ATTACHMENT D

Marked-Up Draft Safety Evaluation for Quad Cities Nuclear Power Station

(Non-Proprietary)

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

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1.0 OVERVIEW

1.1 Introduction

By letter dated December 27, 2000 (Reference 1), Commonwealth Edison Company (ComEd), requested amendments to Facility Operating Licenses DPR-29 and DPR-30 for the Quad Cities Nuclear Power Station, Units 1 and 2 (QCNPS). The proposed amendments would allow an increase in the maximum authorized operating power level from 2511 megawatts thermal (MWt) to 2957 MWt. These proposed changes would represent an increase of approximately 17.8 percent above the current rated thermal power (RTP) and is considered an extended power uprate (EPU). 2511 MWt is the original rated thermal power (ORTP) for QCNPS. These amendments would change the Technical Specifications (TS) appended to the operating licenses to allow plant operation at 2957 MWt. These amendments would also modify license conditions and request additional license conditions to support the power uprate.

The original application was submitted by ComEd, which merged to form Exelon Generation Company, LLC (EGC, the licensee). By letter dated February 7, 2001, EGC assumed responsibility for all pending Nuclear Regulatory Commission (NRC) actions requested by ComEd. EGC later supplemented the original license amendment application by letters dated February 12; April 6 and 13; May 3, 18, and 29; June 5, 7, and 15; July 6 and 23; August 7, 8, 9, 13 (two letters), 14 (two letters), 29, and 31 (two letters); and September 5 (two letters), x, y, z, 2001.

1.2 Background

The QCNPS safety analysis of the proposed EPU was provided in Attachments A and E of the licensee's December 27, 2000, submittal. Attachment E of the submittal is the licensee's Safety Analysis Report, General Electric (GE) Nuclear Energy Licensing Topical Report (LTR) NEDC-32961P (Reference 2). Revision 2 of the Safety Analysis Report (Reference 28), submitted

August 31, 2001, changed some proprietary designations and updated the text to reflect information provided to NRC in preceding correspondence or to revise information that does not significantly affect the conclusions of the original submittal. The licensee's submittal contained plant-specific information consistent with the scope and content of the NRC-approved GE LTR NEDC-32424P-A (Proprietary), "Generic Guidelines for General Electric Boiling Water Reactor (BWR) Extended EPU," February 1999 (Reference 3), known as ELTR1, which included the staff's position paper (Reference 4). For some items, the licensee referenced the analyses and evaluations in the NRC-approved GE LTR NEDC-32523P-A (Proprietary), "Generic Evaluation of General Electric Boiling Water Reactor Extended EPU," February 2000 (Reference 5), known as ELTR2. The ELTR2 generic evaluations are based on (a) an increase in the thermal power up to 20 percent above the unit's ORTP, (b) reactor pressure vessel dome operating pressure up to 1095 psia, (c) reactor system temperature up to 556 °F, and (d) a steam and feedwater flow increase of about 24 percent. The licensee stated that the generic system and equipment performance and the generic transient and accident analyses presented in ELTR1 and ELTR2 are applicable to the QCNPS EPU.

As part of the EPU review process, the staff visited the GE facility in Wilmington, North Carolina, from June 18 to 22, 2001, to audit both the Global Nuclear Fuel (GNF) adherence to the NRC-approved analytical methods for performing the EPU safety analyses, and the QCNPS-specific analyses in support of the EPU. The audit findings and their resolutions are discussed in Section 2.6 of this safety evaluation (SE).

1.3 Approach

To accomplish the EPU, the licensee proposed to increase the plant's operating domain by implementing the maximum extended load line limit analysis (MELLLA) power/flow map and to increase core flow along the resulting flow control line extension. The licensee also proposed to partially implement the Average Power Range Monitor (APRM)/Rod Block Monitor (RBM) TS (ARTS) power and flow dependent limits. The QCNPS proposed EPU will not increase the operating pressure or the current licensed core flow. EPU operation will not increase reactor vessel dome pressure because the plant will have (after modifications to power generation equipment) sufficient pressure control and turbine flow capabilities to control the pressure at the turbine inlet. Higher steam flow will be generated though a more uniform (flattened) core power distribution and an increase in the corresponding feedwater flow to match the higher steam flow. The licensee also plans to revise the loading pattern of the core, use larger batch sizes, and introduce GE-14 fuel.

1.4 Staff Evaluation

The NRC staff's review of the QCNPS EPU amendment request used applicable rules, regulatory guides, Standard Review Plan (SRP, Reference 7), and NRC staff positions on the topics being evaluated. Additionally, the staff evaluated the QCNPS submittal for compliance with the generic boiling-water reactor (BWR) EPU program as defined in ELTR1 and ELTR2. ELTR1 and ELTR2 have previously been accepted by NRC as acceptable guidelines for EPU applications (References 4 and 6). The staff also used the 1998 Safety Evaluation (SE) for the Monticello Nuclear Generating Plant EPU as a guide for scope and depth of review.

Table 1-3 of the QCNPS Safety Analysis Report (Reference 2) lists the NSSS computer codes used in the EPU safety analyses. The table states that all the applicable codes have been reviewed and approved by the NRC, except for the BILBO code, which is not a safety analysis code, and use of the TASC code for application to ECCS-LOCA analyses. The licensee stated that TASC is an improved version of the NRC-approved SCAT code, which has the capability to model advanced fuel features (partial length rods and new critical power correlation). The code has been accepted for transient analyses and TASC is currently under staff review for LOCA analysis. (The staff is currently completing its review of TASC.) Based on the status of the review, we believe that the use of the TASC code would have an insignificant effect on the consequence of the relevant accident analyses. Therefore, we believe that the analysis results on which it is based are valid.

The QCNPS EPU transition reload cores contain both the existing Siemens Power Corporation ATRIUM-9B (9x9) fuel coresident with fresh GNF GE-14 (10x10) fuel, while the equilibrium EPU core will consist exclusively of GE-14 fuel. The EPU safety analyses and the cycle-specific reload analyses were performed in accordance with NRC-approved GE analytical methodologies described in the latest version of NEDE-24011-P-A-10-US, "General Electric Standard Application for Reactor Fuel (GESTAR II)" (Reference 35). The licensing topical reports specifying the codes and methodologies used for performing the safety analyses are documented in Section 5 of the QCNPS TS. The limiting anticipated operational occurrences (AOO) and accident analyses are reanalyzed or confirmed to be valid for every reload and the nonlimiting safety analyses of record are documented in Chapter 15 of the QCNPS updated final safety analysis report (UFSAR). Limiting transient or accident analyses are generally defined as analyses of events that could potentially affect the core operating and safety limits that ensure the safe operation of the plant.

Detailed discussions of individual review topics follow. Since the licensee's submittal and Safety Analysis Report follow the format of the previously reviewed generic ELTRs, the evaluations below are presented in (mostly) the same format and section numbering scheme.

2.0 REACTOR CORE AND FUEL PERFORMANCE

The core thermal-hydraulic design and fuel performance characteristics are evaluated for each fuel cycle. The following sections address the effect of the EPU on fuel design performance, thermal limits, power/flow map, and reactor stability.

2.1 Fuel Design and Operation

Fuel bundles are designed to ensure that: (a) the fuel bundles are not damaged during normal steady state operation and anticipated operational occurrences (AOOs); (b) any damage to the fuel bundles would not be so severe as to prevent control rod insertion when required; (c) the number of fuel rod failures during accidents is not underestimated during accidents; and (d) the coolability of the core is always maintained. For each fuel vendor, use of NRC-approved fuel design acceptance criteria and analysis methodologies assure that the fuel bundles perform in a manner that is consistent with the objectives of Sections 4.2 and 4.3 of the standard review plan (Reference 7) and the applicable general design criteria (GDC) of 10 CFR Part 50, Appendix A. The fuel vendors perform thermal-mechanical, thermal-hydraulic, neutronic, and material analyses to ensure that the fuel system design can meet the fuel design limits during steady state, AOO, or accident conditions.

The licensee's Safety Analysis Report (Reference 2) states that the requested EPU would increase the average power density proportionally to the power increase, but the increased power density would be within the power density of existing GE-supplied BWRs. The increased operating power would affect the operating flexibility and the reactivity characteristics. The EPU is achieved by design changes to the core loading pattern, by using larger reload batch sizes, and by introducing new fuel designs (GE-14).

The licensee's Safety Analysis Report states that, for operation at the currently licensed power or at the proposed EPU, the fuel and core design limits will continue to be met by varying the fuel enrichment and burnable poisons, supplemented by control rod pattern management. The reload core design will flatten the radial power distribution while limiting the absolute power in individual fuel bundles to currently allowable values. NRC-approved core design methods are used to analyze the core performance at the proposed EPU operation.

The EPU fuel cycle calculations were performed using a representative "bounding unit" equilibrium GE-14 core design to demonstrate the feasibility of operation at the higher thermal power, and with the MELLLA rod line while maintaining the fuel design limits. Limits on the fuel rod linear heat generation rates (LHGR) ensure compliance with the fuel mechanical design bases. The thermal-hydraulic design and the operating limits ensure an acceptably low probability of boiling-transition-induced fuel cladding failure in the core in the event of an AOO. The licensee stated that the EPU fuel cycle design calculations demonstrated that these fuel design limits would be maintained and the subsequent reload core designs at the EPU power level will take into account these limits to ensure acceptable differences between the licensing limits and their corresponding operating values.

[redacted]

2.2 <u>Thermal Limits Assessment</u>

General Design Criterion (GDC) 10 of 10 CFR Part 50, Appendix A, requires that the reactor core and the associated control and instrumentation systems be designed with appropriate margin to ensure that the specified acceptable fuel design limits (SAFDLs) are not exceeded during normal operation, including AOOs. Operating limits are established to assure that regulatory and/or safety limits are not exceeded for a range of postulated events (transients and accidents).

The effects of the higher MELLLA rod line and power on the thermal limits are discussed in the following sections. Thermal limits management with ARTS power and flow dependent limits is discussed in Section 9.2 of the licensee's Safety Analysis Report.

2.2.1 Minimum Critical Power Ratio (MCPR) Operating Limit

The safety limit minimum critical power ratio (SLMCPR) ensures that 99.9 percent of the fuel rods are protected from boiling transition during steady state operation. The operating limit minimum critical power ratio (OLMCPR) assures that the SLMCPR will not be exceeded as result of an AOO.

Table 9-1 of the licensee's Safety Analysis Report provides plant parameters used for the current rated power, and for the representative equilibrium GE-14 core at the QCNPS EPU power level of 2957 MWt. Table 9-2 presents the EPU transient analyses results based on the calculated SLMCPR of 1.09, which is slightly lower than the value for the SLMCPR (1.10) for the current cycle. Note that the SLMCPR is established or confirmed every reload, based on the actual core configuration and operating conditions.

The licensee analyzed the limiting transients for operation at the EPU operating domain, based on the GE-14 equilibrium core. Table 9-2 of the licensee's Safety Analysis Report provides the operating limit (OL) MCPR for the limiting transients. The licensee stated that the required OLMCPR is not expected to change significantly from the results shown in Table 3-1 of ELTR1 and Figure 5-3 of ELTR2.

During a previous EPU audit conducted in March 2001, the staff reviewed the experimental database used for the development of the GEXL14 critical power ratio (CPR) correlation for the GE-14 (10x10) fuel lattice design. The QCNPS EPU reload cores introduce GE-14 fuel and the resolution of the audit findings ensures the CPR correlations used to determine the MCPR are properly developed and experimentally benchmarked.

The summary of the staff's finding and the GNF corrective action to resolve the findings are discussed in Section 2.6 of this SE. The ARTS power and flow dependent MCPR limits are discussed in Section 9.2.

2.2.2 Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) and Maximum LHGR operating limits

The MAPLHGR operating limit is based on the most limiting LOCA and ensures compliance with the ECCS acceptance criteria in 10 CFR 50.46. For every new fuel type, the fuel vendors perform LOCA analyses to confirm compliance with the LOCA acceptance criteria, and for every reload licensees confirm that the MAPLHGR operating limit for each reload fuel bundle design remains applicable.

The licensee performed the LOCA evaluation as discussed in Section 4.3, based on the representative GE-14 equilibrium core, operating at the EPU power level. The licensee stated that the LOCA analysis shows no change in the MAPLHGR or the LHGR limits for normal two recirculation loop operation (TLO) and for single recirculation loop operation (SLO). The LOCA analyses are required to account for the increased thermal power. The licensee revised the MAPLHGR multipliers to account for SLO in the higher MELLLA region. The licensee stated that the LHGR limits are fuel dependent and apply regardless of the power level, but added that changes to the GNF advanced core methods will ~~require~~ allow the MAPLHGR and the LHGR limits to be monitored independently. The licensee stated that separate MAPLHGR and maximum linear heat generation rates (LHGR) will be maintained for each GNF fuel type as described in Section 5.7.2.2 of ELTR1. ARTS power and flow dependent LHGR limits are discussed in Section 9.2.

The licensee evaluated the plant's response to operation at the higher MELLLA rod line and power level based on representative bounding equilibrium GE-14 core. Although the initial transition reload cycle specific analysis will not be based on the final EPU conditions, the final transition cycle reload analysis would be based on the MELLLA/ EPU operating conditions and cycle-specific core design. The flatter radial power distribution will result in more fuel bundles operating at or near the boiling transition and this could result in slight increase in the SLMCPR. However, any SLMCPR change would constitute a TS change and the licensee would submit an amendment request for NRC review. As stated above, the audit team reviewed the GE-14 CPR correlation database, used to develop the GEXL14 CPR correlation for GE-14 fuel, which affects the accuracy of the TS SLMCPR calculations. The licensee will specify the other thermal limits in the cycle-specific core operating limit report (COLR), as required in Section 5 of the TS. Also, the licensee cannot exceed the NRC-approved burnup limits. The staff concludes that the licensee has appropriately considered the effects of the MELLLA/EPU operation on the fuel design performance, and the staff concludes that the thermal limits are acceptable.

2.3 Reactivity Characteristics

The licensee stated that for a given core design, operation at higher power could reduce the hot excess reactivity, typically by about 0.2 to 0.3 percent delta K for each 5 percent power increase. The loss of reactivity is not expected to affect the ability to manage the power distribution needed to meet the target power through the cycle. The lower hot excess reactivity can result in an earlier all-rod-out condition during the operating cycle, however, through reload fuel cycle-specific core analyses, the core can be designed with sufficient excess reactivity to maintain the fuel cycle length. [redacted] Sidebarred in the PUSAR The licensee added that the reload core analysis will ensure that the minimum shutdown margin requirements are met for each core design and that the current design and TS cold shutdown margin will be met. Since the licensee will continue to confirm that the TS cold shutdown requirements will be met for each reload core operation, the staff finds this acceptable.

2.3.1 Power/Flow Operating Map

To achieve the 17.8 percent increase from the current rated power (CRP), the licensee proposes to operate at the MELLLA rod line. The EPU operating domain will be defined by: (a) the MELLLA upper boundary line extended up to the EPU rated thermal power, (b) the maximum EPU power level corresponding to 117.8 percent of the CRP, and (c) the existing 100 percent core flow line continued up to the EPU power. The previously analyzed core flow range will be extended so that the RTP will correspond to the EPU power level and the maximum core flow will not be increased. The submittal contains the proposed EPU operating domain power/flow map as Figure 2-1.

The MELLLA upper boundary line replaces the current extended load limit line analysis (ELLLA) upper boundary for single recirculation loop operation. The licensee stated that the maximum power statepoint for the SLO corresponding to the MELLLA upper boundary and recirculation pump speed of 102.5 percent would be 70.2 percent of the EPU RTP (2076 MWt). The associated SLO core flow would then be 55.1 percent core flow (54 Mlbm/hr). The licensee would perform the EPU SLO safety analysis based on the MELLLA statepoint for SLO. The licensee stated that EPU operation at the higher rod line would also require rescaling of the associated protection system setpoints, which are discussed in Section 5.2.5 of this evaluation.

2.4 Stability

QCNPS is currently operating under the requirements of reactor stability Interim Corrective Actions (ICA) and is in the process of implementing long-term stability solution (LTS) Option III hardware changes, but has not yet armed the system. The long-term stability solutions for BWRs are discussed in licensing topical report, NEDO-32465-A, "BWR Owners' Group Stability Solutions Licensing Basis Methodology and Reload Application," (Reference 37).

If the Option III system is declared inoperable, the ICA procedures (Reference 38) are initiated to restrict plant operation in the high power, low core flow region of the BWR power/flow operating map. The procedures contain specific operator actions in response to reactor operation in the defined restricted regions. This generic interim solution is approved to cover all operations and accident scenarios. ICA stability boundaries remain the same in terms of absolute power and core flow for extended power uprate. The power levels, reported as a percentage of rated power, are rescaled to the uprated power.

However, this does not prevent the utility from validating the ICA region boundaries using the ODYSY code. The ODYSY stability application licensing topical report (NEDC-32992P) has been reviewed and accepted by the staff in an April 20, 2001, safety evaluation report (Reference 45). The decay ratio adder of 0.15 will not be applied as this represents stability validation similar to the Enhanced Option I-A.

~~The current ICA solution (Reference 38) uses an administratively controlled exclusion region. The license uses the ODYSY code to establish the exclusion region, which is defined by a curved line that provides a constant margin to the occurrence of anticipated reactor instability. Decay ratios are calculated based on ODYSY stability criteria. The licensee stated that the ICA exclusion region boundary covers those areas of the operating domain where the core decay ratio is 0.8 or greater. The ODYSY code calculates a best estimate core and channel decay ratio and adds 0.15 to the core decay ratio for added conservatism. In addition, the decay ratios are calculated for various statepoints on the power/flow map to determine the intersection of the exclusion region boundary with the natural circulation line and with the MELLLA boundary. The ODYSY stability application licensing topical report (NEDC-32992P) has been reviewed and accepted by the staff in an April 20, 2001, safety evaluation report (Reference 45).~~

~~The licensee stated that the exclusion region is core and fuel dependent and is also affected by the rated core power and the corresponding operating conditions. The exclusion region was calculated for the EPU fuel cycle conditions and the applicability of the exclusion region would be evaluated for each subsequent fuel cycle until the LTS Option III is fully implemented.~~

Maintaining adequate SLMCPR protection is assured using the OPRM scram available in

Option III. The application of the Delta critical power ratio (CPR) over Initial minimum critical power ratio (IMCPR) Versus Oscillation Magnitude (OM) [DIVOM] curve was audited in the June 2001 visit. The DIVOM curves represent normalized curves of CPR performance versus hot bundle oscillation. GNF has generated two generic curves for core wide and regional mode oscillations which are intended to be used in the stability licensing methodology during the reload analysis. During a prior EPU audit, the staff reviewed internal GENE documentation questioning the applicability of the generic DIVOM curves for EPU operation using GE-14 fuel.

The June audit of GE Wilmington covered the pre-EPU and EPU operation. The staff reviewed the design record files (DRF) for the EPU equilibrium core and for the first transition reload cycle stability calculations. The staff review further questioned whether the generic DIVOM curve for ~~the core wide mode and~~ the regional mode oscillation specified in NEDO-32465-A (Reference 37) can be met for the EPU/MELLLA operation. The licensee stated that the Option III will not be used until resolution of use of the generic DIVOM curve for the EPU operating condition is resolved, as discussed in Section 2.6 of this SER.

On June 29, 2001, GE Nuclear Energy (GENE) submitted a 10 CFR Part 21 notification regarding the use of the DIVOM curve. GENE reported that stability reload licensing calculations using the generic DIVOM curve may be nonconservative for plants using the stability detect and suppress trip systems. For Option III stability solution, the trip system setpoints, which ensure adequate MCPR safety limit protection from regional mode instability, may be nonconservative. ~~For Options II and I-D, the Part 21 report stated that flow-biased APRM flux scram may not provide adequate MCPR safety limit margin.~~ This report stated that there is a deficiency for high peak bundle power-to-flow ratios for the regional mode DIVOM curve. ~~and for high core averaged power-to-flow ratios for the core wide mode DIVOM curve.~~ GENE provided a figure of merit for ~~each~~ the generic regional DIVOM curve, which licensees could use to determine the applicability of the existing generic DIVOM curve for their units.

[* *insert QCNPS response to the Part 21 and further staff evaluation here*]**

2.5 Reactivity Control

2.5.1 Control Rod Drive System

The control rod drive (CRD) system controls gross changes in core reactivity by positioning neutron-absorbing control rods within the reactor. The CRD system is also required to scram the reactor by rapidly inserting withdrawn rods into the core. The scram, rod insertion, and withdrawal functions of the CRD system depend on the operating reactor pressure and the pressure difference between the CRD system hydraulic control unit (HCU) and the reactor vessel bottom head pressure.

The licensee stated that since there is no increase in the reactor operating pressure, the CRD scram performance and compliance with the current TS scram requirements are not affected by the operation at the EPU power level. The CRD system was generically evaluated in Section 5.6.3 and J.2.3.3 of ELTR1 and Section 4.4 of Supplement 1 to ELTR2. The licensee stated that since the generic evaluation concluded that the CRD systems for BWR/2-6 designs are acceptable for EPU as high as 20 percent above the original rated power, no additional plant-specific calculations are required beyond confirmatory evaluation. The licensee performed confirmatory evaluations of the performance of the CRD system at the EPU conditions based on a reactor dome pressure of 1005 psig with an additional 35 psid added to account for the static head of water in the vessel.

The licensee stated that for CRD insertion and withdrawal, the required minimum pressure between the HCU and the vessel bottom head is 250 psid. The licensee evaluated the CRD pump capability and determined that the CRD pumps have sufficient capacity to provide the required pressure difference for operation at the EPU conditions. The licensee also evaluated the required CRD cooling and drive flows for EPU operation and stated that the cooling and drive flows are assured by the automatic operation of the CRD system flow control valve, which would compensate for any changes in the reactor pressure. The licensee determined that the operation of the QCNPS CRD system is consistent with the generic evaluations in ELTR1 and ELTR2, and that the CRD system is, therefore, capable of performing its design functions of rapid rod insertion (scram) and rod positioning (insertion/withdrawal) function.

During scrams at low reactor pressure, the accumulator provides the pressure for the scram. However, at higher power, such as during isolation events, the accumulator pressure may not be sufficient due to the system losses. The CRD system is designed to use the reactor pressure to assist the scram for high reactor pressure scrams. ~~In addition, scram time testing verifies the scram time for individual control rods.~~ Therefore, the higher pressures that might occur as a result of EPU operations during isolation events will not have a significant affect on the scram function of the CRD system. <u>In addition, scram time testing verifies the scram time for individual control rods.</u> -The licensee has also evaluated the performance of the CRD insert, withdraw, cooling and drive functions. The staff concludes that the CRD system will remain acceptable at the EPU condition.

2.6 <u>EPU On-Site Audit Reviews</u>

During the weeks of March 26, and June 16, 2001, members of the NRC Reactor Systems Branch (SRXB) staff visited the Global Nuclear Fuel (GNF) engineering and manufacturing facility at Wilmington, North Carolina. The purpose of these visits was to perform on-site audit reviews of selected safety analyses and system and component performance evaluations used to support extended power uprate (EPU) license submittals. The March audit focused on the Duane Arnold Energy Center (DAEC) EPU, and the June audit was related to the EPU submittal for QCNPS and Dresden Nuclear Power Station. The areas covered by these audits are related to the following sections of the licensee's Safety Analysis Report and are discussed accordingly:

2 <u>Reactor Core and Fuel Performance</u>

2.1 <u>Fuel Design and Operation</u>
2.2 <u>Thermal Limits Assessment</u>
2.3 <u>Reactivity Characteristics</u>
2.4 <u>Stability</u>

9 <u>Reactor Safety Performance Evaluations</u>

9.1 <u>Reactor Transients</u>
9.3 <u>Design Basis Accidents</u>

9.4 Special Events

The staff's audit report is attached (Attachment 1.)

The SRXB staff audit, conducted during the week of June 16, 2001, covered the areas of the licensee's Safety Analysis Report being reviewed by SRXB. As stated in Attachment 1, most questions were resolved during the audit, and the rest were covered by RAIs and the licensee responses. With the exception of the GEXL14 correlation re-evaluation and the ATWS questions in Attachment 1, all open items were resolved.

[input information on resolution of GEXL14 and ATWS issues]

3.0 REACTOR COOLANT SYSTEM (RCS) AND CONNECTED SYSTEMS

The staff's review of the reactor coolant system and connected systems focused on the effects of the power uprate on the structural and pressure boundary integrity of the piping systems and components, their supports, the reactor vessel and internal components including the control rod drive mechanism (CRDM), certain pumps and valves, and balance-of-plant (BOP) piping systems.

The GE generic guidelines for BWR power uprate were based on a 24% higher steam flow; an operating temperature increase to 556°F; and an operating pressure increase to 1095 psia. For QCNPS, the maximum reactor vessel dome pressure is unchanged (remains at 1005 ~~psia~~psig) from the current rated power level, and the dome temperature is also unchanged (remains at 547°F). The steam flow rate will increase from 9.76 $\times 10^6$ lb_m/hr to 11.71 $\times 10^6$ lb_m/hr (increase of approximately 20%) for QCNPS. The maximum core flow rate remains unchanged for the proposed power uprate conditions at QCNPS.

3.1 Nuclear System Pressure Relief

The safety and relief valves (S&RV) provide overpressure protection for the nuclear steam supply system (NSSS), preventing failure of the nuclear system pressure boundary and uncontrolled release of fission products. Each unit has eight spring-actuated safety valves (SSV) (unpiped) which discharge directly into the drywell, rather than the suppression pool. Each unit also has, four relief valves (RV), and a single dual function safety/relief valve (SRV), which are piped to the suppression pool. These S&RVs, together with the reactor scram function, provide the overpressure protection. The S&RV setpoints are established to provide the overpressure protection function while ensuring that there are adequate pressure differences (simmer margin) between the reactor operating pressure and the S&RV actuation setpoints. The S&RV setpoints are also selected to be high enough to prevent unnecessary S&RV actuations during normal plant maneuvers.

[redacted] Sidebarred in PUSAR The licensee evaluated the capabilities of the S&RVs to provide overpressure protection based on the current setpoints and tolerances for operation at the EPU power level and determined that the pressure relief system has the capability to provide sufficient overpressure protection. The analytical limits, using the upper tolerance limits of the valve setpoints, are shown in Table 5-1. The licensee also stated that the EPU evaluation is consistent with the generic evaluations and discussions provided in Section 5.6.8 of ELTR1 and Section 4.6 of ELTR2.

Table 5-1 of the licensee's Safety Analysis Report lists the analytical limits of the SRV, SSVs, and RVs, using the +/-1 percent tolerance. QCNPS has a total of thirteen safety and relief valves, with one SRV set to 1135 psig, two SSVs set to actuate at 1240 psig, two SSVs set at 1250 psig, and four SSVs set at 1260 psig. Two RVs are set to actuate at 1101 psig, and two are set at 1124 psig.

Since the licensee performed limiting ASME overpressure analyses (discussed in Section 3.2) based on 102 percent of the EPU power level, and the current SRV, SSV, and RV setpoints and upper tolerance limits will not change, the staff accepts the licensee's assessment that the S&RVs will have sufficient capacity to handle the increased steam flow associated operation at the EPU power level. The ASME overpressure situation is evaluated during each cycle-specific reload analysis. Therefore, the capability of the S&RVs to ensure ASME overpressure protection will be confirmed in the all subsequent reload analysis.

3.2 Reactor Overpressure Protection Analysis

The design pressure of the reactor vessel and reactor coolant pressure boundary (RCPB) remains at 1250 psig. The ASME Code allowable peak pressure for the reactor vessel and the RCPB is 1375 psig (110 percent of the design pressure of 1250 psig), which is the acceptance limit for pressurization events. The most limiting pressurization transient is analyzed on a cycle specific basis and this approach would be applicable for each EPU reload cycle. Section 5.5.1.4 and Appendix E of ELTR1 evaluated the ASME overpressure analysis in support of a 20 percent power increase, stating that the limiting pressurization transients events are the main steam isolation valve (MSIV) closure with failure of the valve position scram and turbine trip with bypass failure (TTNBP). The licensee analyzed both events based on an initial dome pressure of 1005 psig with one SRV out of service (OOS), with 102 percent of the EPU rated thermal power, 108 percent core flow, and a representative GE-14 equilibrium core. The licensee determined that MSIV closure with valve position scram failure was the most limiting pressurization transient, relative to the TTNBP calculation. The MSIV closure event resulted in a maximum reactor dome pressure of 1336 psig, which corresponds to vessel bottom head pressure of 1358 psig. Therefore, the peak calculated dome pressure (1336 psig) remains below the TS 1345 psig Safety Limit and the peak reactor vessel pressure (1358 psig) remains within the ASME limit of 1375 psig. The licensee concluded that there is no decrease in safety margin and the EPU overpressure protection analysis (given in Figures 3-1 and 3-2 of Reference 2) is consistent with the generic analysis in Section 3.8 of ELTR2.

The maximum calculated pressure in the current ASME overpressure transient analysis meets both the ASME and the TS pressure limits. Therefore, the staff concludes that the licensee has demonstrated an acceptable plant response to overpressure conditions for EPU operation.

3.3 Reactor Pressure Vessel (RPV) and Internals

The staff had previously reviewed and accepted the QCNPS pressure-temperature (P-T) limits. Subsequently, the staff has identified technical issues with the methodology used to derive the fluence values used in the P-T limits evaluation. The original fluence estimate was based on early dosimetry and associated analysis which does not satisfy the guidance of Regulatory Guide (RG) 1.190. New fluence estimates calculated for the EPU amendment use the fluence methodology of General Electric (GE) topical report NEDC-32983P, "General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluations," which is currently under review by the staff. However, technical issues must be resolved to justify applying the fluence values for a full 32 effective full power years (EFPY). As an interim solution, the licensee proposed that NRC approve the P-T limits for a shorter, more defensible period. Specifically, by Reference 14, the licensee requested interim approval of the P-T curves until November 30, 2004, for Unit 1 and March 10, 2004, for Unit 2. This corresponds to approximately one cycle of EPU operation.

The licensee estimates the peak inside surface vessel fluence value for QCNPS to be 4.5 $\times 10^{17}$ n/cm^2 for 32 effective full power years (EFPYs) of operation (including the power uprate). The original estimate for 32 EFPY was 5.1$\times 10^{17}$ n/cm^2 (without the power uprate). The new estimate appears lower than expected by the staff, and it is also lower than the original estimate. The licensee justified the lower fluence value based on: (1) the fact that the QCNPS vessel has a larger diameter than BWRs with comparable power level, (2) the fact that the power density is lower than that in comparable power pressurized water reactor plants, and (3) the fact that the licensee practiced low leakage loadings (and will continue the practice in the future). The preceding material was used to justify why it is OK to use the PT curves for one cycle – not necessarily to show why the new calculated fluence value is low.

Based on the licensee's analysis and the staff's review of previously supplied fluence information, the staff finds the licensee's proposed justification acceptable because: (1) the larger diameter increases the neutron flux attenuation, (2) the lower power density will decrease the neutron leakage, and (3) the core loading scheme will further decrease neutron leakage. The recalculation of the peak 32 EFPY fluence indicates that the existing value which was used for the calculation of the P-T curves is conservative. The staff finds the justification for low absolute peak inside vessel value reasonable, based on known physical parameters, and providing adequate assurance of safety for the proposed time limit (e.g, one cycle of EPU operation). However, new fluence predictions using staff-accepted methodologies are required to justify continued operation beyond the proposed time limit, as discussed below.

