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September 18, 1987

Mr. Thomas E. Murley, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Quad Cities Station Unit 1
Proposed Technical Specification
Amendment - Unit 1 Cycle 10 Reload
NRC Docket No. 50-254

Dear Mr. Murley:

Pursuant to 10 CFR 50.90, Commonwealth Edison Company (CECo) proposes to amend Facility Operating License DPR-29, Appendix A, Technical Specifications, to support operation of Quad Cities Unit 1 during Cycle 10. The changes involve the addition of Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limits for the reload fuel and adjustment of the Minimum Critical Power Ratio (MCPR) limit to reflect the results of Cycle 10 transient analyses. The transient and accident analyses have been performed to support several Equipment Out-Of-Service and Expanded Operating Domain modes and the respective Technical Specifications changes. In addition, we propose to delete the provisions for Single Loop Operation (SLO) from the license and incorporate SLO in the body of the Technical Specifications.

The following documents are provided as attachments to support our proposed amendment:

Attachment 1: Summary of Unit 1 Cycle 10 Reload and Proposed Technical Specification changes.

Attachment 2: Evaluation of Significant Hazards Consideration

Attachment 3: Proposed License and Technical Specification Changes for Quad Cities Unit 1.

Attachment 4: General Electric Supplemental Reload Licensing Submittal for Unit 1 Cycle 10.

Attachment 5: NEDE-31345P "Quad Cities Station Units 1 and 2 SAFER/GESTR - LOCA Loss-of-Coolant Accident Analysis," dated July, 1987.

Attachment 6: NEDE-31449, "Extended Operating Domain and Equipment Out-of-Service for Quad Cities Nuclear Power Station Units 1 and 2," dated June, 1987.

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Rec'd w/check $150.00

Please note that Attachment 5 is considered proprietary to General Electric and the associated affidavit has been included.

The analyses discussed in Attachments 4, 5 and 6 apply improved GE methods which have been generically reviewed and approved by the Staff and have been previously utilized on other reload applications at non-CECo plants.

The proposed amendment has been on-site and off-site reviewed and approved. We have reviewed this amendment request and determined that No Significant Hazards Consideration (NSHC) exists. CECo has notified the State of Illinois of our request and our determination by NSHC by providing a copy of this letter and the attachment to the designated State Official.

In accordance with 10 CFR 170, a fee remittance in the amount of $150.00 is enclosed.

We request your approval of this amendment by December 1, 1987. If you have any further questions regarding this submittal, plesae contact this office.

Very truly yours,

J A Silady

J. A. Silady
Nuclear Licensing Administrator

lm

Enclosure: Fee Remittance $150.00

Attachments: As Indicated Above

cc: Region III Inspector - Quad Cities
T. Ross - NRR
M. C. Parker - IDNS

SUBSCRIBED AND SWORN to
before me this 18th day
of September, 1987

Lelia F. Mayo
Notary Public

3596K

ATTACHMENT

SUMMARY OF
QUAD CITIES UNIT 1 CYCLE 10 RELOAD
AND PROPOSED TECHNICAL SPECIFICATIONS

A. BACKGROUND

Quad Cities Unit 1 Cycle 10 will use 120 BD300A and 80 BD300B fuel bundles. Both reload fuel types are GE8X8EB fuel, which is discussed in section B. Additional information on this reload may be found in the Supplemental Reload Licensing Submittal (Reference 1).

This reload was performed by GE using their new advanced reload licensing methods. These new methods are known as the GEMINI methods and replace the GENESIS methods. This reload was analyzed with GE's SAFER/GESTER-LOCA methods rather than the SAFE/REFLOOL LOCA methods.

Included as part of this reload are analyses for the following Equipment Out-Of-Service and Extended Operating domain operating modes (EOOS/EOD): increased core flow, feedwater heater(s) out-of-service, feedwater temperature reduction, relief valve out-of-service, and single loop operation. In addition, the Extended Load Line Limit Analysis (ELLLA) region of the power/flow map continues to be supported for QC1C10. Table A-1 summarizes the analyzed combined modes of operation.

The following sections provide a discussion on the key features of this reload, and a summary of the proposed Technical Specification changes for Unit 1 Cycle 10.

B. GE8X8EB FUEL

The reload fuel for Cycle 10 is of the GE8X8EB fuel design including the option for up to four water rods. Previous cycles only used one or two water rods. The GE8X8EB fuel design has been reviewed and approved generically by the NRC (References 2, 3, and 4) and has been incorporated into Revision 8 of GESTAR-II (Reference 5). GE has submitted to the NRC Amendment 18 (October 31, 1986) to GESTAR-II which addresses QC1C10 reload bundle types, i.e., BD300A and BD300B. Bundle specific information for these fuel types is also provided in Reference 9.

