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U. S. Nuclear Regulatory Commission
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Quad Cities Nuclear Power Station, Units 1 and 2
Facility Operating License Nos. DPR-29 and DPR-30
NRC Docket Nos. 50-254 and 50-265

Subject: **Response to NRC Request for Additional Information for Technical Specification Change Request**

References:

(1) J. P. Dimmette letter to USNRC dated August 31, 1998, "Application for Amendment to Appendix A, Technical Specifications Section 3/4.7.D, Primary Containment Isolation Valves."

(2) GE Topical Report NEDC-31858P, Revision 2, "BWROG Report for Increasing MSIV Leakage Rate Limits and Elimination of Leakage Control Systems," dated September 1993.

(3) E. D. Swartz (ComEd) letter to D. G. Eisenhut (USNRC) dated December 17, 1981, "Supplemental Response to NUREG 0737."

(4) D. B. Vassallo (USNRC) letter to D. L. Farrar (ComEd) dated May 17, 1983, "Resolution of NUREG-0737, Item III.D.3.4, Control Room Habitability."

(5) M. D. Lynch (USNRC) letter to D. L. Farrar (ComEd) dated April 5, 1996, "Issuance of Amendments (TAC NOS. M93597 AND M93598)."

(6) R. M. Pulsifer letter to O. Kingsley dated November 3, 1998, "Request for Additional Information – Quad Cities Nuclear Power Station, Units 1 and 2."

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In Reference (1), ComEd proposed a change to the Technical Specifications (TS) at Quad Cities Nuclear Power Station, Units 1 and 2, that involved a nominal increase in allowable Main Steam Isolation Valve (MSIV) leakage from 11.5 scfh to 30 scfh per valve. In Reference (6), the NRC requested additional information regarding the subject TS change. ComEd's response to this Request for Additional Information (RAI) may be found in the Attachment.

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,

Joel P. Dimmette, Jr.

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

Attachment: Response to Request for Additional Information

cc: Regional Administrator - NRC Region III
NRC Senior Resident Inspector - Quad Cities Nuclear Power Station

Introduction

In Reference (1), ComEd proposed a change to the Technical Specifications that involved a nominal increase in the allowable Main Steam Line Isolation Valve (MSIV) leakage from 11.5 scfh to 30 scfh per valve. The Quad Cities Nuclear Power Station does not employ a leakage control system to treat MSIV leakage following a design basis Loss of Coolant Accident (LOCA). The supporting radiological assessment for the proposed change credited iodine plateout in the main steam piping system and main condenser to further reduce dose to the control room operators and off-site personnel. ComEd considers the use of iodine plateout in this manner to be consistent with the current licensing basis for Quad Cities Nuclear Power Station, Units 1 and 2 (UFSAR Section 15.6.5.5.3.1). Iodine plateout in the main steam piping system and main condenser was used as a dose mitigation mechanism in our control room habitability studies performed in response to NUREG 0737, Item III.D.3.4. Control room dose is the limiting radiological consequence following a LOCA at Quad Cities Nuclear Power Station. The NRC approved ComEd's control room habitability analysis in Reference (4).

In addition, as part of the Integrated Plant Safety Assessment - Systematic Evaluation Program (SEP) performed for Dresden Unit 2, the NRC reviewed the consequences of LOCA events to ensure regulatory compliance with 10 CFR 50.46, 10 CFR 50, Appendix K, and 10 CFR 100 requirements. The NRC's independent assessment included the treatment of MSIV leakage in the main steam piping. The NRC "estimated a thirty hour delay time for the MSIV portion of the leakage, based upon at least an eighty foot length of seismically qualified main steam line downstream of the leaking MSIV." The total offsite radiological consequences were found to be well within 10 CFR 100 limits (see D. Crutchfield (USNRC) letter to L. Del George (ComEd) dated January 5, 1982). Although Quad Cities was not included in the SEP review process, due to the design similarities between Dresden and Quad Cities, ComEd considers the SEP evaluation at Dresden to provide additional justification for crediting the main steam piping system as an iodine removal mechanism. In order to further minimize the radiological impact of the proposed change, ComEd did not propose an increase in allowable primary containment leakage rate which are maintained at their current Technical Specification requirements specified in TS 6.8.D.5, "Primary Containment Leakage Rate Testing Program."

