Commonwealth Edison Company
Quad Cities Generating Station
22710 206th Avenue North
Cordova, Il. 61242-9740
Tel 309-654-2241

ComEd

SVP-98-291

August 31, 1998

U. S. Nuclear Regulatory Commission
Washington, D C 20555

Attention: Document Control Desk

Subject: Quad Cities Nuclear Power Station Units 1 and 2
Application for Amendment to Appendix A, Technical Specifications Section 3/4.7.D, Primary Containment Isolation Valves
NRC Docket Numbers 50-254 and 50-265
Facility Operating License Numbers DPR-29 and DPR-30

Reference: (a) E.S. Kraft (ComEd) letter, SVP-97-105, to USNRC dated May 19, 1997, "Revised Control Room Radiological Assessment"

(b) R. Pulsifer (USNRC) letter to I. Johnson (ComEd) dated March 27, 1997, "Issuance of Amendments 175 and 171 for Quad Cities Nuclear Power Station Units 1 and 2"

Pursuant to 10 CFR 50.90, Commonwealth Edison (ComEd) Company proposes to amend Appendix A, Technical Specifications Section 3/4.7.D, Primary Containment Isolation Valves, of Facility Operating Licenses DPR-29 and DPR-30. The purpose of this amendment request, which is provided in Attachments A through C, is to increase the maximum allowable Main Steam Isolation Valve (MSIV) leakage from 11.5 standard cubic feet per hour (scfh) to 30 scfh per valve when tested at 25 psig, in accordance with Surveillance Requirement 4.7.D.6. Although ComEd is requesting an increase to the MSIV allowable leakage rate, the total primary containment leakage rate as tested in accordance Technical Specification Administrative Requirement 6.8.D.5 will be maintained at the current requirements.

The radiological evaluation supporting the proposed increase in MSIV leakage was performed in accordance with the methodologies contained in Reference (a). The Main Steam Line Break and Loss of Coolant design basis accidents were evaluated to determine the radiological impact on the Control Room operators and offsite dose. The radiological assessment demonstrated release values below the regulatory requirements provided in 10 CFR 50, Appendix A, General Design Criteria 19 and a small fraction of 10 CFR 100 limits.

The proposed changes represent potential outage schedule savings in excess of 5 days; dose savings from 3 Rem to 11 Rem per valve; and financial savings; therefore, ComEd requests NRC approval of this amendment request by November 13, 1998, to support the upcoming Quad Cities Unit 1 refueling outage. ComEd will implement the amendment no later than 30 days following approval.

This proposed Technical Specification amendment has been reviewed and approved by ComEd On-Site and Off-Site Review in accordance with ComEd procedures

If there are any questions or comments concerning this letter, please refer them to Mr. Charles Peterson, Regulatory Assurance Manager, at (309) 654-2241, extension 3609.

Sincerely,

R.W. Pence for JPD

Joel P. Dimmette, Jr.
Site Vice President
Quad Cities Station

OFFICIAL SEAL
ALLEN J CHERNICK
NOTARY PUBLIC, STATE OF ILLINOIS
MY COMMISSION EXPIRES: 10/06/01

Subscribed and Sworn to before me

on this 31 day of August, 1998

Notary Public Allen J Chernick

Attachments: A. Description and Safety Analysis for Proposed Changes
B. Marked-Up Page for Proposed Change
C. Evaluation of Significant Hazards Considerations and Environmental Assessment Applicability Review

cc: J. L. Caldwell, Acting Regional Administrator, Region III
R. M. Pulsifer, Project Manager, NRR
C. G. Miller, Senior Resident Inspector, Quad Cities
W. D. Leech, MidAmerican Energy Company
D. C. Tubbs, MidAmerican Energy Company
INPO Records Center
Office of Nuclear Facility Safety, IDNS

ATTACHMENT A
DESCRIPTION AND SAFETY ANALYSIS
FOR PROPOSED CHANGES
SVP-98-291

INTRODUCTION

The following proposed Technical Specification (TS) amendment for Quad Cities Nuclear Power Station Units 1 and 2 involves an increase in the maximum allowable leakage rate per Main Steam Isolation Valve (MSIV) from 11.5 scfh to 30 scfh. The benefits realized from this proposed change are significant. A greater allowable MSIV leak rate will reduce the amount of unplanned MSIV maintenance improving outage performance and reducing radiation exposure to plant personnel. The expected dose savings is estimated between 3 and 11 Rem per valve.