This fuel type has several mechanical and nuclear features not previously used at Quad Cities but which have been approved for other plants. Mechanical improvements include increased pre-pressurization for increased exposure capability, an increased pellet diameter (resulting in a smaller pellet-clad gap), variable number of water rods, single diameter upper end plug shafts, and a streamlined upper tie plate to reduce the two-phase pressure drop. Some of the nuclear features include higher bundle enrichments for longer operating cycles and increased discharge burnup, axially zoned gadolinia and the variable number of water rods mentioned earlier. Overall these features allow for improved fuel cycle costs, increased flexibility and improved operating margins including an increase in the LHGR limit from 13.4 kw/ft to 14.4 kw/ft.

B.1 LHGR Limit

The purpose of the thermal limits on the linear heat generation rate (LHGR) are to prevent plastic strain of the cladding from exceeding 1% and to prevent fuel melting. Since the LHGR's are a function of fuel type, exposure and gadolinia, GE has calculated a new LHGR limit of 14.4 kw/ft for the GE8X8EB fuel, which conservatively applies to any exposure or gadolinia content. This LHGR limit was calculated using GESTR-MECHANICAL, the GE fuel rod thermal-mechanical performance model. The NRC has found the use of the GESTR-MECHANICAL code acceptable for determining this limit (References 2 and 3). The NRC has also indicated in Reference 5 that GE has demonstrated, using the GESTR-MECHANICAL code, compliance with the fuel design basis criteria of:

1. No fuel melting during normal steady-state operation and whole core anticipated operational occurrences.

2. A small amount of fuel melting not exceeding 1% cladding strain for local anticipated operational occurrences.

These are satisfied for GE8X8E and GE8X8EB throughout the NRC approved burnup range.

The Technical Specifications have been revised to indicate the new LHGR limit for the GE8X8EB fuel.

B.2 MAPLHGR Curves

For GE fuel, maximum average planar heat generation rate (MAPLHGR) curves have served to provide secondary limits in fuel mechanical design supplementing the traditional LHGR limit. This is in addition to their original purpose in assuring that initial conditions of the ECCS analyses remain valid. With the improved SAFER/GESTR-LOCA analyses, LOCA initial condition MAPLHGRs allowed are 12.8 kw/ft for P8X8R and 13.8 kw/ft for GE8X8EB fuel. These values are independent of nodal exposure. More restrictive MAPLHGR curves, however, are utilized to assure fuel rod mechanical integrity. These curves are provided in Appendix B of Reference 9, and are included in the proposed Technical Specification Figure 3.5-1.

C. SAFER/GESTR-LOCA

GE has re-analyzed Quad Cities units with an improved ECCS analysis code package called SAFER/GESTR-LOCA. GESTR-LOCA is a variation of the GESTR-MECHANICAL fuel mechanical design code and is used to calculate the initial fuel assembly stored energy at the beginning of the LOCA event. SAFER is the combination of previous GE ECCS codes SAFE and REFLOOD along with some modeling improvements. CHASTE is also used for fuel heatup calculations. The GE ECCS methodology is described in detail in Reference 10 Volumes I, II, and III. In the SAFER/GESTR-LOCA methodology, the break spectrum is modelled using "best estimate" input parameters in accordance with the Reference 10 methods. Once the entire break spectrum has been analyzed using nominal parameters, the most limiting break is re-analyzed using input parameters fully consistent with 10 CFR 50 Appendix K criteria. This (when combined with a small uncertainty added) provides the licensing basis peak clad temperature (PCT). Additionally, experimental, code, benchmark, modeling and measurement uncertainties are combined with the nominal PCT calculation to provide an upper bound "95/95" PCT. The licensing basis PCT must be greater than the upper bound PCT. For Quad Cities:

Nominal PCT	828°F
Licensing Basis PCT (Appendix K + Adder)	1382°F
Upper bound (95/95) PCT	1275°F

As can be seen 818°F margin exists to the 10 CFR 50.46 requirement of 2200°F. Additionally 325°F margin exists between the internal NRC requirement that the upper bound PCT be below 1600°F. It should be noted that under SAFER/GESTR-LOCA the limiting break/single failure is now a design basis (double-ended guillatine) break in the recirculation suction line with battery failure. This is different than the previous SAFE/REFLOOD analysis limiting break scenario of a design basis break in the recirculation suction line with failure of the LPCI injection valve.