The following provides our response to the Reference (6) Request for Additional Information.

1. *Describe how the Iodine deposition constant assumed in the August 31, 1998, request was derived from the deposition velocity assumed in the referenced (December 17, 1981) analysis. Provide length and diameter of main steam lines (MSL) at Quad Cities.*

In summary, the iodine removal rates were derived using the data from the 1981 Control Room habitability submittal except that it was treated as a single volume as opposed to three separate volumes. It is our understanding from the recent conference call held on November 2, 1998 concerning this subject, that the objective of the question is to understand how a single lambda was derived from the data for three volumes. Additional discussion is provided to help identify the nuances of the modeling.

Background

The analysis prepared in 1981 assumed that the leakage down the steam lines was subjected to plateout and delay within the lines. As discussed in the 1981 Control Room habitability submittal, the iodine removal rates were obtained from NUREG/CR-009 (the removal rates are discussed in question 2). The original analysis considered the transfer of activity through three separate volumes, i.e., the volume between the inboard and outboard MSIVs, the volume between the outboard MSIV and the turbine stop valve, and the volume in the turbine condenser complex. In the 1987 timeframe, the Control Room radiological consequence model for a loss-of-coolant accident was revised to correct several discrepancies specifically, to correct the Standby Gas Treatment System (SBGTS) efficiency in the analysis and accounted for unfiltered inleakage.

Analysis

The leak rate from the containment to the MSIV path in the original analysis (1981 model) did not extrapolate the leak rate from the test pressure of 24 psig to the accident pressure of 48 psig. In both the original and the revised analysis used to support the Reference (1) Technical Specification proposed amendment, the ultimate leakage to the environment is at a rate of 1% per day (note –the input for the code used is fraction based and not flow/volume based).

Another change in the revised analysis was the treatment of the release path as a single volume. With a single volume, the results are much more conservative due to the reduction in hold-up. This is especially true for the doses received during the first 24 hours of an accident. For the three-volume model, the activity in the last volume will have experienced a significant delay as it passes through the first and second volume. The magnitude of the hold-up can be appreciated by considering that in the first volume it takes 3.8 hours for one air change rate and for the second volume it takes 16.5 hours for one air change (using the current leak rate).

The removal rate for the single volume was calculated from the data in the 1981 submittal as follows:

$$\lambda_N = \frac{k_g A}{V} \quad where$$

λ_N = Removal rate due to surface deposition (sec^{-1})
k_g = Average mass transfer coefficient, cm/sec
A = Surface Area
V = Volume of contained gas

The λ_N for each volume were calculated and averaged as follows:

$$\lambda_{N1} = \frac{k_g A}{V} = \frac{(0.012)\ cm/\sec \quad (470)\ ft^2}{(176)\ ft^3 \quad (30.48) cm/ft} = 1.051E-3 \sec^{-1}$$

$$\lambda_{N2} = \frac{k_g A}{V} = \frac{(0.012)\ cm/\sec \quad (1693)\ ft^2}{(761)\ ft^3 \quad (30.48) cm/ft} = 8.76E-4 \sec^{-1}$$

$$\lambda_{N3} = \frac{k_g A}{V} = \frac{(0.012)\ cm/\sec \quad (6.5E5)\ ft^2}{(1.7E5)\ ft^3 \quad (30.48) cm/ft} = 1.50E-3 \sec^{-1}$$

The average λ_N was calculated as follows:

$$\lambda_N = \frac{\sum \lambda_N A_N}{\sum A_N} = \frac{(1.051E-3 \sec^{-1})(470) + (8.76E-4 \sec^{-1})(1693) + (1.50E-3 \sec^{-1})(6.5E5)}{(470 + 1693 + 6.5E5)}$$

$$= 1.503E\text{-}3 \sec^{-1}$$

The Main Steam line piping from the secondary containment penetration to the high pressure turbine including the piping to the Turbine Stop Valves (TSV) and Control Valves (CV) is approximately 200 feet in length (per line) and is 24 inches in diameter.