3.3.1 Reactor Vessel Fracture Toughness

(The proper material that goes with this title is under 3.3.3. This Material should be titled RPV Integrity)

The licensee evaluated effects of the QCNPS power uprate on the reactor vessel and internal components. The loads considered in the evaluation include reactor internal pressure difference (RIPD), loss-of-coolant-accident (LOCA), flow loads, acoustic loads, thermal loads, seismic loads, and dead weight loads. The licensee indicated that the load combinations for normal, upset, and faulted conditions were considered consistent with the current design basis analysis. In the evaluation, the licensee compared the proposed power uprate conditions (pressure, temperature, and flow) against those used in the design basis. For cases where the power uprate conditions are bounded by the design basis analyses, no further evaluation was performed. If the power uprate conditions are not bounded by the design basis, new stresses were determined by scaling up the existing design basis stresses proportionate to the proposed power uprate conditions. The resulting stresses are shown to be less than the applicable allowable values, consistent with the design basis. Based on the licensee's evaluation, the staff finds that the methodology used by the licensee is consistent with the NRC-approved methodology in Appendix I of ELTR1 (Reference 3), and is therefore acceptable.

The stresses and cumulative fatigue usage factors (CUFs) for the reactor vessel components were evaluated by the licensee in accordance with the ASME Boiler and Pressure Vessel Code (Code), Section III, 1965 Edition, which is the code of record at QCNPS. The assessment is performed consistent with the current design basis. Based on the licensee's evaluation, the staff finds the licensee's assessment acceptable and in compliance with the Code of record at QCNPS.

The staff concurs with the licensee's conclusion that the reactor vessel internal components will continue to maintain their structural integrity for the extended power uprate condition.

3.3.2 Reactor Vessel Internals and Pressure Differentials

The licensee provided the calculated maximum stresses and CUFs for the reactor vessel components (Table 3-3 of Reference 2). The stresses and CUFs were evaluated by the licensee in accordance with the ASME Code, Section III, 1965 Edition, which is the code of record at QCNPS. The licensee indicated that for QCNPS, the reactor internal components are not ASME Code components. However, ASME Code requirements have been used as guidelines in the design basis documents. The assessment is performed consistent with the current design basis. The reactor vessel components not listed in Table 3-3 have maximum stresses and CUFs that are either not affected by the power uprate or are already bounded by those listed in the table. The maximum calculated stresses shown in the table are within the allowable limits, and the CUFs are less than the code limit of unity. The licensee evaluated the reactor internal components for QCNPS by comparing the changes in loads that are affected by the power uprate against the margins available in the design basis analysis. Reference 22 shows that the existing margins are sufficient to accommodate the increase in loads for the power uprate. For some cases, the licensee compared the affected loads (i.e., reactor internal pressure differential (RIPD)) on certain components against their design basis loads. Reference 22 shows that the design basis loads are bounding for the power uprate. The maximum stresses for certain critical components of the reactor internals were also provided in Reference 22 for the power uprate conditions. The calculated stresses are shown less than the allowable Code limits.

The licensee assessed the potential for flow-induced vibration on the reactor components. The licensee determined that the EPU has the greatest effect on the steam separators and dryers in the upper portion of the reactor vessel. This is due to the increase steam flow that results from the proposed power uprate. The effects of the power uprate on flow-induced vibrations for other components in the reactor annulus and core regions are less significant because the proposed power uprate conditions do not require any increase in core flow and very little increase (less than 2.2%) in the drive flow. The evaluation of flow-induced vibration for the reactor internal components was performed based on the vibration data recorded during startup testing at QCNPS, the GE prototype BWR/4 plant vibration data, and on operating experience from other similar GE BWR plants. The vibration levels were calculated by extrapolating the recorded vibration data to power uprate conditions and compared with the plant allowable limits. The stresses at critical locations were calculated based on the extrapolated vibration peak response displacements and found to be within the GE allowable design criteria of 10 ksi. Stress values less than 10 ksi are within the endurance limit; therefore, there is no need to compute the cumulative fatigue usage for the component due to flow-induced vibration. The licensee concluded that vibration levels of all safety-related reactor internal components are within the acceptance criteria. Based on the licensee's evaluation, the staff finds the licensee's conclusions acceptable and consistent with the ASME limit of 13.6 ksi for the peak vibration stress.

The licensee indicated in Reference 22 that the steam dryers and separators are not safety-related components; however, their failure may lead to an operational concern. The licensee also indicated that, although the design basis criteria do not require evaluation of the flow-induced vibration or determination of cumulative fatigue usage for the steam separators and dryers, the maximum vibration level for the separators is small in comparison to the allowable limit. The licensee also indicated that the dynamic pressure loads, which may induce vibration for the dryers, are small in comparison to loads for the design basis faulted condition. Accordingly, stresses in the dryers due to vibration associated with the proposed uprated condition are estimated to be less than the allowable limit. In addition, the dryers will be visually inspected during removal in each refueling outage, and any significant cracking can be detected and repaired. The design basis for the steam dryers specifies that the dryers maintain their structural integrity when subjected to a steam line break occurring beyond the main steam isolation valves. Since the dome pressure is not changed, the current steam dryer analysis remains bounding for the proposed power uprate conditions. On the basis of information provided by the licensee in Reference 22, the staff concludes that the licensee has reasonably demonstrated that the steam dryers and separators will meet their design basis requirements and maintain their structural integrity following the proposed extended power uprate.

Based on its review of the licensee's evaluation of the reactor vessel internals, the staff finds that the maximum stresses and fatigue usage factors are within the Code-allowable limits. The staff concurs with the licensee's conclusion that the reactor vessel internal components will continue to maintain their structural integrity for the extended power uprate condition.

The licensee indicated that the code of record for the CRDMs is the ASME Code, Section III, ~~1968~~ 1965 Edition with addenda up to and including ~~Winter 1968~~Summer 1965. The components of the CRDMs which form part of the primary pressure boundary have been designed for a bottom head pressure of 1250 psig, which is higher than the analytical limit of 1095 psig for the reactor bottom head pressure. The licensee's evaluation indicated that the maximum calculated stress for the CRDMs is less than the allowable stress limit. The analysis for cyclic operation of the CRDMs resulted in a maximum CUF of 0.15 for the limiting location, the CRDM main flange, at the extended power uprate condition. This is less than the Code-allowable CUF limit of 1.0.

On the basis of the licensee's evaluation, the staff concurs with the licensee's conclusion that the CRDMs will continue to meet their design basis and performance requirements at extended power uprate conditions.

3.3.3 Reactor Vessel Integrity

This material relates to Fracture Toughness 3.3.1

In Sections 3.3.1 and 3.5 of Reference 2, the licensee assessed the effects of the EPU on the reactor pressure vessel (RPV) and reactor coolant pressure boundary (RCPB) piping of each unit. With regard to the RPV, the licensee provided an assessment of the impact of the EPU on the RPV wall fluence, the need to revise the P-T limit curves, and the validity of previously approved upper shelf energy (USE) equivalent margins analyses. The licensee stated that for EPU, the 32 EFPY shift in nil-ductility reference temperature (RT_{NDT}) resulting from neutron irradiation decreases (see Section 3.3 of this SE) and consequently there is no change required in the adjusted reference temperature. EPU does not affect the existing surveillance program schedule.

For analyzing the RPV, the licensee examined the extended power uprate's effect on the RPV belt line fluence. The analyses addressed the expected RPV material embrittlement since it is directly related to the RPV neutron fluence, which is in turn related to the reactor operating power. The licensee stated that the estimated fluence for the EPU decreases from the updated final safety analysis report (UFSAR) end-of-license value because the pre-EPU fluence is based on conservative dosimetry values and the pre-EPU fluence bounds the fluence calculated for the EPU evaluations. This lower fluence was used to evaluate the RPV against the requirements of 10 CFR Part 50, Appendix G. The results of the licensee's evaluation indicate that:

- The (USE) remains bounded by the equivalent margins analysis for the design life of the vessel and maintains the margin requirements of 10 CFR 50, Appendix G.

- The P-T curves contained in the current Technical Specifications remain bounding for EPU operation up to 32 EFPY.

- For EPU, the 32 EFPY shift in RT_{NDT} resulting from neutron irradiation decreases and consequently requires no change in the adjusted reference temperature (ART), which is the initial RT_{NDT} plus the shift and a margin term.

- The maximum RV dome operating pressure for EPU operation is unchanged from that for current operation. Therefore, the current hydrostatic and leakage test pressures are acceptable for the EPU.

The licensee concluded that the vessel remains in compliance with the regulatory requirements during EPU conditions.

The staff concludes that many of the existing RPV-related evaluations and analyses remain valid and applicable for the EPU, under the conditions described below. This is based on (1) current design assessments that show significant design margins in reactor integrity analyses which are not affected by the proposed power uprate, (2) the loading conditions are either unchanged or are bounded by the analyzed loading conditions, and (3) because the licensee is not predicting an increase in end-of-life fluence. The staff concurs that the USE remains bounded by the equivalent margins analysis for the design life of the vessel and maintains the margin requirements of 10 CFR 50, Appendix G. The staff also concludes that, since the maximum dome operating pressure for EPU is unchanged from that for current operation, the current hydrostatic and leakage test pressures are acceptable for the extended power uprate.

However, as mentioned in Section 3.3 of this SE, the NRC staff has technical issues with the methodology used to derive the fluence values which form the basis for the evaluating reactor vessel integrity and fracture toughness, including P-T limits. The **licensee commits** to revise the fluence predictions using an acceptable methodology before the end of the first cycle of EPU operation on each unit, or to provide justification for continued use of the existing fluence estimate. The staff evaluated the RV integrity and fracture toughness for EPU conditions based on the fluence provided by the licensee, 4.5×10^{17} n/cm^2. If the fluence is projected to increase, the licensee shall re-evaluate the P-T limits and the RV integrity issues before vessel fluence exceeds 4.5×10^{17} n/cm^2.

3.3.4 Steam Separator and Dryer Performance

The steam separators and dryers do not perform a safety-related function other than structural integrity; however their operational performance is important to equipment design and steam moisture ~~carryover~~ content is a factor in design inputs such as transport of particulate radioactive material from the reactor. The steam separator and dryer performance evaluation have been generically described in Section 5.5.1.6 of ELTR1. A plant-specific performance evaluation determined that hardware modifications are required to reduce the moisture content. As noted in the licensee's letter dated May 18, 2001, (Reference ~~12~~3), a startup test will evaluate the performance of the steam separator-dryers and demonstrate the moisture levels are within appropriate limits. In their letter dated August 7, 2001, (Reference 19) the licensee noted the design criteria for the planned modification was established to maintain ~~carryover~~ moisture content $\leq$0.2 wt.% under most operating conditions~~;~~. Acceptable moisture content will be demonstrated based on actual moisture carryover data collected at both Dresden and Quad Cities stations.

Based on the licensee's commitment to perform moisture ~~carryover~~ content testing, the staff concurs with the licensee's conclusion that the moisture content of the steam at EPU conditions will be acceptable.

3.4 Reactor Recirculation System

QCNPS is currently licensed to operate at a maximum core flow of 98 Mlb/hr (100 percent of the rated flow) and the EPU does not require an increase in the maximum allowable core flow. Future application of the increased core flow (ICF) option may increase the maximum core flow to 108 percent of the current rated value, so some analyses are performed at this value. The primary function of the recirculation system is to vary the core flow and power during normal operation. However, the recirculation system also forms part of the reactor coolant system (RCS) pressure boundary.

The licensee evaluated the changes in the system operating pressure and temperature at the EPU conditions and determined that changes are small and result in conditions less than the current design ~~rated~~ conditions. The QCNPS EPU will not involve any increase in the steady state dome pressure. However, operation at the EPU power level would increase the two phase flow resistance, requiring a slight increase in the recirculation system drive flow. The licensee estimated the required pump head and pump flow at the EPU conditions and determined that the power demand of the recirculation motors will increase slightly. The increased drive flow will require increasing the pump speed. The licensee stated that the QCNPS recirculation system and its components are capable of providing the core flow required for operation at the EPU conditions. The recirculation system evaluations are consistent with the generic evaluation in Section 4.5 of ELTR2, Supplement 1. Section 4.5 of ELTR2, Supplement 1, evaluated the recirculation system performance for a 20 percent power uprate with a 75 psig increase in the normal dome operating pressure and concluded that the recirculation system design can accommodate the operating condition associated with the power uprate.

The staff reviewed the impact that a recirculation pump trip would have on ~~the~~ plant safety. The plant is analyzed for decreases in the reactor core coolant flow rate, which depend on the operation of the recirculation pumps and motors. The transient events in this category are: (a) single and multiple recirculation pump trips, (b) recirculation flow controller failure malfunction, (c) recirculation pump shaft seizure (normal and SLO), and (d) recirculation pump shaft break. Core flow is reduced in these events, resulting in a corresponding decrease in the reactor power. For QCNPS, these transients are nonlimiting in terms of thermal limits and are not reanalyzed in cycle-specific reload analysis, except for the SLO pump seizure event. The SLO pump seizure is not analyzed in cycle-specific analyses (Reference UFSAR Chapter 15). This is not mentioned in the EPU submittals. EPU operation is not expected to make these transients limiting.

Chapter 15 of the QCNPS UFSAR states that the pump seizure event during single loop operation is analyzed at every reload to determine the impact on the MCPR, specifically to ensure that this event does not violate the TS SLMCPR for the cycle. QCNPS is licensed to operate with SLO, and the licensee stated that SLO operation would be limited to 70.2 percent of the EPU power level (2076MWt) at 55.1 percent core flow (54Mlb/hr). This power level corresponds to the MELLLA upper boundary at the maximum recirculation pump speed of 102.5 percent.

[redacted] Sidebarred in the PUSAR The cavitation interlock, shown in the lower portion of the power/flow map, ensures that sufficient subcooling is available to prevent cavitation of the recirculation pumps. This is consistent with the evaluation in Section F.4.2.6 of ELTR1.

The licensee will not change the values (percent flow) of the recirculation pump flow mismatch specification in the TS.

The staff finds the licensee's assessment of the changes to the cavitation interlock, the recirculation pump mismatch power basis, and the jet pump SR acceptable. (the jet pump SR does not appear to be discussed in the SE).

Section 4.5.3 of Supplement 1 to ELTR2 discussed the impact of 20 percent power uprate on the recirculation system safety function for the: (a) closure of the discharge valve during low pressure coolant injection (LPCI), (b) pump trip in transients and anticipated transient without scram (ATWS), and (c) measurement of the drive flow used in the average power range monitor (APRM) flow-biased setpoint and rod blocks. For LOCA response, one or both recirculation system discharge valves must close to ensure LPCI injection into the core. Since the QCNPS power uprate does not involve an increase in the operating pressure, the discharge valve closure permissive pressure would not be changed.

The recirculation system drive flow is measured and used as an input to the APRM for the flow-biased APRM scram and rod blocks. According to Supplement 1 to the ELTR2, the recirculation system fast transient analysis is necessary to support EPU operation for the plants that have adopted the Average Power Range Monitor/Rod Block Monitor (ARTS) feature to ensure adequate protection during the transient. The ARTS program replaces the flow-biased APRM trip setdown during operation at off-rated conditions. Under these conditions, ARTS plants use power and flow dependent MCPR and LHGR limits for operation at the off-rated conditions. Table 9-2 of the QCNPS Safety Analysis Report provides the delta-CPR value for the fast recirculation flow transient and confirms that the ARTS multipliers used to develop the power dependent MCPR(P) and shown in Table 9-3 remain bounding. This is acceptable to the staff.

3.5 Reactor Coolant Piping and Components

The licensee evaluated the effects of the power uprate condition, including higher flow rate, temperature, pressure, fluid transients, and vibration effects on the RCPB and the BOP piping, systems, and components. The components evaluated included equipment nozzles, anchors, guides, penetrations, pumps, valves, flange connections, and pipe supports (including snubbers, hangers, and struts). The licensee indicated that the original codes of record as referenced in the original and existing design basis analyses, and analytical techniques were used in the evaluation. No new assumptions were introduced that were not in the original analyses. Based on this information, the staff finds the licensee's evaluation and findings to be acceptable.

3.5.1 Pipe Stresses

The RCPB piping systems evaluated include the reactor recirculation, main steam (MS), main steam drains, reactor core isolation cooling (RCIC), high pressure coolant injection (HPCI), feedwater (FW), reactor water cleanup, core spray, standby liquid control, residual heat removal (RHR), low pressure coolant injection (LPCI)/containment spray, RPV head vent line, and relief valve/safety relief valve (RV)/(SRV) discharge line systems using the present code(s) of record. The licensee indicated that the evaluation follows the process and methodology defined in Appendix K of ELTR1 (Reference 3) and in Section 4.8 of Supplement 1 of ELTR2 (Reference 5). In general, the licensee compared the increase in pressure, temperature and flow rate due to the power uprate against the same parameters used as input to the original design-basis analyses. The comparison resulted in the bounding percentage increases in stress for affected limiting piping systems. The bounding percentage increases are compared to the design margin between calculated stresses and the code allowable limits. As a result of such comparison, the licensee concluded that there are sufficient design margins to justify operation at the power uprate condition. The bounding percentage increases were also applied to the original calculated stresses for the piping to determine the stresses at the proposed power uprate condition. The staff finds the licensee's methodology to be acceptable considering the conservatism in the calculation of the scaling factors for the power uprate stress and loads.

In its response to the staff's request for additional information (Reference 20), the licensee indicated that the majority of the RCPB piping systems at QCNPS are designed to American National Standards Institute (ANSI) B31.1-1967, which does not require a fatigue analysis.

Other codes were used during the plant operation: ASME Code, Section I, 1965 Edition, through Summer 1966 Addenda including Code Cases N-1 thru N-3 and N-7 thru N-11, and ASME Code Section III, Sub-section NC (Class 2), 1977 through 1978 Winter Addenda and ASME Code Section III, Sub-section ND (Class 3), 1974 through 1976 Summer Addenda. These codes do not include requirements to evaluate fatigue. As a result of its evaluation, the licensee concluded that for all RCPB piping systems, the original piping design has sufficient design margin to accommodate the slight changes due to the proposed power uprate. The staff reviewed relevant portions of the evaluation provided by the licensee in Reference 22 and finds the licensee's evaluation acceptable.

The licensee evaluated the stress levels for BOP piping and appropriate components, connections, and supports in a manner similar to the evaluation of the RCPB piping and supports, based on increases in temperature and pressure from the design basis analysis input. The evaluated BOP systems include lines which are affected by the power uprate, but not evaluated in Section 3.5 of Reference 2, such as the LPCI/containment cooling water, feedwater condensate and heater drain, main steam drain lines, and portions of the main steam, feedwater, RCIC, HPCI, and RHR systems outside the primary containment. The existing design analyses of the affected BOP piping systems were reviewed against the uprated power conditions. As a result, the licensee indicated that some main steam and torus attached piping was found not to have a sufficient margin in the original design analyses to accommodate the changes due to the proposed power uprate. For these piping systems, the licensee performed detailed analysis that, in most cases, demonstrated the adequacy of the existing piping design for the power uprate condition. However, in some cases, piping modifications are required to bring the piping within the code allowable stress limits. The licensee committed that the piping modifications will be completed prior to implementation of the power uprate at QCNPS. The licensee provided the calculated stresses (Reference 20), assuming the required modifications were completed. Based on this information, the staff has concluded that the stresses and stress ratios provided in the tables are within the code-allowable limits and are, therefore, acceptable. **The required modifications represent a confirmatory item No. 1 that must be verified prior to power uprate at QCNPS.**

The licensee evaluated pipe supports such as snubbers, hangers, struts, anchorages, equipment nozzles, guides, and penetrations by evaluating the piping interface loads due to the increases in pressure, temperature, and flow for affected limiting piping systems. The increase in pipe support loads due to the power uprate conditions are similar to the increase in piping stresses. However, when combining these increases with the loads such as seismic and deadweight, that are not affected by the power uprate, the overall combined support load increases are generally insignificant except for the main steam and torus attached piping. The licensee indicated that as a result of the evaluation, there are supports, structural attachments and supporting steel that require modifications to meet requirements of the code and code allowable stress limits. The licensee reviewed the original postulated pipe break analysis and concluded that the existing pipe break locations were not affected by the power uprate, and that no new pipe break locations were identified. Based on the licensee's evaluation, the staff concurs with the licensee's findings. **The required torus attached piping modifications represent a confirmatory item No. 2 that must be verified prior to power uprate at QCNPS.**

The licensee indicated that the flow-induced vibration (FIV) levels for the safety-related MS and FW piping systems will increase in proportion to the increase in the fluid density and the square of the fluid velocity following the proposed power uprate. To ensure that the vibration level will be below the acceptable limit, the **licensee is committed** to perform a piping vibration startup test program, as outlined in Section 10.4.3 of the amendment submittal. The startup testing would include monitoring and evaluating the flow induced vibration during the plant start-up for the proposed uprated power operation. Vibration data will be collected at interim test conditions, which correspond to 50 percent, 75 percent and 100 percent of the original rated thermal power (ORTP), to each 5 percent step increase in power level above 100 percent of ORTP, up to the final proposed uprated power level. The vibration at the new higher power uprate level may be determined based on extrapolation of the vibration data taken at the lower power levels. The measured vibration levels are compared against the acceptance criteria, where the allowable vibration stress levels are set by the design fatigue endurance stress intensity limits established by the ASME code~~GE~~ for stainless and carbon steel. Based on the licensee's review, the staff finds the licensee's methodology in assessing the FIV to be acceptable.

Based on the above review, the staff concurs with the licensee's conclusion that the design of piping, components, and their supports, including the required modifications discussed above, is adequate to maintain structural and pressure boundary integrity at the proposed extended power uprate condition.

3.5.2 Flow Accelerated Corrosion

For the RCPB piping, the licensee provided an assessment of changes in the potential for flow accelerated corrosion (FAC) damage due to the EPU. The licensee evaluated the effect of the EPU on FAC in the following systems: recirculation system, main steam and associated piping systems, feedwater system, and other RCPB piping. The licensee's evaluation of the reactor coolant piping confirmed that changes in the flow parameters associated with the EPU would have little or no significant effects on the potential for FAC in those systems which might be susceptible to the phenomenon (e.g., feedwater or main steam systems).

The components in the recirculation system are made from stainless steel which is immune to FAC. FAC damage will not, therefore, occur in this system after power uprate.

The main steam and associated piping system contains components made from carbon steel which is prone to FAC. However, these components are exposed to steam having a 95% quality level and in this environment no FAC damage will occur. Since the power uprate is expected to result in some change in moisture content, there is a possibility of the formation of a FAC active environment. In order to prevent this, **the licensee committed**, as a part of the power uprate implementation, **to modify the reactor vessel moisture separation equipment**. This modification will maintain carryover levels consistent with values before the power uprate and will prevent FAC caused damage.

The feedwater system has carbon steel components which are affected by FAC. After the power uprate there will be some changes in operating conditions caused by the operation of an additional pump. Also, system pressure and temperature are expected to change. These changes will affect the amount of material loss due to FAC. However, the licensee will account for these changes by modifying its CHECWORKS predictive code. The predictions obtained from this modified code will be used by the licensee to assess wear rates and to schedule inspections for the components currently included in the program. It also will serve to include other components which may become susceptible to FAC after power uprate.

The power uprate will only slightly affect the inlet temperature in the other RCPB pipes and will not change their operating environment. Therefore, no potential will exist for the FAC damage to these pipes.

The staff reviewed and evaluated the licensee's analyses of the systems where power uprate may have an effect on FAC. The staff concludes that the licensee has demonstrated that EPU will have a very small effect on FAC. The licensee will account for the FAC changes ~~either~~ by ~~plant modifications, operating procedure changes, or~~ changes to the predictive FAC model (and corresponding changes to inspections) so that timely corrective measures can be implemented.

3.6 Main Steam Flow Restrictors

The licensee stated that there is no impact on the structural integrity of the main steam flow restrictor for the extended power uprate. In Section 3.2 of the power uprate license amendment request, the licensee indicated that a higher peak RPV transient pressure of 1336 psig results from the proposed QCNPS plant power uprate conditions, but this value remains below the ASME code limit of 1375 psig. Therefore, the main steam line flow restrictor will maintain its structural integrity following the power uprate since the restrictor was designed for a differential pressure of 1375 psig. Based on the licensee's evaluation, the staff concurs with the licensee's conclusions.

3.7 Main Steam Isolation Valves (MSIVs)

The main steam isolation valves (MSIVs) are part of the reactor coolant pressure boundary and perform a safety function to isolate the main steam line. The MSIVs must be able to close within the specified time limits at all design and operating conditions upon receipt of a closure signal. They are designed to satisfy leakage limits set forth in the plant technical specifications.

The licensee indicated that the MSIVs have been generically evaluated, as discussed in Section 4.7 of ELTR2. This evaluation covers both the effects of the changes to the structural capability of the MSIV to meet pressure boundary requirements, and the potential effects of EPU-related changes to the safety functions of the MSIVs. The generic evaluation is based on (1) a 20% thermal power increase, (2) an increased operating reactor dome pressure to 1095 psia, (3) a reactor temperature increase to 556° F, and (4) steam and feedwater increase of about 24%. The licensee stated that the conditions for QCNPS are bounded by those in the generic analysis. The dome pressure and temperature does not increase with the EPU. The increase in flow rate assists MSIV closure, which results in a slightly faster MSIV closure time. Technical Specification MSIV closure timing requirements will continue to be met. The licensee concluded that the existing design pressure and temperature for the MSIVs are bounding for the proposed power uprate and that the ability of the MSIVs to perform their isolation function is not affected following the power uprate condition.

The licensee did request an increase in the setpoint for initiation of MSIV closure on high flow to be equivalent to 140% of uprated steam flow in each steamline; consistent with ELTR1 section

F.4.2.5. This setpoint change is evaluated in Section 5.3 item 6 of this SE. The licensee noted that the new break flow setpoint will remain below the analyzed choked flow through the steam line flow restrictors. For lower magnitude breaks, the licensee noted (Reference 19) that breaks between 120%-140% flow will result in a low pressure isolation signal, and additionally a break in the steam tunnel would actuate the high temperature switches. Both of these actuations will also isolate the MSIVs. Therefore, EPU operation as indicated above remains bounded by the conclusion of the generic evaluation in Section 4.7 of ELTR2, and the MSIVs are acceptable for EPU operation.

The staff accepts the licensee assessment that the MSIV closure time will be maintained as analyzed and specified in the TS. In addition, various technical specification surveillances require routine monitoring of MSIV closure time and leakage to ensure that the licensing basis for the MSIVs is preserved. Based on the review of the licensee's rationale and evaluation, the staff concurs with the licensee's conclusion that the plant operations at the proposed EPU level will not affect the ability of the MSIVs to perform their isolation function.

3.8 RCIC System

The QCNPS units have a reactor core isolation cooling system (RCIC). The QCNPS RCIC system provides makeup coolant ~~core cooling~~ (the term is revised to maintain consistency with the TS bases. The intent of the statement is not changed) in the event of a transient where the reactor pressure vessel (RPV) is isolated from the main condenser, concurrent with the loss of all feedwater flow (LOFWF), and when the RPV pressure is greater than the maximum allowable for the initiation of a low pressure core cooling system.

Section 5.6.7 of ELTR1 provides the scope of the RCIC system evaluation. The maximum injection pressure for RCIC is conservatively based on the upper analytical setpoint for the lowest available group of SRVs operating in the spring safety mode. For the QCNPS EPU, the reactor dome pressure and the SRV setpoints are unchanged, and there is no change to the RCIC high pressure injection parameters. In addition, the results of the plant specific LOFWF transient evaluation indicate that the RCIC system design flow rate (400gpm) is sufficient to meet the acceptance criterion (maintain reactor water level above top of active fuel) for EPU conditions. ~~Also, the calculated minimum RCIC injection rate required at EPU conditions is below the system design flow rate (400gpm). The licensee states that the RCIC turbine operation at EPU will not change any startup transient or system reliability.~~

GE services information letter (SIL) No 377, "RCIC Startup Transient Improvement with Steam Bypass," describes startup control modifications intended to improve RCIC startup reliability. The licensee states that the RCIC turbine operation at the no pressure increase EPU conditions will not result in changes to the startup transient characteristics. ~~However, the licensee states that~~Since a re-evaluation of the QCNPS RCIC turbine startup performance indicates acceptable transient speed peaks without performing the SIL 377 modifications, no changes are needed for EPU. The licensee further states that EPU operation does not decrease the NPSH available for the RCIC pump, nor does it increase the NPSH required above the system design value. The required EPU surveillance testing and system injection demands would occur at the same reactor operating pressures, so there would be no change to existing system and component reliability. ~~The LOFWF transient event was evaluated, and the acceptance criterion, (maintain reactor water level above top of active fuel) continues to be met for EPU conditions.~~

Because the licensee has analyzed the LOFWF transient for EPU operation, consistent with the ELTR1 guidelines, and has conservatively evaluated the pressure performance requirements of the QCNPS RCIC system, the staff accepts the licensee's assessment.

3.9 Residual Heat Removal System

The generic residual heat removal (RHR) capability evaluation process is described in Section 5.6.4 of ELTR1. The RHR/Containment Cooling (CC) mode ~~LPCI/Containment Cooling (CC)~~ system is designed to restore and maintain the coolant inventory in the reactor vessel while the Shutdown Cooling (SDC) mode ~~system~~ provides primary system decay heat removal after reactor shutdown for normal and post-accident conditions. The RHR~~LPCI/CC~~ system is designed to operate in the low-pressure coolant injection (LPCI) mode, suppression pool cooling mode (SPC), and containment spray cooling (CSC) mode. The SDC mode ~~system~~ is designed to provide Shutdown cooling, or fuel pool cooling (FPC) assist heat removal. The LPCI mode is discussed in Section 4.2.2, while the effects of the EPU on the other modes are described below. The results of the following evaluations are consistent with the generic evaluation in Section 4.1 of ELTR2.

3.9.1 Shutdown Cooling Mode

[redacted]

Since the SDC evaluation at the EPU condition demonstrated that the plant can meet this cooldown time, the staff finds it acceptable.

3.9.2 Suppression Pool Cooling Mode

During normal plant operation, the SPC function is to maintain the suppression pool temperature below the TS limit. Following abnormal events, the SPC function controls the long-term suppression pool temperature such that the design temperature limit of 281F is not exceeded. Following a LOCA, the increase in decay heat due to EPU increases the heat input to the suppression pool, resulting in a slightly higher peak containment temperature and pressure, as discussed in Section 4.1.1. The analysis at 102 percent of EPU power discussed in Section 4.1.1 results in only an 8F increase in the peak temperature and confirms that the suppression pool temperature remains below its design limit. The higher temperature reduces the NPSH available to the LPCI/CC pumps during operation; the increased pressure partially offsets this effect. ~~however,~~ Section 4.2.5 shows that adequate NPSH margin remains under post-LOCA operating conditions.

3.9.3 Containment Spray Cooling Mode

The containment spray cooling mode of the RHR system is designed to provide water from the suppression pool via spray headers into the drywell and suppression chamber air spaces to reduce the long-term containment pressure and temperature during post-accident conditions. The power uprate slightly increases the containment spray water temperature. This increase has a negligible effect on the ability of the containment spray cooling mode to maintain containment pressure and temperature within design limits, as the peak pressure and temperatures are reached well before the use of containment spray is assumed to occur.

Based on the review of the licensee's rationale and evaluation, the staff concurs with the licensee's conclusion that plant operations at the proposed EPU level will have an insignificant impact on the containment spray cooling mode.

3.9.4 Fuel Pool Cooling Assist Mode

As a result of plant operations at the proposed EPU, the decay heat load for specific fuel discharge scenarios will increase. In the event that the spent fuel pool (SFP) heat load exceeds the heat removal capability of the fuel pool cooling and cleanup (FPCC) system (e.g., during full-core offload events), the RHR will be operated in the fuel pool cooling assist mode to provide supplemental cooling to the SFP, and to maintain the SFP temperature within acceptable limits. The adequacy of the combined heat removal capability of the FPCC system and the RHR system operating in the fuel pool cooling assist mode to meet the increases in SFP heat loads resulting from the proposed EPU is addressed in Section 6.3.

