above, the staff concludes that the impact on plant safety due to the proposed change is small and, therefore, the change is acceptable.

TS 3.7.7 Main Turbine Bypass System

SR 3.7.7.2 and 3.7.7.3

These SRS ensure that the Main Turbine Bypass System will function with the required response as assumed in the transient analysis such as the turbine generator load rejection and feedwater transients in order to mitigate the increase in reactor vessel pressure.

The licensee provided the following justification for concluding that the effect on safety due to the extended surveillance frequency for each of the above surveillances will be small:

- The licensee reviewed historical surveillance data which shows that no failures have occurred that would invalidate the conclusion that the impact, if any, on system availability is small due to this change.
- The main turbine bypass logic which is being tested is part of the Main Turbine Control System which is in continuous operation at power and most malfunctions that would

impact the main turbine bypass system would also impact the main turbine control system and be readily apparent during plant operation.
- The weekly test of the turbine bypass valves (SR 3.7.7.1) will also detect problems since the test uses a fast open signal for the last 10% of valve travel.

Based on the information above, the staff concludes that the impact on plant safety due to the proposed change is small and, therefore, the change is acceptable.

TS 5.5 Programs and Manuals

SR 5.5.2.b

This SR 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 licensee provided the following justification for concluding that the effect on safety due to the extended surveillance frequency for each of the above surveillances will be small:

- The licensee reviewed historical surveillance data which shows that no failures have occurred that would invalidate the conclusion that the impact, if any, on system availability is small due to this change.
- Most portions of the subject systems included in this program are visually walked down during plant testing, and/or operator/system engineer walkdowns which would detect gross leakage.
- Plant radiological surveys will identify any potential sources of leakage.

SR 5.5.8

This SR ensures that the standby gas treatment (SGT) system and control room area filtration (CRAF) system inplace charcoal adsorbers, HEPA filters, and heaters perform their safety function.

The licensee provided the following justification for concluding that the effect on safety due to the extended surveillance frequency for each of the above surveillances will be small:

- The licensee reviewed historical surveillance data which shows that no failures have occurred that would invalidate the conclusion that the impact, if any, on system availability is small due to this change.
- 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 systems. These tests would detect potential changes in HEPA filter efficiency, carbon adsorber bypass leakage, or filter pressure drop,
- The SGT and CRAF system active components and power supplies are designed with redundancy to meet the single active failure criteria.

Based on the information above, the staff concludes that the impact on plant safety due to the proposed change is small and, therefore, the change is acceptable.

Additional TS Changes and Beyond-Scope Items

<<<To be provided later.>>>

IV. STATE CONSULTATION

In accordance with the Commission's regulations, the Illinois State official was notified of the proposed issuance of the amendments. The State official had no comments.

V. ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an Environmental Assessment and Finding of No Significant Impact was published in the *Federal Register* date on **date** (**citation**). Accordingly, based upon the environmental assessment, the Commission has determined that issuance of this amendment will not have a significant effect on the quality of the human environment.

VI. CONCLUSION

The LaSalle ITS provide clearer, more readily understandable requirements to ensure safe operation of the plant. The NRC staff concludes that they satisfy the guidance in the Commission's policy statement with regard to the content of TS and conform to the model provided in NUREG-1434 with appropriate modifications for plant-specific considerations. The NRC staff further concludes that the LaSalle ITS satisfy Section 182a of the Atomic Energy Act, 10 CFR 50.36, and other applicable standards. On this basis, the NRC staff concludes that the proposed LaSalle ITS are acceptable.

The NRC staff has also reviewed the plant-specific changes to CTS as described in this evaluation. On the basis of the evaluations described herein for each of the changes, the NRC staff concludes that these changes are acceptable.

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner; (2) such activities will be conducted in compliance with the Commission's regulations; and, (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

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CTS Discussion of Change Tables