2. *The iodine deposition velocity used is based on the models in NUREG/CR-0009 for determining deposition in the reactor containment, post-LOCA. However, as described in Section 6.1.9 of NUREG/CR-0009, these models rely on the turbulent mixing of the containment atmosphere. Since the 30 cfh leakage per MSL will most likely result in laminar flow (therefore increasing the importance of Iodine transfer through the bulk gas phase), demonstrate that the assumed value is conservative and bounding.*

The iodine deposition factors utilized in the current design basis analyses (derived in support of the 1981 Control Room Habitability submittal) were based on the work contained in NUREG/CR-0009, "Technological Bases for Models of Spray Washout of Airborne Containment Vessels," (issued in 1978). Section 5.1.2 of this report addresses the deposition of iodine on interior containment surfaces. The NUREG, in general, identifies that the transport of particles across the bulk gas would be dominant if the bulk gas were stagnant. The report also concludes that the bulk gas in the containment would be well mixed. As such, it is agreed that much of the experiments reviewed and correlations that are developed in the NUREG are based on a well-mixed volume. It is also agreed that much of work reviewed in the NUREG is also based on large volumes representative of containments. It should be noted however, that the NUREG does state *"As shown on large scale experiments only very small thermal sources are required to mix a large vessel to the point where boundary layer transport dominates."* While the flow along the length of the pipe is extremely slow, the bulk gas is expected to be well mixed due to the residual heat in the piping (the heating/condensing of the gas/vapor mixture will create ample natural circulation to mix the volume). The bulk gas in the condenser is also expected to be well mixed due to the circulation induced by the condensing environment.

Bechtel Power Corporation determined the deposition velocity of 0.012 cm/sec as a part of the 1981 Control Room habitability analysis. The actual method used to calculate the deposition factor is not known. However, based on a review of the NUREG and the resulting factor, it appears that the mass transfer coefficient was conservatively calculated using the laminar flow equation that is presented on page 16 and 88, Equation 13 or 71, respectively. A review of the NUREG shows mass transfer coefficients an order of magnitude higher than that used in the analysis (i.e., 0.13 as opposed to 0.012). For example, the graph on page 92 of the NUREG, for turbulent conditions, shows that the mass transfer coefficient is on the order of 0.15 with 10 °F thermal gradients and 0.05 with a 1 °F thermal gradient. As can be seen, a mass transfer rate of 0.012 is indicative of stagnant volume with a negligible thermal gradient.

The NUREG also summarizes the results of tests performed on small volumes and unheated volumes (i.e., stagnant volumes). The results of these tests summarized in the NUREG are commensurate with the values used in the 1981 submittal.

In addition, sensitivity studies were performed to assess the impact of the removal rate on the analysis results. The control room dose analysis was performed using the current removal rate of 1.503E-3 sec^{-1} (5.41 hr^{-1}) which is based on a deposition rate of 0.012

cm/sec and a removal rate of 1.75E-4 sec^{-1} (0.63 hr^{-1}) which would be based on a deposition rate on the order of 0.0015 cm/sec. With this significant reduction in the removal rate, the 30-day Control Room dose increased by only 0.2 rem or 0.9%. This value is well below removal rates that would be calculated using the guidance in NUREG/CR-0009.

In summary, the removal rates used in the analysis are justifiable and are substantiated by NUREG/CR-0009. Due to the thermal environment in the steam pipes and the condenser, there will be a sufficient thermal gradient to result in a well mixed volume. Even though the volumes are expected to be well mixed, the deposition factors used are commensurate with stagnant volumes. Furthermore, a sensitivity study performed demonstrated that the CR dose is not highly dependant on this value. Reducing the mass transfer rate by a factor of around 8 (0.012 cm/sec to 0.0015 cm/sec) yielded a minor increase in the radiological consequences (0.9%).

3. *The August 31, 1998, request states that a suppression pool time-integrated DF of five was assumed consistent with the Standard Review Plan (SRP) 6.5.5.III.1.*

   a. *Consistent with SRP 6.5.5.II.1 verify that all releases from the reactor core pass into the suppression pool, except for small bypass.*

   b. *Specify what pool bypass was used in the analysis and compare it to the minimum assumed bypass as discussed in SRP 6.5.5.II.2.*

   c. *Provide calculation for determining the overall decontamination factor (DF adjusted for bypass) consistent with SRP 6.5.III.2. What fraction of the pool bypass also bypasses secondary containment?*