3.10 Reactor Water Cleanup System

Evaluation of the reactor water cleanup system (RWCU) is included in Section 3.5 of this SE. There does not appear to be a reference to RWCU in Section 3.5 of the SE.

3.11 Main Steam, Feedwater, and Balance-of-Plant Piping

The main steam, feedwater, and balance-of-plant piping evaluation is addressed along with reactor coolant piping in Section 3.5 of this SE.

4.0 ENGINEERED SAFETY FEATURES

4.1 Containment System Performance

The QCNPS Updated Final Safety Analysis Report (UFSAR) provides the results of analyses of the containment response to various postulated accidents that constitute the design basis for the containment. Operation with 17.8 percent MWt power uprate from 2511 MWt to 2957 MWt would change some of the conditions and assumptions of the containment analyses. ELTR1 (Reference 3) Section 5.10.2 requires the power uprate applicant to show the acceptability of the effect of the uprate power on containment capability. These evaluations will include containment pressures and temperatures, LOCA containment dynamic loads, safety relief valve containment dynamic loads and subcompartment pressurization. Appendix G of ELTR1 prescribes the generic approach for this evaluation and outlines the methods and scope of plant-specific containment analyses to be done in support of power uprate. These analyses will cover the response through the time of peak drywell pressure throughout the range of power/flow operating conditions with power uprate. Appendix G states that the applicant will analyze short term containment pressure and temperature response using the previously applied GE code, M3CPT code. The QCNPS EPU analyses uses LAMB code with Moody's Slip Critical flow model to generate the blowdown flowrates used as inputs to M3CPT. This approach, using a code with a more detailed reactor pressure vessel model, results in more realistic break flows for input to M3CPT, and differs from the current UFSAR analyses. Plant-specific use of the LAMB code, which has been previously reviewed by the NRC for Appendix K LOCA analyses, was addressed in ELTR1, Appendix G.

Appendix G of ELTR1 also requires the applicant to perform long-term containment heat up (suppression pool temperature) analyses for the limiting UFSAR events to show that pool temperatures will remain within limits for suppression pool design temperature, emergency core cooling system (ECCS) net positive suction head (NPSH) and equipment qualification temperatures. These analyses can be performed using the GE computer code SHEX. SHEX is partially based on M3CPT and is used to analyze the period from when the break begins until after peak suppression pool heat up (i.e., the long-term response). The SHEX computer code has been used by GE on all BWR power uprates. QCNPS provided additional details of the confirmatory calculations performed to validate use of this code. The licensee chose a benchmark case of a double-ended break of the recirculation line depicted in UFSAR Table 6.2-3, Case E. Assumptions were adjusted for the SHEX analysis to match those used for original licensing. The HXSIZ codes show that the SHEX code conservatively overpredicts (4° F) peak suppression pool temperature. Based on the licensee's evaluation, the staff concurs that the use of SHEX code is acceptable for EPU containment analyses.

In a letter dated August 13, 2001, providing additional information (Reference 23), the licensee addressed the EPU effect on Technical Specification 3.6.2.1, "Suppression Pool Average Temperature." This Technical Specification is applicable in Modes 1, 2, and 3 with limits varying above and below 1% RTP. The licensee noted that the 1% RTP value is approximately equal to heat losses, such that the reactor is effectively shutdown. The licensee indicated that the number is based on engineering judgement and would remain applicable with the new EPU RTP, which is 17.8% higher.

Based on the licensee's rationale, the staff concurs with the licensee's conclusion that the references to 1% RTP should be retained for Technical Specification 3.6.2.1.

4.1.1 Containment Pressure and Temperature Response

Short-term and long-term containment analyses results following a large break inside the drywell are documented in the QCNPS UFSAR. The short-term analysis was performed to determine the peak drywell and wetwell pressure response during the initial blowdown of the reactor vessel inventory into the containment following a large break inside the drywell (design basis accident (DBA-LOCA)), while the long-term analysis was performed to determine the peak suppression pool temperature response considering decay heat addition. In Reference 19, the licensee provided both short-term and long-term curves for parameters of interest for containment response for a DBA-LOCA, including temperature and pressure for the drywell and wetwell atmosphere, and suppression pool temperature. Reference 19 also included appropriate curves for parameters used in the net positive suction head calculations, which use different assumptions that are conservative for determining available suction pressure for the ECCS pumps. These curves, including the statements of assumptions used and explanatory notes, clarify the containment response and analysis results for the effect of the EPU.

The licensee indicated that the containment analyses were performed in accordance with NRC guidelines using GE codes and models. As noted above, the M3CPT code was used to model the short-term containment pressure and temperature response. The licensee also indicated that the SHEX code was used to model the long-term containment pressure and temperature response for EPU.

4.1.1.1 Long-Term Suppression Pool Temperature Response

(a) Bulk Pool Temperature

The licensee indicated that the long-term bulk suppression pool temperature response with the EPU was evaluated for the DBA-LOCA. The bounding analysis was performed at 102% of EPU rated thermal power (RTP). The analysis was performed using the SHEX code and a more realistic decay heat model. The staff has determined that the model used, the ANS/ANS 5.1-1979 decay heat model with an uncertainty adder of two sigma, is acceptable.

The peak bulk suppression pool temperature was calculated to be 199° F, based on revised EPU methodology, which is an increase of 22° F in peak pool temperature from 177° F over the current licensing basis. However, a portion of that increase is caused by the change in methodology. The EPU results in a 9° F increase in peak pool temperature over the current power level temperature, using EPU methodology and input assumptions. The peak suppression pool temperature remains below the wetwell structure design temperature of 281° F.

Based on the staff's review of the licensee's analyses, and experience gained from review of power uprate applications for other BWR plants, the staff concludes that the peak bulk suppression pool temperature response remains acceptable for the extended power uprate.

(b) Local Pool Temperature with RV plus SRV Discharge

QCNPS is equipped with four relief valves (RVs) and one safety relief valve (SRV) per unit. Because of concerns resulting from unstable condensation observed at high pool temperatures, the local pool temperature limit for RV/SRV discharge is specified in NUREG-0783. Elimination of this limit for plants with quenchers on the RV/SRV discharge lines is justified in GE report NEDO-30832, "Elimination of Limit on Local Suppression Pool Temperature for SRV Discharge with Quenchers." In a safety evaluation report dated August 29, 1994, the staff eliminated the maximum local pool temperature limit for plants with quenchers, provided that steam entrainment in the ECCS suction is not a concern. The licensee indicated that for QCNPS, an evaluation for the worst case geometry, where the quencher and the ECCS suction strainers are located in the same sections (i.e., bays) in the torus, has been performed. The licensee provided details of the EPU evaluation of the likelihood of steam ingestion in Reference 19. The evaluation conservatively assumed that the water is locally saturated in the vicinity of the SRV quenchers and ECCS suction strainers, that all ECCS pumps were operating, and that there was full SRV discharge flow. The licensee quantified the size of the SRV steam plume and envelope of flow drawn into the strainer. Since the evaluation shows that the steam plume will not intersect the flow envelope, steam ingestion into the ECCS suction is not a concern.

Based on the review of the licensee's rationale and evaluation, the staff concurs with the licensee's conclusion that the plant operations at the EPU will have no impact on the local pool temperature with RV/SRV discharge.

4.1.1.2 Containment Airspace Temperature Response

The containment airspace temperature limit of 340° F was based on a bounding analyses of the superheated gas temperature that can be reached with blowdown of steam to the drywell during a DBA-LOCA. Using a new methodology for EPU, the licensee calculated the peak DBA-LOCA drywell gas temperature to be 291° F at the EPU conditions, which remains below the drywell airspace design temperature of 340° F. The current licensing basis analysis had calculated a temperature of 290° F. Using the EPU methodology, the peak drywell air temperature for current rated power is 289° F, which is 2° F lower than the calculated temperature at EPU power. The EPU peak DBA-LOCA drywell air temperature is 10° F above the shell design temperature of 281° F; however, the brief~~10 second~~ duration above design temperature (less than 10 seconds) is not long enough to bring the shell temperature above its design value.

The licensee indicated that the limiting design basis accident with respect to peak drywell temperature is a steam line break. A steam line break produces a higher drywell temperature response than the DBA-LOCA (liquid line break) because the steam has a higher energy content than liquid at the same pressure. The licensee provided additional detail describing the limiting steam line break in their letter dated August 14, 2001 (Reference 24). The licensee analyzed four break sizes ranging from .01 to 0.75 ft^2 for the EPU conditions. The peak drywell airspace temperature was determined to be 337.9° F, which remains below the 340° F temperature limit, and the peak drywell shell temperature was determined to be 277.9° F, which remains below its 281° F design temperature. The peak drywell airspace temperature occurs early in the event of a steam line break and before drywell spray initiation at 600 seconds; therefore, the licensee stated it is relatively insensitive to power level. The drywell shell temperature is calculated to rapidly rise to the saturation temperature for the steam partial pressure in the drywell (around 277° F), and continue a slower increase due to natural convection from the hotter drywell airspace temperature. The temperature rise is terminated with the initiation of drywell sprays.

The licensee stated that review of results for DBA-LOCA and steam line breaks analyzed at EPU conditions shows that the DBA-LOCA is the limiting event for the wetwell airspace and suppression pool temperatures. The analyses for DBA-LOCA calculated a peak wetwell air space temperature of 257° F, which occurs during the blowdown period. In the early phase of the DBA-LOCA, non-condensable gas in the drywell is transported to the wetwell. Compression effects cause the airspace temperature to increase above the suppression pool temperature. Previous UFSAR analyses had assumed thermal equilibrium. The peak calculated wetwell airspace temperature remains below the wetwell structural limit of 281° F for the EPU; and is unchanged from that temperature calculated with current power levels and current methods.

Therefore, the drywell and wetwell air temperature response has no adverse impact on the containment.

Based on the review of the licensee's evaluation, the staff concurs with the licensee's conclusion that the drywell and wetwell air temperature response will remain acceptable after the extended power uprate.

4.1.1.3 Containment Pressure Response

The licensee indicated that the short-term containment response analysis was performed for the limiting DBA-LOCA, which assumes a double ended guillotine break of a recirculation suction line, to demonstrate that operation at the EPU level does not result in exceeding the containment design pressure limits. The short-term analysis covers the blowdown period during which the maximum drywell pressures and maximum differential pressures between the drywell and wetwell occur. These analyses were performed at 102% of EPU RTP per Regulatory Guide 1.49, with the break flow calculated by using a more detailed model than used for previous licensing basis analyses. Use of the GE NEDE-20566-P-A model for LOCA Analyses in accordance with 10 CFR 50 Appendix K was addressed in topical report ELTR1. These analyses calculated a peak drywell pressure of 43.9 psig at EPU, which remains below the containment design value of 62 psig. The licensee noted that this represents a reduction from the current UFSAR analysis results of 47 psig. However; comparing the analyses results obtained from the current rated power using the current methods (42.8 psig) shows an increase of only 1.1 psig peak drywell pressure resulting from the EPU.

The DBA-LOCA analysis wetwell pressure peaks at 36.7 psig during the early phase of the transient due to compression effects of non-condensable gases. This is well below the maximum allowable internal pressure of 62 psig. The peak is 9.7 psig higher than that calculated with current UFSAR methods, because those methods assumed thermal equilibrium between the wetwell pool and associate airspace. However; comparing the analyses results obtained from the current rated power using the same methods as for EPU (36.6 psig) shows an insignificant increase of only 0.1 psig in the peak wetwell pressure resulting from the EPU power.

The current value of peak calculated primary containment internal pressure for the design basis accident (P_a) used for containment testing is 48.0 psig. The licensee has proposed technical specifications changes to decrease this value to 43.9 psig based on the above pressure response for EPU per 10 CFR Part 50 Appendix J. In response (Reference 19) to the staff's request for additional information, the licensee provided a draft of proposed UFSAR Section 6.2.1.3 which is consistent with the EPU application for this change and that is referenced as the basis for Technical Specification B 3.6.1.4. The staff finds the proposed technical specification change acceptable.

Based on the review of the licensee's evaluation, the staff concurs with the licensee's conclusion that the containment pressure response following a postulated LOCA will remain acceptable after the extended power uprate.

4.1.2 Containment Dynamic Loads

4.1.2.1 LOCA Containment Dynamic Loads

The licensee indicated that the LOCA containment dynamic loads analysis for the EPU is based primarily on the short-term recirculation suction line break DBA-LOCA analyses. These analyses were performed similarly to the analysis described above in Section 4.1.1.3 using the Mark I Containment Long Term Program method, except the break flow is calculated using the more detailed reactor pressure vessel model of NEDE-20566-P-A, GE model for LOCA Analyses in accordance with 10 CFR 50 Appendix K. These analyses provide calculated values for the controlling parameters for the dynamic loads throughout the blowdown. The key parameters are the drywell and wetwell pressures, vent flow rates, and suppression pool temperature. The LOCA dynamic loads with the EPU include pool swell, condensation oscillation, and chugging. For a Mark I plant like QCNPS, the vent thrust loads are also evaluated.

The licensee stated that the short-term containment response conditions with the EPU are within the range of test conditions used to define the pool swell and condensation oscillation loads for the plant. The long-term response conditions with EPU for times beyond the initial blowdown period, in which chugging would occur, are within the conditions used to define the chugging loads. The licensee also indicated that the vent thrust loads with the EPU are calculated to be less than the plant-specific values calculated during the Mark I Containment Long Term Program. Therefore, the pool swell, condensation oscillation, chugging loads, and vent thrust loads for the EPU remain bounded by the existing load definitions.

Based on the review of the licensee's rationale and evaluation, the staff concurs with the licensee's conclusion that the LOCA containment dynamic loads will remain acceptable after the extended power uprate.

4.1.2.2 Relief Plus Safety Relief Valve Loads

The relief valve/safety relief valve (RV/SRV) air-clearing loads include discharge line loads, suppression pool boundary pressure loads, and drag loads on submerged structures. These loads are influenced by the RV/SRV opening pressure setpoint, the initial water leg height in the discharge line, the discharge line geometry, and suppression pool geometry. For the first RV/SRV actuations, the only parameter change which can affect the SRV loads that could be introduced by the EPU is an increase in the opening pressure setpoint. This EPU does not include an increase in the opening setpoint pressures; therefore, it has no effect on the loads from the first RV/SRV actuations.

After RV/SRV closure, water refloods the discharge line; condenses steam; creates a low pressure which causes the vacuum breaker to open, allowing water level in the discharge line to decrease. The licensee indicated that to mitigate the effects of subsequent SRV actuations for the existing design, a timer setting (longer than the minimum time) has been built into the QCNPS RV/SRV control logic. This timer extends the time between the SRV closure and subsequent re-opening, ensuring that the water column height during subsequent actuations has been reduced such that re-actuation loads are acceptable. The EPU has no impact on the calculated minimum time intervals between RV/SRV openings; which is based on time, vacuum breaker capacity, and reflood height. Therefore, RV/SRV loads remain bounded by the existing load definition.

Based on the its review of the licensee's rationale and evaluation, the staff concludes that plant operation at the EPU will have insignificant or no impact on the SRV containment loads.

4.1.2.3 Subcompartment Pressurization

The licensee indicated that because the EPU does not include a reactor operating pressure increase, there is only a minor increase in the asymmetrical loads on the vessel, attached piping and biological shield wall, due to a postulated pipe break in the annulus between the reactor vessel and biological shield wall. The results of the updated calculations including the effects of the EPU indicate that the biological shield wall and component designs remain adequate, because there is sufficient pressure margin available.

Based on the its review of the licensee's rationale and evaluation, the staff concludes that plant operation at the EPU will have an insignificant impact on the subcompartment pressurization.

4.1.3 Containment Isolation

The licensee indicated that the system designs for containment isolation have been evaluated for the EPU conditions. The capability of the actuation devices to perform with the higher flow and temperature during normal operations and under post-accident conditions has been determined to be acceptable.

Based on the its review of the licensee's rationale and evaluation, the staff concludes that plant operations at EPU will have an insignificant or no impact on the containment isolation system.

4.1.4 Generic Letter 96-06

The licensee indicated that a review of the plant's past response to Generic Letter (GL) 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions" was conducted to assess the impact of the EPU. The containment analyses demonstrates that the original post-accident containment temperatures increase slightly.

Based on review of the containment pressure and temperature conditions during design-basis accidents, the staff concurs with the licensee's conclusion that the past response to GL 96-06 remains valid for the EPU. **Update per 9/27/1 letter**

4.2 Emergency Core Cooling System

The emergency core cooling system (ECCS) components are designed to provide protection in the event of a loss-of-coolant accident (LOCA) due to a rupture of the primary system piping. Although design basis accidents (DBAs) are not expected to occur during the lifetime of a plant, plants are designed and analyzed to ensure that the radiological dose from a DBA will not exceed the 10 CFR Part 100 limits. For a LOCA, 10 CFR 50.46 specifies design acceptance criteria based on: (a) the peak cladding temperature (PCT), (b) local cladding oxidation, (c) total hydrogen generation, (d) coolable core geometry, and (e) long-term cooling. The LOCA analysis considers a spectrum of break sizes and locations, including a rapid circumferential rupture of the largest recirculation system pipe. Assuming a single failure of the ECCS, the LOCA analyses identifies the break sizes that most severely challenge the ECCS systems and the primary containment. The maximum average planar linear heat generation rate (MAPLHGR) operating limit is based on the most limiting LOCA analysis, and licensees perform

LOCA analyses for each new fuel type to demonstrate that the 10 CFR 50.46 acceptance criteria can be met.

The ECCS for QCNPS includes the high pressure coolant injection system (HPCI), the low pressure coolant injection system (LPCI) mode of the RHR~~LPCI/CC~~, the core spray (CS) system and the automatic depressurization system (ADS). ECCS performance is discussed in Section 4.3.

4.2.1 High Pressure Coolant Injection System.

The HPCI system (in conjunction with with other ECCS systems ~~as backup~~) is designed to maintain reactor water level inventory during small and intermediate break LOCA, isolation transients and LOFWF. For a large break LOCA, the reactor will depressurize rapidly, thereby rendering the HPCI system inoperable.

The HPCI system is required to start and operate reliably over its design operating range. During the LOFWF event, and isolation transients, ~~either~~ the ~~IC or~~ HPCI will maintain water level above the TAF ~~in the event that the RCIC system is unavailable~~. For the MSIV closure, the RVs open and close as required to control pressure and HPCI will eventually restore water level.

The licensee evaluated the capability of the HPCI system, for operation at the EPU power level, to provide core cooling to the reactor to prevent excessive fuel PCT following small and intermediate break LOCA, and to ensure core coverage up to the TAF in isolation transients and LOFWF transients. The licensee stated that the HPCI evaluation is applicable to and is consistent with the evaluation in Section 4.2 of ELTR2. The licensee determined that the HPCI system is acceptable for the EPU.

The generic evaluation in Section 4.2 of the supplement to ELTR2 is based on typical HPCI pump design pressures. The licensee evaluated the capability of the HPCI system to perform as designed and analyzed its performance at the EPU conditions, and concluded that HPCI system can start and inject the required amount of coolant into the reactor for the range of reactor pressures associated with LOCA and isolation transients. The staff concludes that the HPCI system is acceptable for EPU conditions.

4.2.2 Low Pressure Coolant Injection

The Low Pressure Coolant Injection (LPCI) mode of the RHR~~LPCI/CC~~ system is automatically initiated in the event of a LOCA and, in conjunction with other ECCS systems, the LPCI mode is required to provide adequate core cooling for all LOCA events. The licensee further stated that the existing system has the capability to perform the design injection function of the LPCI mode for operation at the EPU condition and that the generic evaluation in Section 4.1 of ELTR2 bounds the QCNPS LPCI system performance. The staff finds the evaluation acceptable.

4.2.3 Core Spray System

The core spray (CS) system initiates automatically in the event of a LOCA and in conjunction with other ECCS systems, the CS system provides adequate core cooling for all LOCA events.

The licensee stated that, as indicated in the ECCS performance discussion in Section 4.3, the calculated LOCA PCT could increase slightly at the EPU. However, the existing CS system, combined with other ECCS systems, will provide adequate long term post-LOCA core cooling. The licensee added the existing CS system hardware has the capability to perform its design injection function at the EPU conditions and that the generic evaluation in Section 4.1 of ELTR2 bounds the QCNPS CS system performance. The staff finds this acceptable.

4.2.4 Automatic Depressurization System (ADS)

The Automatic Depressurization System (ADS) uses the safety/relief valves (S&RVs) to reduce reactor pressure after a small-break LOCA, allowing the LPCI and CS systems to provide cooling flow to the vessel. The plant design requires the RVs and the SRV to have a minimum flow capacity. After a specified delay, the ADS actuates either on low water level plus high drywell pressure or on sustained low water level alone. The licensee stated that the ability of the ADS to initiate on appropriate signals is not affected by the power uprate. The EPU decay heat is higher, increasing the required flow capacity. The licensee stated that the increase in the required flow capacity requires five ADS valves to be operable, instead of the current requirement of four ADS valves. The staff accepts the licensee's evaluation.

4.2.5 Net Positive Suction Head

The licensee indicated that the containment analysis for the net positive suction head (NPSH) was performed for DBA-LOCA at 102% of EPU RTP, using the ANS 5.1+ two sigma decay heat with fuel exposure applicable for GE14 fuel with a 24-month fuel cycle. The results of the analysis determined that additional credit for containment overpressure, as compared with the current license condition B.2, is required because the suppression pool temperature increases at a faster rate and peaks at a higher value compared with the pre-EPU conditions during a LOCA. The increase in suppression pool temperature from EPU decay heat load results in increased vapor pressure, reducing the available suction head available for the ECCS pumps.

In their letter dated August 13, 2001, (Reference 23) in response to the staff request for additional information, the licensee stated that overpressure credit for Quad Cities Units 1 and 2 would be handled in a **future submittal**. The extended power uprate for these units should not be authorized until the staff has reviewed and approved the overpressure credit submittals. Staff review of **adequacy of NPSH is an open review item.** **Update; the information was provided by 9/25/1 submittal.**

4.3 <u>ECCS Performance Evaluation</u>

The Emergency Core Cooling System (ECCS) is designed to provide protection against postulated LOCAs caused by ruptures in the primary system piping. The ECCS performance under all LOCA conditions and the analysis models must satisfy the requirements of 10 CFR 50.46 and 10 CFR Part 50, Appendix K.

The licensee performed the LOCA analysis at 102 percent of the EPU RTP, using GE-14 fuel. The ECCS-LOCA analysis was based on the NRC-approved methodology (SAFER/GESTR).

The licensee determined the licensing basis PCT at the current rated core operating conditions using the standard adder required by the SAFER/GESTR methodology~~with an adder to account for the uncertainties~~. For the EPU conditions, the licensing basis PCT, based on the limiting GE-14 fuel design, is less than 10F different at rated core flow in comparison with the pre- EPU PCT.

For SLO conditions, the licensee applied a multiplier to the normal two loop operation MAPLHGR limits. The licensee stated that the multiplier to the MAPLHGR for the SLO operation ensures that the SLO nominal PCT is less than the PCT for the nominal two loop operation. ~~Section 12~~ Attachment 1 discusses the findings from the staff audit of these calculations and the licensee response.

The licensee determined that the ECCS performance under LOCA conditions, and the analysis models, satisfy the requirements of 10 CFR 50.46 and Appendix K.

As part of the EPU review process, the NRC staff audited the QCNPS LOCA analysis. The staff focused on the GNF use of the LOCA codes and their applicability to the QCNPS EPU. The staff examined design record files (DRF) describing the LOCA analyses, covering both the pre- and post-EPU analyses, and made the following observations:

1. The analyses were based on the NRC approved SAFER/GESTR methodology and GNF followed NRC approved process in performing the ECCS-LOCA analysis.

2. QCNPS was closely involved in the development of -the plant-specific information required by GNF in developing the model.

3. The ECCS-LOCA analyses results showed compliance with the requirements of 10CFR50.46.

4. The GNF method for single loop operation uses statistical adders derived from RTP operation. The staff had questioned this approach in a prior audit and GNF had responded that any uncertainty introduced by using these values will be compensated for by the conservative nature of the SLO application procedure. This procedure leads analysts to derive conservative SLO multipliers. After further review, the staff accepts this explanation and concurs with the GNF conclusion.

The staff concludes that the QCNPS ECCS-LOCA performance complies with 10 CFR 50.46 and Appendix K requirements, and that the analyses were performed with NRC-approved methods and codes.

4.4 Standby Gas Treatment System

The standby gas treatment system (SGTS) is designed to process the secondary containment atmosphere and exhaust it through the plant chimney to limit the release to the environment of radioisotopes that may leak from primary containment under accident conditions. The capacity of the SGTS was selected to provide a negative differential pressure between secondary containment and the outside environment of at least 0.25-inch of water. The licensee stated that this capability is not affected by the EPU.

The licensee stated that the SGTS charcoal filter removal efficiency of 95% for radioiodine is not affected by the EPU. Post-LOCA total iodine loading increases from 6.0 mg/gm to 11.8 mg/gm of activated carbon at EPU conditions, using conservative RG 1.3 assumptions for the iodine chemical form and transport within containment. Despite the increase in iodine loading as a result of the EPU and 24-month fuel cycle, test work at high iodine loading supports filter removal efficiencies in excess of 99% at 60 mg/gm. The chemical form of iodine release based on RG 1.3 is assumed to be primarily composed of elemental and organic iodine that require treatment using activated carbon filtration.

In response to the staff, the licensee stated (Reference 24) that an industry study demonstrated charcoal filter removal efficiencies of over 99% for elemental iodine (which comprises 91% of the evaluated inventory) can be achieved with iodine loading as high as 60 mg/gm, even under adverse waterlogged conditions. The licensee further stated that for organic iodine (which comprises 4% of the evaluated inventory), an industry study demonstrated filter removal efficiencies of 99% with iodine loading as high as 4.4 mg/gm. This is approximately a factor of ten higher than the organic iodine loading of 0.47 mg/gm for the EPU. Therefore, the charcoal loadings from both elemental and organic iodine at EPU conditions are well below values that yield a filter removal efficiency of at least 99%. In addition, the design-basis HEPA filter efficiency of 99% for removal of particulate iodine is not affected by the small increase in iodine loading at EPU conditions.

In order to obtain reasonable assurance of the licensee's assertions, the staff reviewed Oak Ridge National Laboratory (ORNL) Reports ORNL-4180, "Removal of Radioactive Methyl Iodide from Steam-Air Systems (Test Series II)," dated October 1967, and ORNL-TM-2040, "Removal of Elemental Radioiodine from Flowing Humid Air by Iodized Charcoals," dated November 2, 1967. The staff found that the licensee's assertions are consistent with industry studies.

The licensee stated (Reference 24) that the testing and maintenance criteria for SGTS filters based on RG 1.52 (Revision 2) continue to be met in accordance with plant regulatory requirements.

The licensee stated that the amount of cooling airflow needed to limit the temperature increase of the charcoal adsorber due to fission product decay heating is affected by the EPU. However, although the minimum cooling airflow increased from 48 scfm to 74 scfm, it is well below the available design flow of 300 scfm. The licensee stated that no other SGTS parameter is affected by the EPU.

Based on the staff's review of the licensee's rationale, and the experience gained from review of power uprate applications for other BWR plants, the staff concludes that the EPU does not adversely affect operation of the SGTS.

4.5 Other Engineered Safety Features Systems

4.5.1 Post-LOCA Combustible Gas Control System

The licensee indicated that the post-LOCA control of hydrogen and oxygen concentrations is provided by the combustible gas control system (CGCS). The CGCS consists of several subsystems: the primary containment inerting system, the nitrogen containment atmosphere dilution system, the containment atmosphere monitoring system, and the augmented primary containment venting system. The CGCS is designed to maintain the post-LOCA containment atmosphere below hydrogen flammability limits by controlling the concentration of oxygen to not exceed 5% by volume. Design of the system is based on the production of hydrogen from 1) metal-water reaction of active fuel cladding, 2) corrosion of zinc and aluminum exposed to water during a postulated LOCA, and 3) radiolysis of water. The EPU only affects post-LOCA production of hydrogen and oxygen from radiolysis, which will increase in proportion to the EPU power level. The hydrogen contribution from metal-water reaction of fuel cladding is additionally affected by the fuel design change. Therefore, the analysis considers the impact of GE14 fuel introduction on metal-water hydrogen production.

In Reference 19, the licensee supplemented their initial application with five graphs depicting parameters related to CGCS operation varying with time after a LOCA. Parameters depicted included hydrogen generation rates, hydrogen and oxygen concentrations (with and without nitrogen dilution), cumulative nitrogen usage, and containment pressure (with and without nitrogen injection).

The licensee indicated that the time required to reach the 5% oxygen limit following the LOCA, based on 1 percent per day containment leakage, decreases from 25 hours for current reactor power to 19 hours for EPU reactor power with GE-14 fuel. This reduction in time required for nitrogen containment atmosphere dilution (CAD) system initiation does not affect the ability of the operators to respond. Therefore, the CGCS retains its capability of meeting its design basis function of controlling oxygen concentration following the postulated LOCA.

Evaluation of the nitrogen requirements to maintain the containment atmosphere below the 5% flammability limit for seven days post-LOCA shows that the minimum stored volume increases to 141,000 scf for EPU reactor power. The licensee indicated that the CAD system has a minimum stored nitrogen capacity of 200,000 scf, which is sufficient to accommodate seven days of post-LOCA operation. The licensee additionally calculated that the containment pressure build-up as a result of CAD system operation shows that the operating pressure limit of 31 psig (50 percent of the design pressure) is not reached until 32 days after the LOCA. This satisfies the minimum 30-day acceptance limit for containment pressure buildup.

In a letter dated August 13, 2001, providing additional information (Reference 23), the licensee addressed why technical specifications should not be added for the CAD system. The licensee noted that during the recent conversion to BWR improved Standard Technical Specifications (iSTS), the staff did not require addition of the iSTS 3.6.3.4 on CAD System because it was not part of the current licensing basis for Quad Cities. Additionally, the licensee noted that no new technical specifications requirements were deemed necessary since the staff had previously determined, in a safety evaluation dated July 8, 1996, that it was acceptable to control operability requirements for the Dresden Units ~~1~~2 and 3~~2~~ CAD system in licensee-controlled procedures and administrative controls. The staff concludes, notwithstanding the slightly increased oxygen generation rate following EPU and the increased hydrogen generation associated with GE-14 fuel, that technical specifications are not needed for the CAD system.

In response to the staff, the licensee addressed the capacity of the containment hardened vent considering EPU conditions (Reference 19). One of the design inputs for the hardened wetwell vent was the ability to exhaust energy equivalent to 1% RTP. The design of the hardened wetwell vent was based on the current power level. Based on the as-built design, the hardened wetwell vent will exhaust approximately 0.85% at 2957 MWt. The licensee indicated that the primary objective of the hardened wetwell vent is to preclude primary containment failure due to overpressurization, given a loss of decay heat removal event. Using the EPU decay heat curve, 0.85 RTP is reached at approximately 5.6 hours. Based on loss of decay heat removal event, the containment will not reach the primary containment pressure limit until 20 hours. Therefore, the design of the existing hardened wetwell continues to be acceptable for preventing containment overpressure at the EPU conditions.

In Reference 23, the licensee addressed the EPU effect on technical specification section 3.6.2.5 on drywell-suppression pool differential pressure and on technical specification limiting condition for operation 3.6.3.1 on primary containment oxygen concentration. The technical specifications are applicable in Mode 1 from 24 hours after exceeding 15% RTP on a startup and until 24 hours before reducing RTP below 15% for a shutdown. The licensee noted that the 15% RTP value relates to the window for relaxed de-inerting requirements for the primary containment. The basis for the relaxation remains the low probability of an event that generates hydrogen during these time periods and would remain applicable with the new EPU RTP which is 17.8% higher.