All of the Quad Cities analyses were performed with one relief valve out of service (RVOOS); thus, no MAPLHGR penalty is required for one (1) RVOOS.

Additionally, single loop operation (SLO) was analyzed with SAFER/GESTR-LOCA. The new nominal PCT for SLO increased by 186°F to 1014°F while the Appendix K calculation increased 111°F to 1488°F. Thus, no MAPLHGR penalty is required during SLO.

D. CORE WIDE TRANSIENTS

D.1 Relief valve Out-Of-Service

All Core Wide Transients and ECCS analyses were performed with the most restrictive relief valve, i.e. the Target Rock S/RV, out-of-service. This reload package therefore includes a Technical Specification change to allow operation with a relief valve out-of-service.

D.2 MCPR Safety Limit

The current MCPR fuel cladding integrity safety limit of 1.07 is maintained for Cycle 10. Two new fuel types are being introduced, BD300A and BD300B, as described in Section B. However, the new fuel types have the same safety limit MCPR (Reference 5 Table 5.2-3a).

D.3 Limiting MCPR Transient

For previously approved operating modes, the Cycle 10 MCPR operating limit required to preclude violation of the fuel cladding integrity limit would be 1.28 for P8X8R, BP8X8R and GE8X8EB fuel. This is the cycle specific value based on the load reject without bypass (LR w/o BP) event and was calculated using GE's advanced reload methods (GEMINI). A cycle independent MCPR LCO of 1.32, however, would be required to allow continued operation with up to a 100°F loss in feedwater heating using the standard Option B approach. For the GEMINI methods, an Option B average 20% scram time (τave) of 0.68 secs would be the threshold for additional MCPR limit adjustment. In order to provide additional scram time margin, however, a final LCO value of 1.33 has been chosen which allows τave to reach 0.71 secs prior to further MCPR adjustment. This MCPR limit bounds all the analyzed combinations of Expanded Operating Domain and Equipment Out-Of-Service shown in Table A-1. The current MCPR LCO of 1.35 has therefore been changed to 1.33 with a scram time adjustment based on τave value of 0.71 secs.

D.4 Compliance to ASME Pressure Vessel Code

The results of the Q1C10 analyses for the postulated MSIV closure with indirect scram and no Relief Valve credit, provided in Reference 1, indicate that the peak steamline pressure will be 1295 psig and the peak vessel pressure will be 1319 psig. These values are within the Technical Specification safety limit of 1345 psig for steam dome pressure and the ASME vessel over-pressurization limit of 1375 psig (110% of design pressure) for anywhere in the primary system.

D.5 ATWS Recirculation Pump Trip (RPT)

This reload analysis again includes the ATWS mitigating Recirculation Pump Trip (RPT) system with a trip setpoint of 1250 psig.

E. LOCAL TRANSIENTS

E.1 Rod Withdrawal Error

For the past several cycles a plant/cycle specific Rod Withdrawal Error (RWE) event has not been analyzed, but rather a statistical generic analysis was referenced. Due to the incorporation of the new GE8X8EB fuel, GE performed a cycle specific analysis. The results of this analysis calculated a ΔCPR of 0.16 for the current rod block setpoint of 107%, which when added to the safety limit provides an event MCPR LCO of 1.23. The Δ for a rod block monitor (RBM) setpoint of 108 was calculated to be 0.19, which provides an event MCPR LCO of 1.28. This is bounded by both the LR w/o BP event and the proposed technical specification operating limit of 1.33. Therefore, it is proposed to revise the RBM setpoint from 107 to 108. The Technical Specification for the RBM upscale trip level setting has been revised from $0.65W_D+42$ to $0.65W_D+43$ so that at 100% drive flow the rod block setting is equal to 108%.

E.2 Fuel Loading Error Event

The worst case bundle misorientation for Q1C10 results in MCPR equal to or less than the 1.07 safety limit when its initial MCPR is less than or equal to 1.20 (1.18 + 0.02, GE calculation plus an NRC imposed variable water gap penalty). This is bounded by the initial CPRs required by the Q1C10 LR w/o BP analysis and is further bounded by the proposed Q1 Technical Specification operating limit of 1.33.