The proposed increase in the maximum allowable MSIV leakage rate is supported by a revision to the Quad Cities Nuclear Power Station Units 1 and 2 Control Room (CR) radiological assessment. The revised radiological assessment was submitted to the NRC on May 19, 1997 (Reference (a)), and includes significant enhancements such as credit for suppression pool scrubbing, updated iodine dose conversion factors (DCFs), and an assessment of higher burnup fuel designs. The Reference (a) report also clearly defines the design inputs, methodologies and assumptions used in calculating the control room and offsite dose levels following a postulated design basis accident.

The Reference (a) topical report presented a parametric radiological assessment as a function of several key input parameters. The following proposed license amendment establishes the key input parameters and results that clearly define the revised licensing bases radiological impact following a design basis Loss of Coolant Accident (LOCA) and Main Steam Line Break (MSLB) at Quad Cities Nuclear Power Station, Units 1 and 2. With respect to control room dose, the LOCA and MSLB accidents are the limiting radiological events at Quad Cities Nuclear Power Station Units 1 and 2. Both events have been evaluated using the methodology provided in Reference (a) to determine the radiological impact on the CR operators and offsite personnel. By refining the radiological design basis and updating the design inputs, assumptions, and methodologies, sufficient margin to regulatory limits has been demonstrated. The dose assessment includes the 30-day control room operator dose level in accordance with General Design Criteria (GDC) 19, and the 2-hour Exclusion Area Boundary (EAB) and the 30-day Low Population Zone (LPZ) offsite dose levels in accordance with 10 CFR 100.

BACKGROUND

The radiological consequences of design basis accidents were assessed during the original Quad Cities Nuclear Power Station Units 1 and 2 licensing process using many of the conservative methodologies employed by General Electric during that time frame.

As part of the licensing process, the NRC performed a number of conservative confirmatory analyses, which demonstrated the adequacy of the facility design.

The CR dose acceptance limits are established by 10 CFR 50 Appendix A, GDC-19. The requirements of GDC-19 became part of the licensing basis for Quad Cities Nuclear Power Station Units 1 and 2 as a result of the commitments made to NUREG 0737, Item III.D.3.4, "Control-Room Habitability Requirements." This post-TMI action item required a review of the Control Room Ventilation Systems which included an assessment of the system's ability to adequately protect the CR operators against the effects of an accidental release of radioactive gases and verification that the plant can be safely operated and shut down under design basis accident conditions. ComEd also made commitments to upgrade the Control Room Emergency Filtration System (CREFS) at Quad Cities Nuclear Power Station Units 1 and 2 to comply with the 30-day post-accident dose limits. These upgrades included the installation of a new safety related air handling unit (AHU) to serve as the primary train during an accident condition and makeup air charcoal filtration units.

Subsequently, two design inputs to the original NUREG-0737 analyses were changed. The first input was the assumption that only 10 scfm of unfiltered air entered the CR emergency zone. As a result of an airborne event at Zion in 1986, walkdowns of the Quad Cities Nuclear Power Station Units 1 and 2 CREFS were performed. The walkdowns revealed that the CR emergency zone unfiltered inleakage would be in excess of the 10 scfm assumed in the original analyses. This infiltration is in the form of inleakage through the ductwork and components under a negative pressure located outside of the CR emergency zone. The revised inleakage was calculated to be 260 scfm for Quad Cities Nuclear Power Station Units 1 and 2. The second input to the original NUREG-0737 analysis to be corrected was the use of a Standby Gas Treatment System (SBGTS) methyl iodide removal efficiency of 99%. This input was based on the UFSAR analysis description. However, the plant Technical Specification value was specified as 90%. The original NUREG-0737 design was based on allowing the system to supply unfiltered air for the first 8 hours of the accident.

The reanalysis of the post-accident operator dose, with these revised input parameters, resulted in a 30-day thyroid dose in excess of the GDC-19 limits. The operation of the CREFS was revised so that the dose was within GDC-19 acceptance limits. The dose was reduced to within acceptance limits by requiring manual initiation of the CREFS within 110 minutes for Quad Cities Nuclear Power Station Units 1 and 2.

ComEd discovered in late 1996 and early 1997 that the Quad Cities Nuclear Power Station Units 1 and 2 Secondary Containment volume was actually less than that documented in the UFSAR and Technical Specifications. The NRC issued an

As part of the licensing process, the NRC performed a number of conservative confirmatory analyses, which demonstrated the adequacy of the facility design.