Based on a review of the licensee's rationale, the staff concurs with the licensee's conclusion that the references to 15% RTP should be retained for technical specifications 3.6.2.5 and 3.6.3.1.

Based on the review of the licensee's rationale and evaluation, the staff concurs with the licensee's conclusion that plant operations at the proposed uprate power level, combined with use of GE-14 fuel, will have a minor impact on the post-LOCA combustible gas control system and the nitrogen containment atmosphere dilution system will remain acceptable.

4.5.2 Main Control Room Atmosphere Control System

The main control room atmosphere control system (MCRACS) processes the control room intake atmosphere to limit the release of radioisotopes to the control room that may leak from containment under DBA-LOCA conditions. The capacity of the MCRACS (also designated control room emergency ventilation system in plant technical specifications) provides a positive differential pressure between the control room and the outside environment to minimize the potential for unprocessed in-leakage into the control room.

The licensee stated that the increase in heat gain to the control room resulting from the EPU for both normal and emergency modes is insignificant. By letter date August 14, 2001, in response to the staff (Reference 24), the licensee explained that the heat load increases resulting from the EPU do not adversely impact the MCRACS, since these increases occur outside the control room areas. Major control devices in the control room remain unchanged. The small electrical currents transmitted to some indicating devices in the control room increase because of higher process temperature and electrical loads. The associated minor heat load increases from these electrical signals have an insignificant effect on the pre-EPU design margin of the MCRACS in both the normal and emergency modes.

The licensee stated that the only EPU effect on the MCRACS results from an increase in the radioiodine released during a DBA-LOCA. The licensee evaluated the effect of the EPU, in combination with a 24-month fuel cycle, on the post-LOCA iodine loading of the control room charcoal filters. The post-LOCA iodine releases collected on the control room intake filters at EPU conditions was estimated using the 0-2 hour X/Q values for the entire duration of the event, assuming no deposition or holdup of iodines in the main steam lines or in the secondary containment. Despite the increase in iodine loading as a result of the EPU and 24-month fuel cycle, the iodine loading on the control room filters remains a small fraction of the allowable limit of 2.5 mg of total iodine (radioactive plus stable) per gram of activated carbon, identified in RG 1.52. Therefore, the control room filter efficiency is not affected by the EPU. The licensee stated that the technical support center is not affected by the EPU.

In response to the staff (Reference 24), the licensee described the evaluation and its assumptions utilized in determining the effect of the EPU and 24-month fuel cycle on the post-LOCA iodine loading of the control room charcoal filters. Based on docketed information provided by the licensee, the staff concludes that the evaluation and its assumptions are acceptable. The licensee also pointed out that the iodine loading on the control room filters for QCNPS is calculated to be 2.26E-3 mg of total iodine per gm of activated carbon, and that this iodine loading is a small fraction of the above design limit identified in RG 1.52. The licensee further stated that the control room filter efficiency of 99% associated with the MCRACS HEPA and charcoal filters continues to be effective under EPU conditions.

The licensee also stated in Reference 24 that the existing commitments to regulatory requirements and guidelines included in the design-basis for the MCRACS are unchanged for the EPU. The requirements and guidelines include 10 CFR Part 50, Appendix A, General Design Criteria 19, Regulatory Guide 1.52 (Revision 2) and Standard Review Plan 6.4. The regulatory requirements of Generic Letter 99-02 are also met. <u>We did not discuss compliance with GL 99-02 on the docket for EPU. I assume this was obtained from other sources, such as the GL response. We do meet GL 99-02.</u>

Based on the staff's review of the licensee's rationale, and the experience gained from review of power uprate applications for other BWR plants, the staff concludes that the EPU does not adversely affect the operation of the MCRACS.

4.5.3 Standby Coolant Supply System

Update per 9/19/1 submittal

5.0 <u>INSTRUMENTATION AND CONTROL</u>

5.1 <u>Nuclear Steam Supply System (NSSS) Monitoring and Control Systems</u>

[redacted]

[redacted] Sidebarred in the PUSAR The submittal used GE[redacted] methodologies (References 39 and 40) to generate the allowable values and the nominal trip setpoints, based on the changes in the analytical limits shown in Table 5-1. The table lists the instrument setpoint analytical limits for pre-EPU and EPU operation and identifies changes based on results from the EPU safety analyses.

The following section discusses the effect of the EPU on the analytical limits and the instrumentation setpoints.

5.1.1 Suitability Of Existing Nuclear Steam Supply System (NSSS) and Balance-of-Plant (BOP) Instruments

For the proposed power uprate, each existing instrument of the affected nuclear steam supply systems (NSSS) and balance-of-plant (BOP) systems was evaluated by the licensee to determine its suitability for the revised operating range of the affected process parameters. Where operation at the power-uprated condition impacted safety analysis limits, the evaluation verified that the acceptable safety margin continued to exist under all conditions of the power uprate. Where necessary, setpoint and uncertainty calculations for the affected instruments were revised. Apart from the few devices that needed change, the licensee's evaluations found most of the existing instrumentation acceptable for the proposed power uprate operation. The evaluations resulted in the following changes:

- Modify the tripping logic of the fourth condensate pump on LOCA to allow the continued use of the feedwater pumps.

  Implement reactor recirculation pump runback on loss of feedwater flow or loss of a condensate pump to reduce the potential for a scram on reactor low water level and allow continued operation.
- Replace the average power range monitor (APRM) flow control trip reference card to add the clamp function for the APRM flow-biased rod block.
- Install an additional steam line steam resonance compensator card designed to attenuate third-order harmonics in the electro-hydraulic control system to reduce electrical noise in the system.
- Replace the main steam line flow-high differential pressure indicating switches to accommodate the new setpoint.
- Expand the indicating range on various control room and in-plant instrumentation.
- Replace the offgas condenser outlet gas temperature switches to accommodate the new span.

These changes will be made to accommodate the revised process parameters. The staff concludes that **when the above-noted modifications and changes are implemented** during the next refueling outage, the Quad Cities instrumentation and control systems will accommodate the proposed power uprate without compromising safety.

5.1.2 Neutron Monitoring System

The licensee will rescale the fixed averaged power range monitor (APRM) power signal to the EPU RTP, such that the indications will read 100 percent at uprated power level. The licensee stated that the EPU would have little effect on the intermediate range monitor (IRM) overlap with the source range monitor (SRM) and the APRMs. The licensee will use the normal plant surveillance procedures to adjust the IRM overlap with the SRMs and APRMs. The licensee's Safety Analysis Report (Reference 2) further stated that the APRM downscale setting does not need to change. The EPU would affect the neutronic life of the local power range monitor (LPRM) detectors and the radiation levels of the TIPs, but the effects would be expected to be very small. Operation at the higher MELLLA rod line will affect the IRM overlap and the staff accepts the licensee's plan to adjust the overlap for the EPU condition to ensure adequate reactor monitoring.

The rod block monitor (RBM) is required to be operable at 30 percent of the rated power and limits the effects of local reactivity insertion due to erroneous control rod withdrawal, by inhibiting further withdrawal. As shown in Table 5-1 of the licensee's Safety Analysis Report, the RBM upscale trips are based on thermal power ranges, as discussed below.

5.2 Reactor Protection System/Engineered Safety Features Actuation System Instrumentation Trip Setpoint and Allowable Values

The instrumentation setpoints are determined based on plant operating experience, conservative licensing analysis, and/or limiting design/operating value. The licensee stated that the instrumentation setpoints in the QCNPS TS are established using the GENE setpoint methodology for the APRM setpoint and the ComEd setpoint methodology for the others. Each setpoint is selected with sufficient margin between the actual trip setting and the value used in the safety analysis (the analytical limit) to allow for instrument accuracy, calibration and drift. To avoid inadvertent initiation of the protective actions (spurious trip avoidance), sufficient margin is established, whenever possible, between the actual trip setting and the normal operating limit (Table 5-1 of Reference 2). If the analytical limit does not change based on the results of the EPU safety analyses, then the associated plant setpoints and the nominal trip setpoints do not change.

The staff has previously reviewed both of these instrument setpoint methodologies and found them acceptable for establishing new setpoints in power uprate applications. However, the staff was concerned about the reduction of margin between the instrument setpoints, allowable values, and analytical limits and in a conference call on May 16, 2001, the staff requested the licensee to provide the changes in instrument setpoints and allowable values together with the analytical limits provided in Table 5-1 of the licensee's Safety Analysis Report (Reference 2). In its response dated June 15, 2001, (Reference 16) the licensee provided the table containing instrument setpoints, allowable values and analytical limits. Based on the review of this table the staff has determined that the proposed power uprate will not result in any significant reduction of margin. Therefore, the staff finds the licensee's response acceptable.

The proposed setpoint changes resulting from the power uprate are intended to maintain existing margins between operating conditions and the reactor trip setpoints and do not significantly increase the likelihood of a false trip or failure to trip upon demand. Therefore, the existing licensing basis is not affected by the setpoint changes to accommodate the power uprate.

5.2.1 High-Pressure Scram

If a pressurization transient is not terminated by direct valve position scram or high flux scram, the reactor would scram on the high pressure. The reactor vessel high-pressure scram signal setting is established slightly above the reactor vessel maximum normal operating pressure and below the specified analytical limit. This setting prevents spurious scrams during normal operation but provides adequate margin to the maximum allowable reactor vessel pressure. Since the steady state reactor pressure would not change with EPU, the reactor high-pressure scram analytical limit would also not change (Table 5-1).

5.2.2 High-Pressure Recirculation Pump Trip

The anticipated transient without scram (ATWS) recirculation pump trip (RPT) action trips the pumps for transients that result in increase in reactor vessel dome pressure and/or low reactor water level. The ATWS analyses assume that the recirculation pumps will trip at the analytical limit of 1250 psig. The primary function of the ATWS-RPT is to reduce core flow, insert negative reactivity through the generation of steam voids and reduce the core power during the initial portion of the ATWS event. Therefore, the high pressure ATWS-RPT setpoint is based on the value used in the ATWS analysis. The licensee also stated that the ATWS-RPT low reactor level setpoint is ~~a~~ not a significant factor for the limiting analyzed ATWS events.

The licensee used the current high-pressure ATWS-RPT setpoint in the EPU ATWS evaluation in Section 9.4.1 and stated that the calculated peak vessel pressure remains below the ASME upset limit of 1500 psig. The licensee concluded that the current EPU high-pressure ATWS-RPT setpoint is acceptable for the EPU and remains unchanged as indicated in Table 5-1. Since the EPU ATWS analyses demonstrated that the peak vessel pressure will remain below the ASME upset limit, assuming the current setpoint of 1250 psig, as stated by the licensee, the staff accepts that the setpoint limit does not need to change.

5.2.3 Safety Relief Valve

The licensee states that the SSV, SRV and RV analytical limits will not change, since the reactor operating pressure will not change, and therefore the ASME overpressure protection and pressurization transients are based on the current setpoints (Table 5-1). The staff accepts this evaluation.

5.2.4 Main Steam Line High Flow Isolation

Anything?????

5.2.5 Neutron Monitoring System

The licensee states that the analytical limit (AL) (as a percentage of RTP) for the fixed APRM high power scram will not change. Therefore, the licensee will maintain the values in the TS for the allowable value and for the nominal trip setpoints. The AV for the fixed APRM trip is being changed, as noted in Section 5.3.4. The AL will not change. This is consistent with Section F.4.2.2 of ELTR1. The licensee also evaluated all of the limiting transients that rely on the fixed APRM trip at the EPU conditions.

Since EPU operation will include implementing the MELLLA operational region, the licensee has developed new equations for the flow-biased APRM scrams, both for normal two recirculation loop and SLO operation. The licensee stated that the design bases for the MELLLA operating regime uses a linear relationship for all analytical limits versus the recirculation drive flow, which is consistent with the APRM hardware design and licensing analyses. According to the licensee, the ALs for the flow-biased APRM scrams are straight line equations, in which the slope was changed consistent with other BWR MELLLA applications. The licensee also maintains equivalent margins between the rod blocks and scram trip setpoints to avoid spurious protective actions. The flow-biased APRM scram analytical limits are also specified in Table 5-1 of the EPU submittal.

The staff reached the same conclusion.

The RBM limits erroneous control rod withdrawal by supplying a trip signal to the Reactor Manual Control System to block further withdrawal. The trip signal is initiated when the RBM output exceeds the rod block setpoint. The licensee stated that the setpoints are determined on a fuel cycle-specific basis and will be modified as needed. The TS SR threshold is unchanged at 30 percent RTP.

5.2.6 Reactor Water Level Instruments

Anything ????

5.2.7 Low Condenser Vacuum

Anything ?????

5.2.8 Turbine Stop Valve Closure and Turbine Control Valve Fast Closure Scram Bypass

Anything ?????

5.2.9 Rod Worth Minimizer

Anything ???

5.2.10 Pressure Regulator

Anything ?? ?

5.3 TS Setpoint Changes Related to the Power Uprate

The following TS changes have been proposed by the licensee:

1. TS Section 3.3.1.1, SR 3.3.1.1.2

   The licensee has proposed to remove the reference to the TS Section 3.2.4 which requires gain adjustment. The APRM gain and setpoint adjustment requirements are superceded by the APRM/ Rod Block Monitor (RBM) TS (ARTS) power and flow dependent limits. The staff's evaluation of the removal of TS Section 3.2.4 is discussed in **Section (XXX)** of this safety evaluation. On this basis, the staff finds the licensee's proposed TS change to be acceptable.
2. TS Section 3.3.1.1, SR 3.3.1.1.13, Required Action E.1, and TS Table 3.3.1.1-1, Functions 8 and 9

   The licensee has proposed to reduce from 45~~30~~% to 38.5~~26~~% the percentage-of-RTP value corresponding to the power level where the reactor protection system (RPS) trip on turbine stop valve (TSV) or on turbine control valve (TCV) fast closure is automatically bypassed ~~from 30% to 26%~~. The licensee's justification of this change is that these scram signals are automatically bypassed at a low power level when the turbine bypass steam flow capacity is sufficient to mitigate a TSV or TCV closure transient. Because the turbine bypass capacity is not being changed by this EPU, the corresponding percentage of RTP is being revised to maintain the current thermal power value in MWt, corresponding to the existing bypass steam flow capacity. On this basis, the staff finds the licensee's proposed TS change to be acceptable.

3. TS Table 3.3.1.1-1, Function 2.b

   The licensee has proposed to revise the APRM flow-biased scram equations for reactor recirculation two-loop and single-loop operation. The licensee has also raised the allowable value for the clamped portion of the APRM flow-biased neutron flux high from $\leq$120% to $\leq$122%. The staff's evaluation of the clamped portion of the allowable value is discussed in the next item of this safety evaluation. The APRM flow-biased trip function provides protection against transients where thermal power increases slowly. This function also protects fuel cladding integrity by ensuring that the minimum critical power ratio (MCPR) safety limit is not exceeded. Because of the lower scram trip setpoint, the APRM flow-biased trip will initiate a scram before the clamped allowable value is reached during any transient event that occurs at a reduced recirculation flow. These changes are necessary to ensure consistent operation with the maximum extended load line limit analysis (MELLLA) power/flow map. The staff's review of the MELLLA power/flow map is documented in **Section XXX** of this safety evaluation. Based on the acceptance of the operation with the MELLLA power/flow map, the staff finds the licensee's proposed TS change to be acceptable.

4. TS Table 3.3.1.1-1 Functions 2.b and 2.c

The licensee has proposed to revise the clamped portion of the allowable value for the APRM flow biased neutron flux high from ≤120% to ≤122%. The transient analysis for the power uprate is based on the analytical limit of 125% RTP. The APRM setpoint calculations determined that based on this analytical limit, an allowable value of 122% is appropriate and ensures that the analytical limit is maintained. On this basis, the staff finds the licensee's proposed TS change to be acceptable.

5. TS Table 3.3.1.1-1, Function 10

The licensee has proposed to revise the allowable value for the turbine condenser vacuum - low scram setpoint. The licensee has not revised the analytical limit for this function. Since the staff-accepted setpoint methodology has been used to calculate the allowable value, the transient analyses are not affected by this change. On this basis, the staff finds the licensee's proposed TS change to be acceptable.

6. TS Table 3.3.6.1-1, Function 1.d

The licensee has proposed to increase the allowable values for the main steam line flow-high isolation function. ~~The difference in the allowable value between the units is due to physical differences in the flow restrictors.~~ (This is a cut and paste from Dresden. There is no difference between the QC units.) Since the flow restrictors do not change the maximum steam flow, the proposed change decreases the difference between the allowable value and the maximum flow. The purpose of this instrumentation is to provide protection against pipe breaks in the main steam line outside the drywell. For a complete severance of one main steam line, steam flow increases almost instantaneously to the maximum rated steam flow as limited by the flow restrictors. Thus the present and proposed setpoint would be attained virtually at the same time and the consequences of the main steam line break remain unchanged. On this basis, the staff finds the licensee's proposed TS change to be acceptable.

7. TS Table 3.3.6.1-1, Functions 2.a, 5.b, and 6.b

The licensee has proposed to revise the allowable value for Reactor Vessel Water Level - Low from greater than or equal to 11.8 inches to greater than or equal to 3.8 inches for the following functions: primary containment isolation, Reactor Water Cleanup (RWCU) System isolation, and shutdown cooling isolation system isolation. **Add Low level amendment conclusions.** On this basis, the staff finds the licensee's proposed TS change to be acceptable.

8. TS Table 3.3.6.2-1, Function 1 and TS Table 3.3.7.1-1, Function 1

The licensee has proposed (Reference 26) to revise the allowable value for secondary containment isolation and control room emergency ventilation (CREV) system isolation on reactor vessel water level - low signal from ≥11.8 inches to ≥3.8 inches. The secondary containment isolation is initiated in order to minimize the potential of an offsite release and CREV system isolation is initiated to minimize the potential dose to control room operators. The licensee has chosen the allowable value for these functions to be the same as the allowable value for the reactor protection system and therefore has not analyzed it separately. Since the proposed change in the reactor scram setpoint does not result in a change to the current safety analyses, the change in the allowable value for the secondary containment isolation function continues to ensure that any offsite releases are within the limits calculated in the safety analysis. Also, the change in allowable value for the CREV system isolation function continues to ensure that the radiation exposure to control room operator does not exceed the limits set by General Design Criterion 19 "Control Room." On this basis, the staff finds the licensee's proposed TS change to be acceptable.

9. TS Table 3.3.7.1-1, Function 3

   The licensee has proposed (Reference 26) to revise the allowable value for main steam line flow - high from percent of rated steam flow to units of psid. The proposed change preserves the same allowable value in terms of percent of rated steam flow. However, because of the increase in rated steam flow at extended power uprate, the proposed change increases the actual mass flow rate of steam required to actuate the isolation function. Also since the maximum steam flow following steam line break does not change due to the flow restrictors, the proposed change results in a decrease in the difference between the allowable value and the maximum flow. However, because the purpose of the main steam line flow - high is to isolate main steam line for pipe break outside the drywell, the steam flow increases almost instantaneously to the maximum flow limited by the flow restrictors. Thus the change in setpoint does not impact the allotted time and the consequences of a design basis main steam line break remain unchanged with the change in high flow setpoint. On this basis, the staff finds the licensee's proposed TS change to be acceptable.

Based on the above review and justifications, the staff concludes that the licensee's instrument setpoint methodology and the resulting TS setpoint changes for the power uprate are consistent with the Quad Cities licensing basis and are, therefore, acceptable.

6.0 <u>ELECTRICAL POWER AND AUXILIARY SYSTEMS</u>

6.1 <u>AC Power</u>

6.1.1 Offsite Power System

The staff has reviewed information provided by the licensee to determine the impact of the power uprate on offsite power. Areas included in the review were grid stability analysis, and related electrical systems.

6.1.1.1 Grid Stability and Reliability Analysis

The licensee performed a grid stability uprate review to determine the adequacy of grid stability for the Quad Cities Unit 1 and 2 power uprate. The grid stability studies, considering the increase in electrical output, demonstrated conformance to 10 CFR 50, Appendix A, General Design Criteria (GDC) 17. GDC 17 addresses onsite and offsite electrical supply and distribution systems for safety-related components. There is no significant effect on grid stability or reliability. There is no modification associated with EPU that would increase electrical loads beyond those levels previously included, or revise the control logic of the distribution systems.

The staff requested that the licensee provide details about the grid stability analysis including major assumptions and results and conclusions of the analysis. In response to the staff request, the licensee stated (Reference 9) that GE PowerSystems Energy Consulting performed a study using a relative approach to determine the impact of the proposed plant uprates on the performance of the power system. System performance at the current plant outputs was determined first in order to establish the benchmark. Then the system performance with both units uprated was determined and compared to the benchmark. Both power flow and stability analyses were performed. The power flow analyzed the branch loading and bus voltage levels under normal and contingency operating conditions. The stability analysis evaluated both first swing stability and system damping. A variety of disturbance scenarios were analyzed, including single transmission line outages, single generating unit outages, double transmission line outages, double generating unit outages, and combined transmission line and generating unit outages. Additionally, the amount of reactive power (i.e., MVAR) support available in the system was also studied. It is expected that compensating measures will be required for MVAR support at certain times. Implementation of these compensating measures will be in accordance with the interconnection agreements and will be accomplished following completion of the current study by the Transmission and Distribution entity of the Exelon Energy Delivery Company (EDC).

The GE study for transient stability concluded that for all fault scenarios, system performance was stable with damped oscillation. The GE study for power flow analysis concluded that the EDC power grid will accommodate the uprate power flows for the planned 100 percent summer and winter peaks. As the power uprate implementation approaches, the Transmission and Distribution entity of EDC is reviewing the impact of the uprate on the power grid as currently configured. Resolution of any issues discovered during these reviews **will be accomplished prior to operation** at power uprate. EDC System Planning and Operating Guide ensures that adequate voltage is maintained at the QCNPS switchyard with either or both units shutdown. This assures that offsite power will be available to the units to meet the requirements of Appendix A to 10 CFR Part 50, "General Design Criteria for Nuclear Power Plants."

The licensee stated (Reference 18) that transmission and distribution entity of EDC has approved the connection of the uprated Dresden Nuclear Power Station (DNPS) Unit 2 and Quad Cities Nuclear Power Station Unit 2 to the power grid. These are the units that will connect to the grid under EPU conditions in the years 2001 and early 2002. The approval shows that sufficient MVAR support will be available. The approval of the remaining units will be obtained before the additional load is supplied to the grid. Additional MVAR support can be accomplished by having any of the generating units on the system (i.e., either Exelon Generation Company (EGC), LLC units or other units) reduce their MW output and increase their MVAR output.

Based on its review of the licensee's evaluation, the staff concludes that the proposed power uprate at QCNPS will not adversely affect the grid stability and reliability.

6.1.1.2 Related Electrical Systems

The licensee performed a power uprate review to determine the adequacy of electrical systems associated with the main turbine-generator auxiliary systems.

6.1.1.2.1 Main Generator

The existing main generator is rated at 960 MVA (912 MW), 0.95 power factor, 18 kV. With uprate the expected generator output is 912 MWe at 0.95 power factor which is within the capability of the generator. The review determined that the electrical system's configuration and operating voltage ranges are unchanged and remain adequate for operation at the higher output.

6.1.1.2.2 Isolated Phase Bus Duct

The existing isolated phase bus duct rating is 33000 amps for the main section and 2000 amps for the branch section. The maximum current output is 32,413 amps [960MVA/ (1.7321x18x0.95)] using generator output of 960 MVA and 95 percent of 18 kV. The review determined that the isolated phase bus duct is adequate for both rated voltage and low voltage current output.

6.1.1.2.3 Main Transformer

The existing main transformer rating is 985 MVA for Unit 1 and 952 MVA for Unit 2. The main power transformers and the associated switchyard components are adequate for the uprated output.

Thus, the turbine/generator and major electrical components extending from the isolated phase bus to the switchyard remain adequate for operation at the higher output.

6.1.2 Onsite Power Distribution System

The onsite power distribution system consists of transformers, buses, switchgear, and distribution panels. The alternating current (ac) power to the distribution system is provided from the transmission system or the onsite emergency diesel generators. Station batteries provide direct current (dc) power to the dc distribution system. Station loads under normal operation/distribution conditions are computed based on equipment nameplate data and calculated brake horsepower with actual diversity factor applied. The only significant change in electrical load demand is associated with condensate and booster pumps, reactor recirculation pumps, reactor feedwater pumps, and condensate demineralizers. The increased flow due to uprate conditions requires energizing the installed spare (third) reactor feedwater pump, energizing the installed spare (fourth) condensate and booster pump, and the increase of the operating point for the two reactor recirculation pumps. These additional loads when evaluated by design basis calculations result in acceptable operation of the electrical auxiliary system during normal startup and operation with two auxiliary transformers in service. However, operation at EPU conditions on a single transformer exceeds the non-safety 4 kV switchgear short-circuit rating, transformer winding rating, and bus duct rating. Also, in the event of a fast transfer to single transformer operation at EPU conditions, the same situation exists. To address this potential operational problem, the licensee will **institute a procedurally controlled load shedding scheme** to be implemented within **one hour** after a fast transfer. (This approach will be confirmed by thermal analysis or an engineering evaluation to address the overload conditions for the auxiliary transformers, the bus duct, and related connections. To address the potential operational problem due to the switchgear overduty condition, a test to upgrade the switchgear and breakers to a higher momentary current rating will be performed and a time delay of about 6 cycles on the short circuit interrupting will be implemented. In response to the staff's concern regarding a test to upgrade the switchgear and breaker to a higher momentary current rating, the licensee stated that non-safety-related 4 kV switchgear is manufactured by GE. The licensee contracted Pacific Breaker Systems, Inc. to specify the testing, procure the equipment, and perform the tests. The licensee provided adequate details regarding the tests. The licensee is currently working with GE Industrial Systems Division to provide the modifications and perform the final momentary test. After successful tests, the bracing in the field will be modified.) Update the preceding material. The licensee stated (Reference 18) that GE Industrial Systems Division performed the momentary rating test. The test applied current that had a first peak of ~~154.3~~154.8 kA for 17 cycles before being interrupted by the station breaker. The test was successful in demonstrating that, with the modified bracing, the switchgear and the breaker can meet the EPU momentary current requirements of 151.5 kA for the first peak. The bracing of the switchgear for the load cubicles will be modified to reflect the tested configuration. The six cycle time delay will be accomplished by disconnecting the instantaneous trip from the overcurrent protection for the load breakers. This modification will not affect the existing coordination.

Additionally, **a review** of the 4160 V bus and auxiliary transformer overcurrent relay setpoints **will be performed** (review was completed) to ensure proper settings for operation at EPU conditions. In response to the staff's request for additional information regarding relay setpoints and coordination, the licensee provided details (Reference 9). The licensee stated that the existing settings will remain the same and no changes are required.

The licensee stated that no increase in flow or pressure is required of any ac powered ECCS equipment for the EPU. Therefore, the amount of power required to perform safety-related functions (pumps and valves loads) is not increased with the EPU. The existing diesel generator load calculations are unchanged by the uprated conditions, and the current emergency power system design remains adequate. The system has sufficient capacity to support the required loads for safe-shutdown, to maintain a safe-shutdown condition, and to operate the required engineered safety feature equipment following a postulated accident.

Thus, the power uprate has no impact on the emergency onsite power system.

6.2 DC Power

The staff has reviewed information provided by the licensee to determine the impact of the EPU on the DC power system. The DC power distribution system provides control and motive power for various systems and components within the plant. The licensee noted that system loads are computed based on equipment nameplate data. Operation at the EPU RTP level does not increase any loads beyond nameplate rating or revise any control logic. The licensee stated that the DC power distribution system is adequate.

Based on the licensee's information, the staff concludes that the proposed EPU at QCNPS has no impact on the DC power system.

6.3 Fuel Pool Cooling

The fuel pool cooling and cleanup system (FPCCS) is important to safety in that it removes the decay heat released from stored irradiated fuel assemblies to maintain the pool water temperature at or below design temperature under normal operating conditions. For limiting conditions, the residual heat removal system can be aligned to the spent fuel pool (SFP) to provide supplemental cooling or rapid makeup water addition. Other makeup water systems are also available to maintain sufficient coolant inventory for operation of the cooling systems and to protect the fuel from damage following a sustained loss of forced cooling.

By increasing the amount of power produced in each fuel assembly and, therefore, the decay heat generated in each assembly, the EPU directly affects the decay heat generation rate in the SFP, the rate of temperature increase following a loss of cooling, and the rate of coolant loss if the pool reaches bulk boiling conditions. In their response dated August 13, 2001, (Reference 23) the licensee described changes in operating assumptions (i.e., rate of fuel transfer) and evaluation methods (i.e., credit for evaporative cooling) relative to those described in the QCNPS UFSAR. The increase in the rate of fuel transfer increases the peak decay heat rate in the SFP, while the credit for evaporative cooling reduces the conservatism in the evaluation of SFP conditions.

The licensee's bounding evaluation of SFP conditions for planned partial-core discharges was based on the decay heat calculated for [redacted] The staff found this method of decay heat rate determination acceptable, and the staff concluded that the applied assumptions were likely to bound future planned partial-core discharges. Because a full-core offload closer to the previous refueling discharge would produce a higher peak decay heat rate, the evaluated full-core offload would not bound all potential discharge scenarios. However, the calculated decay heat rate for the full-core offload and its associated boiloff rate of 78.5 gpm would likely be bounding for planned full-core offloads for refueling.

The licensee's evaluation also considered the following heat removal paths: [redacted] Because shutdown safety management [redacted] QCNPS maintain [redacted] the two units cross-connected, the licensee considered failure of an FPCCS pump the limiting single failure for planned offloads. All four FPCCS pumps were considered available for unplanned offloads. In their letter dated September 5, 2001, (Reference 31) **the licensee committed** to perform a cycle-specific analysis of SFP cooling capability if the two SFPs are not inter-connected and to implement procedural controls to ensure reactor building conditions are consistent with conditions assumed in the evaluation of credited evaporative cooling capacity. The staff concluded that the credited heat removal capability was sufficiently reliable for both planned refueling and unplanned maintenance offloads.

The licensee maintained SFP temperature acceptance criteria of 140° F for planned offloads and 150° F for unplanned offloads. The credited heat removal capacity was adequate to satisfy these acceptance criteria for the evaluated planned and unplanned offloads. In Reference 31, the **licensee also committed** to apply the same methods and acceptance criteria in evaluating planned offloads that are not bounded by the existing analysis. The staff found the analytical methods and SFP temperature limits acceptable for evaluation of refueling outage specific evaluations of SFP cooling capability.

Available makeup water capacity from each evaluated source continues to exceed the maximum calculated boiloff rate. The licensee stated (Reference 31) that, based on pump performance curves and estimated system resistance, the condensate transfer pump is capable of delivering over 275 gpm to the skimmer surge tank, which would overflow into the SFP if the FPCCS was not operating. The licensee also described the capability to deliver over 90 gpm through hoses on the refueling floor to the SFP from either the condensate transfer system, the clean demineralized water system, or the fire water system. The capability of these sources exceeds the peak calculated boiloff rate of 78.5 gpm, and the calculated minimum time for the SFP temperature to increase from 150□ F to 212□ F of 13.5 hours allows adequate time to align any of the above makeup sources. Therefore, the staff found the existing makeup water systems adequate for the EPU conditions.

Based on the staff's review of the licensee's rationale and evaluation, and the experience gained from review of power uprate applications for other BWR plants, the staff concludes that operation of the SFP cooling system at EPU conditions is acceptable.

6.4 Water Systems

6.4.1 Service Water Systems

The service water systems are designed to provide cooling water to various systems (both safety-related and non-safety-related systems).