F. STABILITY ANALYSIS

NRC approval of GE's amendment 8 to GESTAR II (Reference 5) stated that a cycle specific stability analysis was not required for BWR 3's since they have been shown to have adequate stability margins. As a result, GE did not provide a stability analysis in the supplemental reload licensing submittal for Q1C10. However, GE was later requested and did provide a stability analysis for Q1C10.

The Q1C10 decay ratio at the intersection of the natural recirculation line and the extrapolated APRM rod block line power level is 0.59. Since proposed Technical Specifications do not allow continued operation on natural circulation, combinations of low flow and high power sufficient to produce high decay ratios are not permitted. GE has also confirmed that the reduced slope of 0.58 for APRM Rod Block flow biasing was used in the Q1C10 stability analysis. This assures the continued acceptability of operating in the expanded power/flow region previously approved by the NRC.

G. ACCIDENTS

G.1 Loss of Coolant Accident

See Section C for the Peak Cladding Temperature results of the new SAFER/GESTR-LOCA analyses. Compliance with other 10 CFR 50.46 ECCS criteria was also demonstrated. Please note that NEDO-24146A "Loss of Coolant Accident Analysis for Dresden Units 2/3 and Quad Cities Units 1/2 Nuclear Power Stations" is therefore replaced by NEDC-31345P "Quad Cities Nuclear Power Station Units 1 and 2 SAFER/GESTR - LOCA Loss of Coolant Accident Analysis" (Reference 9) as the primary reference for Quad Cities new ECCS licensing basis.

G.2 Rod Drop Accident

The Rod Drop Accident (RDA) event has been statistically analyzed on a generic basis and is no longer analyzed on a plant/cycle specific basis. The generic analysis provides assurance that the 280 cal/gram enthalpy deposition limit will not be violated. GE supplemented the generic analyses with PRC 86-07 (Reference 12), which looked at different scenarios. The highest deposition of enthalpy calculated was 171 cal/gram. This provides confidence on the 95/95 level that the Technical Specification limit will not be violated in the unlikely event of the postulated Design Basis RDA. The generic RDA Analysis has been approved by the NRC, Quad Cities On-Site Review 84-3 and Quad Cities Off-Site Review 84-2. This is supplemented with a three one-notch error analysis which CECo has implemented for consistency with the new Rod Worth Minimizer.

H. SINGLE LOOP OPERATION (SLO)

Quad Cities has operated for the last 6 years with SLO capability (cross tie closed). In 1981, NRC approval was granted and an amendment was placed in the operating license to incorporate required SLO restrictions. At this time, it is desired to delete the restrictions from the license and place them in the body of the Technical Specifications. The restrictions to be incorporated are those placed in Unit 2 Technical Specifications (OSR 86-32) without the MAPLHGR reduction factor which is no longer required. To support the new GE8X8EB fuel design, and the increased operating domain, the Quad Cities SLO analysis previously performed (Reference 6) was reviewed to verify that it remains applicable. In Reference 11, GE considered the MCPR safety and operating limits, stability margin and LOCA analyses. They concluded that the prior SLO analyses (Reference 6) remain applicable for the new fuel type and with one relief valve out of service. The following sections summarize previous analyses performed in support of SLO at Quad Cities.

H.1 Plant Transients During SLO

In Section 3.1 of Reference 6 GE has summarized their review of abnormal operating transients during Single Loop Operation (SLO). In response to NRC questions raised during their review of Cooper Station's request for SLO Technical Specifications, GE completed specific analyses of numerous plant transients initiated during SLO. The results demonstrate the applicability of the Reference 6 analyses for the CECo BWR3's.

The four most important aspects of plant transients and SLO are:

(1) They are initiated from less than rated power due to the lower core flow.

(2) The safety limit MCPR and LCO MCPR have been increased (0.01) to account for increased uncertainties in core flow and TIP readings during SLO.

(3) The APRM Scram and Rod Block and RBM flow biased setpoints are adjusted to preserve the relationship normally existent between the setpoints and operating points in the power/flow map during two loop operation.

(4) The idle recirculation (drive) loop is effectively isolated by closing and electrically disarming of either the discharge or suction valve, and by closing of the crosstie (equalizer) line; the internal (driven) jet pump loop is either in forward or reverse flow, depending on the speed of the operating pump.

Because the highest power attainable during SLO is expected to be some 18 to 28% less than rated two loop thermal power, plant transients become less severe. as demonstrated in GE analyses in support of the improved power/flow map, the most limiting power/flow condition is the 100% power/100% core flow point.