Pressure suppression pool scrubbing with an NRC recommended minimum decontamination factor of 5 was utilized to reduce the amount of particulates and elemental iodine available for leakage in the primary containment in accordance with SRP 6.5.5.III.1. The SRP notes that, for a Mark I containment, the applicant's decontamination factor of 5 or less may be accepted without any need to perform calculations. Under REVIEW PROCEDURES, SRP 6.5.5.III states: "*If the suppression pool is intended as an engineered safety feature for mitigation of radiological doses, then the reviewer estimates its effectiveness in removing fission products from fluids expelled from the drywell or directly from the pressure vessel through the depressurization system.*"

The activity released from the core during the blowdown phase of a LOCA will be mixed in the drywell atmosphere. As a result of the pressure buildup in the drywell, the steam/air mixture in the drywell will be forced through the downcomers into the suppression pool (which is at a lower pressure) where condensibles are removed. In the process of passing through the suppression pool water, iodine fission products are scrubbed. The scrubbing of iodine is limited to particulate and elemental iodine because organic iodine is more subject to dissolution. Based on pool DF data presented in NEDO-25420, suppression pool scrubbing factors of 30 to 1000 are justifiable for elemental and particulate iodine species. With an instantaneous release of fission products postulated at the start of a LOCA, a large break LOCA would result in most of the blowdown and activity passing rapidly through the suppression pool.

A slower, more mechanistic activity release would result in less activity being available instantaneously for release to the reactor enclosure. However, the slow release would be accompanied by steam and hydrogen (e.g., NUREG/CR-2540) which would pressurize the drywell and force flow through the suppression pool where significant quantities of iodine would still be removed. In addition, emergency cooling water circulating from the reactor to the drywell through the suppression pool and back to the core by core spray and Low Pressure Coolant Injection (LPCI) would contribute to scrubbing of iodine being released from the core long after blowdown. As a result, the application of the minimum decontamination factor of 5 to the MSIV leak path which may bypass the suppression pool but is still in contact with suppression pool water, is reasonable. Furthermore, considering the extremely conservative application of the DF of 5 to the primary containment leakage path activity, the net effect of both leak paths on the control room operator dose justifies the use of MSIV leak path scrubbing with a DF of 5.

For the analysis performed, a scrubbing factor or decontamination factor of 5 for elemental and particulate iodine and 1 (i.e., no removal) for organic iodine is conservatively used for both the MSIV and primary to secondary leakage. With a hypothetical blowdown lasting 30 seconds, and a total release of 25 percent of the iodine in the core to the drywell atmosphere, 20 percent of the elemental and particulate iodine and 100 percent of the organic iodine would remain airborne in the primary containment after scrubbing. This accounts for re-evolution of iodine from the emergency cooling/suppression pool water. The dose reduction factor used specifically in this analysis is derived as follows.

SRP 6.5.5 recognizes removal of particulate and elemental iodines in the suppression pool during an accident using the following formula:

$$D = DF/((1 + B(DF - 1))$$

where; D = Decontamination Factor
DF = Maximum Allowable Decontamination Factor (Mark 1 = 5)
B = Bypass Factor of Fraction that Bypasses the Suppression Pool

Per the plant Technical Specifications (Sections 3.7.K.3, 4.7.K.5, and 3.7.K Bases), the bypass area is periodically tested by performing a pressure decay test to ensure that the area is not in excess of a 1 inch orifice. This 1 inch orifice bypass area in comparison to the downcomer area is extremely small. Thus as B approaches zero, the equation reduces to D = DF, or D = 5 for a Mark 1 containment. However, since 4 percent of the iodine is organic (methyl iodine) and is unaffected by scrubbing, the dose reduction factor of 0.232 used in this analysis is calculated as follows.

$$\text{Dose}_{(\text{With SRP 6.5.5 Reduction})} = 0.96D_o/5 + 0.04D_o$$

$$= 0.232D_o$$

where; D_o = Dose Without SRP Reduction

The above equation was applied to the results of leakage that passes through SBGTS. Since the MSIV leakpath considers plateout of particulate and elemental iodines, the application of the above equation would be incorrect (with the plateout in the MSIV path of the elemental and particulate iodine, a composite reduction rate can not be applied to the results). The reduction of the MSIV dose contribution as a result of scrubbing was calculated as follows.

$$\text{Dose}_{(\text{With SRP 6.5.5 Reduction})} = D_{part}/5 + D_{el}/5 + D_{org}$$

where; D_{part} = Particulate Iodine Dose
D_{el} = Elemental Iodine Dose
D_{org} = Organic Iodine Dose

In essence, since the plateout removes most of the elemental and particulates in the MSIV path, the application of the suppression pool scrubbing to this path has a negligible affect.