6.4.1.1 Safety-Related Loads

The safety-related service water systems provide cooling water to the following essential components/systems: residual heat removal (RHR) heat exchangers, RHR pump seal coolers, RHR pump motor coolers, RHR heat exchanger room coolers, core spray room coolers, residual heat removal service water (RHRSW) pump cubicle coolers, diesel generator cooling water (DGCW) pump cubicle coolers, SFP emergency makeup (if needed), diesel generator cooling water heat exchangers, high pressure coolant injection (HPCI) room cooler, and the control room emergency ventilation system refrigeration condensing unit. All heat removed by these systems is rejected to the ultimate heat sink (section 6.4.5).

During shutdown cooling with the RHR system, heat loads on the RHR heat exchangers will increase proportionally to the increase in reactor operating power level, thus, increasing the time required to reach the shutdown temperature. The staff's evaluation of the effect of plant operations at the proposed EPU on shutdown cooling with the RHR system is addressed in **Section 3.9.1.**

The licensee performed evaluations and stated that the performance of the safety-related service water systems during and following a LOCA with loss of offsite power has been found acceptable. The licensee noted that the EPU results in an increase of 8 MBTU/hr, resulting in a peak heat load of 98 MBTU/hr for the RHRSW. Additional details are provided in the licensee's letter dated August 13, 2001 (Reference 23).
The RHRSW provides cooling water to the RHR heat exchangers under normal or post-accident conditions. The long-term containment pressure and temperature responses following a LOCA are governed by the ability of the RHR system to remove the decay heat from the suppression pool. The licensee performed containment pressure and temperature response analyses which demonstrate that the capability of the containment system is adequate to operate at the proposed EPU. In the containment pressure and temperature response analyses, the post LOCA RHRSW cooling was assumed to be unchanged for power uprated conditions. Therefore, the RHRSW cooling remains adequate for plant operations at the proposed EPU to perform its safety function during and following a LOCA. The staff's evaluation of the containment system performance for plant operations at the proposed EPU is addressed in Section 4.1.

Based on the review of the licensee's rationale, the staff finds that QCNPS operations at the proposed EPU maintain the containment temperature and pressure response at acceptable levels and do not change the operations of the safety-related service water systems, and otherwise have an insignificant or minor impact. Therefore, the staff concludes that the safety-related service water systems at QCNPS remain adequate for plant operations at the proposed EPU to perform their safety function during and following a LOCA.

6.4.1.2 Non-Safety-Related Loads

Several service water heat loads will increase as a result of the EPU. The licensee stated that the major heat load increases from the EPU reflect an increase in main generator heat losses rejected to the stator water coolers and hydrogen coolers in addition to increases in turbine building closed cooling water and reactor building closed cooling water heat loads. The licensee performed evaluations which demonstrate that the temperature of the service water temperature discharged to the circulating water system is slightly increased at the proposed EPU.

Since the service water system does not perform any safety-related function, the impact of the proposed EPU on the designs and performances of this system was not reviewed.

6.4.2 Main Condenser, Circulating Water, and Normal Heat Sink System Performance

The main condenser, circulating, and normal heat sink systems are designed to provide the main condenser with a continuous supply of cooling water for removing heat rejected to the condenser thereby maintaining condenser pressure as recommended by the turbine vendor. The licensee stated that the EPU operation increases the heat rejected to the condenser, and therefore, increases the condenser backpressure. If condenser pressures approach the backpressure limitation, then the licensee must reduce reactor power to maintain an adequate vacuum.

Since the main condenser, circulating water, and normal heat sink systems do not perform any safety-related function, the impact of the proposed EPU on the design and performance of these systems was not reviewed.

6.4.3 Reactor Building Closed Cooling Water System

The reactor building closed cooling water (RBCCW) system is designed to remove heat from various auxiliary plant equipment housed in the reactor building during normal plant operations. The licensee performed evaluations and stated that the increase in heat loads on this system due to plant operations at the proposed EPU are not significant. These increases arise due to running the reactor recirculation pumps at a higher speed and additional decay heat load for the fuel pool coolers. The operation of the remaining equipment cooled by the RBCCW system is not power ~~dependant~~ dependent and is not affected by EPU. The licensee provided additional details of the EPU effect on the RBCCW heat loads (Reference 23) in response to the staff. The licensee's reevaluation of RBCCW system heat loads for EPU was based on a revised service water design temperature of 90° F (original design 95° F). This was based on a review of historical service water temperatures at QCNPS. This design temperature change; together with the swing RBCCW pump and heat exchanger aligned to the unit with an emergency full core offload, results in heat transfer capability exceeding the required heat load for all operating modes.

Based on the review of the license's rationale, the staff finds that the heat loads in equipment cooled by the RBCCW system have been evaluated for power uprate operations and the loads remain within system capability. Therefore, the staff concludes that the impact of plant operations at the proposed EPU on the RBCCW system is acceptable.

6.4.4 Turbine Building Closed Cooling Water System

The turbine building closed cooling water system (TBCCW) system supplies cooling water to many of the non-safety HVAC units and other turbine building equipment. Principal heat loads that increased include the bus duct coolers and the added heat from operation of a fourth condensate/booster pump and a third reactor feed pump. Other loads do not increase significantly due to the EPU. The licensee evaluations of the increased TBCCW system heat loads demonstrated a coolant increase of less than 0.5° F. The TBCCW system has adequate heat removal capability for plant operations at the proposed EPU.

Since the TBCCW system does not perform any safety-related function, the impact of the proposed EPU on the designs and performances of this system was not reviewed.

6.4.5 Ultimate Heat Sink (UHS)

The Mississippi River is the normal heat sink via the intake and discharge canals, providing essential cooling water for QCNPS at the EPU conditions. However, in the event of loss of the downstream Lock and Dam No. 14, water trapped in the intake and discharge bay becomes the UHS. In this event, make-up water is required to the UHS for decay heat removal. This make-up activity is currently required for plant operation. The licensee stated that sufficient time is available to replenish water in the UHS following a loss of the dam from EPU conditions.

In their August 7, 2001, response to the staff (Reference 19), the licensee provided additional information regarding the impact of EPU on the ability of the QCNPS UHS to maintain the suppression pool below its acceptance limit of 177° F. The licensee's analyses assume the use of the main condenser for 24 hours after shutdown; and use of 3 portable pumps supplying 5100 gpm to the residual heat removal service water intake. The licensee stated that the time to perform manual actions to provide makeup water from the river for a dam failure is unaffected by EPU operation as the time depends only on the approximate 2-day interval to reach separation between the UHS and the river. Under these conditions the suppression pool temperature reached is 166° F, which is an increase of 10° F from current conditions. Similarly the maximum cribhouse intake temperature increases 1.5° F to 108° F, yet remains below the acceptance value of 109° F.

Based on the review of the licensee's rationale, the staff finds that QCNPS operations at the proposed EPU will have an insignificant impact on the UHS.

6.5 Standby Liquid Control (SLC) System

The licensee evaluated the effect of the EPU on the SLC system injection and shutdown capability. The QCNPS SLC is a manually operated system that pumps ~~concentrated~~ a sodium pentaborate solution into the vessel in order to provide neutron absorption and is capable of bringing the reactor to a subcritical shutdown condition from rated thermal power.

The licensee stated that an increase in the core thermal power does not by itself directly affect the ability of the SLC boron solution to bring the reactor subcritical and to maintain the reactor in a safe shutdown condition. Operating at the EPU condition does not affect the required boron solution. Implementation of a higher fuel batch fraction, a change in fuel enrichment, or a new fuel design are the conditions that might affect the shutdown concentration. The SLC system shutdown capability is reevaluated for each reload core. The new fuel design combined with a planned extension in the fuel cycle operating time does not currently require an increase in the minimum reactor boron concentration of 600 ppm, and therefore no increase in the volume of the stored boron solution for the EPU cycle.

~~According to the licensee, the SLC system is designed to inject at a maximum reactor pressure equal to the upper analytical setpoints for the second lowest group of SRVs operating in the relief modevalves. The licensee stated that since the reactor dome pressure and the SRV setpoints will not change, the current SLC system process parameters will not change. The licensee added that the SLC pumps are positive displacement pumps, where small changes in the SRV setpoint would have no affect on the SLC system capability to inject the required flow rate.~~ This section needs to be rewritten to reflect the RAI response on ATWS.

The SLC ATWS performance is addressed in Section 9.4.1 and the licensee has stated that the evaluation is based on a representative core design at the EPU condition. The licensee determined that the ATWS analysis showed that there is no adverse effect on the ability of the SLC system to mitigate an ATWS. Therefore, the capability of the SLC system to provide its backup function is not affected by the EPU.

During June 2001 QCNPS EPU audit, the staff asked GNF and the licensee to confirm that for all limiting ATWS analysis, the SLC system would be able to inject the required flow rate at the required time in the analyses, without lifting the SLC system bypass relief valve. The GNF screening process in use at the time of the QCNPS audit did not specifically identify the QCNPS bypass relief valve capability. This issue is being treated by GNF generically, as well as on a plant specific basis. As described in Section 2.6, **the staff has requested additional information from the licensee on this issue.**

During the staff audit, the Project Task Report T0902, "Anticipated ~~t~~Transient Without Scram," was reviewed and GNF and licensee staff discussed the QCNPS bounding loss of offsite power (LOOP) ATWS event. For this event, the calculated peak vessel pressure would reach a maximum of 1316 psig at about 9.2 seconds from the start of the event, before the initiation of the SLC system at 96 seconds. The SRVs would open to relieve the pressure during any further pressure spikes, which can result from calculated reactor vessel level undershoots. The calculated undershoot is caused by a computer code (ODYN) limitation in modeling the HPCI and RCIC system. The undershoot of the water level results in over correction of the level, and the resulting overshoot of the level generates a high core flow and core power, and eventually generation of excessive steam. This artifice of the calculation can result in increased vessel pressure.

Considering that the ODYN calculation is conservative, i.e, plant response to the water level transient is expected to be considerably faster than the ODYN model, and that there would still be sufficient margin to lifting the SLC relief valve, and that the pressure spikes that are calculated occur for a short duration, the staff concludes that the SLC system will be able to inject boron into the RCS as required by 10CFR50.62.

6.6 Power-Dependent Heating Ventilation And Air Conditioning Systems

The heating, ventilation, and air conditioning (HVAC) systems consist mainly of heating or cooling supply, exhaust and recirculation units in the turbine building, reactor building, and drywell. The EPU is expected to result in a small increase in the heat loads caused by slightly higher process temperatures and higher electrical currents in some motors and cables.

The affected areas are the steam tunnel, ECCS pump rooms, and drywell in the reactor building; the feedwater heater bay and condenser area, feedwater pumps, condensate/condensate booster pumps and the motor-generator set areas in the turbine building. Other areas are unaffected by the EPU because the process temperatures remain relatively constant.

In the steam tunnel, the heat load increases due to the increase in the feedwater process temperature. The maximum area temperature increase is 0.5° F.

In the drywell, the increase in feedwater process temperature and the increase in the recirculation pump motor horsepower are within the margins of~~in~~ the system capacity. By letter dated August 14, 2001, in response to the staff, the licensee stated (Reference 24) that the HVAC system is designed for heat loads from the recirculation pumps at the QCNPS of 1,870,000 BTU/hr. At EPU conditions, the expected heat load from pump motors is 1,573,840

BTU/hr, providing a margin of approximately 296,000 BTU/hr. At EPU conditions, the feedwater temperature increase is 13.8° F. The associated increase in feedwater piping heat load is 10,439 BTU/hr. The feedwater piping and the recirculation pump motors are in the same space and are cooled by the same cooling system. The margin in the HVAC design for the recirculation pump motor heat load is sufficient to compensate for the increase in feedwater piping heat load.

In the ECCS pump rooms, the heat loads increased as a result of a higher suppression pool temperature. The ECCS pump room coolers have adequate cooling capacity to maintain the design ECCS room temperature. By letter dated August 14, 2001, in response to the staff's Request for Additional Information (RAI), the licensee stated that the QCNPS residual heat removal (RHR) heat load increases from 319,798 BTU/hr to 335,800 BTU/hr due to the EPU, well within the room cooler capacity of 570,000 BTU/hr. The high pressure coolant injection (HPCI) room at QCNPS is not affected by the EPU since there is no process temperature, electrical, or other heat load changes that affect the pre-EPU design heat loads.

In the turbine building, the maximum temperature increase in the feedwater heater bay and condenser areas is approximately 4° F due to the increase in the feedwater process temperatures. The feedwater pump motors and motor-generator sets are internally cooled by separate dedicated once-through ventilation systems. The heated ventilating air is directly exhausted to the atmosphere without mixing with the room air; thus, the effect on area temperature is negligible. The effects of the higher internal temperature in the pump motors and motor-generator sets have been evaluated and <u>shown to be acceptable for operation during the remaining plant life.</u> ~~accounted for in the plant environmental qualification (EQ) program.~~ The temperature in the condensate pump area increases by approximately 9° F caused by the operation of the fourth condensate pump.

In response to the staff, the licensee stated (Reference 24) that the operation of the fourth condensate/booster bump, as required for EPU operation, causes an increase in the heat load. Since the cooling capacity of the ventilation system is not being changed, the pre-EPU design room temperature may be exceeded during times when the outdoor air is at the design temperature (i.e., periods of expected seasonal high temperatures), but this does not continue for extended periods of time. The normal operation of the non-safety-related pumps in this area is not affected, based on a review of the motor insulation ratings, which exceed the EPU temperatures. The licensee stated that all equipment in the EQ program affected by such temperature increases has been evaluated and is acceptable.

Based on the licensee's review of design basis calculations and EQ design temperatures, the design of the HVAC is adequate for the EPU. The licensee stated (Reference 24) that in several reactor building areas, the post-LOCA temperature increase is a few degrees due to higher EPU heat loads. The secondary containment is isolated and the HVAC systems for the general areas do not operate post-LOCA. The licensee stated that the equipment in all such areas in the EQ program has been evaluated and found acceptable, as provided in the site EQ program documentation.

The licensee stated (Reference 24) that a separate EPU evaluation for the ECCS related HVAC systems was performed for QCNPS. Therefore, any site differences were captured in the evaluations. The licensee further explained that the other HVAC systems are similar enough between sites for normal operations that they could be evaluated together. The evaluations determined that no changes in the operation or configuration of these systems were required for the EPU, and that all of the systems continued to meet design requirements.

Based on the staff's review of the licensee's rationale, and the experience gained from review of power uprate applications for other BWR plants, the staff concludes that the EPU does not adversely affect the operation of HVAC.

6.7 Fire Protection Program

The staff finds that the operation of the Quad Cities Nuclear Power Station at the EPU will have no impact on the existing fire detection or suppression systems, the existing fire barriers provided to protect safe shutdown capability, or the administrative controls that are specified in the plant's fire protection plan required by 10 CFR 50.48(a). The NRC requirements for achieving and maintaining safe shutdown following a fire require that: (1) one train of systems necessary to achieve and maintain hot shutdown be maintained free of fire damage, and (2) that the systems necessary to achieve and maintain cold shutdown can either be repaired within 72 hours if using redundant systems, or that the systems can be repaired, and that cold shutdown can be achieved within 72 hours if using alternative or dedicated shutdown capability. While Section 6.7, "Fire Protection", of the licensee's Safety Analyses Report (Reference 2) only addresses cold shutdown capability and is silent concerning hot shutdown capability, Table 6-3 of the report indicates that the limits for the important reactor process variables (i.e., peak cladding temperature, primary systems pressure, primary containment pressure, and suppression pool bulk temperature) are not exceeded following a fire event using the reactor core isolation cooling (RCIC) system. The staff has accepted the use of RCIC for providing reactor coolant makeup to achieve hot shutdown when those systems are protected in accordance with the requirements specified in Section III.G of Appendix R to 10 CFR Part 50. While the higher decay heat associated with the EPU may reduce the time available for the operators to achieve cold shutdown, it should not impact the time required to repair those systems necessary to achieve and maintain cold shutdown, and would therefore only affect those fire areas in the plant where alternative or dedicated shutdown systems are relied upon to satisfy NRC requirements (i.e., those plant areas that must achieve cold shutdown within 72 hours following a fire). The licensee has stated that the safe shutdown systems and equipment used to achieve and maintain cold shutdown conditions do not change, and are adequate for the EPU conditions. The staff finds this acceptable.

The EPU may affect systems necessary to achieve and maintain hot shutdown for those plant areas that rely upon the use of safety relief valves in conjunction with the use of low pressure systems, such as core spray and low pressure coolant injection, to provide reactor coolant makeup, or those plant areas that rely on alternative or dedicated shutdown capability as defined in Section III.G of Appendix R. For example, Section 4.2.4 Automatic Depressurization System, notes that to achieve the required flow capacity for the EPU conditions, five automatic depressurization system valves must be operable and that prior to the EPU only four automatic depressurization system valves were required to be operable. However, the licensee has not credited ADS operation in conjunction with low pressure systems for Appendix R hot shutdown operations.~~not indicated that systems other than RCIC are used for a fire at Quad Cities.~~ The

EPU has reduced the time available for the operators to stabilize the plant in hot shutdown using RCIC. The licensee has stated that the operator actions required to mitigate the consequences of a fire are not affected by the EPU, sufficient time is available for the operators to perform the necessary actions, and any necessary changes to procedures will be accomplished prior to the EPU implementation. The staff finds this acceptable.

An evaluation of the effect of the EPU on the top ten fire scenarios in terms of core damage frequency contribution was performed by the licensee. The licensee concluded that the EPU would have only a minor effect of the fire risk estimated in the licensee's Individual Plant Examination of External Events. The staff finds this acceptable.

Therefore, based on a review of the information provided by the licensee in Reference 2, the staff concludes that the EPU will not adversely affect the safe shutdown capability in the event of a fire and is, therefore, acceptable.

6.8 Systems Not Impacted or Insignificantly Impacted by EPU

The licensee identified those systems which are not affected or insignificantly affected by plant operations at the proposed EPU. The staff has reviewed those systems (i.e., auxiliary steam, instrument air, service air, miscellaneous HVAC, diesel generator and their associated supporting systems, etc.). Based on the staff's review of the systems identified by the licensee and the experience gained from review of EPU applications for other BWR plants, the staff concludes that plant operations at the proposed uprate power level has no or insignificant impact on these systems.

7.0 POWER CONVERSION SYSTEMS

7.1 Turbine-Generator

The turbine-generator was originally designed to have the capability to operate continuously at 105% of rated steam flow conditions with a degree of margin to allow control of important variables such as steam inlet pressure. As a result of the proposed plant operations at the EPU, the high pressure turbine will be modified to maintain the GE standard flow margin of 3% of the EPU rated steam flow.

The licensee performed evaluations to verify the mechanical integrity of the turbine-generator and components, under plant operations at the proposed EPU. These included both stationary and rotating components and valves, control systems, and other support systems. The licensee stated that results of the evaluations showed that modification of the high pressure turbine to increase flow passing and some other non-safety modifications to the turbine-generator are needed for the EPU.

The licensee described some of these changes in response to the staff (Reference 19). These include new boreless high pressure turbine rotors and nozzle diaphragms for increased volumetric flows, new setpoints for the cross around relief valves, stator water cooling alarm and runback setpoint changes and various changes to the electro-hydraulic control/turbine supervisory instrumentation.

The licensee further stated that they evaluated the probability of a turbine overspeed and its associated turbine missile production due to plant operations at the proposed EPU. In response to the staff they noted that since the geometry of the LP rotors and blading is not changing as a result of the EPU the existing analysis remains valid. There is sufficient design margin in the current turbine overspeed protection trip settings to limit overspeed to the current limit of 120% of rated speed. For QCNPS, the backup overspeed trip setpoint will be changed in accordance with the original equipment manufacture's recommendation. The previous sentence is not correct. The 120% overspeed limit is not changing, but the overspeed trip settings will be reduced to ensure that the 120% overspeed is not exceeded. Therefore, the turbine could be continuously operated safely at the proposed EPU.

The staff requested additional information regarding the implications of the increase in reactor power which may be limited by the main generator capability of 912 MWe following EPU. The licensee's response (Reference 19), stated that due to the change in plant efficiency over the operating cycle; reactor power could vary from approximately 96% of thermal power under optimal conditions in the cold winter to 100% of power on warm summer days. The licensee stated that these variations in reactor power do not approach the magnitude of changes required for surveillance testing and rod pattern alignments. Additionally, the licensee stated that the effect of having thermal power limited by main generator capacity (load follow) on radioactive waste generation will be minimal in that the major change for such generation at EPU conditions is an increase in feedwater flow and conductivity.

(SE regarding turbine component integrity and turbine missile generation would be provided **by EMEB**. SE regarding turbine overspeed protection would be provided **by EEIB.**)

7.2 Miscellaneous Power Conversion Systems

The licensee evaluated miscellaneous steam and power conversion systems and their associated components, including the condenser, air removal system and steam jet air ejectors, for plant operations at the proposed EPU. The licensee stated that the existing equipment for these systems is acceptable for plant operations at the proposed EPU. Modification to some non-safety-related equipment, such as steam dilution modifications to the condenser air removal systems, is necessary to provide adequate capacity for the proposed uprated core thermal power.

Since these systems do not perform any safety-related function, the staff has not reviewed the impact of plant operations at the proposed EPU on the design and performance of these systems.

7.3 Turbine Steam Bypass

The turbine bypass valves were initially rated for a steam flow of 40% of the original rated steam flow. For EPU conditions, the resulting bypass capability will be 33.3% of EPU steam flow. The licensee has proposed revisions to technical specifications reflecting the revised percentage of rated steam flow. Transient analyses remain based on actual mass flow rates which are not changed for EPU.

Since the turbine bypass system does not perform any safety-related function, the staff has not reviewed the impact of plant operations at the proposed EPU on the design and performance of this system.

7.4 Feedwater and Condensate Systems

The licensee noted that EPU operation requires modifications related to these systems, such as feedwater pump low suction pressure staggered trips and recirculation system runbacks; as well as alteration of operating system lineups, such as running all three feedwater pumps (versus two previously) and all four condensate/condensate booster pumps (versus three previously). As stated by the licensee, the feedwater and condensate systems do not perform system level safety-related functions. Therefore, the staff performed a limited review of the impact of plant operations at the proposed EPU on the design and performance of these systems.

In response to a ~~the~~ staff question, the licensee addressed various changes that are planned to improve plant trip avoidance capability under EPU conditions (Reference 19). A reactor recirculation pump runback is being added to reduce potential for reactor scrams on low water level following a loss of either a feedwater or condensate pump. The runback is enabled whenever main steam flow exceeds the capability of two feedwater pumps. When enabled, a runback will rapidly reduce the core flow to the equivalent for 82% power if less than three feedwater pumps are running coincident with a reactor low level alarm setpoint or if all condensate pumps aren't running. The licensee's analyses indicates that these changes should reduce core flow and reactor power to within the capability of the running feedwater/condensate pumps to avoid reduction in the reactor water level to the revised (lower) scram and isolation setpoints.

The licensee is also implementing staggered trips of the feedwater pumps on low suction pressure, considering the increased potential for such trips on a loss of a condensate pump when running all four condensate pumps for EPU conditions. The existing low suction feedwater pump trips are being modified to trip one feedwater pump if suction pressure drops to the low suction trip for 3 to 5 seconds; then trip a second feedwater pump if the suction pressure remains low for 12 to 15 seconds. For equipment protection, all pumps will continue to trip if suction pressure decreases to the low-low suction setpoint. The licensee will also scale feedwater control and indication loops and adjust feedwater pump runout logic to accommodate EPU flow rates.

The licensee stated that proper operation of the runback and feedwater pump suction trip logics will be verified in post-modification testing. The feedwater control system response and feedwater pump performance will be verified at various power levels during post-EPU startup testing.

The staff requested additional information regarding the effect of the EPU on the feedwater system specifically related to the capability to handle additional flow in the feedwater heater drains to avoid challenges to operators and safety systems potentially caused by loss of feedwater heaters. In their response dated August 7, 2001, (Reference 19) the licensee stated that an evaluation of the feedwater heater level control and drain valves was performed to assess flow passing capabilities. The licensee determined that the EPU operating conditions do not significantly challenge the capability of the level control valves~~, with the exception of the QCNPS Units 2 and 3 feedwater heater "B" level control valves which require valve trim replacement. QCNPS previously made similar changes to these valves and no changes are required for EPU.~~ The licensee also reviewed thermal-hydraulic conditions and determined that shell modifications were required to support a re-rate of the "C" and "D" feedwater heaters for increased EPU design pressure conditions; the "C" feedwater heaters are re-rated to 100 psig from 83 psig and the "D" feedwater heaters are re-rated to 178 psig from 150 psig.

Based on the review of the licensee's rationale, the staff finds the feedwater and condensate systems acceptable for extended power uprate operations.

8.0 RADWASTE SYSTEMS AND RADIATION SOURCES

QCNPS uses waste treatment systems designed to collect and process gaseous, liquid, and solid waste that might contain radioactive material. These radioactive waste treatment systems were evaluated in the Final Environmental Statement (FES) dated September 1972. The proposed 17.8% extended power uprate (EPU) will not involve any significant physical changes in the waste treatment systems, nor will it affect the environmental monitoring of any waste stream described in the FES. For normal operations, no new or different radiological waste streams are created as a result of the proposed power increase.

8.1 Liquid and Solid Waste Management

The major impact of the power uprate on the station's solid radioactive waste production involves the increased generation of spent feedwater cleanup resins (SFCR), the major component of low level radioactive waste (LLRW). LLRW also includes filter sludge, dry active waste, metals, etc. Because of the estimated increased levels of activated corrosion products in the feedwater system, SFCR quantities should increase as a result of increased changeout frequency for resin bed media. Due to the increases in condensate/feedwater flow and temperature, the licensee expects that the increase in solid waste production (chiefly resins) will be proportional to (but no more than) the power uprate. This estimate is supported by experience gained at other BWR facilities, now operating with smaller power uprates (2-5%). Based on this estimate, the overall increase in solid radioactive wastes is expected to be a small percentage (approximately 10%) of the station's yearly projected low-level waste burial volume for the year 2000 (144 cubic meters). This amount is bounded by the FES.

The volume of liquid radioactive waste released should not be impacted by the power uprate. The site recycles a substantial fraction of the water used to process liquid radioactive material waste streams. However, due to the expected increased presence of fission products and activated corrosion and wear products in the reactor condensate, feedwater, and coolant, and increased flow through the condensate and reactor water cleanup demineralizers, more liquid backwashes of these demineralizers will be necessary. However, since the water quality of these backwashes is high, these waters will be recycled and thereby will not add to the volume of water released offsite.

Since the amount of activity (number of curies) of radioactive material contained in the liquid effluents is expected to increase in proportion to the 17.8% power uprate, the concentration of radioactive materials released as liquid wastes is expected to increase by that same amount. From 1995 to 1999, the average offsite calculated doses to the public from the liquid release pathway were very small fractions of the Part 50, Appendix I numerical standards and the limits of 40 CFR Part 190. From 1995 to 1999, the highest calculated whole body dose component was 0.03% of Appendix I criteria, while the highest calculated critical organ dose component was less than 0.01% of the 40 CFR 190 limit. For that same period, the average calculated dose from liquid effluents for all liquid release pathways was about 0.003% of Appendix I guidelines. A projected 17.8% increase of these very small values results in a negligible increase in calculated public dose, and the overall contribution to the public dose from the liquid effluent pathway would remain a very small fraction of the regulatory limits.

8.2 Gaseous Waste Management

The Gaseous Waste Management System (GWMS) consists of the main offgas system and various building (turbine, reactor and radwaste) ventilation systems. Airborne radioactive material releases are controlled, processed, filtered and monitored, and include gaseous and particulate forms. Gaseous fission products such as Krypton-85 and Iodine-131 are produced by the fuel in the core during reactor operation. A small percentage of these fission gases are released to the reactor coolant from the small number of fuel assemblies which are expected to develop leaks during normal reactor operation. The main offgas system removes these fission gases directly from the plant main condenser, and these gases are processed before release. These offgas effluent release quantities are greater than the sum of all other gaseous release streams. Thus, the effluent release rate (and resultant public dose) depend primarily on fuel defect rate. Current and expected fuel performance for Quad Cities Units 1 and 2 has been significantly better than the original design. The licensee conservatively assumed a 18~~7~~% increase in gaseous effluents (as a linear function of the power increase). Using the highest calculated dose over the period 1995 to 1999, this assumed effluent increase would result in the worst case offsite pathway dose (in terms of percentage of the 15 mrem limit) of 0.33% of the 10 CFR Part 50, Appendix I numerical design objectives. For that same period, the average calculated dose from gaseous effluents for all designated airborne dose pathways was 0.16% of the Appendix I guidelines. Therefore, as a result of the 17.8% power rate increase, the resultant calculated dose to the public from the overall release of gaseous effluents will remain a very small fraction of the dose limits of 10 CFR 20.1301.

8.2.1 Offgas System

Radiolysis of water (i.e., formation of H_2 and O_2) in the core increases linearly with power, thus increasing the heat load on the offgas recombiner and related components. The licensee evaluated the impact of the increases of these offgases resulting from plant operation at the proposed EPU on the offgas system, and provided additional information in a letter dated August 7, 2001 (Reference 19). The licensee calculated that the heat load for the offgas recombiner will increase from approximately 83% to 97.5% of the system design, with a radiolytic hydrogen flow rate of 30.9 lbs/hr post-EPU. The licensee stated that this is a bounding case using hydrogen water chemistry, which requires hydrogen injection into the feedwater system at close to the upper limit of the normal injection range. The licensee stated that they intend to operate with low levels of~~without~~ hydrogen injection in combination with noble metal application. Since the hydrogen injection rate decreases considerably when using noble metal injection, the hydrogen mass flow rate will be considerably less than the bounding value.

The offgas system processes air in-leakage evacuated from the main condenser. The system also processes non-condensable radioactive gases in the main condenser which consist of activation gases and fission product noble gases transported through the steam lines and turbine. The rate of main condenser in-leakage is unaffected by EPU. This in-leakage determines the hold-up time for radioactive decay as the increased radiolytic hydrogen flow from EPU power increase will be removed as noted above in the offgas recombiner, before the hold up volume. The design basis noble gas release rate (0.2 Ci/s) for QCNPS is independent of power level and referenced to a 30-minute hold-up time which is not affected by EPU conditions. Expected offgas releases will be a fraction of the design basis release rate which bounds the effect of increased power.

The licensee assumed that the radioactive gases will increase proportionally to the EPU increase. In Reference 19 the licensee corrected a statement in Section 8.4.3 of their safety evaluation (Reference 2) to note that an increase of 12% in fission product activity is expected for the EPU. The concentration of coolant activation products and fission products in the steam lines will remain unchanged as the linear increase in production is diluted by the increase steaming rate post-EPU. The licensee stated that the gaseous effluents are well within limits at original power and remain well within limits following EPU implementation. The system radiological release rate is administratively controlled, and does not change with operating power. Therefore, EPU does not significantly affect the offgas system design or operation.

Based on the review of the licensee's rationale and the experience gained from the staff's review of power uprate applications for similar BWR plants, the staff concludes that plant operations at the proposed EPU will have an insignificant impact on the offgas system.

8.3 Radiation Sources

The staff has reviewed the licensee's plan for power uprate with respect to its effect on the facility radiation levels and on the radiation sources in the core and coolant. The radiation sources in the core include radiation from the fission process, accumulated fission products, and neutron reactions as a secondary result of reactor power. The radiation sources in the core are expected to increase in proportion to the increase in power. This increase, however, is bounded by the existing safety margins of the design basis sources. Since the reactor vessel (inside fully-inerted primary containment) is inaccessible during operation, a 17.8% increase in the radiation sources in the reactor core will have no affect on occupational worker personnel doses during power operations. Due to design shielding and containment surrounding the reactor vessel, worker occupational doses are largely unaffected, and doses to the public from radiation shine from the reactor vessel remain essentially zero as a result of the extended power uprate (EPU).

