SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. TO FACILITY OPERATING LICENSE NO. DPR-29

AND AMENDMENT NO. TO FACILITY OPERATING LICENSE NO. DPR-30

EXELON GENERATION COMPANY, LLC

AND

MIDAMERICAN ENERGY COMPANY

QUAD CITIES NUCLEAR POWER STATION, UNITS 1 AND 2

DOCKET NOS. 50-254 AND 50-265

I. INTRODUCTION

Quad Cities Nuclear Power Station, Units 1 and 2 (Quad Cities), has been operating with Technical Specifications (TS), issued on June 28, 1996, that were developed during the Technical Specification Upgrade Program (TSUP), as amended from time to time. The TSUP was a partial adoption of the TS found in NUREG-0123, "Standard Technical Specifications General Electric Plants BWR/4," Revision 4. The TSUP was initiated as a result of findings by a Diagnostic Evaluation Team inspection performed at Dresden Nuclear Power Station (Quad Cities' sister site) in 1987.

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

— NUREG-1433, "Standard Technical Specifications - General Electric Plants, BWR/4" 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 Quad Cities 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 Quad Cities TS are referred to as the Current TS (CTS), and the TS in NUREG-1433 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 Quad Cities 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 Quad Cities operating license were approved, as follows:

Amendment No. (Unit 1, Unit 2)	Description of Change	Issue Date
196 192	Remove Main Steam Line Radiation Monitor SCRAM and Isolation Functions	10/13/2000
197 193	Transfer of Operating License to EGC	01/12/2001
No., No	Increase Surveillance Test Interval and Allowed Outage Time for Assorted Instrumentation	**date**

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.

In addition, the ITS conversion was supported by two other license amendment requests, dated August 30, 2000, which the licensee identified as being required to implement the ITS. One request related to the Emergency Diesel Generator Cooling Water Pump Allowed Outage Time, and the other changed the surveillance requirements for the Emergency Diesel Generator. The August 30, 2000, applications provided additional supporting information for changes that had already been incorporated into the March 3, 2000, application. The review of the August 30, 2000, applications is included in this safety evaluation.

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 Quad Cities ITS is acceptable for continued plant operation. It is acceptable that the ITS differs from STS, since the ITS reflects Quad Cities'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-1433, 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 Quad Cities's CTS to those based on STS as modified by plant-specific changes is consistent with Quad Cities'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 (not used for Quad Cities)

— Technical Changes - Less Restrictive, (LC), i.e., eliminating instrumentation requirements for alarm and indication only functions (not used for Quad Cities)

— 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 Quad Cities 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, LD, LE and LF)

L, 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. 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 Quad Cities 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 Quad Cities 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.

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

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.

Type 2 Descriptions of Systems Operation

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.

3/4.2.E Control Rod Block Actuation

The CTS requires the control rod block actuation channels shown in Table 3.2.E-1 to be operable with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.2.E-1. Several control rod block actuation functions are relocated to the TRM.

3/4.2.E.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.

3/4.2.E.3 Source Range Monitors (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.

3/4.2.E.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.

3/4.2.E.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.

3/4.2.F 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 3.2.F-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 Dresden 2 and 3 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: Drywell air temperature, safety and relief valve position indicators - acoustic and temperature, neutron monitoring (source range), and torus air temperature. Those instruments meeting the criteria are retained by the ITS criteria.

3/4.2.H Explosive Gas Monitoring Instrumentation

Explosive gas monitoring instrumentation are relocated to the TRM. The CTS require explosive gas monitoring instrumentation channels shown in Table 3.2.H-1 to be operable with their Alarm/Trip setpoints set to ensure that the limits of specification 3.8.H 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.

3/4.2.I Suppression Chamber and Drywell Spray Actuation

Suppression chamber and drywell spray actuation instrumentation are relocated to the TRM. CTS require the suppression chamber and drywell spray actuation instrumentation channel(s) shown in Table 3.2.I-1 to be operable with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.2.I-1. The suppression chamber and drywell spray actuation instrumentation preclude inadvertent actuation of containment and suppression pool sprays during a LOCA. In the presence of a LOCA signal, the spray valves can not open unless 1) the reactor vessel water level is above the 2/3 core height level, to preclude diversion of LPCI when water inventory is needed for core flooding, and 2) the drywell pressure is between 0.5 psig and 1.5 psig, to ensure a line break is detected.

The operability of the suppression chamber and drywell spray actuation instrumentation does not affect the operability of LPCI. If either of the two instruments trip too soon, the other instrument Function still ensures that flow is not diverted away from core flooding. While tripping of both the instruments allow the permissives for opening drywell and suppression pool spray valves to be met, inadvertent operation does not automatically result, since manual actions must still be taken to open the valves. In addition, if a LOCA signal is not present, this instrumentation does not preclude operation of the drywell and suppression pool spray valves. Therefore, inadvertent operation of drywell spray has been analyzed at Dresden 2 and 3 **(applicable to Quad Cities?)** and does not result in containment failure due to operation of the reactor building-suppression chamber and the suppression chamber-drywell vacuum breakers. These vacuum breakers are controlled by both CTS and ITS. Therefore, operability of the Drywell Spray System and the Suppression Chamber Spray System are not impacted.