The increase in the MCPR safety limit for SLO by 0.01 to account for increased uncertainties in core flow and TIP readings necessitated an increase in the MCPR LCO for SLO. This provides the same protection on SLO as on two loop operation against penetration of the safety limit. Because of the lower initial power on SLO, the resulting CPRs are less for transients; hence, the LCO determined for two loop transients only needs to be increased by the 0.01 uncertainties adder which was applied to the MCPR Safety Limit.

During SLO the reverse flowing idle loop jet pumps non-conservatively alter the normal two loop drive flow to core flow relationship by diverting flow away from the core. Because the flow-biased setpoints are referenced to drive flow, the setpoints must be reduced during SLO to re-establish the two loop relationship between drive flow and core flow. Should this correction not be made and a transient initiate during SLO, the new drive flow to core flow relationship would result in a flow biased trip occurring non-conservatively at a higher neutron flux to core flow ratio than planned.

GE has analytically calculated the magnitude of the setpoint reduction to be 3.5% (based upon a slope of 0.58) for D 2/3 and Q 1/2. This correction has been applied in the proposed Technical Specifications. The 3.5% reduction in APRM setpoints corresponds to a 6.1% reduction in core flow.

Of the flow biased trips, only the RBM is actually credited in safety analyses. Modifying all of the flow-biased trips is conservative, appropriate, and provides the same degree of protection during SLO as two loop operation. The steeper slope of the RBM requires a slightly larger reduction in the RBM set point (6.1%) x (.65) = ~ 4%.

In summary, abnormal plant transients initiated from SLO are conservatively bounded by two loop analyses providing the above mentioned adjustments are made to the MCPR safety limit and LCO, APRM Scram and Rod Block and RBM flow-biased setpoints, and the idle loop is sufficiently well isolated.

H.2 Accidents During SLO

The Q 1/2 FSAR identifies 4 categories of design basis events: Rod Drop Accident (RDA), Main Steamline break (MSLB), Refueling Accident, and Loss-of-Coolant Accidents (LOCA). In addition, GESTAR identifies the Fuel Assembly Loading error and Recirculation Pump Seizure as design basis events. The consequences should one of these accidents occur during single loop versus two loop operation have been addressed by GE in GESTAR and Reference 1. A discussion of the LOCA and Recirculation Pump seizure events follow. Dual Loop Operation events bound Single Loop Operation events in the other four categories.

1. Recirculation Pump Seizure during SLO

   This accident is defined as the "...instantaneous stoppage of one recirculation pump shaft while the reactor is operating at full power." GE has considered this accident to be mild by comparison to a LOCA but analyzed the event for SLO for Browns Ferry Unit 1. The results of this analysis demonstrate that the MCPR Safety Limit will not be penetrated, hence, no fuel failures are expected. The analysis was performed at power/flow ratios up to 82%/56%. Because BF1 is a large core, BWR-4, high power density plant, this analysis is conservatively applicable to SLO on Q 1/2.

2. Loss-of-Coolant Accident during SLO

   For SLO, in addition to the crosstie closed requirement GE requires either the suction or discharge valve in the idle loop recirculation line be closed and disarmed to prevent LPCI flow back through the idle pump and into the annulus through the suction pipe should a LOCA occur and LPCI be injected in the idle loop. This provision has been included in the proposed Technical Specifications.

H.3 Core Stability during SLO

The USNRC has addressed the issue of reactor stability in SLO in Generic Letter 86-09 (Reference 7).

The following is an excerpt from that letter:

"...In low flow operating regions, it has been necessary to develop special operating procedures to assure that General Design Criteria 10 and 12 are satisfied in regard to thermal-hydraulic instabilities. Technical Specifications consistant with these procedures have been accepted by the staff for reactors which are not demonstrably stable based on analyses using approved analytical methods;..."

The USNRC has approved GE's stability methodology (Reference 8). The acceptance criteria is 0.80 versus a Q1C10 calculated decay ratio of 0.59. This decay ratio was calculated at the intersection of the natural circulation line and the extended APRM Rod Block line. This represents a point which is less stable than the regions of allowed operation at Quad Cities Unit 1 under the proposed Technical Specifications, which continue to prohibit continued operation with natural circulation.

Based upon the approved GE methodology and the NRC generic letter, no Technical Specifications incorporating stability monitoring are required for SLO at Quad Cities Unit 1 Cycle 10.