The CR dose acceptance limits are established by 10 CFR 50 Appendix A, GDC-19. The requirements of GDC-19 became part of the licensing basis for Quad Cities Nuclear Power Station Units 1 and 2 as a result of the commitments made to NUREG 0737, Item III.D.3.4, "Control-Room Habitability Requirements." This post-TMI action item required a review of the Control Room Ventilation Systems which included an assessment of the system's ability to adequately protect the CR operators against the effects of an accidental release of radioactive gases and verification that the plant can be safely operated and shut down under design basis accident conditions. ComEd also made commitments to upgrade the Control Room Emergency Filtration System (CREFS) at Quad Cities Nuclear Power Station Units 1 and 2 to comply with the 30-day post-accident dose limits. These upgrades included the installation of a new safety related air handling unit (AHU) to serve as the primary train during an accident condition and makeup air charcoal filtration units.

Subsequently, two design inputs to the original NUREG-0737 analyses were changed. The first input was the assumption that only 10 scfm of unfiltered air entered the CR emergency zone. As a result of an airborne event at Zion in 1986, walkdowns of the Quad Cities Nuclear Power Station Units 1 and 2 CREFS were performed. The walkdowns revealed that the CR emergency zone unfiltered inleakage would be in excess of the 10 scfm assumed in the original analyses. This infiltration is in the form of inleakage through the ductwork and components under a negative pressure located outside of the CR emergency zone. The revised inleakage was calculated to be 260 scfm for Quad Cities Nuclear Power Station Units 1 and 2. The second input to the original NUREG-0737 analysis to be corrected was the use of a Standby Gas Treatment System (SBGTS) methyl iodide removal efficiency of 99%. This input was based on the UFSAR analysis description. However, the plant Technical Specification value was specified as 90%. The original NUREG-0737 design was based on allowing the system to supply unfiltered air for the first 8 hours of the accident.

The reanalysis of the post-accident operator dose, with these revised input parameters, resulted in a 30-day thyroid dose in excess of the GDC-19 limits. The operation of the CREFS was revised so that the dose was within GDC-19 acceptance limits. The dose was reduced to within acceptance limits by requiring manual initiation of the CREFS within 110 minutes for Quad Cities Nuclear Power Station Units 1 and 2.

ComEd discovered in late 1996 and early 1997 that the Quad Cities Nuclear Power Station Units 1 and 2 Secondary Containment volume was actually less than that documented in the UFSAR and Technical Specifications. The NRC issued an

amendment to Quad Cities Nuclear Power Station Units 1 and 2 to revise the Technical Specifications to specify a SBGTS filter methyl iodide removal efficiency of 95% (Reference (b)) thereby offsetting the effect of the smaller secondary containment volume. This efficiency is credited in the current and revised analyses for Quad Cities Nuclear Power Station Units 1 and 2 and is confirmed periodically by testing in accordance with Technical Specification Surveillance Requirement 4.7.P.

Recently, a number of BWR licensees have requested and received revisions to their Technical Specifications (TSs) to increase the allowed Main Steam Isolation Valve (MSIV) leakage. Those requests were based, in part, on credit for suppression pool scrubbing in accordance with Standard Review Plan (SRP) 6.5.5. The following amendment request proposes to increase the maximum allowed MSIV leakage, per valve basis, from 11.5 scfh to 30 scfh. The methodology used to calculate the resulting impact on offsite dose and control room dose is detailed in the Reference (a) topical report.

A. SUMMARY OF PROPOSED CHANGES

Pursuant to 10 CFR 50.90, ComEd proposes to amend the TSs for Quad Cities Nuclear Power Station Units 1 and 2. The proposed amendment request increases the maximum allowable MSIV leakage rate specified in TS Surveillance Requirement 4.7.D.6, per valve, from 11.5 scfh to 30 scfh at 25 psig.

To support the increase in allowable MSIV leakage, ComEd has performed a radiological assessment using the methodologies and inputs described in the Reference (a) topical report. The methodology significantly enhances our ability to perform radiological assessments by adopting updated radiological methodologies, revised DCFs, suppression pool scrubbing effects, and an assessment of higher burnup fuel. In particular, the enhancements described in Reference (a) include:

- Added credit for suppression pool scrubbing,
- Credit for International Commission on Radiological Protection (ICRP-30) DCFs,
- Reduction in the amount of mixing in the reactor building from 100% to 50%,
- Assessment of higher burnup fuel, and
- An assessment of secondary containment bypass leakage.