If the instruments trip too late or not at all, then no flow can be diverted by the drywell and suppression chamber sprays; thus LPCI is not affected. The only Technical Specification system affected in this case are the Drywell Spray System and the Suppression Chamber Spray System. A failure of the instrumentation to function would preclude the spray valves from being opened from the control room. However, these systems are manually controlled systems that are not needed for a minimum of 10 minutes following a DBA LOCA, and the valves could still be opened locally at the valve operator. In addition, the instruments could be overridden to allow operation from the control room. Therefore, operability of these instruments are not an initial assumption of any design basis accident (DBA) or transient analysis.

3/4.2.K Toxic Gas Monitoring

The Toxic Gas Monitoring System is relocated to the TRM. CTS require the toxic gas monitoring system to be operable with their alarm/trip setpoints adjusted to actuate at an ammonia concentration of less than or equal to 50 ppm. This system ensures sufficient capability is available to promptly detect and initiate protective action in the event of an accidental ammonia release. This capability protects control room personnel; however, the instruments are not assumed to mitigate a design basis accident (DBA) or transient since an accidental ammonia release is not a DBA or transient.

3/4.3.N Economic Generation Control System

The Economic Generation Control System limits are relocated to the TRM. CTS 3/4.3.N specify that the economic generation control system (EGCS) may be in operation with automatic flow control provided that core flow is $\geq$ 65% and $\leq$ 100% of rated core flow, and thermal power is $\geq$ 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.

3/4.6.N Structural Integrity

The CTS requirements that the structural integrity of ASME Code Class 1, 2 and 3 components (pumps and valves) be maintained operable in accordance with Specification 4.6.N are relocated to the TRM. Specification 4.6.N 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.L Drywell Spray

CTS require the Drywell Spray function of the low pressure coolant injection (LPCI)/containment cooling systems to be operable with two independent subsystems, each subsystem consisting of one operable LPCI pump, and an operable flow path capable of recirculating water from the suppression pool through a heat exchanger and the drywell spray nozzles. These requirements are relocated to the TRM.

The drywell spray function of the LPCI/containment cooling systems is utilized in post-LOCA conditions to condense steam in the drywell, thereby further lowering containment pressure. Emergency operating procedures direct manual initiation of the drywell spray function of the LPCI/containment cooling systems. However, in the analysis of the bounding event for containment pressurization due to the DBA, the drywell spray function of the LPCI/containment cooling systems was not utilized for mitigation of the event. The drywell spray function is not required for proper performance of the containment pressure suppression system and is not an initial assumption of any design basis accident (DBA) or transient analysis.

3/4.8.E Flood Protection

Flood protection requirements are relocated to the TRM. Flood protection shall be available for all required safe shutdown systems, components and structures. This Technical Specification has provisions for high river level. A high river water level is a preliminary indication of flood conditions. Flooding is not a design basis accident (DBA) or transient. In addition, flooding is not postulated to occur during any DBA or transient, thus river water level (as it pertains to flooding) is not credited in any safety analysis. The Flood Protection Technical Specification requirements were put in place to ensure that facility protective actions will be taken and operation will be terminated in the event of flood conditions. This requirement is adequately controlled in plant emergency procedures.

3/4.8.G 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 $\geq$ 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.

3/4.10.E 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.

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 Quad Cities 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.

<u>Conversion to ITS Section 3.6.1.3</u>

CTS 4.7.A.2 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 3.7.D. In the ITS, this surveillance is relocated from the CTS Primary Containment Integrity specification (CTS 3/4.7.A) 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 Quad Cities CTS would allow several interpretations of how CTS 4.7.A.2 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 Quad Cities 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.

<u>Conversion to 24 Month Surveillance Interval (LD, LE, LF)</u>

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 Quad Cities), leak rate testing pursuant to Appendix J to 10 CFR Part 50 (which is not applicable to Quad Cities 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 Quad Cities 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 (SLC) System</u>

The SLC system is a backup to the control rod drive system and designed to be single-failure proof.

<u>SR 3.1.7.8</u>   Verify flow through one SLC subsystem from pump into reactor pressure vessel

This SR ensures that the SLC system is capable of injecting into the reactor pressure vessel by verifying a flow path and also by firing one of the explosive valves.

<u>SR 3.1.7.9</u>   Verify all heat traced piping between storage tank and pump suction is unblocked

This SR ensures that the SLC system is capable of injecting into the reactor pressure vessel by verifying a flow path through the heat traced piping.

System availability during the operating cycle is assured by:

- The SLC system is designed so that all active components are single failure proof.
- Each SLC pump is tested during the operating cycle in accordance with the Inservice Testing Program.
- Daily SRs verifies that temperatures in the SLC system tank and the SLC pump suction piping precludes boron precipitation.

- Monthly SRs ensure the continuity of the explosive charge on the discharge valves.
- The explosive valves are designed to be highly reliable.
- The licensee's review of surveillance test history did not identify any test 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.

Based on this information, the staff concludes that the proposed change on plant safety is small and, therefore, acceptable.

TS 3.1.8 Scram Discharge Volume (SDV) Vent and Drain Valves

SR 3.1.8.3 Verify each SDV vent and drain valve:

(A) Closes in $\leq$ 30 seconds after receipt of an actual or simulated scram signal; and
(B) Opens when the actual or simulated scram signal is reset

This SR ensures that the SDV vent and drain valves close in $\leq$ 30 seconds after receipt of an actual or simulated scram signal and open when the actual or simulated scram signal is reset.