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 secondary bypass leakage path existed. The evaluation, which is provided in detail in E.S. Kraft (ComEd) letter to USNRC dated May 19, 1997, "Control Room Radiological Assessment," concluded that the only credible secondary containment bypass leakage path was through the MSIVs. The revised radiological assessment does not include any bypass leakage paths other than MSIV leakage.

4. *The August 31, 1998, request includes a revised analysis of the MSL break accident. Since an increase in allowable MSIV leakage (the subject of the request) is not a parameter in the analysis, justify revising it as a basis for this request.*

   The Main Steam Line Break (MSLB) calculation was included since the control room unfiltered inleakage rate is being increased from 260 to 400 cfm. In the Topical report, it was noted that the unfiltered inleakage has no affect on the radiological consequences (the adjacent areas are considered clean during the brief passage of the MSLB cloud). Activity transport is via the intake flow. As a part of this effort, the MSLB analysis was conservatively revised to consider that the infiltration is at the same concentration as the cloud.

5. *Justify changing the MSLB design basis to a "puff" release model that is not consistent with the guidance in Regulatory Guide 1.5, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors." Provide the technical basis for the cloud diameter assumed. Were the cloud diameters and wind speeds assumed for calculating the EAB and LPZ doses the same as that assumed at the control room intake?*

   The radiological consequences of the Main Steam Line Break (MSLB) are not impacted by the Reference (1) proposed TS amendment. The licensing basis Control Room Habitability evaluations at Quad Cities Station do not consider the impact of a MSLB (only the LOCA is assumed). However, in order to confirm the acceptability of increasing control room inleakage from adjacent areas, ComEd performed a conservative MSLB accident evaluation. This analysis included the conservative assumption that the areas adjacent to the control room were assumed to contain activity as a result of the MSLB.

   The release model used to perform this confirmatory study used a release model consistent with the guidance in Regulatory Guide 1.5 and SRP 15.6.4. For the offsite dose analysis, the atmospheric dispersion factors were determined using the methodologies in Regulatory Guide 1.78 and SRP 2.3.4. As a note, this methodology was used to successfully recreate the MSLB analysis for Dresden. For the Control Room analysis, the equations and modeling techniques of Regulatory Guide 1.78 and SRP 2.3.4 were utilized to define the cloud with a Gaussian distribution and standard deviation as defined in the guidance documents. This cloud distribution was input into the Control Room dose model.

6. *In order to credit iodine plateout, the main steam line piping, the bypass/drain lines, the interconnected piping and the condenser will need to retain their structural integrity following a Safe Shutdown Earthquake (SSE). The licensee is requested to demonstrate the structural integrity of the entire leakage treatment path, including the above stated leakage path piping, the associated supports, the condenser structural members, and the associated anchorage, using seismic input data and analytical methodologies acceptable to the NRC.*

ComEd has reviewed the Reference (2) Boiling Water Reactor Owners Group (BWROG) Topical Report which evaluated the capability of main steam piping and main condensers to process MSIV leakage following a design basis LOCA. Based on this comprehensive evaluation (which included Quad Cities as a typical General Electric BWR plant), the BWROG concluded there is reasonable assurance that the main steam piping and main condenser is sufficiently rugged and will remain functionally intact following a design basis LOCA and concurrent seismic event to mitigate the consequences of an accident. Main steam piping and main condensers exhibit substantial seismic ruggedness. Comparisons of piping and condenser designs in GE plants with those in the earthquake experience data base reveals that the GE plant designs are similar to and more rugged than those that have exhibited good earthquake performance. The possibility of a significant failure in GE BWR main steam piping or condenser in the event of a design basis earthquake is highly unlikely and any such failure would also be contrary to a large body of historical earthquake experience data and thus unprecedented.