- Updated input parameters (See Table 2 for all input parameters):
  - MSIV leakage from 11.5 scfh to 30 scfh per valve
  - Operator action time to initiate emergency ventilation from 110 minutes to 40 minutes (consistent with Dresden design basis)
  - Control Room unfiltered inleakage from 260 scfh to 400 scfh
  - Containment free air volume form 275,000 ft^3 to 269,000 ft^3 which is the minimum value reported in the UFSAR (Table 6.2-1)

As discussed in Section F below, the increase in allowable MSIV leakage was evaluated using the methodologies described in the Reference (a) topical report. The results establish a new licensing basis for both the control room and offsite dose values. The methodology was used to determine the 2-hour EAB, the 30-day LPZ and the 30-day CR dose. The resulting dose levels are below regulatory requirements.

B. DESCRIPTION OF THE CURRENT REQUIREMENTS

The current TS Surveillance Requirement 4.7.D.6 requires a periodic verification of MSIV leakage to ensure the leakage per MSIV, when tested at 25 psig, is less than or equal to 11.5 scfh. The MSIVs are tested at a pressure lower than P_a (48 psig) per an approved Appendix J exemption dated June 12, 1984 for Quad Cities Nuclear Power Station, Units 1 and 2. The MSIV leakage is included as Type C leakage. The MSIVs are tested at a frequency specified in the Primary Containment Leakage Rate Testing Program, Technical Specification Administrative Requirement 6.8.D.5.

C. BASES FOR THE CURRENT REQUIREMENTS

Technical Specification Surveillance Requirement 4.7.D.6 limits the leakage rate per MSIV to $\leq$ 11.5 scfh. The MSIV valves are periodically leak tested in accordance with the Primary Containment Leakage Rate Testing Program to verify the adequacy of their containment isolation function. The maximum allowed leakage rate, 11.5 scfh, is a component in both the offsite and control room dose radiological evaluations performed in accordance with GDC 19 and 10 CFR 100 respectively. The MSIVs are tested at lower pressures in accordance with an approved exemption, but the leakage rate is included in Type B and C test totals.

In addition, the Primary Containment Leakage Rate Testing Program specifies the maximum allowable primary containment leakage rate be maintained less than 1.0 L_a. Note that L_a is equal to 1% containment air weight per day for Quad Cities Nuclear

Power Station, Units 1 and 2. Although ComEd is requesting an increase to the MSIV allowable leakage rate, the total primary containment leakage rate as tested in accordance Technical Specification Administrative Requirement 6.8.D.5 will be maintained at the current requirements.

A summary of the current and revised assumptions in radiological assessment for Quad Cities Nuclear Power Station is provided in Table 2, "Key Inputs Into Radiological Assessment."

D. NEED FOR REVISION OF THE REQUIREMENT

The proposed TS amendment for Quad Cities Nuclear Power Station, Units 1 and 2 involves an increase in the maximum allowable leakage rate per MSIV from 11.5 scfh to 30 scfh when tested at 25 psig. The benefits realized from this proposed change are significant. A greater allowable MSIV leak rate will reduce the amount of unplanned MSIV maintenance improving outage performance and reducing radiation exposure to plant personnel.

The proposed increase in the maximum allowable MSIV leakage rate is supported by a revision to the Quad Cities Nuclear Power Station, Units 1 and 2, control room radiological assessment. The revised radiological assessment was submitted to the NRC Staff on May 19, 1997 (Reference (a)) and includes significant enhancements such as credit for suppression pool scrubbing, updated iodine DCFs, and allowance for higher burnup fuel designs. The report also clearly defines the design inputs, methodologies and assumptions used in calculating the control room and offsite dose levels following a postulated design basis accident.

E. DESCRIPTION OF THE PROPOSED CHANGES

The proposed amendment request increases the maximum allowable MSIV leakage rate specified in TS Surveillance Requirement 4.7.D.6, per valve, from 11.5 scfh to 30 scfh when tested at 25 psig.

Current TS Surveillance Requirement 4.7.D.6:

> "At the frequency specified by the Primary Containment Leakage Rate Testing Program, verify leakage for any one main steam line isolation valve when tested at P_t (25 psig) is
> ≤ 11.5 scfh."

Proposed TS Surveillance Requirement 4.7.D.6:

> "At the frequency specified by the Primary Containment Leakage Rate Testing Program, verify leakage for any one main steam line isolation valve when tested at P_t (25 psig) is
> ≤ 30 scfh."