System availability during the operating cycle is assured by:

- SR 3.1.8.2 requires that the SDV vent and drain valves be cycled fully closed and fully open once every 92 days during the operating cycle.
- Performance of SR 3.1.8.2 demonstrates that mechanical components and portions of the valve logic remain operable.
- The licensee's review of surveillance test history did not identify any test 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

Based on this information, the staff concludes that the proposed change on plant safety is small and, therefore, acceptable.

<u>TS 3.4.5</u> <u>RCS Leakage Detection Instrumentation</u>

<u>SR 3.4.5.3</u> Perform a CHANNEL CALIBRATION of required leakage detection instrumentation.

This SR ensures that the required primary containment atmosphere particulate monitoring system and drywell floor drain sump monitoring systems are operable and within the established calibration requirements.

System availability during the operating cycle is assured by:

- The primary containment atmosphere particulate monitoring system and the drywell floor drain sump monitoring system provide independent methods to detect RCS leakage.
- Instrument drift occurs incrementally over an extended time frame whereas the detection of RCS leakage would be represented by a sudden step change. Therefore, the detection of RCS leakage would not be precluded from the long-term affects of instrument drift.

- The primary containment atmosphere particulate monitoring system and the drywell floor drain sump monitoring system only provides a monitoring function and is not relied upon for actuation of safety-related equipment.
- SR 3.4.5.1, which is performed once every 12 hours, verifies operation of the primary containment atmospheric monitoring system.
- SR 3.4.5.2, which is performed once every 31 days, requires a channel functional test which would detect significant system degradation.
- The licensee's review of historical calibration records concludes that the impact, if any, of this change on system availability is minimal.

Based on this information, the staff concludes that the proposed change on plant safety is small and, therefore, acceptable.

<u>TS 3.5.1</u> <u>ECCS - Operating</u>

The staff's evaluation for TS 3.5.1 separately groups ECCS systems (i.e., SRs 3.5.1.7 and 3.5.1.8) and the ADS (i.e., SRs 3.5.1.9, and 3.5.1.10).

<u>SR 3.5.1.7</u> Verify, with reactor pressure ≤ 180 psig, the HPCI pump can develop a flow rate of ≥ 5000 gpm against a system head corresponding to reactor pressure.

This SR ensures that the high pressure coolant injection system (HPCI) can perform its design function by developing the appropriate system flow.

<u>SR 3.5.1.8</u> Verify each ECCS injection/spray subsystem actuates on an actual or simulated automatic initiation signal

This SR includes the HPCI system, the low pressure coolant injection (LPCI) system, and the core spray (CS) system. The ECCS functional test ensures that a system initiation signal (actual or simulated) to the automatic initiation logic 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 automatic valves to their required positions.

HPCI, LPCI, and CS system availability during the operating cycle is assured by:

- The ECCS network has built-in redundancy so that no single-failure prevents starting of the ECCS.
- Extensions of the calibration cycle and logic system functional test frequency have previously been justified.
- Pumps and valves associated with these systems are tested in accordance with the Inservice Testing program. These tests will detect significant failures in the ECCS subsystems.
- SR 3.5.1.1, which is performed once every 31 days, ensures that the ECCS piping systems are filled with water to prevent water hammer affects.
- SR 3.5.1.2, which is performed once every 31 days, ensures that the ECCS valves are in the correct position.
- SR 3.5.1.3, which is performed once every 31 days, verifies the correct breaker alignment to the LPCI swing bus.

- The licensee’s review of surveillance test history for the ECCS system did not identify any test 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.

Based on this information, the staff concludes that the proposed change on plant safety is small and, therefore, acceptable.

SR 3.5.1.9 Verify the ADS actuates on an actual or simulated automatic initiation signal

This SR verifies that the ADS operates as designed when initiated either by an actual or simulated initiation signal and that the valve and solenoid are functioning properly.

SR 3.5.1.10 Verify each required ADS valve opens when manually actuated

This SR verifies that the ADS function operates as designed when manually actuated and also ensures the valve actuator and solenoids operate properly.

System availability during the operating cycle is assured by:

- The ADS has built-in redundancy so that no single-failure prevents the opening of the required number of ADS valves.
- The relief valves associated with the ADS are equipped with remote manual switches so that the entire system can be operated manually as well as automatically.
- The licensee’s review of surveillance test history for the ADS system did not identify any test 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.

Based on this information, the staff concludes that the proposed change on plant safety is small and, therefore, acceptable.

TS 3.5.2 ECCS - Shutdown

SR 3.5.2.5 Verify each required ECCS injection/spray subsystem actuates on an actual or simulated automatic initiation signal

This SR includes the HPCI system, the LPCI system, and the CS system. The ECCS functional test ensures that a system initiation signal (actual or simulated) to the automatic initiation logic 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 automatic valves to their required positions.

HPCI, LPCI, and CS system availability during the operating cycle is assured by:

- The ECCS network has built-in redundancy so that no single-failure prevents starting of the ECCS.
- Extensions of the calibration cycle and logic system functional test frequency have previously been justified.

- Pumps and valves associated with these systems are tested in accordance with the Inservice Testing program. These tests will detect significant failures in the ECCS subsystems.
- SR 3.5.2.1, which is performed once every 12 hours, verifies for each required ECCS subsystem that (a) suppression pool water level is ≥ 8.5 ft, or (b) contaminated condensate storage tank water level is ≥ 7.5 ft.
- SR 3.5.2.2, which is performed once every 31 days, verifies for each required ECCS subsystem that the piping is filled with water.
- The licensee's review of surveillance test history for the ECCS systems did not identify any test 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.