H.4 Core Monitoring Uncertainties during SLO

GE has identified larger measurement uncertainties during SLO in core flow and TIP readings (Reference 6 Section 2). The overall effect of factoring the new values for these uncertainties (6% and 2.85%, respectively) into the process computer uncertainty has resulted in an increase in the latter from 8.7% to 9.1% for SLO on reload cores. This in turn resulted in an increase of 0.01 in the MCPR Safety Limit. In order to maintain the two loop margin between the MCPR LCO and Safety Limit, the LCO should also be increased by 0.01 during SLO. This value is incorporated in the proposed Technical Specifications in lieu of the previous conservative valve of 0.03.

H.5 APRM Noise and Core Plate ΔP Fluctuations during SLO

Single Loop Operation tests performed at Brown's Ferry Nuclear Power Plant on February 9, 1985 have demonstrated that the increased core plate ΔP and APRM noise experienced during SLO is caused by increased flow noise associated with back flow through the idle loop jet pumps and the resultant head differences and flow behavior in the annulus between the idle and active loops, and does not represent a less stable mode of operation.

GE has analyzed the effects of such fluctuations on core materials. Using 2.9 psi peak-to-peak fluctuations in core plate Δp and 20% of scale peak-to-peak noise in APRMs, fuel cladding and channel duty and crack propagation in core materials are not adversely affected (throughout the design lifetime of the channels, cladding, or core materials). Thus, the core plate ΔP and APRM noise surveillance required in the current Quad Cities Unit 1 SLO licensing restriction is no longer needed and has therefore been deleted in the proposed License Amendment. The station will continue to monitor core plate ΔP which provides an early indication of potential Jet Pump performance problems.

I. INCREASED CORE FLOW/FINAL FEEDWATER TEMPERATURE REDUCTION

Operation at greater than rated core flow is supported by the analysis in Reference 11. Reference 11 reviewed the considerations of the reload licensing submittal (Reference 1) and further considerations, such as feedwater nozzle fatigue.

The Increased Core Flow (ICF) analysis was performed at the bounding condition of rated thermal power and 108% rated core flow. Where reduced feedwater temperature might affect the analysis, cases were run with a 100°F reduction in temperature, as well as the normal temperature. This assures that the analysis is valid for cycle extension using Final Feedwater Temperature Reduction (FFWTR). An exposure greater than that expected for the End-of-Cycle is assumed. The analysis concludes that operation within the ICF region will have no impact on safe plant operation, with or without FFWTR.

The operating restriction in the license for coastdown to 40% and off normal FW heating during coastdown is no longer required. The FFWTR analyses analyzed coastdown to 20% with a FW temperature reduction of 100°F, thereby bounding the license restriction and all expected coastdown needs.

As part of the Reference 11 evaluation of ICF, the following areas were determined to be bounded by other modes, within acceptable limits, or unaffected in some cases (such as Fuel Loading Error):

1) Limiting MCPR transients
2) ASME Pressure Vessel Code Overpressurization Compliance
3) Rod Withdrawal Error
4) Fuel Loading Error
5) Stability
6) LOCA
7) RDA
8) Fuel and reactor internals mechanical loadings
9) Reactor internal vibration
10) Feedwater nozzle fatigue
11) Containment LOCA response

J. FEEDWATER HEATER OUT-OF-SERVICE

Reference 11 reviewed operation with a 100°F reduction in feedwater temperature, which could occur due to FW heater operational problems during the cycle prior to the start of coastdown. The expanded operating domain, with the exception of the Increased Core Flow (ICF) region, was examined. (Planned use of FW temperature reduction was evaluated in ICF as a cycle extension strategy. See Section I.) The reference concludes that a cycle independent MCPR limit can be established, that the effect on LOCA response is negligible, and that feedwater nozzle refurbishment requirements increase if feedwater heater out-of-service (FWHOOS) is used for greater than 10% of each cycle.

J.1 Transient Analysis

To establish cycle independent MCPR limits with FWHOOS, a bounding End-of-Cycle (EOC) exposure was assumed. Since transient mitigation depends strongly on control rod action (scram reactivity), a top-peaked axial power shape was selected. This conservatively increases the time until the control rod action becomes effective. Comparison with CPR results using a nominal power shape show that the top-peaked shape yields a 0.04 Δ CPR conservatism.

The load reject without bypass (LR w/o BP) and the feedwater controller failure (FWCF) transients were analyzed with the FWHOOS. These transients were selected because LR w/o BP is the limiting transient event for Q1C10 (Reference 1) and FWCF may become more severe with FWHOOS. Transients were run with 100°F feedwater temperature reduction at 100% core power/100% core flow and 100% core power/87% core flow. The latter point accounts for operation above the rated rod line and represents the lowest flow allowed at rated power in the ELLLA region.