In addition the BWROG evaluated a total of 131 full load turbine trips. None of the units reported having experienced piping or pipe support problems in the main steam line bypass or smaller interfacing systems attributed to these events. These operational occurrences generate loads that are comparable to seismic loads. The good performance of BWR systems subject to these relatively frequent operational transient events provides additional assurance that the piping and support systems designs have adequate flexibility and clearances to preclude adverse seismic events. The BWROG study also evaluated the availability of the main steam piping and condenser alternate treatment pathway for processing MSIV leakage. It was determined that the probability of a near coincident LOCA and seismic event is much smaller than other plant safety risk.

In addition to the assessment presented in Reference (2), ComEd has conducted a review of the main steam piping systems and condensers at the Quad Cities Nuclear Power Station. The conclusions based on this review are consistent with the conclusions reached in the GE BWROG Topical Report, that the main steam line piping and condensers in BWR reactors are sufficiently rugged to preclude gross failures during a seismic event. A summary of this review is provided below.

Main Steam Piping Analysis

The main steam piping system is described in UFSAR section 10.3 and is comprised of the following subsystems:

(1) Main Steam Piping (High Pressure Turbine Supply, 4-lines)
(2) Main Turbine Bypass Piping (tied to equalization header, 9-lines)
(3) Auxiliary Steam Loads Piping (tied to equalization header, 3-lines)
(4) Main Steam Drain Piping

In the early 1980's Quad Cities performed an analysis of the Main Steam Piping and associated Main Steam Drain Piping in response to NRC IE Bulletin 79-14, "Seismic Analysis For As-Built Safety-Related Piping Systems."

The 24" Main Steam Piping outside containment, from primary containment penetration X-7 to the high-pressure turbine, was analyzed in Calculation Nos. Q1-MS-01C (Unit 1) and Q2-MS-01C (Unit 2). The piping was analyzed seismically throughout. The piping beyond the outboard MSIV was analyzed to determine the effects on the safety-related piping and to ensure that the piping is seismically rugged and will remain functional following a LOCA and concurrent seismic event.

The evaluations performed in response to NRC IE Bulletin 79-14 did not include the Main Turbine Bypass Piping; however, these lines are constructed using the same materials and standards as the Main Steam Piping (ASA B31.1). In addition, the bypass piping is attached to the main steam equalization header, which has been evaluated and found to be acceptable under seismic conditions. The configuration and matierials, as described above, are consistent with the configurations in the earthquake experience database; therefore, it is reasonalbe to assume the Main Turbine Bypass Piping is seismically rugged and will remain functional following a LOCA and concurrent seismic event.

The Auxiliary Steam Loads Piping were also not included in the seismic evaluations performed in response to NRC IE Bulletin 79-14. These lines provide steam to various house loads including the Turbine Gland Seal and Steam Jet Air Ejector systems. The configuration and construction of the Auxiliary Steam Loads Piping is consistent with the earthquake experience database; therefore, it is reasonalbe to assume the Auxiliary Steam Loads Piping is rugged and will remain functional following a LOCA and concurrent seismic event.

The Main Steam Drain Piping outside containment, from the primary containment penetration X-8 to valve MO-1(2)-0220-4, was analyzed seismically in calculation nos. Q1-MS-02B(C) (Unit 1) and Q2-MS-0B(C) (Unit 2) to ensure primary containment would be maintained during a seismic event. The computer models used in the calculations included a few supports past the -4 valve to ensure an adequate seismic model. Walkdowns were performed for Unit 1 and determined that the piping past the -

4 valve is approximately 90 feet long to line no. 1-3017-12" (condenser header). The piping appears to be adequately supported with rigid supports spaced approximately every ten feet. The longest span is about 14½ feet which is below the maximum span (rigid range) of 15½ feet as specified in UFSAR Section 3.9.3.1.3 for seismic design curves. Therefore the piping is rigidly supported and will not rupture due to an earthquake. Line no. 1-3017-12" also appears to be adequately supported based on walkdowns of the piping and is expected to remain functional following a LOCA and concurrent seismic event. Based on a review of Unit 2 drawings, the Unit 2 Main Steam Drain Piping is configured similar to Unit 1. Therefore, the piping on Unit 2 is also expected to remain functional.