Based on this information, the staff concludes that the proposed change on plant safety is small and, therefore, acceptable.

<u>TS 3.5.3</u> <u>Reactor Core Isolation Cooling (RCIC) System</u>

<u>SR 3.5.3.4</u> Verify, with reactor pressure ≤ 180 psig, the RCIC pump can develop a flow rate of ≥ 400 gpm against a system head corresponding to reactor pressure.

<u>SR 3.5.3.5</u> Verify the RCIC System actuates on an actual or simulated automatic initiation signal.

SR 3.5.3.4, the RCIC low pressure flow test, ensures 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, the RCIC system functional test, ensures that a system initiation signal to the automatic initiation logic will initiate the system to respond as designed.

System availability during the operating cycle is assured by:

- SR 3.5.3.1, which is performed once every 31 days, verifies that the RCIC system piping is filled with water to avoid effects of water hammer.
- SR 3.5.3.2, which is performed once every 31 days, verifies valve position in the RCIC system.
- SR 3.5.3.3, which is performed once every 31 days, requires a RCIC pump test to verify pump capacity at system pressure.
- The RCIC system is a backup to the HPCI system. In addition, technical specifications prohibit extended plant operation with both the RCIC and HPCI systems inoperable. Finally, the safety analysis does not take credit for operation of the RCIC system.
- The licensee's review of surveillance test history did not identify any test 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.

Based on this information and the fact that the RCIC system is not relied upon in the safety analysis, the staff concludes that the proposed change on plant safety is small and, therefore, acceptable.

TS 3.6.1.1 Primary Containment

SR 3.6.1.1.2 Verify drywell-to-suppression chamber bypass leakage is less than or equal to the bypass leakage limit. However, during the first unit startup following bypass leakage testing performed in accordance with this SR, the acceptance criterion is ≤ 2% of the drywell-to suppression chamber bypass leakage limit.

The drywell-to-suppression chamber bypass leak test ensures that the boundary between the drywell airspace and the suppression chamber airspace is maintained to ensure the pressure suppression function is operable by limiting the amount of bypass steam leakage which would not be directed through the suppression pool water.

System availability during the operating cycle is assured by:

- The suppression chamber-to-drywell vacuum breakers are the only active mechanical devices in the boundary between the drywell air space and the suppression chamber. The vacuum breakers are verified to be in the closed position once every 14 days through performance of proposed SR 3.6.1.8.1. In addition, a functional test of each required vacuum breaker is performed once every 31 days through performance of SR 3.6.1.8.2. These tests ensure that the valves are functional and closed.
- The suppression chamber-to-drywell vacuum breakers include a passive design which does not appear to be subject to any time-based changes that would be affected by the change to a 24-month operating cycle.
- The licensee's review of surveillance test history did not identify any test 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.

Based on this information, the staff concludes that the proposed change on plant safety is small and, therefore, acceptable.

TS 3.6.1.3 Primary Containment Isolation Valves (PCIVs)

SR 3.6.1.3.7 Verify each automatic PCIV actuates to the isolation position on an actual or simulated isolation signal

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

System availability during the operating cycle is assured by:

- The PCIVs, including the actuating logic, are designed to be single-failure proof and, therefore, are highly reliable.
- Extension of the logic system functional test has been previously justified.
- During the operating cycle the PCIVs are either exercised (closed or open) or partially stroked (open or closed) in accordance with the Inservice Testing program or have justifications and reliefs to document why testing on an extended frequency is acceptable. The exercise or partial stroke testing of these PCIVs tests a significant portion of the PCIV's circuitry and will detect failures of this circuitry or failures with valve movement.

- The licensee's review of surveillance test history did not identify any test 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.

Based on this information, the staff concludes that the proposed change on plant safety is small and, therefore, acceptable.

<u>SR 3.6.1.3.8</u> Verify each reactor instrumentation line excess flow check valve (EFCV) actuates to the isolation position on an actual or simulated instrument line break signal.

This SR provides assurance that the instrumentation line EFCVs will perform as designed by actuating to their isolation position on an actual or simulated instrument line break signal. The 24-month surveillance frequency is based on the need to perform the SR under conditions that apply during a plant outage and the potential for an unplanned transient if the SR were performed with the reactor at power.

System availability during the operating cycle is assured by:

- The instrument lines are seismic category 1 and terminate in instruments that are seismic category 1. The instrumentation piping is composed of quarter-inch piping in the secondary containment that is sized to assure that a postulated failure would limit any offsite exposure to substantially below the standards of 10 CFR Part 100.
- Due to the mechanical nature of the check valves and instrumentation piping system, there are no definable drift components or any time-based conditions that could appreciably change during the operating cycle.
- The licensee's review of surveillance test history did not identify any test 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.

Based on this information, the staff concludes that the proposed change on plant safety is small and, therefore, acceptable.

<u>SR 3.6.1.3.9</u> Remove and test the explosive squib from each shear isolation valve of the traversing incore probe (TIP) system.

The SR requires that the explosive squib be removed and tested for the shear isolation valve of the TIP system. An in-place functional test is not possible with this design.