During operations, the reactor coolant passing through the reactor core region becomes radioactive as a result of nuclear reactions. The activation product concentrations in the steam will remain nearly constant following the power uprate since the increase in activation production in the steam passing through the core will be balanced by the increase in steam flow through the core. The activation products in the reactor water, however, will increase in approximate proportion to the increase in thermal power. The installed shielding at Quad Cities was conservatively designed so that the increase in activation products in the reactor coolant resulting from the proposed power uprate will not affect radiation zoning in the plant.

Activated corrosion products (ACP), which are the result of the activation of metallic wear materials in the reactor coolant, could increase as a result of the proposed power uprate. The equilibrium level of ACP in the reactor coolant is expected to increase in proportion to both the increase in feedwater flow rate and the increase in neutron flux in the reactor, while the increased feedwater flow will likely reduce the efficiency of the reactor water cleanup system (RWCU). However, the expected ACP increase should not exceed the design basis concentrations. Most of the areas (e.g., recirculation pumps and the RWCU) that would be affected by this increase in activated corrosion products are located in locked areas or areas, such as the drywell (primary containment), that are inaccessible during plant operation. Since these areas are usually high dose rate areas, personnel access to these areas will continue to be restricted during plant operations as required by 10 CFR Part 20 high radiation area (HRA) requirements, and in accordance with plant technical specifications and required licensee implementing procedures.

In an effort to reduce the occupational worker dose (and the radiation skyshine public dose component), the licensee initiated the noble metal injection process (NIP), consistent with their implementation of as low as is reasonably achievable (ALARA) principles. By injecting small quantities of noble metal into the reactor feedwater system, the level of highly-activated radioactive material deposited as crud on primary coolant piping sources and fuel is reduced. Additionally, NIP provides another dose reduction benefit, as it effectively reduces the radiation skyshine from the steam-side turbine building components that contributes to public doses. Main Steam Line dose rates at Quad Cities have decreased by as much as a factor of four as a result of the NIP process.

Fission products in the reactor coolant result from the escape of minute fractions of the fission products which are contained in the fuel rods. Fission product release into the primary coolant is dependent on the nature and number of fuel defects and is approximately linear relative to core thermal power. Using ANSI/ANS 18.1-1999 normal operations source term methodology, the licensee calculated about a 12% increase in fission product concentration in the reactor coolant from the fuel (assuming no increase in fuel cladding defects). However, the fission product concentration in the steam should remain nearly constant following the power uprate, given the proportional increase in steam flow (dilution) through the core. Given that current levels of fission product activity in the reactor coolant and steam are small fractions of the design basis data, a 12% increase should have a minimal impact.

8.4 Radiation Levels

Radiation sources in the reactor coolant contribute to the plant radiation levels. As discussed previously, the proposed 17.8% power uprate will result in a proportional increase in certain radiation sources in the reactor coolant. This increase in reactor coolant activity will result in some increases (up to about 18%) in plant radiation levels in most areas of the plant. This increase in plant radiation levels may be higher in certain areas of the plant (e.g., inside the drywell and near the RWCU) due to the presence of ACP. Some post-operational radiation levels may also be higher in those areas of the plant where accumulation of corrosion product crud (activated corrosion and wear products) is expected (i.e., near the spent fuel pool cooling system piping and the reactor coolant piping as well as near some liquid radwaste equipment). Many of these areas are normally locked, controlled in accordance with Part 20 HRA requirements, and require infrequent access.

The licensee has stated that many aspects of the plant were originally designed for higher-than-expected radiation sources. Therefore, the small potential increase in radiation levels resulting from the proposed power uprate will not affect radiation zoning or shielding in the various areas of the plant that may experience higher radiation levels. The purpose of the licensee's ALARA program is to ensure that doses to individual workers will be maintained within acceptable limits by controlling access to radiation areas. The licensee will continue to use procedural access, work planning and controls, and pre-job worker training/briefings to compensate for any increased radiation levels and to maintain occupational doses ALARA. As part of the overall EPU test program, during the incremental 3% power step increases the licensee will perform special surveys of area external radiation levels to assure that the radiation areas are properly designated, posted and controlled as required by Part 20 and plant technical specifications.

The proposed power uprate will also cause a small increase in post-accident radiation levels. Item II.B.2 of NUREG-0737 states that the occupational worker dose guidelines of GDC 19 (10 CFR Part 50, App. A) shall not be exceeded during the course of an accident. Compliance with Item II.B.2. ensures that operators can access and perform required duties and actions in designated vital areas. GDC 19 requires that adequate radiation protection be provided such that the dose to personnel should not exceed five rem whole body, or its equivalent to any part of the body (extremity limit is 75 rem), for the duration of the accident. The licensee has determined that, based on conservative calculations, the post-accident radiation levels will increase by 11-45% (as a function of plant location) as a result of the proposed power uprate. Based upon this analysis, the calculated post-accident vital area worker doses (for coolant and air sampling activities) to the whole body and extremities are less than 1 and ~~1.7~~ 1.8 rem respectively. Therefore, personnel access to and work in designated vital areas for accident mitigation following a loss-of-coolant-accident (LOCA) can still be accomplished without exceeding the dose requirements of GDC 19. Additionally, the calculated dose estimates for personnel performing required post-LOCA duties in the plant's Technical Support Center (TSC) remain within GDC limits. The site's Emergency Operations Facility (EOF) is over 100 miles from the site, and therefore the EOF habitability is unaffected by the power uprate.

The licensee has calculated the impact on operator doses in the control room from the following design basis accidents (DBA): LOCA, main steam line break accident, fuel handling accident and control rod drop accident. In the worst case, the LOCA provides a 20% increase to the operator's whole body dose, which includes ~~assuming all~~ the dose ~~is~~ from direct radiation shine external to the control room (dose is 0.377~~142~~ rem, versus the 5 rem limit). ~~The control rod drop accident results in 0.266 rem, the highest calculated operator whole body dose for the DBA analyzed for the EPU.~~ See Section 9.3.2 for additional discussion of control room doses from DBAs.

Several physical plant modifications will need to be completed prior to full implementation of the power rate increase. The reactor vessel steam dryer/separator will be modified to compensate for the increase in moisture carryover from the reactor to the steam lines. These modifications will be planned and conducted in accordance with the station ALARA program. This expected one-time occupational dose to modify these and other systems should be a small fraction of the average yearly worker collective dose for the units.

Direct radiation (skyshine) from the main steam system components in the turbine building provides another offsite public dose pathway from an operating BWR. The licensee has calculated the public dose from radiation sources in reactor steam from coolant activation products (chiefly Nitrogen-16). Nitrogen-16 production is increased by routine hydrogen gas injection into the reactor feedwater in an effort to prevent intergranular stress corrosion cracking of reactor internals. The licensee also uses the noble metal injection process (NIP) primarily to maintain worker doses ALARA. Additionally, NIP provides another dose reduction benefit, as it allows for a significant reduction in hydrogen injection rates, thus effectively reducing the direct radiation shine from the steam-side turbine building components. Main steam line dose rates have decreased by as much as a factor of four at the Quad Cities units as a result of the NIP process. While this skyshine dose is not expected to actually increase as a result of the power uprate, the station's required calculation methodology conservatively assumes the skyshine dose is directly proportional to reactor power. Given a 17.8% increase in reactor power, the licensee conservatively estimates that the skyshine dose would be about 44% of the 25 mrem dose limit of 40 CFR 190 (using the highest calculated dose during 1995 to 1999).

On the basis of the staff's review of the Quad Cities Unit 1 and Unit 2 license amendment, the staff concludes that the 17.8 percent power uprate will have little effect on personnel occupational doses and that these doses will be maintained ALARA in accordance with the requirements of 10 CFR 20.1101. Additionally, the operator calculated doses from external radiation exposures during a DBA will be less than the GDC 19 criteria, and will allow operators access into vital areas for needed emergency activities. The staff, therefore, finds the proposed power uprate at QCNPS to be acceptable from a normal operations, occupational, and GDC 19 accident dose perspective.

9.0 REACTOR SAFETY PERFORMANCE EVALUATION

9.1 Reactor Transients

AOOs are abnormal transients which are expected to occur one or more times in the life of a plant and are initiated by a malfunction, a single failure of equipment, or a personnel error. The applicable acceptance criteria for the AOOs are based on 10 CFR Part 50, Appendix A, General Design Criteria (GDC) 10, 15, and 20. GDC 10 requires that the reactor core and associated control and instrumentation systems be designed with sufficient margin to ensure that the specified acceptable fuel design limits (SAFDLS) are not exceeded during normal operation and during AOOs. GDC 15 stipulates that sufficient margin be included to ensure that the design conditions of the reactor coolant pressure boundary are not exceeded during normal operating conditions and AOOs. GDC 20 specifies that a protection system be provided that automatically initiates appropriate systems to ensure that the specified fuel design limits are not exceeded during any normal operating condition and AOOs.

The standard review plan (Reference 7) provides further guidelines that: (1) pressure in the reactor coolant and main steam system should be maintained below 110 percent of the design values according to the ASME Code, Section III, Article NB-7000, "Overpressure Protection;" (2) fuel cladding integrity should be maintained by ensuring that the reactor core is designed to operate with appropriate margin to specified limits during normal operating conditions and AOOs; (3) an incident of moderate frequency should not generate a more serious plant condition unless other faults occur independently; and (4) an incident of moderate frequency, in combination with any single-active component failure or single operator error, should not result in the loss of function of any fission product barrier other than the fuel cladding. A limited number of fuel cladding perforations are acceptable.

The QCNPS UFSAR evaluates a wide range of potential transients. Chapter 15 of the UFSAR contains the design basis analyses that evaluate the effects of an AOO resulting from changes in system parameters such as: (1) a decrease in core coolant temperature, (2) an increase in reactor pressure, (3) a decrease in reactor core coolant flow rate, (4) reactivity and power distribution anomalies, (5) an increase in reactor coolant inventory, and (6) a decrease in reactor coolant inventory. The plant's responses to the most limiting transients are analyzed each reload cycle and are used to establish the thermal limits. A potentially limiting event is an event or an accident that has the potential to affect the core operating and safety limits.

The generic guidelines for EPU evaluation (Appendix E of ELTR1) identified: (a) the limiting transient to be considered in each event category, (b) the analytical methods to be used, (c) the operating conditions assumed in the generic evaluation presented in the report, and (d) the criteria that was applied. The licensee stated that in support of the EPU, each limiting transient analysis for each category of the transients listed in Table E-1 of ELTR1 was analyzed. Table 9-1 of the licensee's Safety Analysis Report (Reference 2) describes the reactor operating conditions used in analyzing the limiting transients for the current pre-EPU fuel cycle and for the EPU representative core. The table also lists the nominal dome operating pressure and the SLMCPR used in the transient analyses and in calculating the MCPR operating limits. The EPU transients analyses were based on a representative GE-14 core and the calculated SLMCPR value of 1.09 for the core.

The licensee stated that input parameters related to performance improvement program (PIP) features or equipment out of service (OOS) have been included in the safety analyses for the EPU. QCNPS is currently licensed or seeks to implement for EPU operation MELLLA, end-of cycle-coastdown, SLO, final feedwater temperature reduction (FFWTR), increased core flow (ICF), and ARTS power and flow dependent limits. Therefore, the EPU transient analyses that were performed considered these operating constraints. According to the licensee, most of the transient events are analyzed at full power and at the maximum allowed core flow operating point on the power/flow map (Figure 2-1). The licensee also included the 2 percent power uncertainty in the analyses either directly or statistically. The licensee performed the following limiting transients and Table 9-2 of the licensee's Safety Analysis Report provides the results. For all events in Table 9-2, the SRV is assumed to be OOS.

- load rejection with bypass failure (LRWOB)

- turbine trip with bypass failure (TTWOB)

- feedwater controller failure (FWCF) - maximum demand
- loss of feedwater heating (LFWH)
- inadvertent HPCI actuation (bounded by LFWH)
- rod withdrawal error (RWE)
- fast recirculation increase
- slow recirculation increase
- load reject with bypass
- MSIV closure-all valves
- MSIV closure-one valve

The licensee determined that, as shown in Table 9-2 and in Figures 9-1 through 9-4, there are no changes to the basic characteristics of any limiting events due to the EPU operating conditions. The severity of transients at less than rated power are not significantly affected by EPU, due to the protection provided by the adoption of ARTS power and flow dependent limits, as discussed in Section 9.2.

In its evaluation of ELTR1 (Reference 4), the staff concluded that the minimum set of limiting transients described in Appendix E of the topical should be included in the uprate amendment request. The staff also stated that a list of all of the transients analyzed in support of the power uprate should be included, with an explanation of how the limiting transients were selected. The QCNPS submittal did not provide the bases for selecting the EPU limiting transients. However, it was confirmed that GNF selects the limiting EPU transients by evaluating the seven categories of transient events based on the EPU parameters to ensure that: (a) the UFSAR events remain bounded by the reload transient events, (b) no non-limiting events become limiting in terms of thermal limits due the power uprate, and (c) no additional limiting events impacting thermal limits are caused by the EPU operating conditions. Appendix E.2.2 of ELTR1 also discusses the bases for selecting the limiting transients to analyze in support of the EPU and the stated justifications are applicable to QCNPS.

In support of operation at the higher MELLLA rod line and at the EPU power level, the licensee analyzed the limiting transients using a representative equilibrium GE-14 core. The current EPU analyses are based on NRC-approved analytical methods and codes. The transient evaluations also take into account the impact of the performance improvement programs or special features in establishing the thermal limits for the EPU operation. The staff concludes that the EPU transient analyses did not identify any major changes to the basic characteristics of any of the limiting events due to the EPU operating conditions. The staff finds this acceptable.

In the current TS, some LCOs and SRs use 25 percent of the RTP to determine when to apply the corresponding requirement. The value of 25 percent of RTP is based on generic analyses conducted for a bundle power of 4.8 MWt. Since the EPU evaluations show less than 4.8MWt/bundle, the 25 percent threshold remains valid. The staff finds this acceptable.

The recirculation system drive flow is measured and used as an input to the APRM for the flow-biased APRM scram and rod blocks. According to Supplement 1 to the ELTR2, the recirculation system fast transient analysis is necessary to support EPU operation for the plants that have adopted the ARTS feature to ensure adequate protection during the transient. The APRM/Rod Block Monitor TS (ARTS) program replaces the flow-biased APRM trip setdown during operation at off-rated conditions. Under these conditions, ARTS plants like QCNPS use power and flow dependent MCPR and LHGR values for operation at the off-rated conditions. Table 9-2 of the EPU submittal provided the changes in the CPR for the fast recirculation flow increase transient and confirmed that the ARTS multipliers used to develop the power dependent MCPR(P) are bounding. This is acceptable to the staff.

9.2 Transient Analysis For ARTS Power And Flow Dependent Limit

One of the restrictions on the operating flexibility of a BWR, during power ascension from the low-power/low-core flow condition to the high-power/high-core flow condition, is the Average Power Range Monitor (APRM) scram and flow-referenced rod block setdown requirements. The APRM/ Rod Block Monitor TS (ARTS) power and flow dependent limits improvement program objectives are to provide adequate fuel thermal limits while increasing plant operating efficiency and flexibility. The licensee states that use of the ARTS power and flow dependent limits ensures that the plant does not exceed any fuel thermal limit and, thus, the margin of safety is not affected. The ARTS program utilizes the results of the AOO analyses to define initial condition operating thermal limits which conservatively ensure that all licensing criteria are satisfied without setdown of the flow-referenced APRM scram and rod block trips. The specific objective of the associated APRM changes is to justify replacement of the APRM trip setdown (gain and setpoint) requirement with the more meaningful ARTS power-dependent and flow-dependent thermal limits. The licensee stated that this change reduces the need for manual setpoint adjustments and allows a more direct thermal limits administration, increases reliability, and provides more accurate protection of plant safety. The QCNPS ARTS power and flow dependent program is essentially the same as the Partial ARTS program previously implemented at the LaSalle County Station units (References 46 and 47).

The elimination of the APRM gain and setpoint requirement can affect fuel thermal-mechanical integrity ~~and ECCS-LOCA performance~~ (ARTS does not affect LOCA performance). The acceptability requirements for this change are that:

- The Safety Limit MCPR (SLMCPR) shall not be violated as a result of any AOOs.
- All fuel thermal-mechanical design bases shall remain within the GE generic fuel licensing limits described in GESTAR-II (Reference 35).
- ~~The peak cladding temperature and the maximum cladding oxidation fraction following a LOCA shall remain within 10 CFR 50.46 limits.~~

The safety analyses used to evaluate and establish the Operating Limit MCPR (OLMCPR), such that the SLMCPR is not violated and to ensure that the fuel thermal-mechanical design bases are satisfied, are discussed in Section 9.2 of the Safety Analysis Report. ~~Section 4.3 discusses the effect on ECCS-LOCA response of the ARTS implementation along with the expansion of the power/flow map using the EPU MELLLA regime.~~ The PUSAR is misleading. APRM setdown was never credited in the lOCA analysis – either before or after ARTS.

The ARTS-specific changes are:

1. The requirement for setdown of the APRM scram and rod blocks is deleted.

2. New power-dependent MCPR adjustment factors, MCPR(P), are added.

3. New flow-dependent MCPR adjustment factors, MCPR(F), replace the K_F multiplier.

4. New power-dependent LHGR adjustment factors, LHGRFAC(P), are added.

5. New flow-dependent LHGR adjustment factors, LHGRFAC(F), are added.

6. The affected Technical Specifications and associated Bases are modified or deleted, as required.

As discussed in the subsections below, the ARTS limits are generally determined or confirmed using bounding QCNPS-specific analyses, although it is stated that cycle-specific limits may be developed and used.

9.2.1 Elimination of APRM Gain and Setpoint Requirement

The original ARTS development program included generic evaluations over a wide range of plant configurations, operating parameters, and power and flow conditions to generate a large database of limiting transients, which can also be applied to QCNPS operation in the MELLLA power/flow map region. This generic database was used to develop a methodology for specifying the MCPR and LHGR plant operating limits, which assures that margins to fuel safety limits are equal to or larger than those achieved with the APRM gain and setpoint requirements. These generic evaluations also determined the adequacy of power dependent limits developed for two power ranges:

- between rated power and the power level (Bypass) where the reactor scram on turbine stop valve closure or turbine control valve fast closure is bypassed, and

- between Bypass and 25 percent of the rated power.

Bypass is 38.5 percent of EPU rated thermal power.

The licensee stated that the generic power-dependent (and flow-dependent) MCPR and LHGR limits developed for use in the first power range were verified by QCNPS-specific analyses of the limiting transients. Between Bypass and 25 percent power, QCNPS-specific analyses were performed to establish unique limits for the low power range. The licensee stated that these QCNPS-specific limits were developed with sufficient conservatism to cover future reloads of GE14 fuel, using the GEXL-PLUS correlation form and the GEMINI analysis methods, although cycle-specific limits may be used in the future for any portion of the range (statement added for clarity).

9.2.1.1 ARTS AOO Analysis Assumptions

To develop and verify the plant-specific, but cycle-independent ARTS thermal limits, the AOO transient analyses were performed using the EPU thermal power of 2957 MWt and 108 percent rated core flow (ICF option), as shown on the licensee's Safety Analysis Report power flow map of Figure 2-1. The plant EPU conditions and system setpoints are summarized in Tables 1-2 and 5-1. The Feedwater Controller Failure (FWCF) event is analyzed with a feedwater temperature of 256 F at rated power (equivalent to a reduction of 100F).

9.2.1.2 Power-Dependent MCPR Limit, MCPR(P)

From Bypass to rated power, bounding power-dependent trend functions (K_P) are used as multipliers to the rated operating limit (OL) MCPR values to determine the MCPR(P) limits. The licensee stated that the FWCF event is more limiting than the generator load reject without bypass as the initiating power is reduced. The QCNPS-specific calculated values were compared with the generic limits in Table 9-3 of the licensee's Safety Analysis Report, verify the applicability of the generic limits.

The licensee noted that the QCNPS ARTS program is a partial application (like LaSalle), in that QCNPS is not implementing hardware changes to the RBM system, which would provide protection for an off-rated rod withdrawal event (RWE). Instead, analyses of the off-rated RWE event with no rod block were performed to verify that the combination of the generic K(P) and the QCNPS-specific MCPR(P) limits bound the SLMCPR limit requirement.

The licensee stated that the idle recirculation loop startup (IRLS) was considered generically for ARTS and that the assumption of an initial 50 F delta-T between loops is appropriate and consistent with the QCNPS TS requirements.

Below Bypass and above 25 percent power, bypass of the direct scram on closure of the turbine stop valve and turbine control valve change the characteristics of the FWCF and load reject without bypass (LRNBP) transient events. Both events were analyzed and the MCPR(P) limits are calculated as OLMCPR bounding values for both initial high-flow and low-flow conditions. The calculated and limiting values are shown in Table 9-4 and Figure 9-5 of the licensee's Safety Analysis Report.

9.2.1.3 Power-Dependent LHGR Limit, LHGRFAC(P)

Power-dependent LHGR limits are achieved by a LHGRFAC(P) multiplier derived from the generic database. The licensee states that, for GNF fuel designs, both incipient centerline melting of the fuel and the plastic strain of the cladding are considered. QCNPS-specific transient analyses were performed to confirm the applicability of the generic LHGRFAC(P) limits above Bypass, as shown in Table 9-5 of the licensee's Safety Analysis Report. Below Bypass, both high and low core flow multipliers were calculated by QCNPS-specific analyses to establish limits with sufficient margin to apply to future GE14 reloads, as shown in Table 9-6 of the licensee's Safety Analysis Report. Figure 9-6 shows the bounding QCNPS-specific power dependent LHGRFAC(P) multipliers for both power ranges and for both low and high initial core flow.

9.2.1.4 Flow-Dependent MCPR Limit, MCPR(F)

The licensee stated that the flow-dependent MCPR(F) limits ensure that the Safety Limit MCPR is not violated during recirculation flow increase transient events. The design basis event is a slow-flow increase which is not terminated by a scram, but which stabilizes at a new higher power corresponding to the maximum possible core flow. The generic flow dependent MCPR limits were verified by performing flow runout at a typical mid-cycle exposure plant condition (at constant xenon), along a rod line bounding the MELLLA power up to the maximum core flow runout at 108 percent core flow. The bounding generic and cycle-independent ARTS MCPR(F) limits are shown in Figure 9-7 of the licensee's Safety Analysis Report.

9.2.1.5 Flow-Dependent LHGR Limit, LHGRFAC(F)

The licensee stated that the flow-dependent LHGRFAC(F) limits assure that all fuel thermal-mechanical design bases are met for a slow recirculation flow runout event. The same generic transient analyses were statistically evaluated for the bounding overpower as a function of the initial and maximum core flow, to ensure that the peak transient LHGR would not exceed fuel mechanical limits. These bounding flow dependent limits, as shown in Figure 9-8 of the licensee's Safety Analysis Report, are generic and cycle-independent.

9.2.2 Overall Governing MCPR and LHGR Limits

The licensee stated that for a given power/flow statepoint (P,F) all four limits (MCPR(P), LHGRFAC(P), MCPR(F), and LHGRFAC(F) are determined and the most limiting MCPR (maximum value) and most limiting LHGR (minimum value) will be the governing limits. Note that the MCPR curves have to be adjusted if the assumed SLMCPR value of 1.09 (Table 9-1) is to be changed. Changing the TS SLMCPR would require a separate submittal.

9.3 Design Basis Accidents

9.3.1 Background to Evaluation of Radiological Consequences of Design Basis Accidents

ELTR1 provides generic guidelines for justifying operation at up to 20% increased core thermal power. The guidelines for the performance of radiological evaluations are contained in Section 5.4 and Appendix H of ELTR1. Section 5.4 provides that the magnitude of the potential radiological consequences of a design basis accident (DBA) is proportional to the quantity of fission products released to the environment. This release is a product of the activity released from the core and the transport mechanisms between the core and the effluent release point. In general, the inventory of fission products in the fuel rods, the creation of radioactive materials outside of the fuel by irradiation, and the concentration of radioactive material in the reactor coolant system are directly proportional to the rated thermal power. Thus, an increase in the rated thermal power can be expected to increase the inventory of radioactive material that is available for release. The previously analyzed transport mechanisms could be affected by plant modifications associated with the power uprate, potentially resulting in a larger release rate. The ELTR1 provides that the EPU application will provide justification that current radiological consequences are still bounding and within applicable criteria, or will provide re-analysis of any areas adversely affected by the proposed uprate.

Appendix H of the ELTR1 describes the generic bases to be used in the generic radiological evaluations or in re-analysis of any areas adversely affected by the EPU. ELTR1 is based, in part, on two limitations: (1) the reactor core design undergoes only small modifications by the change in power, and (2) the core design is accomplished with fuel bundles of the same type. If there are significant changes to the fuel loading or design parameters, the EPU application will need to re-assess changes to the isotopic concentrations in the fuel. Also, the impact of increased fuel enrichment and burnup would need to be addressed if these parameters exceed any of the requirements of 10 CFR 51.52(a).

Appendix H of the ELTR1 provides that existing calculations as shown in the current UFSAR are valid and that, with few exceptions, the postulated results are changed by the magnitude of the change in radiation source. The increased consequences can be resolved on a ratio of the sources basis. Exceptions are associated with changes in radioactive material transport assumptions and methods caused by modifications to the plant pursuant to the uprate. The appendix provides that new calculations will be carried out only as necessary. There are some design basis events, such as a main steam line break, which release the radioactive materials in reactor coolant to the environment. Since the evaluations for these events utilize the reactor coolant concentrations established by the technical specifications, the consequences of these events will not change unless the mass of coolant lost changes.

Section 2.8 of the NRC staff position on ELTR1 (Reference 4) provided that the existing calculations found in the SAR should remain valid as a result of the EPU and that the doses will be increased by the magnitude of the change in the source term. The staff noted that the increased doses must meet the dose acceptance criteria in the plant's licensing basis and that the licensee will demonstrate assumptions and conditions stated in the ELTR1 are met. If these assumptions are not met, applicants will be expected to recalculate the affected radiological analyses.

ELTR2 presents specific evaluations of areas of licensing review that are generically applicable to some or all of the BWR product lines. Section 5.3.2.2.3 of ELTR2 addresses the EPU impact on radiological consequences of design basis accidents and provides information comparable in scope and detail to that provided in Section 5.4 and Appendix H in ELTR1.

9.3.2 Plant-Specific Evaluation

Section 9.3 in the licensee's safety analysis (Reference 2) addresses the impact of the EPU on the previously analyzed radiological consequences of DBAs for QCNPS. This section is based on the guidelines in Section 5.4 of ELTR1. The plant-specific radiological assessments were evaluated at 102% of the proposed rated thermal power, consistent with the guidance of Regulatory Guide 1.49, "Power Levels of Nuclear Power Plants."

Development of Plant-Specific Scaling Factors

The core fission product inventory used in performing the existing, pre-EPU, radiological consequence analyses is based on the curies per megawatt-thermal (Ci/MWt) constants provided in TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites." This document, published in 1962, provides Ci/MWt values for several reactor fission products. These values are representative of the low burnup fuels considered at that time and the fission product generation and depletion analysis methodology then available. These inventories were dominated by fission product yields from uranium-235 (U-235) fission. During power operation, U-239 is produced by the irradiation of U-238, with the U-239 subsequently decaying to plutonium-239 (Pu-239), which is fissionable. At current high fuel burnup levels, the fission of Pu-239 contributes significantly to the fission product inventory in the core. Also, the fission product yields from Pu-239 differ from those for U-235, resulting in changes in the fission product composition. In order to address these impacts of the EPU, EGC had a re-calculation performed of the core fission product inventory for GE14 fuel and a 24 month fuel cycle using the industry-accepted ORIGEN2 code. This code utilizes updated fission product yields and decay chains and includes the fission product contributions of Pu-239 and other transuranic nuclides. In re-calculating the fission product inventory, EGC has addressed the ELTR1 guidelines regarding the assessment of the impacts of the EPU and higher burnup fuel impact on radionuclide composition and inventory. The staff finds this approach acceptable.

The scaling factor used to correct the previously analyzed thyroid doses for the impact of the EPU is the ratio of the ORIGEN2 iodine inventories at the EPU power level to the previous TID-14844 iodine inventories at the pre-EPU power level, weighted for the iodine dose factors. Similarly, the scaling factor used to correct the previously analyzed whole body doses for the impact of the EPU is the ratio of the ORIGEN2 noble gas inventories at the EPU power level to the previous TID-14844 noble gas inventories at the pre-EPU power level, weighted for the whole body dose factors. The resulting scaling factors for the thyroid dose and the whole body dose due to the change in core inventory are 1.27 and 1.18, respectively.

Since the previous control room dose DBA LOCA analyses were performed using a fission product inventory based on the pre-EPU rated thermal power without the 2 percent margin, EGC increased the scaling factors for the control room to 1.30~~29~~ for thyroid and 1.20~~19~~ for whole body for the DBA LOCA results only.

The staff finds that the method used to determine the scaling factors to be appropriate and consistent with the staff-approved ELTR1 and ELTR2 and the conditions identified in the associated staff SER.

Application of Scaling Factors to Pre-EPU Analyses

EGC considered the plant-specific EPU impact on the following DBA accidents: LOCA, control rod drop accident (CRDA), fuel handling accident (FHA), main steam line break (MSLB) outside containment, instrument line break (ILB) outside containment, and an offgas treatment system component failure. The results of these analyses are tabulated in the Table below. For the LOCA, CRDA, and FHA, the EPU does impact the fission product inventory. As such, the radiological consequences postulated in prior analyses were multiplied by the plant-specific scaling factors described above. For the LOCA and the FHA, there were no plant modifications that would impact the transport of radioactive material to the environment so no further adjustments or re-analyses were necessary. For the mechanical vacuum pump release pathway during a CRDA, the scaling factors were increased to account for the increased main steam line flow into the main condenser at EPU conditions.

For the MSLB and the ILB accidents, the analyses assume that the reactor coolant specific activity is at the maximum value allowed by technical specification, expressed in terms of dose equivalent Iodine-131. As such, these analyses are not affected by the EPU. The source term used pre-EPU analyses for evaluating the offgas treatment system component failure was set conservatively and independently of the reactor thermal power. For the MSLB, offgas, and ILB accidents, the EPU does not affect transport assumptions used in the analyses. Specifically, EGC has proposed to operate at the same reactor dome pressure used pre-EPU for post-EPU operations. While the post-EPU normal operational steam flow will be greater, the flow restrictors in the steam lines establish the maximum flow rate at which steam will flow during MSLB conditions. The pre-EPU analyses were based on the maximum flow rate which is unaffected by the EPU. As a result of these considerations, the EPU has no impact on previously analyzed consequences of the MSLB, ILB, and offgas treatment system component failure events.

QCNPS RADIOLOGICAL ANALYSIS RESULTS, REM

	0-2 hr EAB		30-day LPZ		30-day CR	
Event	Whole Body	Thyroid	Whole Body	Thyroid	Whole Body	Thyroid
Loss-of-Coolant Accident						
Pre-EPU	5.0	120.0	<5.0	<120.0	0.12	22.8
Post EPU	6.0	152.0	<6.0	<152.0	0.14	29.6
Criterion	25.0	300.0	25.0	300.0	5.0	30.0
Control Rod Drop Accident						
Pre-EPU	2.9	9.4	0.5	<1.0	0.22	21.8
Post EPU	3.4	12.1	0.6	<1.3	0.27	28.0
Criterion	6.25	75.0	6.25	75.0	5.0	30.0
Fuel Handling Accident						
Pre-EPU	0.36	9.9	0.04	0.69	0.012	7.66
Post EPU	0.42	12.6	0.05	0.87	0.014	9.73
Criterion	6.25	75.0	6.25	75.0	5.0	30.0

Control Room Doses

As noted above, EGC evaluated the consequences of the EPU on control room habitability, using the scaling methodology presented in the staff-approved ELTR1 and ELTR2 topical reports. The staff is currently evaluating, on a generic basis, deficiencies in the design, operation, and maintenance of control room habitability systems and is pursuing appropriate regulatory action. The staff expects to issue a generic letter and regulatory guidance on these issues in 2001. One of the primary deficiencies identified by the staff involves unsubstantiated assumptions at many plants regarding the amount of unfiltered in-leakage to the control room envelope during accident conditions. Due to the magnitude of the potential increases in post-EPU accidents, the staff reviewed the EGC submittal to determine whether there was reasonable assurance that the QCNPS control room habitability systems could perform their design function to provide plant operators a habitable environment in which to take actions necessary to operate the plant in a safe manner.