Main Condenser

The Quad Cities Nuclear Power Station Main Condenser design is very similar to the design at LaSalle County Station. The LaSalle Main Condenser has been evaluated by the NRC as capable of withstanding a seismic event without gross structural failure (Reference (5)).

The key condenser design parameters for Quad Cities were compared to LaSalle's condenser design parameters. There parameters are listed below for comparison:

Main Turbine Condenser Parameters

Parameter	LaSalle	Quad Cities
Weight (Empty)	2,880,000 lb.	2,589,200 lb.
Weight (Operating)	6,026,000 lb.	4,275,000 lb.
Weight (Test)	12,880,000 lb	11,426,000 lb.
Length	90 ft.	89'-10"
Width	35 ft.	30'
Height (incl. Extension necks)	71 ft.	63'-4"
Number of reinforced concrete piers	8	8
Number of bolts per pier	6	8
Total number of bolts	48	64
Bolt size	1 $^{5}/_{8}$"	2"
Pier spacing (E-W)	24'-4"	25'-11"
Outer pier spacing (N-S)	77'-4"	86'-7"
Seismic Acceleration (Horizontal)	0.32 g	0.30 g
Seismic Acceleration (Vertical)	0.19 g	0.16 g

As can be seen from the above data, the Quad Cities Main Condenser is of comparable size, weighs less, has more and larger anchor bolts and has lower demand seismic accelerations. Additionally, the Quad Cities condenser is specifically addressed and was a part of the BWROG investigation which studied the performance of piping and condensers in response to strong motion earthquakes, the results of which are reported in the Reference (2) Topical Report. The report states "Comparisons of piping and condenser design in example GE Mark I, II and III plants with those in the earthquake experience data base reveal the GE plant designs are similar to those that have exhibited good earthquake experience." It also states "We conclude that the possibility of a failure and significant breach of pressure boundary in GE BWR main steam piping or condensers in the event of a design basis earthquake is highly unlikely and that such failure would also be contrary to a large body of historical earthquake experience data, and thus unprecedented." Therefore, the Quad Cities Main Condenser is considered by ComEd to be seismically acceptable by comparison.

7. *To address the seismic II/I issue, the licensee is requested to address the seismic capability of the turbine building.*

Performance of the Turbine Building during a seismic event is of interest to the issue of MSIV leakage only to the extent that the building structure and internal components should be of sufficient ruggedness and not degrade the capabilities of the selected main steam and condenser pathways. In Reference (2), the BWROG evaluated this type of industrial structure and has confirmed that excellent seismic capability exists. There are no known cases of structural collapse of either turbine buildings at power stations or structures of similar construction.

The Quad Cities Turbine Building was designed in accordance with the required state and local building codes. The Turbine Building is constructed of reinforced concrete to elevation 639' and a steel superstructure to the roof at elevation 700'. The Reactor and Turbine buildings are connected at the Reactor building operating floor elevation 639' and near the steel frame roof at elevation 690'. Both the Reactor and Turbine buildings are founded directly on rock. The Main Turbine is located at elevation 639' and the Main Condenser is located at elevation 569'-8". Ground elevation is located at elevation 595'. The seismic performance of the turbine building is addressed in UFSAR section 3.7.

The seismic capability of the Turbine Building was also assessed as part of the Individual Plant Examination of External Events (IPEEE) program. The Quad Cities IPEEE Submittal Report, Rev. 0, dated February, 13, 1997, Section 3.4.1.4, Turbine Building Complex states that the "entire turbine building complex is a reinforced concrete structure except for the turbine hall superstructure" and that the "turbine building complex, was assigned a seismic capacity of 0.30g pga." Therefore, the Turbine Building is capable of withstanding the safe shutdown earthquake, which has a 0.24g pga (peak ground acceleration).

The earthquake experience database of this type of industrial structure has, in general, been excellent. As indicated in the Reference (2) report, there are no known cases of structural collapse of a turbine building like structure.