System availability during the operating cycle is assured by:

- The replacement charge for the explosive squib is from the same manufactured batch as the one fired or from another batch that has been certified by having one of the batch successfully fired.
- Administrative controls for the explosive charges, such as those that limit shelf life and operating life, are followed
- SR 3.6.1.3.4 verifies the circuit continuity of the TIP shear isolation valve explosive charge once every 31 days.

- The licensee's review of surveillance test history did not identify any test 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.

Based on this information, the staff concludes that the proposed change on plant safety is small and, therefore, acceptable.

<u>TS 3.6.1.7</u> <u>Reactor Building-to-Suppression Chamber Vacuum Breakers</u>

<u>SR 3.6.1.7.3</u> Verify the opening setpoint of each vacuum breaker is ≤ 0.5 psid.

This SR ensures that each reactor building-to-suppression chamber vacuum breaker check valve and vacuum breaker butterfly valve is capable of performing its safety function.

System availability during the operating cycle is assured by:

- The vacuum relief system design for the active components has built-in redundancy.
- SR 3.6.1.7.2 requires that each vacuum breaker be functionally tested once every 92 days by cycling each vacuum breaker check valve and butterfly valve. This surveillance ensures that the valves are capable of being cycled and return to the closed position.
- The licensee's review of surveillance test history did not identify any test 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.

Based on this information, the staff concludes that the proposed change on plant safety is small and, therefore, acceptable.

<u>TS 3.6.1.8</u> <u>Suppression Chamber-to-Drywell Vacuum Breaker</u>

<u>SR 3.6.1.8.3</u> Verify the opening setpoint of each required vacuum breaker is ≤ 0.5 psid.

SR 3.6.1.8.3 verifies the opening setpoint of each suppression chamber-to-drywell vacuum breaker is less than or equal to the specified differential pressure. The 24-month frequency is based on the need to perform this surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the surveillance were performed with the reactor at power.

System availability during the operating cycle is assured by:

- SR 3.6.1.8.1, which is performed once every 14 days, verifies that each vacuum breaker is closed.
- SR 3.6.1.8.2 requires that each vacuum breaker be functionally tested once every 31 days. This surveillance ensures that the valves are capable of being cycled and return to the closed position.
- The licensee's review of surveillance test history did not identify any test 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.

Based on this information, the staff concludes that the proposed change on plant safety is small and, therefore, acceptable.

<u>TS 3.6.4.1</u>　<u>Secondary Containment</u>

<u>SR 3.6.4.1.3</u>　Verify the secondary containment can be maintained $\geq$ 0.25 inch of vacuum water gauge for 1 hour using one standby gas treatment subsystem at a flow rate of $\leq$ 4000 cfm.

This SR ensures secondary containment boundary integrity by demonstrating that secondary containment vacuum can be maintained.

System availability during the operating cycle is assured by:

- Secondary containment is maintained at a negative pressure during normal plant operation. Any significant degradation to the secondary containment barrier would be detected through loss of vacuum.
- SR 3.6.4.1.1, which is performed once every 24 hours, verifies that the secondary containment vacuum is being maintained at $\geq$ 0.25 inch water gauge.
- Secondary containment structural integrity is maintained through administrative controls which ensure that no significant changes will be made to the secondary containment without proper evaluation.
- The licensee's review of surveillance test history did not identify any test 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.

Based on this information, the staff concludes that the proposed change on plant safety is small and, therefore, acceptable.

<u>TS 3.6.4.2</u>　<u>Secondary Containment Isolation Valves (SCIVs)</u>

<u>SR 3.6.4.2.3</u>　Verify each automatic SCIV actuates to the isolation position on an actual or simulated signal.

SR 3.6.4.2.3 ensures that each SCIV is capable of performing its intended function by actuating to the isolation position on an actual or simulated signal.

System availability during the operating cycle is assured by:

- SR 3.6.4.2.2, which is performed once every 92 days, verifies that the isolation time of each power operated, automatic SCIV is within limits. This surveillance cycles each automatic SCIV and would detect significant degradation affecting valve operation.
- The active components and power supplies of the SCIVs are designed to be single failure proof.
- The licensee's review of surveillance test history did not identify any test 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.

Based on this information, the staff concludes that the proposed change on plant safety is small and, therefore, acceptable.

<u>TS 3.6.4.3</u> <u>Standby Gas Treatment (SGT) System</u>

SGT system ensures that radioactive materials that leak from the primary containment into the secondary containment following an accident are filtered and adsorbed prior to being exhausted to the environment.

<u>SR 3.6.4.3.3</u> Verify each SGT subsystem actuates on an actual or simulated initiation signal

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

System availability during the operating cycle is assured by:

- There are two redundant and independent SGT subsystems such that a single-failure will not prevent system operation.
- SR 3.6.4.3.1 is a monthly surveillance that requires each SGT subsystem to be started and operated for ≥ 10 hours with heaters operating. This test verifies system operation and would identify significant system problems or failures.
- The licensee's review of surveillance test history did not identify any test 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.

Based on this information, the staff concludes that the proposed change on plant safety is small and, therefore, acceptable.

<u>TS 3.7.2</u> <u>Diesel Generator Cooling Water (DGCW) System</u>

<u>SR 3.7.2.2</u> Verify each required DGCW pump starts automatically when its associated diesel generator starts

The DGCW System functional test, SR 3.7.2.2 ensures that a system start signal from the associated diesel generator will cause the system to operate as designed, by automatically starting the DGCW pump.