F. SAFETY ANALYSIS OF THE PROPOSED CHANGES

ComEd has evaluated the consequences of the proposed TS amendment, which increases the maximum allowable MSIV leakage per MSIV from 11.5 scfh to 30 scfh, using a revised methodology for calculating the dose to the control room operators during a postulated radiological event at Quad Cities Station Units 1 and 2. The original control room habitability studies were provided in the early 1980s in response to NUREG-0737, Item III.D.3.4, requirements. The revised methodology described in Reference (a) significantly enhances the control room dose assessment by adopting updated radiological methodologies, revised DCFs, suppression pool scrubbing effects, and higher burnup fuel designs. Using a consistent approach, the offsite dose consequences have also been evaluated. The revised methodology and proposed increase in allowable MSIV leakage continues to demonstrate adequate margin to regulatory limits. A summary of the key enhancements is provided as follows:

<u>Pressure Suppression Pool Scrubbing</u>

Pressure suppression pool scrubbing with a decontamination factor (DF) of 5 was applied to the amount of particulate and elemental iodine leaking from primary containment to the secondary containment in accordance with SRP 6.5.5.III.1. The SRP notes that for a Mark I containment, a DF of 5 may be applied without the need to perform calculations. A DF of 1.0 is used for noble gases and organic iodides.

Dose Conversion Factors

The existing licensing basis accident analysis is based on the DCFs from Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors," and TID14844, "Calculation of Distance Factors for Power and Test Reactor Sites," which were developed in the early 1960s. RG 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50," Appendix I, recommends DCFs that are significantly lower than those specified in RG 1.3 or TID-14844 (ICRP-30 provides lower DCFs). Although these DCFs have not been included in a regulatory guide for use in accident analyses, they have been submitted to and approved by the NRC in a number of post-TMI control room dose analyses. The ICRP-30 DCFs have been utilized in this methodology.

Reactor Building Mixing

Regulatory Guide 1.3 recommends that the analysis assume that the primary containment leakage pass directly to the Standby Gas Treatment System (SBGTS) without mixing in the secondary containment. The previous CR dose analysis took credit for mixing in 100% of the secondary containment volume. Mixing with 50% of the secondary containment volume is a conservative adjustment for the current analysis and is consistent with SRP 6.5.3.II. The leakage from primary containment can not "short circuit" to the release point. The SBGTS intakes are located on the fourth floor of the reactor building (secondary containment). Primary containment leakage occurs primarily at lower elevations; therefore, the 50% mixing basis is justified based on the separation between SBGTS intakes and primary containment leakage points.

Source Term

The current licensing basis source terms are based on the guidance of TID-14844, which is consistent with the guidance contained in 10 CFR 100. To maintain consistency with the requirements of 10 CFR 100, the radiological consequences of the proposed increase in MSIV leakage were determined using the source term methodology in TID-14844.

An assessment was also performed to evaluate the impact of higher burnup fuel designs, using a source term provided by Siemens Power Corporation (the current fuel supplier for Quad Cities). The Control Room doses were calculated assuming a 20,000 Mwd/MTU and 60,000 Mwd/MTU burnup source term. This higher burnup source

term increased the Control Room doses by 6.4% and 16.7%, respectively. The assessment of higher burnup fuel demonstrates that adequate margin to regulatory limits is maintained. ComEd maintains that the source term provided in TID-14844 is the licensing basis for Quad Cities Nuclear Power Station Units 1 and 2 and is consistent with 10 CFR 100.

Secondary Containment Bypass Leakage

Quad Cities reviewed the potential for additional secondary containment bypass leakage paths. This review entailed identifying all potential paths which originate in the primary containment or are attached to a system which penetrates the primary containment and that ultimately terminates or passes through an area outside of the secondary containment. Each potential leakage path was reviewed in a realistic manner to determine if a potential bypass leakage path existed. The evaluation, which is provided in detail in Reference (a), concluded that the only credible secondary containment bypass leakage path was through the MSIVs. The revised analysis does not include any bypass leakage paths other than MSIV leakage.

MSIV Leakage

The treatment of MSIV leakage is consistent with our current licensing basis, which allows credit for iodine plateout in the main steam lines and the turbine condenser complex following a LOCA. In order to credit iodine plateout, the main steam line piping and main condenser complex are required to retain sufficient structural integrity following a Safe Shutdown Earthquake (SSE), so that they can act as a holdup volume for fission products. At Quad Cities Nuclear Power Station, the steam lines will retain sufficient structural integrity to transport the relatively low flow rate MSIV bypass leakage throughout the steam lines and condenser (UFSAR Section 16.6.5.5.3.1). The dose assessment assumes the condenser is open to the atmosphere via leakage through the low pressure turbine seals. Thus, it is only necessary to ensure that gross structural failure of the condenser will not occur. The resulting plateout mechanisms described in the Reference (a) methodology are similar to those previously approved by the NRC.