The staff reviewed an earlier license amendment application dated May 19, 1997 for QCNPS. In this application, the then-Commonwealth Edison, described the results of tracer gas testing of the unfiltered in-leakage and stated that the measured unfiltered in-leakage was less than leakage previously assumed in control room habitability analyses. The May 19, 1997 licensing action was retracted by Commonwealth Edison. For the EPU application, the staff requested EGC provide additional information confirming that the in-leakage conclusion was still valid. In their response dated July 6, 2001, EGC asserted that the in-leakage conclusion was still valid and described ongoing programs and surveillance tests that are intended to assure that any degradation in unfiltered control room in-leakage is identified and corrected. While the staff resolution of the control room habitability issue may deem it necessary to generically require periodic boundary integrity re-testing, the staff believes that the earlier testing and ongoing control program at QCNPS provide reasonable assurance that the EPU will not have an adverse impact on control room habitability. The staff's acceptance of EGC's unfiltered in-leakage conclusions does not foreclose on any future generic regulatory actions that may become applicable to QCNPS in this regard.

The staff reviewed the assumptions, inputs, and methods used by EGC to assess the radiological impacts of the proposed EPU at QCNPS. In doing this review, the staff relied upon information provided by EGC, staff experience in doing similar reviews and, the staff-accepted ELTR1 and ELTR2 topical reports. The staff finds that EGC used analysis methods and assumptions consistent with the conservative guidance of ELTR1 and ELTR2. The staff compared the doses estimated by EGC to the applicable criteria. The staff finds, with reasonable assurance, that the licensee's estimates of the EAB, LPZ, and control room doses will continue to comply with 10 CFR Part 100 and 10 CFR 50, Appendix A, GDC 19, as clarified in NUREG-0800 Sections 6.4 and 15. Therefore, QCNPS operation at the proposed EPU rated thermal power is acceptable with regard to the radiological consequences of postulated design basis accidents.

9.4 Special Events

9.4.1 Anticipated Transient Without Scram (ATWS)

The ATWS is defined as an AOO with failure of the reactor protection system to initiate a reactor scram to terminate the event. The requirements for ATWS are specified in 10 CFR Part 62. The regulation requires BWR facilities to have the following mitigating features for an ATWS event:

1. a standby liquid control (SLC) system with the capability of injecting a borated water solution

with reactivity control equivalent to the control obtained by injecting 86 gpm of a 13 weight percent sodium pentaborate decahydrate solution at the natural boron-10 isotope abundance into a 251 inch inside diameter reactor vessel

2. an alternate rod insertion (ARI) system that is designed to perform its function in a reliable manner and that is independent from sensor output to the final actuation devise

(3) equipment to trip the reactor coolant recirculation pumps automatically under conditions indicative of an ATWS.

BWR performance during an ATWS is also compared to other criteria, which were used in the development of the ATWS safety analyses described in NEDO-24222, "Assessment of BWR Mitigation of ATWS, Volume II" (Reference 43). The criteria include: (a) limiting peak vessel bottom pressure to less than the ASME Service Level C limit of 1500 psig, (b) ensuring that the peak cladding temperature remains below the 10 CFR 50.46 limit of 2200F, (c) ensuring that the cladding oxidation remains below the limit in 10 CFR 50.46, (d) limiting peak suppression pool temperature to less than 202F (which is the limiting temperature selected to ensure that the LOCA analysis results remain bounding ~~containment design temperature~~), and (e) limiting the peak containment pressure to a maximum of 62 psig (110 percent of containment design pressure).

The ATWS analyses assume that the SLC system will inject within a specified time to bring the reactor subcritical from the hot full power condition and to maintain the reactor subcritical after the reactor has cooled to the cold shutdown condition. ~~For every reload, the licensee evaluates how plant modifications, reload core designs, changes in fuel design, and other reactor operating changes affect the applicability of the ATWS analysis of record.~~ In accordance with the GESTAR methodology, the licensee re-analyzes the ATWS event if changes to fuel type or significant plant modifications will affect the ATWS response. (The revised words are more in line with the GESAT method.)

The licensee stated that QCNPS meets the ATWS mitigation requirements defined in 10 CFR 50.62, because: (a) an ARI system is installed, (b) the boron injection capability is equivalent to 86 gpm, and (c) an automatic ATWS-RPT has been installed. Section L.3 of ELTR1 discusses the ATWS analyses and provides a generic evaluation of the following limiting ATWS events in terms of overpressure and suppression pool cooling: (a) MSIV closure, (b) PRFO, (c) LOOP, and (4) inadvertent opening of a relief valve (IORV). The licensee performed the ATWS analyses, as discussed in ELTR1, at the MELLLA/EPU operating condition to demonstrate that QCNPS meets the ATWS acceptance criteria. To provide a benchmark for the plant response to limiting ATWS events at EPU~~E~~ conditions, the licensee also performed the ATWS analyses based on the current rated thermal power.

Section 9.4.1 of the licensee's Safety Analysis Report lists the key input parameters used in the ATWS analyses and the corresponding results (peak vessel bottom pressure, peak cladding temperature, peak suppression pool temperature and peak containment pressure). The licensee stated that the results of the ATWS analyses meet the ATWS acceptance criteria. Therefore, the plant's response to an ATWS event for EPU operation is acceptable.

The analysis shows that the ATWS PCT for the current RTP is 1478F and that the EPU PCT is 1418F. The staff confirmed during the audit that the stated PCT values are correct and examined the bases for these values. The staff also found similar trends (pre-EPU PCTs higher than the EPU PCTs) for other licensee calculations. Since the ATWS analyses are based on NRC-approved methods and the licensee performed the ATWS analyses at the MELLLA/EPU conditions, the staff accepts the licensee evaluation.

The staff concludes that QCNPS meets the ATWS mitigating features stipulated in 10 CFR 50.62 and that the results of the ATWS analyses for EPU/MELLLA operation meet the ATWS acceptance criteria. Future reload analyses will confirm that the plant response to an ATWS event, based on the cycle-specific condition, will continue to meet the ATWS acceptance criteria.

9.4.2 Station Blackout (SBO)

The staff has reviewed information provided by the licensee to determine the impact of the power uprate on the existing analysis performed for station blackout (SBO). The licensee stated that SBO evaluation was performed using the guidelines of Nuclear Management and Resources Council (NUMARC)-8700, "Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors," except where RG 1.155 takes precedence. The licensee stated that the plant responses to and coping capabilities for an SBO event are not affected by operation at the EPU level, **because the increase in decay heat for EPU is absorbed by the ~~operation of the isolation condenser~~**torus water inventory. There are no changes to the systems and equipment used to respond to an SBO, nor is the required coping time changed.

The initial conditions and assumptions for SBO under EPU conditions have been revised to be consistent with NUMARC 87-00 and RG 1.155. The EPU decay heat analysis assumes an operating history of 100 days at the full uprated power conditions of 2957 MWt prior to the SBO event.

On April 6, 2001, the licensee provided additional information describing its evaluation of the EPU effect on the dominant areas of concern containing equipment necessary to mitigate the SBO event:

Drywell Temperature

The licensee stated that the reactor pressure vessel temperature and pressure remain the same and there are no significant changes in drywell heat sources. A slight (<17 °F) increase in feedwater temperature occurs post-EPU, however the licensee determined that significant margin (calculated to be 74 °F in pre-EPU calculations) to the drywell design temperature would remain.

Suppression Pool Temperature

The licensee determined that the increase in due to additional decay heat post-EPU would be less than 6 °F. Pre-EPU evaluations determined that suppression pool temperature would not exceed 130 °F in the one-hour period without AC power. The temperature increase is bounded by the containment analysis for LOCA conditions and significant margin to design limits remains.

Control Room Ventilation and Auxiliary Electric Equipment Room Ventilation

The licensee's pre-EPU calculations indicated that the peak one-hour temperatures were acceptable. The heat loads in these areas are primarily related to indicating lights and other non-power dependent electrical equipment and remain the same as before. Therefore, the pre-EPU evaluation remains valid.

RCIC Room Heatup

The licensee noted that the pre-EPU calculation for room temperature assumed a constant heat load from RCIC operation. Therefore; notwithstanding the slightly increased operation time post-EPU, the results remain valid.

~~No HPCI Room Heatup Evaluation ?~~

Contaminated Condensate Storage Inventory

The higher decay heat for the EPU operation would increase the boil-off rate; therefore, the ability of the plant to maintain core coverage, using the available inventory in the CCST could be affected.

The staff has reviewed QCNPS ability to cope during a station blackout and to ensure core cooling and coverage during the event. The staff accepts the licensee's conclusion that the plant's SBO coping capabilities will not be adversely affected by EPU operation. This paragraph may be misplaced. It sounds like a general conclusion about SBO, rather than an evaluation of the CST.

DC Battery Capacity

The licensee stated that pre-EPU battery cell sizing calculations were performed for the 125 volt dc and 250 volt dc batteries. These calculations considered a four-hour load profile based on a combined set of loads from a variety of events. It was determined for both the 125 volt dc and 250 volt dc batteries that adequate margin exists. The battery load demands during the one-hour SBO duration are slightly increased under EPU conditions. However, the current pre-EPU battery load profile remains bounding because it assumes a more restrictive scenario of multiple HPCI initiations during a 4-hour duration.
Based on the review of the licensee's rationale, the staff finds that the impact of plant operations at the proposed EPU on the systems and equipment used to cope with an SBO event is insignificant. The staff concludes that the plant will continue to meet the requirements of 10 CFR 50.63 for EPU conditions.

10. ADDITIONAL ASPECTS OF EXTENDED EPU

10.1 High-Energy Line Breaks

The licensee's plan to achieve the proposed higher power at the QCNPS is to expand the operating envelope on the power/flow map through implementation of maximum load line limit analysis (MELLLA). Operation at the EPU level does not require an increase in the reactor vessel dome pressure over the pre-EPU value to supply more steam to the turbine. Therefore, plant operations at the EPU level will have an insignificant impact (due to changes in the fluid conditions, i.e., pressure or enthalpy, within the system piping) to the mass and energy release rates following a high energy line break (HELB) outside the primary containment.

10.1.1 Temperature, Pressure and Humidity Profiles Resulting From HELB

The licensee performed a HELB analysis for all systems (e.g., main steam system, feedwater system, reactor core isolation cooling system, etc.) evaluated in the UFSAR. The licensee stated that affected buildings and cubicles that support the safety-related functions are designed for the resulting environmental conditions (i.e., pressure, temperature and humidity profiles) due to plant operations at the proposed EPU level. The equipment and systems that support a safety-related function were evaluated and determined to be qualified for the environmental conditions.

Based on the review of the licensee's rationale, the staff concurs with the licensee that the environmental conditions used to qualify equipment and systems that support a safety-related function either remain bounding, or the rooms and equipment have been appropriately evaluated for the EPU effects. The pressure, temperature and humidity profiles resulting from a HELB outside the containment are acceptable for plant operations at the proposed EPU level.

10.1.1.1 Main Steam Line Break

The licensee stated that the critical parameter normally affecting the main steam line break (MSLB) analysis relative to the EPU would be an increase in reactor vessel dome pressure. Since there is no increase in the reactor vessel dome pressure, there is no increase in the blowdown rate following an MSLB in the steam tunnel. Therefore, the pressure and temperature profiles following an MSLB in the steam tunnel are not affected for plant operations at the proposed EPU level. The licensee letter dated August 7, 2001, (Reference 19) provided additional information regarding the effect of increasing the main steam isolation setpoint on high energy line breaks. The MSLB was analyzed with a circumferential rupture, resulting in the flow restrictor choking flow and thus bounding other breaks. Credit was taken for isolation on high flow; however the licensee noted that in the event of smaller breaks not resulting in high steam line flow isolation~~;~~, low steam line pressure or high steam tunnel temperature isolation signals will function to isolate the HELB. These isolation signals are governed by the QCNPS technical specifications.

Based on the review of the licensee's rationale and the experience gained from the staff's review of power uprate applications for similar BWR plants, the staff concurs with the licensee that the existing pressure and temperature profiles following an MSLB in the steam tunnel are not affected and are acceptable for plant operations at the proposed EPU level.

10.1.1.2 Feedwater Line Break

At the EPU level, the feedwater temperature, pressure and flow rate increase slightly, resulting in an increase of 6% in the mass and energy release for a feedwater line break. The licensee performed an analysis for feedwater line break in the steam tunnel. The licensee provided additional details of the feedwater line break analyses in their letter dated August 7, 2001 (Reference 19). The feedwater line break was analyzed with a concurrent main steam line break to establish design basis for QCNPS. For the effect of the EPU, the licensee ran benchmark calculations using both current and EPU conditions to evaluate the effects of the changes. The results were used to estimate that the peak pressure would remain lower than the design basis value of 27.5 psia used for main steam tunnel environmental parameters. The licensee also evaluated the long term temperature profiles using the COMPARE code to calculate current and EPU temperatures. The results indicated that the temperature difference was insignificant and within the accuracy of the calculation. The licensee stated that design margins within the pre-EPU HELB analysis for feedwater line break in the steam tunnel are conservative and remain bounded by the main steam line break with a concurrent feedwater line break.

Based on the review of the licensee's rationale, the staff concurs with the licensee that the pressure and temperature profiles following a feedwater line break in the main steam tunnel have been adequately evaluated.

10.1.1.3 ECCS Line Breaks

Because there is no increase in the reactor dome pressure relative to the current analyses, the mass release rate following a HPCI line break does not increase. The licensee stated that the previous analyses for these line breaks are bounding for the EPU conditions.

Based on the review of the licensee's rationale and the experience gained from the staff's review of power uprate applications for similar BWR plants, the staff concurs with the licensee that the previous analyses for these line breaks remain bounding for the EPU conditions.

10.1.1.4 RCIC System Line Breaks

Because there is no increase in the reactor dome pressure relative to the current analyses, the mass release rate following a RCIC system line break does not increase. The licensee stated that the previous analyses for these line breaks are bounding for the EPU conditions.

Based on the review of the licensee's rationale and the experience gained from the staff's review of power uprate applications for similar BWR plants, the staff concurs with the licensee that the previous analyses for these line breaks remain bounding for the EPU conditions.

10.1.1.5 Reactor Water Cleanup (RWCU) Line Breaks

The licensee performed evaluations and stated that as a result of the small increase in subcooling with no reactor vessel dome pressure increase, the blowdown rate increases slightly. Conservative model assumptions were stated to more than offset the effect of the mass release increase. The sub-compartment pressure increase was evaluated and determined to be acceptable. Therefore, the previous HELB analysis regarding RWCU line breaks remains bounding for the EPU condition.

Based on the review of the licensee's rationale and the experience gained from the staff's review of power uprate applications for similar BWR plants, the staff concurs with the licensee that the previous analysis for RWCU line breaks remains bounding for the EPU condition.

10.1.1.6 Instrument Line Breaks

The licensee evaluated the instrument line break analysis which indicates that the blowdown rate remains the same and there is no pressure increase. Therefore, the previous HELB analysis regarding the instrument sensing line breaks remains bounding for the EPU condition.

Based on the review of the licensee's rationale and the experience gained from the staff's review of power uprate applications for similar BWR plants, the staff concurs with the licensee that the previous analyses for the instrument sensing line breaks remain bounding for the EPU conditions.

10.1.1.7 Internal Flooding From HELB

The licensee stated that the analyses for flooding in the main steam tunnel assumes flooding of the entire below grade volume. This assumption is conservative and bounding for the EPU conditions. In their August 7, 2001, response to the staff, the licensee addressed the effects of plant operations at the proposed EPU on the internal flooding for other systems outside the containment. The licensee stated that other high energy line breaks in the turbine building; such as breaks in the feedwater and condensate systems, are bounded by the worst-case internal flooding from a postulating pipe break in the moderate energy circulating water system inside the turbine building.

Based on the review of the licensee's rationale and the experience gained from the staff's review of power uprate applications for similar BWR plants, the staff concurs with the licensee that the previous analyses regarding internal flooding remain bounding for the EPU conditions.

10.1.2 Moderate-Energy Line Break

The licensee stated that a moderate energy line break (MELB) analysis is based on system parameters not changed with the EPU. The circulating water system can accommodate the EPU heat load at the existing system flow rate; therefore, changes are not planned. In response to the staff's RAI the licensee addressed existing moderate energy flooding analyses and features to protect safety-related equipment from flooding in the turbine building. At QCNPS this includes the residual heat removal service water pumps which are located in watertight vaults. Existing active protective features for circulating water flooding include a trip of the~~se~~ circulating water pumps on high level in the condenser pit area; however the ultimate consequence remains flooding of the building to the level of the river through gravity feed.

With regard to MELB for the proposed EPU conditions, the primary concern is internal flooding resulting from a postulating MELB outside the containment. As indicated in the above Section 10.1.1.7, the worst-case internal flooding is from a postulating pipe break in the circulating water system inside the turbine building. The previous evaluations of internal flooding remain bounding for the proposed EPU as there is no change in the circulating water system. Therefore, the staff concludes (**with respect to the applicable areas for which we have the primary review responsibility**) that MELB is not a concern for QCNPS operations at the proposed EPU conditions.

10.2 Equipment Qualifications

10.2.1 Environmental Qualification of Electrical Equipment

The licensee evaluated the safety-related electrical equipment to ensure qualification for the normal and accident conditions expected in the area in which the devices are located. The licensee applied the margins to the environmental parameters in accordance with the original qualification program and determined that no change is needed for EPU.

EPU is expected to increase both the normal and post-accident radiation conditions (integrated dose) in the plant by no more than the percentage increase in power level. However, the licensee performed EPU assessment in conjunction with the change to a 24-month fuel cycle. The increase in accident conditions resulting from combined effect of EPU and a 24-month fuel cycle is dependent, as a function of time, on the controlling radiation source (i.e., suppression pool water, drywell atmosphere, etc.) and the credited shielding. The increase in radiation levels reflect the use of current computer codes, methodology, and nuclear data in developing the uprated core inventory versus the methodology, computer tools, and nuclear data in the development of the original licensing basis core inventory. The increase reflects the inclusion of several hundred additional isotopes in the new core, as well as a 2 percent margin for instrument error recommended by Regulatory Guide 1.49. For purposes of equipment qualification, the maximum increase in the normal and accident radiation environment applicable to existing safety-related equipment is conservatively evaluated to be 20 and 40 percent respectively.

10.2.1.1 Inside Containment

Environmental qualification (EQ) for safety-related electrical equipment located inside the containment is based on main steam line break and/or design basis accident (DBA)/ loss-of-coolant accident (LOCA) conditions and their resultant temperature, pressure, humidity, and radiation consequences and includes the environments expected to exist during normal plant operation. The maximum accident radiation levels used for qualification of equipment inside containment are from a DBA/LOCA. The review of the EPU conditions identified some equipment located within the containment, which could potentially be affected by the higher accident radiation level. However, the qualification of this equipment was resolved by refined radiation calculations or by the use of new test data. The licensee stated (Reference 9) that the drywell pressure and temperature conditions are impacted for EPU as follows.

- The present drywell peak pressure for qualification of 63 psia is bounding the EPU condition.

- The present and EPU drywell temperature profiles are shown below.

Time (hours)	Present Temperature (° F)	EPU Temperature (° F)
0.01	334	338
0.5	334	338
0.57	287	303
0.8	282	288
61	165	183
588	128	146
8760	112	130

For all equipment inside the containment within the EQ program, evaluations were performed to demonstrate that existing environmental documentation was adequate to meet the revised temperature and pressure values due to EPU. Evaluations were done for each equipment type using the following approach.

1. The qualification test temperature conditions for the required operability period during the first 24 hours following a LOCA were shown to envelop the corresponding EPU temperature profile,

2. The qualification test temperature conditions for the required operability period beyond 24 hours to 1 year following a LOCA were shown to meet the revised EPU temperature profile using the Arrhenius methodology.

3. Maximum test pressure was shown to envelop the revised peak pressure for EPU.

The licensee concluded that EPU did not result in any changes to operating times for equipment required to operate following an accident.

The current EQ for equipment inside the containment is based on a normal relative humidity of 20 percent to 90 percent and an accident relative humidity of 100 percent. This is not changed for the EPU.

Additionally, operations at EPU conditions changes the radiation environments for certain plant areas in which electrical equipment is located. For the EQ equipment, revised radiation values were compared to the existing posted qualified test values. This comparison identified some equipment (electrical penetration assemblies and cables) where the EPU profile exceeded the current posted values. Material analysis and other test report data for the electrical penetration assemblies were utilized to demonstrate qualification to the EPU values. A unique radiation dose analysis was performed to demonstrate qualification to the EPU values for cables.

In summary, the safety-related electrical equipment inside the primary containment are qualified to the new temperature and radiation profiles due to the EPU.

10.2.1.2 Outside Containment

Accident temperature, pressure, and humidity environments used for qualification of equipment outside containment result from a main steam and feedwater line breaks in the steam tunnel, or other high-energy line breaks (HELBs) whichever is limiting for each plant area. The accident temperature, pressure, and humidity conditions outside containment, resulting from a LOCA inside containment, may change with the power levels as a result of the increased suppression pool temperature. The licensee stated (Reference 9) that no changes to pressure or humidity environments result in areas outside containment for a LOCA inside containment. Changes for temperature environments outside containment for a LOCA inside containment are being determined and evaluated for effects on qualification of electrical equipment within the EQ program. Evaluations will be done to show that the existing environmental documentation is adequate to meet the revised temperature profile due to EPU. Evaluation will be done for each equipment type using the following approach.

1. Existing documentation will be used to show that the qualification test temperature profile envelops the revised peak temperature for EPU.

2. The qualification test will be shown to meet the revised Post LOCA conditions outside containment for EPU using the Arrhenius methodology.

The licensee stated (Reference 18) that the reviews of EQ equipment were performed and shown to meet the revised environmental parameters following EPU. Qualification was shown by one or more of the following approaches. These are all industry standard methods for EQ reviews:

1. Existing documentation was used to show that current qualification test temperature profile and radiation dose bound the EPU conditions.
2. An additional test report was obtained for the equipment.
3. New test data on materials was used to demonstrate qualification.
4. An equipment unique radiation calculation was performed.

Most equipment was shown to be qualified for EPU conditions with little or no additional analysis, as identified in item 1 above.

The Rosemount Pressure Transmitter installed outside primary containment which required more rigorous evaluation. Location specific radiation dose calculation to determine specific total dose for the transmitter was used to qualify for the revised EPU environmental conditions.

In summary, the safety-related electrical equipment outside the primary containment are qualified to the new temperature and radiation profiles due to the EPU.

10.2.2. EQ of Mechanical Equipment with Nonmetallic Components

In their August 7, 2001, response to the staff, the licensee stated that the QCNPS plant design control program ensures that non-metallic components (i.e., seals, gaskets, lubricants and diaphragms) are properly specified and procured for the environment in which they are intended to function. The licensee stated that the changes in operating conditions, as well as normal and accident environmental conditions, have been determined for EPU. These changes are minor compared with the range of conditions allowed for mechanical components.

Based on the review of the licensee's rationale, and since the changes for the normal and accident environmental conditions inside and outside the containment, and system process temperatures are negligible, the staff concludes that the environmental qualification of the non-metallic components exposed to the EPU conditions is not adversely impacted.

10.2.3 Mechanical Components Design Qualification

10.2.3.1 Equipment Seismic and Dynamic Qualification

The licensee evaluated equipment qualification for the power uprate condition. The dynamic loads such as SRV discharge and LOCA loads (including pool swell, condensation oscillation, and chugging loads) that were used in the equipment design will remain unchanged as discussed in Section 4.1.2 of Reference 2. This is because the plant-specific hydrodynamic loads which are based on the range of test conditions for the design-basis analysis at QCNPS, are bounding for the power uprate condition.

Based on its review of the proposed power uprate amendment, the staff finds that the original seismic and dynamic qualification of safety-related mechanical and electrical equipment is not affected by the power uprate conditions for the following reasons:

1. The Seismic loads are unaffected by the power uprate;
2. No new pipe break locations or pipe whip and jet impingement targets are postulated as a result of the uprated condition;
3. Pipe whip and jet impingement loads do not increase for the power uprate; and
4. SRV and LOCA dynamic loads used in the original design basis analyses are bounding for the power uprate.

10.2.3.1.1 Safety-Related SRV and Power-Operated Valves

The licensee performed the over-pressure protection analysis at the uprated power condition using the upper tolerance limits of the valve set points. The analysis calculated a peak RPV steam pressure of 1336 psig at the bottom of the vessel. This peak pressure remains below the ASME allowable of 1375 psig (110% of design pressure) and that safety-related SRV operability is not affected by the proposed power uprate. Furthermore, the <u>maximum operation reactor dome pressure remains unchanged for the QCNPS power uprate.</u> [redacted] The staff concludes that the SRVs and the SRV discharge piping will continue to maintain their structural integrity and to provide sufficient over-pressure protection to accommodate the proposed power uprate.

10.2.3.1.2 Safety-Related SRVs and Power-Operated Valves <u>(Same Title as above but different subject)</u>

As discussed in its original request and response to staff questions, the licensee evaluated the effect of the power uprate on the capability of plant mechanical systems, including safety-related pumps and valves, to perform their safety functions at QCNPS. In addition to the review of safety-related pumps, safety relief valves, and other components for their adequate design for operation at the power uprate conditions, the licensee reviewed in more detail the safety-related air-operated valves (AOVs) in its AOV program, and the safety-related motor-operated valves (MOVs) within the scope of the program established in response to Generic Letter (GL) 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance." The licensee evaluated the safety-related AOVs and MOVs for process and ambient condition changes resulting from the power uprate, including parameters such as fluid flow, temperature, pressure, differential pressure, and ambient temperature. In a **supplemental response (Reference 32???)**, the licensee indicated that potential pressure locking and thermal binding of its safety-related power-operated gate valves had been evaluated in light of the proposed power uprate. The licensee determined that the power uprate conditions did not affect the scope of valves evaluated in response to GL 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves." The licensee also determined that the valves previously evaluated in response to GL 95-07 would not be adversely affected by potential pressure locking or thermal binding as a result of the proposed power uprate. The staff finds the licensee's evaluation of the effect of the proposed power uprate on the capability of safety-related pumps and valves at QCNPS to be acceptable.

The licensee confirmed, in Reference 22, that the setpoint of the relief valves installed on the penetration piping and the spring check valves contained in the relief bypass line are not affected by the proposed power uprate. The licensee also indicated that for other water-filled piping, the resulting stresses calculated at the proposed power uprate conditions were found to be within the allowable limit. Therefore, the licensee concluded that the proposed power uprate has no impact on the evaluation in response to GL 96-06 on potential over-pressurization of isolated piping segments for QCNPS. The staff concurs with the licensee's conclusion.

Based on the information provided by the licensee, the staff concludes that the proposed power uprate will not have an adverse effect on the performance of safety-related valves and mechanical components at QCNPS.

10.3 Required Testing

10.3.1 Generic Test Guidelines for GE BWR EPU

Section 5.11.9 of ELTR1 (Reference 3), provides the general guidelines for power uprate testing.

- A testing plan will be included in the uprate licensing application. It will include pre-operational tests for systems or components which have revised performance requirements. It will also contain a power increase test plan.

- Guidelines to be applied during the approach to and demonstration of uprated operating conditions are provided in Section L.2, "Guidelines for Uprate Testing," of ELTR1. The licensee's safety analysis report (Reference 22 Reference 2 indicated in this draft SER for the Dresden EPU (i.e., NEDC-32962P, Revision 0, December 2000) is not the latest version (i.e., NEDC-32962P, Revision 2, August 2001). Revision 2 was used for this review.)) submitted with the licensee's application, provides additional information relative to power uprate testing.

10.3.2 Startup Test Plan

- The licensee will conduct limited startup testing at the time of implementation of power uprate. The tests will be conducted in accordance with the guidelines of ELTR1 to demonstrate the capability of plant systems to perform their designed functions under uprated conditions.

- The tests will be similar to some of the original startup tests, described in Section 14.2.4.2 [14.2.12.2] Incorrect UFSAR section reference for Quad Cities. Correct section number inserted. Note: Correct UFSAR section number appears later in this text (on the next page). of the Quad Cities UFSAR. Testing will be conducted with established controls and procedures, which have been revised to reflect the uprated conditions.

$ [redacted]

[redacted] The tests will be conducted in accordance with a site-specific test procedure currently being developed by the licensee. The test procedure will be developed in accordance with written procedures as required by 10 CFR 50, Appendix B, Criterion XI.

The following power increase test plan is provided in Section 10.4 "Required Testing," of the licensee's safety analysis report (Reference 2).

a. [redacted]

b. [redacted]

c. [redacted]

d. [redacted]

[redacted]

The licensee's test plan follows the guidelines of ELTR1 and the staff position regarding individual power uprate amendment requests (Reference 4).

10.3.3 Systems/Components with Revised Performance Requirements

The guidelines in Section 5.11.9 of ELTR1 specify that pre-operational tests will be performed for systems or components which have revised performance requirements. These tests will occur during the ascension to extended power uprate conditions. [redacted]

$ [redacted]

$ [redacted]

[redacted]

[redacted]

[redacted]

[redacted]

10.4.1.4 Operator Response

10.4.1.5 Summary of Internal Events Evaluation

10.4.2 External Events

10.4.3 Shutdown Risk

10.4.4 Quality of PRA

10.5 Human Factors

This evaluation is limited to the operator performance aspects resulting from the increased maximum power level. It includes required changes to operator actions, human-system interface, procedures and training resulting from the change in maximum power level. The evaluation is based on the licensee's responses to five broad questions regarding human performance.

The staff's guidance for this review includes Information Notice 97-78, "Crediting of Operator Actions in Place of Automatic Actions and Modifications of Operator Actions, Including Response Times," ANSI/ANS-58.8, "Time Response Design Criteria for Safety-Related Operator Actions," 1984, and NUREG-0800, Standard Review Plan, Chapter 18 (draft), "Human Factors Engineering."

Question 1 - Describe how the proposed power uprate will change plant emergency and abnormal procedures.

In its submittal of February 12, 2001 (Reference 8), the licensee stated that emergency operating procedure changes are limited to revisions to numerical values such as maximum core thermal power and heat capacity temperature limit of the Suppression Pool, and that operator actions remain unchanged. Two abnormal operating procedures (AOP) will change as a result of modifications to equipment. One change will be to the required actions following a feedwater pump trip due to a modification which will install an automatic recirculation system runback. The second AOP change is due to the condensate pump circuitry being revised to trip the fourth running pump during a loss of coolant accident to prevent an electrical overload. EGC stated that these emergency and abnormal procedure changes will be addressed during operator training sessions prior to operation at extended power uprate (EPU) conditions. The staff is satisfied with this response as the changes are minimal and **EGC has committed** to provide the necessary training.

Question 2 - Describe any new risk-important operator actions as a result of the proposed power uprate. Describe changes to any current risk-important operator actions that will occur as a result of the uprate. Explain any changes in plant risk that result from changes in risk-important operator actions. That is, identify those operator actions that will require additional response time or will have reduced time available; identify any operator actions that are being automated as a result of the power uprate; and provide justification for the acceptability of these changes.