System availability during the operating cycle is assured by:

- Each of the DGCW pumps are tested in accordance with the inservice testing program to ensure that each subsystem can provide the proper flow against a specified test
- pressure. This test will detect significant failures of the DGCW system to perform its intended function.
- SR 3.8.1.2, which requires monthly testing of the diesel generators, verifies operation of the DGCW system. This testing will detect significant failures affecting system operation.
- The licensee's review of surveillance test history did not identify any test 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.

Based on this information, the staff concludes that the proposed change on plant safety is small and, therefore, acceptable.

TS 3.7.4 Control Room Emergency Ventilation (CREV) System

The CREV system provides a radiologically controlled environment from which the plant can be safety operated following a LOCA. The CREV is designed to maintain the control room emergency zone environment for a 30-day continuous occupancy after a DBA without exceeding dose limits. The CREV System will pressurize the control room emergency zone to about 0.125 inches water gauge to minimize infiltration of air from adjacent zones.

SR 3.7.4.3 Verify the CREV System isolation dampers close on an actual or simulated initiation signal

SR 3.7.4.4 Verify the CREV System can maintain a positive pressure of ≥ 0.125 inches water gauge relative to the adjacent areas during the pressurization mode of operation at a flow rate of ≤ 2000 scfm.

SR 3.7.4.3 and 3.7.4.4 ensures that the CREV System is capable of automatic isolation and that the control room emergency zone boundary leakage is within the capacity of the CREV System by demonstrating that the control room emergency zone can be maintained at a positive pressure with respect to adjacent areas when in the emergency isolation/pressurization mode of operation.

System availability during the operating cycle is assured by:

- SR 3.7.4.1, which is conducted once every 31 days, requires the CREV System be operated for ≥ 10 hours with the heaters operating. These tests would detect significant failures affecting system operation.
- The actual or simulated isolation signal is equivalent to a logic system functional test. Extension of the logic system functional test has been previously justified.
- The control room emergency zone is maintained at a positive pressure during normal operation. Therefore, any substantial degradation of the boundary will be evident and repairs can be accomplished in a timely manner.
- The licensee’s review of surveillance test history did not identify any test 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.

Based on this information, the staff concludes that the proposed change on plant safety is small and, therefore, acceptable.

TS 3.7.5 Control Room Emergency Ventilation Air Conditioning (AC) System

The CREV AC system provides a suitable environment for continuous personnel occupancy and ensures the operability of control room equipment and instruments under normal and accident conditions.

SR 3.7.5.1 Verify the Control Room Emergency Ventilation AC System has the capability to remove the assumed heat load.

The SR verifies that the CREV AC System has the capability to remove the assumed heat load. The CREV AC System auto-starts on control room temperature when the CREV System is operating. Both the CREV and the CREV AC are normally maintained in standby and are operated only for required surveillances.

System availability during the operating cycle is assured by:

- SR 3.7.4.1, which is conducted once every 31 days, requires the CREV System be operated for ≥ 10 hours with the heaters operating. The licensee has confirmed that the CREV AC is verified to be operational during the monthly performance of SR 3.7.4.1. These tests would detect significant failures affecting system operation.
- The licensee's review of the surveillance test history did not identify any test 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.

Based on this information, the staff concludes that the proposed change on plant safety is small and, therefore, acceptable.

TS 3.8.1 AC Sources - Operating

The unit Class 1E AC Electrical Power Distribution Systems AC sources consist of the offsite power sources, and the onsite standby power sources (diesel generators (DGs) 1, 2, and ½).

The Class 1E unit AC distribution system is, for the most part, divided into redundant load groups (Division 1 and 2), so loss of any one group does not prevent the minimum safety functions from being performed. The exception is that the opposite unit's AC Electrical Power Distribution System powers shared loads (i.e., standby gas treatment subsystem, CREV System (Unit 2 only), and CREV AC System (Unit 2 only)). Although shared by both units, the CREV System and the CREV AC System are single train systems that are powered only from a single Unit 1 motor control center. Each unit's load group has connections to physically independent offsite power sources and a single DG.

The staff's evaluation for SRs 3.8.1.9 through and including 3.8.1.19 have been grouped together.

SR 3.8.1.9 Verify manual transfer of unit power supply from the normal offsite circuit to the alternate offsite circuit.

This SR includes the transfer of each unit auxiliary transformer to the associated unit reserve auxiliary transformer and a verification of the cross tie between the unit's 4160 V ESS buses.

SR 3.8.1.10 Verify each DG rejects a load greater than or equal to its associated single largest post-accident load, and:

1. following load rejection, the frequency is ≤ 66.73 Hz;
2. Within 3 seconds following load rejection, the voltage is ≥ 3952 V and ≤ 4368 V; and
3. Within 4 seconds following load rejection, the frequency is ≥ 58.8 Hz and ≤ 61.2 Hz.

This SR demonstrates the DG load response characteristics and capability to reject the largest single load without exceeding predetermined voltage and frequency and while maintaining a specified margin to the overspeed trip.

SR 3.8.1.11 Verify each DG does not trip and voltage is maintained ≤ 5000 V during and following a load rejection of ≥ 2340 kW and ≤ 2600 kW.

This SR ensures proper engine generator load response under the simulated test conditions. This test simulates the loss of the total connected load that the DG experiences following a full load rejection and verifies that the DG does not trip upon loss of the load.