Emergency Safety Feature (ESF) Leakage

The ESF CR dose contribution is subject to the guidance of SRP 15.6.5, Appendix B. Quad Cities Station implemented a program to reduce leakage from ESF components which circulate water outside of the primary containment as a result of NUREG-0737, post-TM action Item III.D.1.1, "Integrity Of Systems Outside Containment To Contain Radioactive Material For Pressurized-Water Reactors and Boiling-Water Reactors." This leakage is typically assessed in the industry in a realistic manner and is treated as an administrative limit, which requires the plants to take action should the limit be exceeded. The current program at Quad Cities ensures that leakage is maintained at a very minimal value. In accordance with TS Administrative Requirement 6.8.D.1, "Reactor Coolant Sources Outside Primary Containment," ESF leakages are located and quantified. The total measured leakage rates have been minimal (less than 5 gallons per hour). Historically, the dose analyses conservatively use the typical industry administrative limit of 5 gallons per hour. The ESF leakrate for the Quad Cities dose assessment is taken as two times the typical industry limit for simultaneous leakage from all components in the ESF systems (10 gals/hr).

The ESF CR dose contribution is modeled separately and added to the doses from the SBGTS and MSIV leakage. Fifty percent of core iodine inventory, based on maximum reactor power level, is assumed mixed in the pressure suppression pool water circulating through the containment external piping systems. Iodine is analyzed as being uniformly mixed in the minimum volume of the pressure suppression pool water in accordance with SRP 15.6.5, Appendix B. This volume is conservatively bounded by a value of 110,000 ft^3. The actual ESF water volume is substantially larger because it would include reactor water and water in piping.

The flash fraction is taken to be 10 percent in accordance with SRP 15.6.5 Appendix B since the temperature of the pressure suppression pool water circulating outside of containment does not exceed 212 degrees F (the Pool Condensation Stability Limit is 205 degrees F). Ten percent of the iodine in the leakage is thus assumed to become airborne. The airborne activity released by flashing ESF water is assumed to mix with half the air in the secondary containment before it is released through the SBGTS to the environment via the stack.

Results

The results of the radiological assessment are provided in Table 3, "Radiological Consequences." The results establish a new licensing basis for both the CR and offsite dose values for the limiting radiological events, Main Steam Line Break and LOCA. The methodology was used to determine the 2-hour EAB, the 30-day LPZ and the 30-day Control Room dose. As can be seen in Table 1 below, "Comparison to Acceptance Criteria," the dose levels are below regulatory requirements of GDC 19 and 10 CFR 100.

TABLE 1

Comparison to Acceptance Criteria

	Current Licensing Basis Dose Level (REM)	Proposed Licensing Basis Dose Level (REM)	Regulatory Limit (REM)
30-Day Control Room Dose – Whole Body	MSLB: Not Evaluated LOCA: 0.118	MSLB: 7.56E-3 LOCA: 0.16	5
30-Day Control Room Dose – Thyroid	MSLB: Not Evaluated LOCA: 22.8	MSLB: 23.6 LOCA: 23.0	30
30-Day Control Room Dose - Beta	MSLB : Not Evaluated LOCA: 1.3	MSLB: 5.54E-2 LOCA: 2.52	30
2-Hour EAB Dose – Whole Body	MSLB: ** LOCA: 6	MSLB: 0.228 LOCA: 4.45 *	25
2-Hour EAB Dose – Thyroid	MSLB: ** LOCA: 150	MSLB: 17.1 LOCA: 4.78*	300
30-Day LPZ – Whole Body	MSLB: ** LOCA: 3	MSLB: 1.49E-2 LOCA: 0.96*	25
30-Day LPZ – Thyroid	MSLB: ** LOCA: 108	MSLB: 1.12 LOCA: 4.63*	300

* - Note the EAB and LPZ analysis does not credit mixing in the secondary containment