With FWHOOS, LR w/o BP becomes less severe due to the lower steam production rate and the reduced void fraction. As mentioned above, FWCF with FWHOOS becomes more severe than with full heating.

Both events are bounded by the selected cycle independent limit representing LR w/o BP with a conservative top-peaked power shape, bounding exposure, and no FWHOOS, i.e. 1.33.

K. CONCLUSION

With approval of the proposed Technical Specifications, Commonwealth Edison concludes that there are no unresolved safety issues for the Quad Cities Unit 1 Cycle 10 reload in that:

a. The probability of an occurrence nor the consequence of an accident nor the malfunction of safety related equipment, as previously evaluated in the safety analysis report, is not increased.

b. The possibility for an accident or malfunction of a different type than previously evaluated in the safety analysis report is not created.

c. The margin of safety, as defined in the basis for any Technical Specification, is not reduced.

TABLE A-1

ANALYZED COMBINED MODES OF OPERATION

Recirculation System Status	Power/Flow EOD	EOOS
DLO	ELLLA	-
DLO	ELLLA	RVOOS
DLO	ELLLA	FWHOOS
DLO	ICF	-
DLO	ICF	RVOOS
DLO	ICF + FFWTR	-
DLO	ICF + FFWTR	RVOOS
DLO	FFWTR	-
DLO	FFWTR	RVOOS
SLO*	ELLLA	-
SLO*	ELLLA	RVOOS

* Crosstie Closed

REFERENCES

1. GE document 23A5831 "Supplemental Reload Licensing Submittal for Quad Cities Nuclear Power Station Unit 1 Reload 9," dated June 1987 (Attached).

2. NRC letter, MFN-148-85, H. N. Berkow to J. S. Charnley, "Acceptance for Approval of Fuel Designs Described in Licensing Topic Report NEDE-24011-P-A-6, Amendment 10 for Extended Burnup Operation," dated December 3, 1985.

3. NRC letter MFN-082-85, C. O. Thomas to J. S. Charnley, "Acceptance for Referencing of Licensing Topical Report NEDE-24011-P-A-6, Amendment 10, 'General Electric Standard Application for Reactor Fuel,'" dated May 28, 1985.

4. GE letter JSC-058-84, J. S. Charnley to C. O. Thomas, "Submittal of Proposed Amendment 10 to GE LTR NEDE-24011-P-A-6," dated November 30, 1984.

5. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel", Revision 8.

6. GE document NEDO-24807, "Dresden Nuclear Power Station Units 2 and 3 and Quad Cities Nuclear Power Station, Units 1 and 2 Single Loop Operation," dated December 1980.

7. NRC Generic Letter No. 86-09, "Technical Resolution of generic Issue No. B-59 (N-1) Loop Operation in BWRs and PWRs," dated March 31, 1986.

8. NRC letter C. O. Thomas to H. C. Pfefferlen, "Acceptance for Referencing of Licensing Topical Report NEDE-24011, Rev. 6, Amendment 8, 'Thermal Hydraulic Stability Amendment to GESTAR II,'" dated April 24, 1985.

9. GE document NEDC-31345P, "Quad Cities Station Units 1 and 2 SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis," dated July 1987 (Attached).

10. NEDC-23785P, "The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident," Volume I, II, and III.

11. GE document NEDE-31449, "Extended Operating Domain and Equipment Out-of-Service for Quad Cities Nuclear Power Station Units 1 and 2," dated June 1987 (Attached).

12. GE letter G-EBO-7-190, J. A. Miller to H. E. Bliss, "GE PRC 86-07 Limiting Control Rod Sequence for CRDA", May 6, 1987.

ATTACHMENT 2

QUAD CITIES UNIT 1 CYCLE 10 RELOAD

SIGNIFICANT HAZARDS EVALUATION

Commonwealth Edison proposes to amend Facility Operating License DPR-29 for Quad Cities Unit 1 to support the Cycle 10 core reload. The proposed revisions include two basic types of changes: (a) changes specific to the cycle 10 reload fuel and (b) analyses and changes resulting from analyses performed to expand the operating region.

Description of Amendment Request

The changes specific to the Cycle 10 reload fuel and analyses include:

1) Incorporation of the Cycle 10 Minimum Critical Power Ratio (MCPR) limit and new T_{AVE} values and references resulting from the new ODYN methods.

2) Addition of Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limits for the reload fuel.