SR 3.8.1.12 Verify on an actual or simulated loss of offsite power signal:

(A) De-energization of emergency buses;
(B) Load shedding from emergency buses; and
(C) DG auto-starts from standby condition and:
   1. energizes permanently connected loads in ≤ 10 seconds,
   2. maintains steady state voltage ≥ 3952 V and ≤ 4368 V,
   3. maintains steady state frequency ≥ 58.8 Hz and ≤ 61.2 Hz, and
   4. supplies permanently connected and auto-connected loads for ≥ 5 minutes

This SR verifies all actions encountered from the loss of offsite power, including shedding of the nonessential loads and energization of the emergency buses and respective loads from the DG. It further demonstrates the capability of the DG to automatically achieve the required voltage and frequency within the specified time.

SR 3.8.1.13 Verify on an actual or simulated Emergency Core Cooling System (ECCS) initiation signal each DG auto-starts from standby condition and:

A. In ≤ 10 seconds after auto-start achieves voltage ≥ 3952 V and frequency ≥ 58.8 Hz;
B. Achieves steady state voltage ≥ 3952 V and ≤ 4368 V and frequency ≥ 58.8 Hz and ≤ 61.2 Hz;
C. Operates for ≥ 5 minutes;
D. Permanently connected loads remain energized from the offsite power system; and
E. Emergency loads are auto-connected to the offsite power system.

This SR demonstrates that the DG automatically starts and achieves the required voltage and frequency within the specified time (10 seconds) from the design basis actuation signal (LOCA signal). In addition, the DG is required to maintain proper voltage and frequency limits after steady state is achieved.

SR 3.8.1.14 Verify each DG's automatic trips are bypassed on actual or simulated loss of voltage signal on the emergency bus concurrent with an actual or simulated ECCS initiation signal except:
A. Engine overspeed; and
B. Generator differential current.

This SR demonstrates that each DG non-critical protective trip is bypassed on an actual or simulated ECCS initiation signal and that critical protective functions trip the DG.

SR 3.8.1.15 Verify each DG operating within the power factor limit operates for ≥ 24 hours:
A. For ≥ 2 hours loaded ≥ 2730 kW and ≤ 2860 kW; and
B. For the remaining hours of the test loaded ≥ 2340 kW and ≤ 2600 kW.

This SR demonstrates that each DG can start and run continuously at full load capability for an interval of not less than 24 hours, 22 hours of which is at a load equivalent to 90% to 100% of the continuous rating of the DG and 2 hours of which is at a load equivalent to 105% to 110% of the continuous rating of the DG.

SR 3.8.1.16 Verify each DG starts and achieves:
A. In ≤ 10 seconds, voltage ≥ 3952 V and frequency ≥ 58.8 Hz; and
B. Steady state voltage ≥ 3952 V and ≤ 4368 V and frequency ≥ 58.8 Hz and ≤ 61.2 Hz.

This SR demonstrates that the diesel engine can restart from a hot condition, such as subsequent to shutdown from normal surveillances, and achieve the required voltage and frequency within 10 seconds.

SR 3.8.1.17 Verify each DG:
A. Synchronizes with offsite power source while loaded with emergency loads upon a simulated restoration of offsite power;
B. Transfers loads to offsite power source; and
C. Returns to ready-to-load operation.

This SR demonstrates that the manual synchronization and automatic load transfer from the DG to the offsite source can be made and that the DG can be returned to ready to load status when offsite power is restored. It also ensures that the auto-start logic is reset to allow the DG to reload if a subsequent loss of offsite power occurs.

SR 3.8.1.18 Verify interval between each sequenced load block is within ± 10% of design interval for each load sequence time delay relay.

This SR verifies that the sequence time is within ±10 percent of the design for each load sequence timer. Under accident conditions, loads are sequentially connected to the bus by the time delay relays. The time delay relays control the permissive and starting signals to motor breakers to prevent overloading of the bus power supply due to high motor starting currents.

SR 3.8.1.19 Verify, on an actual or simulated loss of offsite power signal in conjunction with an actual or simulated ECCS initiation signal:

A. De-energization of emergency buses;
B. Load shedding from emergency buses; and
C. DG auto-starts from standby condition and:
   1. energizes permanently connected loads in $\leq$ 10 seconds,
   2. energizes auto-connected emergency loads,
   3. maintains steady state voltage $\geq$ 3952 V and $\leq$ 4368 V,
   4. maintains steady state frequency $\geq$ 58.8 Hz and $\leq$ 61.2 Hz, and
   5. supplies permanently connected and auto-connected emergency loads for $\geq$ 5 minutes.

This SR demonstrates operation of each DG during a loss-of-offsite power test signal coincident with an ECCS initiation.

AC Source system availability during the operating cycle is assured by:

- SR 3.8.1.2 requires that each DG be tested for operability once every 31 days. This testing, which is not being changed, will provide prompt identification of any substantial DG degradation or failure.
- SR 3.8.1.8 requires that each DG be fast start tested once every 184 days. This test, which is not being changed, will provide prompt identification of any substantial DG degradation or failure.
- DGs are not operated outside of the monthly operability tests in order to minimize wear related degradation.
- DG attributes subject to degradation due to aging, such as fuel oil quality, are subject to its requirements for replenishment and testing.
- An evaluation of known failures did not identify any time-based elements that would invalidate the conclusion that the increased operating cycle will have a small, if any, impact on system reliability.
- The licensee's review of the surveillance test history did not identify any test 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.