** - Note: See UFSAR 15.6.4.5 description of AEC initial licensing evaluation.

TABLE 2
Key Inputs Into Radiological Assessment

Input Value	Current Quad Cities Value	Revised Quad Cities Value	Comment
Core Thermal Power Level	2511	2561	Power Level Increased to 102% to be consistent with SRP 15.6.5
Operating History and Source Term	1000 Days using TID 14844	1000 Days using TID 14844	SRP 15.6.5 Recommendation
Core Radiological Fractions Released to Drywell (%)			
Noble Gases:	100	100	Per Reg. Guide 1.3 and SRP 15.6.5
Halogens:	25	25	Per Reg. Guide 1.3 and SRP 15.6.5
Forms of Iodine Species (%)			
Elemental:	91	91	Per Reg. Guide 1.3 and SRP 15.6.5
Organic:	4	4	Per Reg. Guide 1.3 and SRP 15.6.5
Particulate:	5	5	Per Reg. Guide 1.3 and SRP 15.6.5
Fission Product Scrubbing	Not Credited	DF=5	Fission Product Scrubbing in the suppression pool per SRP 6.5.5 (Mark I).
ESF Leakage Outside Primary Containment	Not Evaluated	10 gph (2 X 5 gph)	ESF leakage included per SRP 15.6.5, Appendix B.
Fission Product Released to Suppression Pool (%)	Not Credited	50	SRP 15.6.5
Suppression Pool Volume	Not Credited	110,000 ft^3	Conservative Bounding Value
Primary Containment Leakrate – total, %/day	1.0	1.0	Technical Specification Requirement - 6.8.D.5
Leak Rate Through Each MSIV, scfh:			Per Reg. Guide 1.3 and SRP 15.6.5 * ORNL NSIG-5 Extrapolation Factor = 1.73.
@25 psig	1[illegible]	30.0	
@48 psig*	1[illegible]	51.9	

Total Leak Rate: 4 Main Steam Lines, scfh @25 psig @48 psig*	 46.0 79.6	 120 207.6	*ORNL NSIG-5 Extrapolation Factor = 1.73.
Containment Free Air Volume (ft3)	275,000	269,000*	*UFSAR Table 6.2-1.
Primary Containment Leak Rate which goes to Secondary, %/Day	0.85	0.55	Calculated - ILRT leak rate less MSIV leakage.
Primary Containment Leak Rate which goes through MSIV, %/Day	0.15*	0.45	*Calculated Value.
Standby Gas Treatment System Efficiency, %			
Organic Iodines:	95	95	Technical Specification Requirement - 4.7.P
Elemental Iodines:	95	95	Technical Specification Requirement - 4.7.P
Particulate Iodines:	95	95	Technical Specification Requirement - 4.7.P
Secondary Containment Release Rate (%/day)	100	297	Considers recirc mixing in 50% of volume, 10% TS tolerance and 10% margin.
Leak Rate from Turbine-Condenser Volume, %/Day	1	1	Original NUREG-0737 III.D.3.4 Submittal
Plateout Removal Constant – MSIV Leak Rate Only, Sec^{-1} Elemental Iodine: Particulate Iodine: Organic Iodine:	 1.503 E-3 1.503 E- 0.0	 1.503 E-3 1.503 E-3 0.0	Calculation based on Description in original NUREG-0737 III.D.3.4 Submittal.

Dispersion Data – at intake, sec/m^3 MSIV Leakage 0-2 hr 2-8 hr 8-24 hr 1-4 days 4-30 days SEGTS 0-2 hr 2-8 hr 8-24 hr 1-4 days 4-30 days	 1.29E-3 1.29E-3 7.61E-4 4.84E-4 2.13E-4 7.00E-4 6.45E-6 3.81E-6 2.42E-6 1.07E-6	 1.29E-3 1.29E-3 7.61E-4 4.84E-4 2.13E-4 7.00E-4 6.45E-6 3.81E-6 2.42E-6 1.07E-6	From Original NUREG-0737 III.D.3.4 submittal – utilizing the Halitsky methodology for ground level release and Reg Guide 1.3 for the SBGTS release.
Control Room Emergency Zone Free Volume (ft^3)	184,000	184,000	None.
Volume of Control Room Proper (ft^3)	58,000	58,000	None.
Control Room Ventilation, cfm: Normal Intake Flow: Emergency Intake Flow:	 2,000 2,000	 2,200 (2000+10%) 1,620 (2000 -10% tolerance, - 10% margin)	10% Technical Specification 4.8.D tolerance for dose under emergency flow (during pressurization); no change to normal flow.
Control Room Intake Charcoal Absorption Efficiencies for Iodines (%): Organic: Elemental: Particulate:	 99 99 99	 99 99 99	Assumption consistent with Reg Guide 1.52 and Technical Specification 4.8.D.
Time following DBA at which normal intake is isolated and AFU is started, minutes.	110	40	Reduced to further lower operator dose.