The licensee responded that no new risk-important operator actions were identified as a result of EPU for QCNPS.

For QCNPS, eight current operator actions were identified in which time available to complete the action will be reduced as a result of EPU. In the worse-case action, the time available to initiate RPV depressurization following a medium LOCA is reduced from 25 minutes to 20 minutes. EGC has calculated that the increase in HEP will result in an increase in CDF of 1.4%. Operator actions of injecting SBLC and controlling reactor vessel level following an ATWS have reduced action time available from 20 minutes two 16 minutes. This will increase QCNPS HEP and result in an increase in CDP of approximately 1%.

Question 3 - Describe any changes the proposed power uprate will have on operator interfaces for control room controls, displays and alarms. For example, what zone markings (e.g., normal, marginal and out-of-tolerance ranges) on meters will change? What set points will change? How will the operators know of the change? Describe any controls, displays, alarms that will be upgraded from analog to digital instruments as a result of the proposed power uprate and how operators were tested to determine they could use the instruments reliably.

The licensee stated in its submittal of February 12, 2001, that no major physical changes to control room controls, displays or alarms are required as a result of the EPU. Some changes are required to indicator spans, alarm settings, and automatic actuation setpoints to accommodate increased process conditions. Existing zone banding on all control board indications will be reviewed for acceptability and revised as necessary prior to EPU operation.

EGC listed the control board changes and additions to be made, and the setpoints to be changed as a result of the EPU. EGC stated that these changes are being implemented as design changes in accordance with approved change control procedures which includes an impact review by operations and training personnel.

The staff is satisfied that the control room changes are minor and that they will be implemented by approved design change procedures including an impact review by operations and training personnel.

Question 4 - Describe any changes the proposed power uprate will have on the Safety Parameter Display System. How will the operators know of the changes?

The licensee stated that the analog and digital inputs to the Safety Parameter Display System (SPDS) are not affected. One alarm changes to reflect the revised low reactor water level scram function. The setpoint changes are listed in Reference 8. EGC has committed to complete these changes to the SPDS prior to power ascension to EPU conditions and to discuss these changes as part of the operator training program for EPU. **Based on these commitments**, the staff finds that the licensee's consideration of the affect of EPU on SPDS is satisfactory.

Question 5 - Describe any changes the proposed power uprate will have on the operator training program and the plant reference control room simulator, and provide the implementation schedule for making the changes.

In its February 12, 2001, submittal, EGC stated that an operator lesson plan will be developed to teach plant changes as a result of the EPU and existing lesson plans will be revised to reflect the changes. The EPU lesson plan will be presented to all licensed/certified operations personnel before startup is initiated for operating at extended power conditions. EPU changes will be incorporated in continuing training lesson plans as applicable.

Operator training for power uprate conditions will be performed on the simulator prior to operating at EPU conditions. This training will consist of comparisons of plant conditions between the current maximum power level and the uprated power level, the normal operating procedure actions to achieve the uprated level, and selected transients and accidents that present the greatest change from previous power levels.

A simulator software module reflecting the major plant systems and reactor changes as a result of the EPU will be implemented prior to the operator training session before the EPU is initiated. Simulator performance validation will be conducted in accordance with ANSI/ANS 3.5-1985. It will be performed in two stages. First, the simulator performance will be validated against the EPU expected system response. Second, post-startup data will be collected and compared with simulator performance data, allowing any necessary adjustments to be made to the simulator model.

Based on these commitments, the staff is satisfied that the operators will be sufficiently trained and qualified in the EPU conditions.

The staff concludes that the review topics associated with the operator's integration into the proposed extended power uprated system have been satisfactorily addressed by the licensee. The staff further concludes that the proposed extended power uprate should not adversely affect operator performance and minimally increases human error probability based on reduced time available on several risk-important operator actions. The impact of these operator actions on plant risk is discussed in Section 10.4.

10.6 Plant Life

Section 10.7, "Plant Life," of the licensee's safety analysis report (Reference 2) states:

The longevity of most equipment is not affected by EPU. There are various plant *programs (i.e., Equipment Qualification, Flow Accelerated Corrosion) that deal with age-related components. Equipment qualification is addressed in Section 10.3, and flow accelerated corrosion is addressed in Sections 3.5 and 3.11. These programs were reviewed and do not significantly change for the EPU. In addition, the Maintenance Rule provides oversight for the other mechanical and electrical components important to plant safety, to guard against age-related degradation.*

The Equipment Qualification, Flow Accelerated Corrosion and Maintenance Rule (10 CFR 50.65) programs detect and mitigate age-related degradation of components at Quad Cities. The staff has reviewed the applicant's submittal regarding plant life and finds that it is consistent with the guidelines of Section 5.11.6, "Plant Life," of ELTR1, which have been accepted by the NRC as the generic basis for extended power uprate amendment requests.

11.0 CHANGES TO FACILITY OPERATING LICENSE AND TECHNICAL SPECIFICATIONS

12.0 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.

13.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, 51.33, and 51.35, a draft environmental assessment on the proposed amendment was published in the Federal Register for a 30-day period. There were no comments on the proposed action. Accordingly, based upon the environmental assessment and final finding of no significant impact, the Commission has determined that issuance of these amendments will not have a significant effect on the quality of the human environment.

14.0 CONCLUSION

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.

15.0 REFERENCES

1. Letter from Commonwealth Edison Company to Nuclear Regulatory Commission, "Dresden Nuclear Power Station, Units 2 and 3, and Quad Cities Nuclear Power Station, Units 1 and 2, Request for License Amendment: Power Uprate Operation," dated December 27, 2000, with attachments, RS-00-0167.

2. GE Nuclear Energy, "Safety Analysis Report for Quad Cities 1 & 2 Extended Power Uprate," Licensing Topical Report, NEDC-32961P (Proprietary), December 2000.

3. GE Nuclear Energy, "Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate, " (ELTR1), Licensing Topical Report NEDC-32424P-A (Proprietary), dated February 1999; and NEDC-32424 (Non-proprietary), dated April 1995.

4. Nuclear Regulatory Commission Letter to General Electric Company, "Staff Position Concerning General Electric Boiling Water Reactor Extended Power Uprate Program," dated February 8, 1996.

5. GE Nuclear Energy, "Generic Evaluation of General Electric Boiling Water Reactor Extended Power Uprate, " (ELTR2), Licensing Topical Report NEDC-32523P-A (Proprietary), dated February 2000; NEDC-32523P-A Supplement 1, Volume 1 (Proprietary), dated February 1999, and NEDC-32523P-A Supplement 1, Volume II (Proprietary), dated April 1999.

6. Nuclear Regulatory Commission Letter to General Electric Company, "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to General Electric Licensing Topical Report NEDC-32523P," dated September 14, 1998.

7. Nuclear Regulatory Commission, "Standard Review Plan," NUREG-0800, dated April 1996.

8. Letter from R. M. Krich, Exelon Generation Company, LLC, to Nuclear Regulatory Commission, "Supplemental Information for Request for License Amendment for Power Uprate Operation," dated February 12, 2001, RS-01-023.

9. Letter from R. M. Krich, Exelon Generation Company, LLC, to Nuclear Regulatory Commission, "Additional Electrical Information Supporting the License Amendment Request to Permit Uprated Power Operation at Dresden Nuclear Power Station and Quad Cities Nuclear Power Plant," dated April 6, 2001, RS-01-052.

10. Letter from R. M. Krich, Exelon Generation Company, LLC, to Nuclear Regulatory Commission, "Supplement to Request for License Amendment for Power Uprate Operation," dated April 13, 2001, RS-01-083.

11. Letter from R. M. Krich, Exelon Generation Company, LLC, to Nuclear Regulatory Commission, "Additional Environmental Information Supporting the License Amendment Request to Permit Uprated Power Operation at Dresden Nuclear Power Station and Quad Cities Nuclear Power Station," dated May 3, 2001, RS-01-089.

12. Letter from R. M. Krich, Exelon Generation Company, LLC, to Nuclear Regulatory Commission, "Additional Testing Information Supporting the License Amendment Request to Permit Uprated Power Operation at Dresden Nuclear Power Station and Quad Cities Nuclear Power Station," dated May 18, 2001, RS-01-104.

13. Letter from R. M. Krich, Exelon Generation Company, LLC, to Nuclear Regulatory Commission, "Additional Health Physics Information Supporting the License Amendment Request to Permit Uprated Power Operation at Dresden Nuclear Power Station and Quad Cities Nuclear Power Station," dated May 29, 2001, RS-01-108.

14. Letter from R. M. Krich, Exelon Generation Company, LLC, to Nuclear Regulatory Commission, "Additional Fluence Information Supporting the License Amendment Request to Permit Uprated Power," dated June 5, 2001, RS-01-107.

15. Letter from J. A. Benjamin, Exelon Generation Company, LLC, to Nuclear Regulatory Commission, "Additional Materials Information Supporting the License Amendment Request to Permit Uprated Power Operation," dated June 7, 2001, RS-01-113.

16. Letter from R. M. Krich, Exelon Generation Company, LLC, to Nuclear Regulatory Commission, "Additional Instrumentation and Controls Information Supporting the License Amendment Request to Permit Uprated Power Operation," dated June 15, 2001, RS-01-116.

17. Letter from R. M. Krich, Exelon Generation Company, LLC, to Nuclear Regulatory Commission, "Additional Offsite Dose Information Supporting the License Amendment Request to Permit Uprated Power Operation," dated July 6, 2001, RS-01-124.

18. Letter from R. M. Krich, Exelon Generation Company, LLC, to Nuclear Regulatory Commission, "Additional Electrical Information Supporting the License Amendment Request to Permit Uprated Power Operation at Dresden Nuclear Power Station and Quad Cities Nuclear Power Station," dated July 23, 2001, RS-01-143.

19. Letter from K. A. Ainger, Exelon Generation Company, LLC, to Nuclear Regulatory Commission, "Additional Plant Systems Information Supporting the License Amendment Request to Permit Uprated Power Operation at Dresden Nuclear Power Station and Quad Cities Nuclear Power Station," dated August 7, 2001, RS-01-151.

20. Letter from K. A. Ainger, Exelon Generation Company, LLC, to Nuclear Regulatory Commission, "Additional Mechanical Information Supporting the License Amendment Request to Permit Uprated Power Operation at Dresden Nuclear Power Station and Quad Cities Nuclear Power Station," dated August 8, 2001, RS-01-157.

21. Letter from K. A. Ainger, Exelon Generation Company, LLC, to Nuclear Regulatory Commission, Additional Reactor Systems Information Supporting the License Amendment

Request to Permit Uprated Power Operation at Dresden Nuclear Power Station and Quad Cities Nuclear Power Station dated August 9, 2001, RS-01-158.

22. Letter from K. A. Ainger, Exelon Generation Company, LLC, to Nuclear Regulatory Commission, "Additional Mechanical Information Supporting the License Amendment Request to Permit Uprated Power Operation at Dresden Nuclear Power Station and Quad Cities Nuclear Power Station," dated August 13, 2001, RS-01-162.

23. Letter from K. A. Ainger, Exelon Generation Company, LLC, to Nuclear Regulatory Commission, "Additional Plant Systems Information Supporting the License Amendment Request to Permit Uprated Power Operation, Dresden Nuclear Power Station and Quad Cities Nuclear Power Station," dated August 13, 2001, RS-01-161.

24. Letter from K. A. Ainger, Exelon Generation Company, LLC, to Nuclear Regulatory Commission, "Additional Plant Systems Information Supporting the License Amendment Request to Permit Uprated Power Operation at Dresden Nuclear Power Station and Quad Cities Nuclear Power Station," dated August 14, 2001, RS-01-167.

25. Letter from K. A. Ainger, Exelon Generation Company, LLC, to Nuclear Regulatory Commission, "Additional Risk Information Supporting the License Amendment Request to Permit Uprated Power Operation at Dresden Nuclear Power Station and Quad Cities Nuclear Power Station," dated August 14, 2001, RS-01-168.

26. Letter from K. A. Ainger, Exelon Generation Company, LLC, to Nuclear Regulatory Commission, "Supplement to Request for License Amendment for Power Uprate Operation," dated August 29, 2001, RS-01-175.

27. Letter from K. A. Ainger, Exelon Generation Company, LLC, to Nuclear Regulatory Commission, "Additional Radiation Dose Information Supporting the License Amendment Request to Permit Uprated Power Operation," dated August 31, 2001, RS-01-183.

28. Letter from K. A. Ainger, Exelon Generation Company, LLC, to Nuclear Regulatory Commission, "Safety Analysis Reports Supporting the License Amendment Request to Permit Uprated Power Operation," dated August 31, 2001, RS-01-180.

29. GE Nuclear Energy, Safety Analysis Report for Quad Cities 1 & 2 Extended Power Uprate NEDO-32961, Revision 1, dated August 2001.

30. GE Nuclear Energy, Safety Analysis Report for Quad Cities 1 & 2 Extended Power Uprate, NEDC-32961P, Revision 2, (Proprietary), dated August 2001.

31. Letter from K. A. Ainger, Exelon Generation Company, LLC, to Nuclear Regulatory Commission, "Additional Plant Systems Information Supporting the License Amendment Request to Permit Uprated Power Operation," dated September 5, 2001, RS-01-186.

32. Letter from K. A. Ainger, Exelon Generation Company, LLC, to Nuclear Regulatory Commission, "Additional Risk Information Supporting the License Amendment Request to Permit Uprated Power Operation at Dresden Nuclear Power Station," dated September 5, 2001, RS-01-187.

33. Letter from R. M. Krich, Exelon Generation Company, LLC, to Nuclear Regulatory Commission, "Request for License Amendment for Pressure Temperature Limits," dated June 26, 2001.

34. **[Sect. 4.2.5, pg 38, BU96-3 audit report]**

35. General Electric, "General Electric Standard Application for Reactor Fuel," GESTAR II, NEDE-24011-P-A-14, July 2000.

36. General Electric, "GE Fuel Bundle Design," NEDE-31152P, Volumes 1, 2, and 3, dated December 1988.

37. General Electric Nuclear Energy, "BWR Owners Group Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology and Reload Applications," Licensing Topical Report NEDO-32465-A, dated August 1996.

38. BWROG-94078, "BWR Owners Group Guidelines for Stability Interim Corrective Action," June 1994

39. General Electric Nuclear Energy, "General Electric Instrument Setpoint Methodology," NEDC-31336P-A, September 1996.

40. Commonwealth Edison Company, "Analysis of Instrument Channel Setpoint Error and Instrument Loop Accuracy," NES -EIC-20.04, Rev. 3.

41. General Electric Nuclear Energy, "ATWS Rule Issues Relative to BWR Core Thermal-Hydraulic Stability," NEDO-32047-A, dated June 1995.

42. General Electric Nuclear Energy, "BWR Owners Group Long-term Stability Solution Licensing Methodology," NEDO-31960-A and Supplement 1, dated April 1996.

43. General Electric, "Assessment of BWR Mitigation of ATWS, Volume II" (NUREG-0460 Alternative No. 3), NEDE-24222, dated December 1979.

44. Nuclear Regulatory Commission, "Power Oscillation in Boiling Water Reactors (BWRs)," Bulletin Number 88-07, Supplement 1, dated December 1988.

45. Letter from Stuart A. Richards, Nuclear Regulatory Commission, "Review of NEDC-32992P ODYSY Application for Stability Licensing Calculation," Safety Evaluation, dated April 20,

2001.

46. Letter from G.G. Benes (ComEd) to USNRC, "LaSalle County Nuclear Power Station Units 1 and 2, Application for Amendment Request to Facility Operating Licenses NPF-11 and NPF-18, Appendix A, Technical Specifications Partial ARTS Implementation NRC Docket Nos. 50-373 and 50-374," June 9, 1994.

47. Letter from W.D. Reckley (USNRC) to D.L. Farrar (ComEd), "Issuance of Amendments (TAC Nos. M89631 and M89632)," April 13, 1995.

48. Letter from T. W. Simpkin, Exelon Generation Company, LLC, to Nuclear Regulatory Commission, "Additional Risk Information Supporting the License Amendment Request to Permit Uprated Power Operation at Dresden Nuclear Power Station," dated September 14, 2001, RS-01-194.

49. Letter from T. W. Simpkin, Exelon Generation Company, LLC, to Nuclear Regulatory Commission, "Additional Information Supporting the License Amendment Request to Permit Uprated Power Operation at Dresden Nuclear Power Station and Quad Cities Nuclear Power Station," dated September 19, 2001, RS-01-200.

50. Letter from K. A. Ainger, Exelon Generation Company, LLC, to Nuclear Regulatory Commission, "Supplement to Request for License Amendment for Power Uprate Operation," dated September 25, 2001, RS-01-206.

51. Letter from K. A. Ainger, Exelon Generation Company, LLC, to Nuclear Regulatory Commission, "Additional Testing Information Supporting the License Amendment Request to Permit Uprated Power Operation at Dresden Nuclear Power Station and Quad Cities Nuclear Power Station," dated September 27, 2001, RS-01-209.

52. Letter from K. A. Ainger, Exelon Generation Company, LLC, to Nuclear Regulatory Commission, "Additional Information Supporting the License Amendment Request to Permit Uprated Power Operation, Dresden Nuclear Power Station and Quad Cities Nuclear Power Station," dated September 27, 2001, RS-01-210.

53. Letter from K. A. Ainger, Exelon Generation Company, LLC, to Nuclear Regulatory Commission, "Additional Information Supporting the License Amendment Request to Permit Uprated Power Operation," dated ......, 2001, RS-01-.....

54. Letter from K. A. Ainger, Exelon Generation Company, LLC, to Nuclear Regulatory Commission, "Additional Information Supporting the License Amendment Request to Permit Uprated Power Operation," dated ......, 2001, RS-01-......

55. Letter from K. A. Ainger, Exelon Generation Company, LLC, to Nuclear Regulatory Commission, "Additional Information Supporting the License Amendment Request to Permit Uprated Power Operation," dated ....., 2001, RS-01-.....

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ATTACHMENT 1

EPU ON-SITE AUDIT REVIEWS

During the weeks of March 26, and June 16, 2001, members of the NRC Reactor Systems Branch (SRXB) staff visited the Global Nuclear Fuel (GNF) engineering and manufacturing facility at Wilmington, North Carolina. The purpose of these visits was to perform on-site audit reviews of selected safety analyses and system and component performance evaluations used to support extended power uprate (EPU) license submittals. The March audit focused on the Duane Arnold Energy Center (DAEC) EPU, and the June audit was related to the QCNPS EPU submittal. The areas covered by these audits are related to the following sections of the licensee's Safety Analysis Report and are discussed accordingly:

2 Reactor Core and Fuel Performance

2.1 Fuel Design and Operation
2.2 Thermal Limits Assessment
2.3 Reactivity Characteristics
2.4 Stability

9 Reactor Safety Performance Evaluations

9.1 Reactor Transients
9.3 Design Basis Accidents
9.4 Special Events

Observations and Resolution:

Review areas from the DAEC audit that also apply to Dresden and Quad Cities are included here. In each section, the areas reviewed are identified and grouped by a bullet listing. The audit reviews resolved a number of questions as discussed below. Several open items were identified, which were addressed by requests for additional information (RAIs) and resolved later, by licensee responses summarized below.

2 REACTOR CORE and FUEL PERFORMANCE

2.1 Fuel Design and Operation

The SRXB staff audit covered the following areas:

- Follow-up issue addressed in RAI Question 3:

  In 1992, following an NRC Team Audit of the GE-11 (9x9, part-length rods) fuel design compliance with Amendment 22 of NEDE-20411-PA, GE (now GNF) was encouraged to develop a procedure for implementing Amendment 22 criteria for new CPR correlation development as defined in GESTAR II. This procedure is documented in GNF technical design procedure TDP-0117, Rev. 2, page 8. Explain how this procedure was applied in the development of the GEXL14 correlation for use with GE-14 (10x10, part-length rods) fuel at Quad Cities and Dresden, especially with regard to items 1 and 2 of the TDP, given the apparent absence of raw experimental data points for upskew and downskew power profiles. Provide technical justification if the criteria of Amendment 22 process criteria were not met.

The licensee response to RAI Question 3 states:

TDP-0117, Rev. 2, Sections 5.3 and 5.4 describes the test matrix for the ATLAS testing for the development of the GEXL correlation. This process was used, as described in "GEXL14 Correlation for GE14 Fuel," NEDC-32851, Revision 1, September 1999. NEDC-32851, Rev. 1 also provides the process that was used to develop the uncertainties for GEXL14, using the COBRAG code to simulate the upskew and downskew power shape effects.

As discussed in the response to RAI Question 1 below, the GEXL correlation will be re-evaluated based on experimental test data alone. This includes data characterizing the trend with axial power shape. See also the response to RAI Question 2. With this action, the GEXL correlations for GE14 10x10 fuel will be in full compliance with Amendment 22 to GESTAR II, and the application of the approved Amendment 22 process documents the safety of the GE14 fuel design.

- GE-14 fuel design compliance with respect to the GESTAR Amendment 22 process and applicable approved topical reports;

  In addition to the follow up issue discussed above, the staff reviewed the GE-14 fuel design compliance with the Amendment 22 process and with the approved topical reports, NEDC-32601P, NEDC-32694P, and NEDC-32502P, Rev. 1. To facilitate the review, the process was compared with previous compliance reviews of the GE-11 and GE-12 fuel designs. The reviewers questioned several aspects of the documentation, but judged that the intent of the process was largely met. There are no remaining issues, and GNF will document the generic responses for future reference to support the TS Amendment.

- Fuel performance information for 10x10 fuel lattice design (GE-14) fuel used for QCNPS, including available post irradiation examination (PIE) data;

  GNF staff presented a summary of recent fuel performance information for 9x9 and 10x10 fuel designs and discussed the schedule for collecting future PIE data for the 10x10 fuel lead use assemblies and reload batches. The results generally showed increased fuel reliability in the recent designs. The staff is satisfied with the results and planned inspection schedules.

- Analyses of QCNPS first transition GE-14 reload core design, in comparison with equilibrium GE-14 core discussed in the licensee's Safety Analysis Report, with respect to operating T/H limits.

Analyses performed for the first GE-14 transition reload core were reviewed by examination of the design record files for QCNPS Cycle 18, and by discussions with Exelon and GNF engineering personnel involved in the analyses.

2.2 Thermal Limits Assessment (Critical Power Performance)

The SRXB audit covered the following areas:

- Experimental data base for 10x10 fuel lattice designs, used to develop the GEXL14 critical power ratio (CPR) correlation for GE14 fuel, for QCNPS reloads;
- Range of CPR experimental data and correlation fit with respect to QCNPS EPU operating power, flow, and temperature requirements;
- Statistical aspects of experimental data base and correlation, (design of experiment, goodness of fit, uncertainty analysis) to support QCNPS applications

Critical Power Performance

The staff reviewed the experimental data base used for the development of the GEXL14 critical power ratio (CPR) correlation for the GE14 (10x10) fuel lattice design,

As indicated, in the followup issue discussion above, the staff questioned the adequacy of the testing of the new 10x10 GE14 fuel (and GE12 fuel) to determine their respective CPR correlations. No power upskew or downskew experimental data was collected to develop and validate the GEXL10 or the GEXL14 correlations for use in the US fuel/spacer designs. The staff requested (RAI Question 2) the licensee to provide additional data and analyses to substantiate and validate the GEXL10 and GEXL14 correlation uncertainties in the upskew and downskew regions. RAI Question 2 stated:

Describe the testing of the new GE14 fuel that was conducted to test the respective CPR correlations. Identify any additional data, available or planned, to substantiate and validate the correlations. Provide upskew or downskew data that has been collected to validate the GEXL10 or the GEXL14 correlations for use at Quad Cities, Units 1and 2, and Dresden, Units 2 and 3.

In response to RAI Question 2, the licensee stated that:

The GEXL14 correlation for GE14 fuel was based on 638 full-scale ATLAS test points, all of which were cosine axial power shape. Since the original GE14 testing was performed, additional testing has been performed in the ATLAS facility for the GE14 fuel design for both cosine and inlet-peaked power shape. An additional 527 test points were obtained, of which 318 points were for a cosine axial power shape and the remaining 209 points were for an inlet-peaked power shape. Thus, there are 1165 experimental data points available to re-evaluate the GEXL14 critical power correlation, as discussed in. The response to RAI Question 1 discusses the re-evaluation.

RAI Question 1 states:

The COBRAG computer code is the critical power ratio (CPR) methodology used to predict critical power behavior throughout the core. The NRC staff has not reviewed this code. We understand that COBRAG uses first principle models to predict boiling transition and the details of the flow field. Justify the adequacy of the COBRAG code in predicting, from "first principles," boiling transition phenomenon in the upper portion of GE14 fuel and, if applicable to Quad Cities or Dresden, for GE12 (10x10) fuel.

The licensee response to RAI Question 1 states:

For GE14, the GEXL14 correlation was developed from full-scale critical power data for cosine axial power shape and COBRAG-predicted critical power trends versus axial power shape. Comparison of the GEXL correlation to more recently performed full-scale testing for GE14 fuel for cosine and inlet peaked power shapes have shown that the GEXL14 correlation predicts the trend with respect to axial power shape and, therefore, the GEXL14 correlation is considered to be adequate. The correlation uncertainty for the GEXL14 correlation is being re-evaluated based on data alone and the COBRAG-generated data is being removed from the correlation uncertainty calculations. The capability of the GEXL correlations for GE14 fuel to predict the axial power shape effect is being re-evaluated based solely on the full scale ATLAS test data.

As indicated in the licensee response above, GNF has agreed to remove the COBRAG calculated points from the GEXL14 data base. This resolves the question of COBRAG applicability.

- The staff reviewed the range of experimental data versus the operating power, mass flow and temperature conditions required for the QCNPS EPU operation.

  The data range for the cosine axial power shape was judged to cover the EPU operating range requirements for QCNPS.

  The staff reviewed the statistical aspects of the CPR experimental data base, the correlation development and validation, and the uncertainty analyses. The statistical techniques and application to the cosine data for the GEXL correlation determination were judged to be sufficient, with the exceptions noted above.

2.4 Stability

The SRXB audit covered the following areas:

$ operating experience relative to T/H compatibility of different QCNPS fuel types at low-flow/high power conditions with off-normal void distribution,

$ clarification of applicability of Solution III to QCNPS transition mixed cores, and

$ evaluation of stability impact of changes due to QCNPS mixed core with respect to restrictions in operating region and scram due to instability;

The application of the ODYSY code to the Interim Corrective Action (ICA) stability solution was reviewed by discussions with GNF staff. At the time of the audit, the ODYSY stability application licensing topical report (NEDC-32992P) was under review and was subsequently approved as discussed in Section 2.4 of this SER.

In reviewing the applicability of the long-term Solution I-D option for DAEC application, the staff questioned whether the generic DIVOM curve for core wide mode and regional mode stabilities was applicable for EPU operation. The DIVOM [Delta critical power ratio (CPR) over Initial minimum critical power ratio (IMCPR) Versus Oscillation Magnitude] curves are normalized curves of CPR performance versus the hot bundle oscillation magnitude. Two generic curves are used to specify core wide oscillation and regional mode oscillation. The regional mode curve is used to determine the Option III trip setpoints against regional mode instability. The core wide curve is used for Option I-D plants to confirm that the flow-biased APRM trip setpoint provides adequate MCPR safety limit protection against core wide instability. The staff reviewed the QCNPS EPU and transition Cycle 18 analyses to determine the applicability of the generic curves for EPU operation. GE provided the staff with a February 19, 2001, "Interim Corrective Action Request," which indicated that for 20 percent EPU, the generic DIVOM curve may not be bounding for regional mode oscillations. The internal corrective action report stated that the generic DIVOM curves are acceptable for 5 percent power uprate. In June 29, 2001, GE issued a 10 CFR Part 21 report on the potential non-conservatism and provided a figure of merit to be applied to the both core wide and regional DIVOM curves. This resolved the staff's questions regarding the applicability of the generic DIVOM curves for EPU operations at QCNPS as discussed in Section 2.4 of this SER.

9 REACTOR SAFETY PERFORMANCE EVALUATIONS

9.1 Reactor Transients

Dresden and Quad Cities design record files and the TaskT0900, "Transient Analysis," report were reviewed during the audit. No problems were found, and the discussion of limiting transients is included in the appropriate sections of this SER.

9.2 Design Basis Accidents

The SRXB audit covered the following area

- QCNPS loss of coolant accident (LOCA) analysis for pre- and post-uprate conditions;

The staff reviewed the QCNPS LOCA analyses for pre- and post-uprate operating conditions by discussions of design record files with Exelon and GNF engineering personnel involved in the analyses. One item was questioned and resolved by RAI.

RAI Question 4 stated:

The LOCA analysis of off-rated conditions (specifically, single loop operation) assumes that the statistical adders developed for the SAFER code at rated conditions will apply. Justify the use of these adders for single-loop operation (SLO) at Quad Cities and Dresden.

The licensee response to RAI Question 4 states:

The maximum average planar linear heat generation rate (MAPLHGR) multiplier for single loop operation (SLO) is set at a value that keeps the nominal SLO peak cladding temperature (PCT) below the nominal two-loop PCT for the design basis accident (DBA). The upper bound PCT is then calculated for the limiting two-loop DBA case. This process assumes that the two-loop upper bound PCT would bound an explicit SLO upper bound PCT calculation. Inherent in this process is the assumption that the upper bound adder terms used in the two-loop calculations are bounding for SLO conditions.

The SLO PCT is first peak limited; the two-loop PCT is second peak limited. There is less uncertainty in the first peak PCT calculation than the second peak PCT calculation. The first peak PCT is governed primarily by the steady-state stored energy in the fuel rod and the time of boiling transition. The phenomena governing the second peak PCT are more complex include core uncovery, vessel refilling, spray and steam cooling, core reflooding, and quenching, along with any residual effects from the first peak heatup. These uncertainties are reflected in the upper bound adder terms used for the first and second peak upper bound PCT calculations. Since the uncertainty is less for the first peak PCT, the first peak upper bound adders are smaller. Therefore, the assumption that the upper bound adder terms used in the two-loop calculation are bounding for SLO is valid.

9.4 Special Events

- Post-uprate anticipated transient without scram (ATWS) analysis for QCNPS EPU operating region;

  Dresden and Quad Cities design record files and the Project Task Report T0902, "Anticipated Transient Without Scram," were reviewed during the audit. The following section, 9.4.1, addresses a generic item identified during the QCNPS audit, regarding SLC system performance.

9.4.1 Anticipated Transients Without Scram (ATWS)
The staff has requested the following additional information from the licensee:

1. What ATWS events were analyzed at EPU equilibrium versus EPU transition cycle conditions

2. Confirm that for all limiting ATWS events, the SLCS will be able to inject at the

appropriate time without lifting the SLCS bypass relief valves, or if the valves lift they are capable of reseating. For example, will the SLCS be able to inject the required flow rate at the assumed time for the ATWS LOOP event without reaching the rated SLCS relief valve setpoint?

3. What are the limiting events for each of the five acceptance criteria in Section 9.4.1 of the LICENSEE'S SAFETY ANALYSIS REPORT?

4. Confirm that the operator response to an ATWS event is not being modified from that described in Section L.3.2 of ELTR1. If the operator requests SLCS actuation before the time assumed in the analyses, will the relief valve be able to lift and reseat when the SLCS injection is required?

Conclusions

The SRXB staff audit, conducted during the week of June 16, 2001, covered the areas of the licensee's Safety Analysis Report being reviewed by SRXB. As stated, most questions were resolved during the audit, and the rest were covered by RAIs and the licensee responses. With the exception of the GEXL14 correlation re-evaluation and the ATWS questions, all open items were resolved.

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