Based on this information, the staff concludes that the proposed changes for SRs 3.8.1.9 through 3.8.1.19 on plant safety are small and, therefore, acceptable.

TS 3.8.4 DC Sources - Operating

The DC electrical power systems provide the AC emergency power system with control power. They also provide both motive and control power to selected safety-related equipment. DC subsystems provide DC electrical power to inverters, which in turn power the AC essential service buses.

The 250 VDC power sources provide motive power to selected safety related larger DC loads such as DC motor-driven pumps and valves. The Division 1 and Division 2 125 VDC power sources provide both motive and control power to selected safety-related equipment, as well as

circuit breaker control power for the non-safety related 4160 switchgear, and all 480 V load centers.

The staff's evaluation for SRs 3.8.4.3 through and including 3.8.4.7 have been grouped together.

SR 3.8.4.3 Verify battery cells, cell plates and racks show no visual indication of physical damage or abnormal deterioration that could degrade battery performance.

SR 3.8.4.4 Remove visible corrosion, and verify battery cell to cell and terminal connections are coated with anti-corrosion material

SR 3.8.4.5 Verify battery connection resistance is
$\leq$ 1.5E-4 ohm for inter-cell connections, and
$\leq$ 1.5E-4 ohm for terminal connections.

SR 3.8.4.6 Verify each required 125 V battery charger supplies:
A. 250 amps at $\geq$ 250 VDC for $\geq$ 4 hours for the 250 VDC subsystems; and
B. 200 amps at $\geq$ 125 VDC for $\geq$ 4 hours for the 150 VDC subsystems.

SR 3.8.4.7 Verify battery capacity is adequate to supply, and maintain in OPERABLE status, the required emergency loads for the design duty cycle when subjected to a battery service test.

System availability during the operating cycle is assured by:

- The design, in conjunction with the technical specification requirements which limit the extent and duration of inoperable DC sources, provides substantial redundancy in DC sources.
- Battery parameters such as float voltage, electrolyte level, and specific gravity are monitored during the operating cycle to verify battery operability and will provide prompt identification of any substantial battery or battery charger degradation or failure. As an example, SR 3.8.4.1, which is performed once every 7 days, verifies that battery terminal voltage on float charge is within limits.
- Batteries are not discharged except for the performance of the operating cycle test demonstrations of operability. Therefore, there is minimal risk of age-related degradation.
- SR 3.8.4.2, which is performed once every 92 days, requires monitoring for visible corrosion at battery terminals and connectors. These examinations will provide prompt identification of any substantial battery
- The licensee's review of surveillance test history did not identify any test 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.

Based on this information, the staff concludes that the proposed change on plant safety is small and, therefore, acceptable.

Administrative Controls

TS 5.5.2 Primary Coolant Sources Outside Containment

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable.

The program includes integrated leak test requirements for each system. The frequency of performing these tests is being changed from once per operating cycle to once every 24 months.

System availability during the operating cycle is assured by:

- Most portions of the subject systems included in this program are visually walked down, while the plant is operating, during plant testing, and/or operator/system engineer walkdowns. In addition, housekeeping/safety walkdowns also serve to detect any gross leakage. If leakage is observed from these walkdowns, corrective actions will be taken for repairs.
- Plant radiological surveys will identify any potential sources of leakage. System walkdowns and surveys provide monitoring of the systems at a greater frequency than once per refueling cycle and would identify any significant system degradation or failures.
- The licensee's review of historical maintenance and surveillance data demonstrates that there is no adverse trend that would invalidate the conclusion that the impact on system availability, if any, is minimal from the proposed change.

Based on this information, the staff concludes that the proposed change on plant safety is small and, therefore, acceptable.

TS 5.5.7 Ventilation Filter Testing Program (VFTP)

The VFTP shall establish the required testing of the engineered safety feature filter ventilation systems. The frequency for filter testing as described in the VFTP for the standby gas treatment (SGT) and control room emergency ventilation (CREV) systems is being changed from 18 to 24 month intervals.

System availability during the operating cycle is assured by:

- Both the SGT and CREV Systems are normally in standby. Therefore, the systems are not subject to degradation due to plant operation.
- Additional system testing is required if the potential for degradation occurs (i.e., following any structural maintenance on the HEPA filter or charcoal adsorber housings, following painting, fire, or chemical release in any ventilation zone communicating with the systems).
- The licensee's review of historical maintenance and surveillance data demonstrates that there are no failures that would invalidate the conclusion that the impact on system availability, if any, is minimal from the proposed changes.

Based on this information, the staff concludes that the impact of the proposed change on plant safety is small and the propose change is, therefore, acceptable.

<u>Additional TS Changes and Beyond-Scope Items</u>

<<<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 Quad Cities 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-1433 with appropriate modifications for plant-specific considerations. The NRC staff further concludes that the Quad Cities 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 Quad Cities 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.

Principal Contributors:	C. Schulten	S. Bailey	A. Chu
	E. Tomlinson	R. Tjader	J. Foster
	H. Garg	R. Giardina	T. Liu
	C. Harbuck	D. Pickett	Z. Abdulahi
	S. Saba	A. Cubbage	M. Waterman
	D. LuRie	T. Dunning	

Date:

CTS Discussion of Change Tables