3) Addition of an LHGR limit specific to the GE8X8EB fuel.

4) Increasing the rod block monitor setpoint.

The Technical Specification changes resulting from analysis performed to expand the operating region and to allow operation with certain equipment out-of-service include:

5) Deletion of existing License Condition addressing Single Loop Operation (SLO) and incorporation of SLO in the body of the Technical Specifications.

6) Changes to the analyzed operating region to include increased core flow (ICF) and feedwater temperature reduction (FTR).

7) Revision of the Automatic Pressure Relief Subsystem Technical Specification to require action only after two or more relief valves are found to be inoperable.

8) Deletion of the license operating restriction for coastdown to 40% power and coastdown with off-normal FW heating.

BASIS FOR PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Commonwealth Edison has evaluated the proposed Technical Specifications and determined that they do not represent a significant hazards consideration. Based on the criteria for defining a significant hazards consideration established in 10 CFR 50.92 (c), operatiON of Quad Cities Unit 1 Cycle 10 in accordance with the proposed changes will not:

a. Involve a significant increase in the probability or consequences of an accident previously evaluated because:

1&2) The incorporation of the MCPR and MAPLHGR limits noted above is explicitly provided to establish limits on normal reactor operation which ensure that the core is operated within the assumptions and initial conditions of the accident analyses. Operation within these limits will assure that the consequences of the affected transient and accidents remain within the results of the analyses. These limits were generated using analytical methods previously approved by the NRC. the probability of an accident is not affected by this change because no physical systems or equipment which could initiate an accident are affected.

3) GE has calculated the LHGR limit for the GE8X8EB fuel using the GESTR-MECHANICAL code, which has been found acceptable by the NRC, and demonstrated that the new LHGR limit (together with the appropriate MAPLHGR limit), assures that the fuel design basis criteria are satisfied for GE8X8EB fuel. The probability of an accident is not affected by this change because no physical systems or equipment which could initiate an accident are affected.

4) GE has performed a cycle specific rod withdrawal error analysis, which demonstrates that the consequences of an accident are not affected since a rod block reading of 108% results in an event MCPR which is bounded by the proposed MCPR LCO and LHGRs within the design basis. The probability of an accident is not affected by this change because no physical systems or equipment which could initiate an accident are affected.

5) The proposed SLO provisions are explicitly based on analyses performed by General Electric using NRC approved methods, to determine required adjustments in operating restrictions for SLO. Operation within the proposed SLO limits has been previously analyzed to assure that the consequences of accidents are not increased. The probability of an accident is not increased because operation in single loop has been previously approved for Quad Cities and has no causal relationship with the equipment or system failures necessary to initiate an accident.

6&8) The proposed changes to increase the allowable operating region, including coastdown to 20% and coastdown with off-normal FW heating, have been analyzed by GE using NRC approved methods to determine the operating restrictions (MCPR, MAPLHGR). GE demonstrated that the MCPR and MAPLHGR limits are bounded by the previous cycle and the proposed values for MCPR and MAPLHGR. The probability of an accident is not increased because operation in the expanded region does not significantly alter the normal operation of the equipment, for which failures have been previously analyzed.

7) GE has performed all cycle transient and LOCA analyses assuming the most limiting relief valve out-of-service.

b. Create the possibility of a new or different kind of accident from any accident previously evaluated because:

1-4) The proposed MCPR, MAPLHGR and LHGR limits and RBM setpoint represent limitations on core power distribution which do not directly affect the operation or function of any system or component. As a result, there is no impact on or addition of any systems or equipment whose failure could initiate an accident.

5) SLO has been previously analyzed and approved for Quad Cities.

6,8) The expanded operating region represents changes to the core power and flow distribution and does not significantly affect the operation or function of any system or component. The major component affected is the recirculation pumps whose failure has been previously analyzed. As a result, there is no significant impact on or addition of any system or equipment whose failure could initiate an accident.

7) GE assumed operation with the most limiting relief valve out-of-service in the transient and LOCA analyses. Therefore, this condition has been analyzed and no new or different accidents are created.

c. Involve a significant reduction in the margin of safety because all of the proposed changes have been analyzed to demonstrate that the consequences of transients or accidents are not increased beyond that previously evaluated and accepted at Quad Cities.

Based on the above discussion, Commonwealth Edison concludes that the proposed amendments do not represent a significant hazards consideration.

ATTACHMENT 3

QUAD CITIES UNIT 1 CYCLE 10

PROPOSED TECHNICAL SPECIFICATIONS

3596K