Unfiltered Control Room Inleakage (scfm)	260	400	Inleakage following control room pressurization.
Control Room Cleanup Recirculation Flowrate (scfm)	0	0	No credit taken for recirculation cleanup.
Occupancy Factors			Per SRP 6.4
0 to 1 day:	1.0	1.0	
1 to 4 day:	0.6	0.6	
4 to 30 day:	0.4	0.4	
Effective X/Qs. sec/m^3 Including occupancy factor			Calculated. X/Q times occupancy factor.
Bypass			
0-2 hr	1.29E-3	1.29E-3	
2-8 hr	1.29E-3	1.29E-3	
8-24 hr	7.61E-4	7.61E-4	
1-4 days	2.90E-4	2.90E-4	
4-30 days	8.52E-5	8.52E-5	
SBGTS			
0-2 hr	7.00E-4	7.00E-4	
2-8 hr	6.45E-6	6.45E-6	
8-24 hr	3.81E-6	3.81E-6	
1-4 days	1.45E-6	1.45E-6	
4-30 days	4.28E-7	4.[illegible]-7	
Dose Conversion Factors	TID DCFs	ICRP 30 DCFs	

Data and Assumptions for Main Steam Line Break

Input Value	Current Quad Cities Value	Revised Quad Cities Value	Comments
Quantity of Reactor Coolant Released	100,000 lbs.	100,000 lbs.	Based on current analyses (USFAR 15.6.4).
Quantity of Reactor Coolant and Fission Products that Flash to Steam	55,000 lbs.	55,000 lbs.	Based on current analyses (UFSAR 16.5.4).
Specific Activity	0.2 and 4.0 μCi/gm.	0.2 and 4.0 μCi/gm.	Technical Specification 3.6.J Limit Measure Using ICRP 2.
Analysis Methodology	*	Puff Release – Uniform Cloud	The steam release is analyzed as a puff release that migrates across the CR intake at a rate of 1 meter per second.
Diameter of Cloud	*	141.1 ft	Calculated value.
Duration of Cloud Exposure at CR Intake	*	43 seconds	1 meter per second wind speed.
Rate at which Activity Enters Control Room.	*	2,600 scfm	Activity enters control room via normal outside air intake (plus 10% for margin). No credit is taken for Quad Cities Auto Isolation of the intake. Also conservatively assume intake of activity via infiltration from adjacent areas.

* - Note: See UFSAR 15.6.4.5 description of AEC initial licensing evaluation.

TABLE 3

Radiological Consequences

Loss of Coolant Accident (Rem)[1]

	EAB		LPZ	
Leakage Path	Thyroid (Rem)	Whole Body (Rem)	Thyroid (Rem)	Whole Body (Rem)
SBGT Contribution	4.26	4.37	3.60	0.944
MSIV Contribution	0.329	2.88E-2	0.867	8.13E-3
ESF Contribution	0.193	4.67E-2	0.165	1.02E-2
Total:	4.78	4.45	4.63	0.962
	Thyroid		Whole Body	
Control Room Dose	23.0		0.16	

[1] Note the EAB and LPZ analysis does not credit mixing in the secondary containment

Main Steam Line Break Accident (REM)

	EAB		LPZ	
Leakage Path	Thyroid (Rem)	Whole Body (Rem)	Thyroid (Rem)	Whole Body (Rem)
At 0.2 μCi/gm	0.855	1.14E-2	5.61E-2	7.46E-4
At 4.0 μCi/gm	17.1	0.228	1.12	1.49E-2
	Thyroid		Whole Body	
Control Room at 0.2 μCi/gm	1.18		3.78E-4	
Control Room at 4.0μCi/gm	23.6		7.56E-3	

G. IMPACT ON PREVIOUS SUBMITTALS

ComEd has reviewed previous submittals and has concluded the proposed increase in MSIV leakage has no impact on previous submittals.

H. SCHEDULE REQUIREMENTS

The proposed changes represent potential outage savings; therefore ComEd requests NRC approval of this amendment request by November 13, 1998, to support the upcoming Quad Cities Unit 1 refueling outage. ComEd will implement the amendment no later than 30 days following approval.

I. REFERENCES

(a) E.S. Kraft (ComEd) letter, SVP-97-105, to USNRC dated May 19, 1997, Revised Control Room Radiological Assessment

(b) R. Pulsifer letter to I. Johnson dated March 27, 1997, Issuance of Amendments 175 and 171 for Quad Cities Nuclear Power Station Units 1 and 2.