8. *The licensee is requested to address the reliability of the entire leakage treatment path, including all of its boundary valves. This may include descriptions and diagrams of the Intended leakage treatment path and boundaries, and assurance that valves required to open the leakage treatment path to the condenser are provided with a highly reliable power source. In addition, confirm that valves which are required to open the leakage treatment path to the condenser will be included as part of the plant inservice testing (IST) program with appropriate testing interval.*

The radiological assessment performed in the Reference (1) submittal takes advantage of the large volumes of the main steam lines and main condenser to provide holdup and plateout of fission products that may leak from closed MSIVs. This method uses the Main Steam Drain Piping to direct leakage to the Main Condenser. In this approach the Main Steam Piping, the Main Steam Drain Piping and the Main Condenser are used to mitigate the consequence of an accident.

Following a LOCA, four potential pathways exist for any leakage past the MSIVs. For more information, please refer to enclosed figure 1, "Main Steam Piping System." ComEd considers the Main Steam Drain Piping (pathway No. 4 below) as the mitigation pathway to the Main Condenser.

1) Main Steam Piping to the High Pressure Turbine, through the High Pressure Turbine seals to the environment, bypassing the Main Condenser.

MSIV leakage through the Main Steam Piping to the High Pressure Turbine is not considered credible. The MSIV leakage would have to pass through the Main Turbine Stop and Control valves before reaching the high-pressure turbine (bypassing the Main Condenser). The Main Turbine Stop and Control valves are designed to close following a design basis LOCA event (normal response following a reactor shutdown). These valves are highly reliable as would be required, if these valves were within the IST program scope, each valve is exercised periodically in accordance with the manufacturer recommendations, approved station procedures and Technical Specifications (Table 3.1.A-1, Reactor Protection System Instrumentation).

2) Main Turbine Bypass to the Main Condenser with delayed release to the environment through the Low Pressure Turbine seals.

MSIV leakage through the Main Turbine Bypass Piping to the Main Condenser is not considered credible. The Main Turbine Bypass Valves are designed to close following a design bases LOCA event (normal response following a reactor shutdown and MSIV closure). These valves are highly reliable as would be required, if these valves were within the IST program scope, each of the Main Turbine Bypass Valves is exercised periodically in accordance with the manufacturer recommendations and approved station procedures. This testing provides reasonable assurance that the Main Turbine Bypass Valves are functional and will close following a design basis accident.

3) Auxiliary Steam Loads (delayed or treated release to the environment)

There are three leakage pathways associated with the Auxiliary Steam Loads.

Offgas Steam Supply system
The Offgas system is designed to evacuate noncondensible gasses from the Main Condenser during normal operation. Although this system would not auto-isolate following a design basis LOCA, the pathway is not considered credible for the following reasons. Any MSIV leakage that reaches the Offgas system could migrate back to the Main Condenser (and be delayed as described in path No. 4 described below). This pathway is not considered a credible treatment pathway because the pathway is much longer and more tortuous than the Main Steam Drain Piping pathway to the Main Condenser, which is discussed in item No. 4 below.

Alternatively, the leakage reaching the Offgas system could continue downstream of the Offgas system. The Offgas system exhaust is a "treated" pathway in that the gasses exhausting from the system enter a 36-inch "hold-up" pipe (normal 4-hour hold up time at full power) and then pass through a set of charcoal adsorbers before being discharged through the 310 ft main chimney. For these reasons, leakage through the Offgas system piping will have no impact on the radiological consequences of the design basis LOCA.

Turbine Gland Seal steam supply.
The second potential leakage pathway associated with the Auxiliary Steam Loads is the steam supply to the turbine gland seal system. This path is not considered credible because the supply steam to the gland seal system is normally closed during full power operation.

Main Condenser Low Load Heat Reheat Coil Supply.
The final potential pathway associated with the Auxiliary Steam Loads is the Main Condenser Low Load Heat Reheat Coil Supply. This path is not considered credible because this system has been isolated and is no longer used at Quad Cities.

4) Main Steam Drain Piping to the Main Condenser with delayed release to the environment through the Low Pressure Turbine seals.

The Main Steam Drain Piping to the Main Condenser pathway is the existing pathway that may be credited as an iodine treatment mechanism to reduce the radiological consequences of a LOCA. This existing flow path from the Main Steam Drain Piping to the Main Condenser requires repositioning of Main Steam Drain Valve 220-03 which is powered from the Class 1E Essential Service Motor Control Center 18-1 and utilizes the existing orificed line, bypassing the closed 220-04 valve to provide a pathway to the condenser.

The 220-03 valve is not currently included in the Quad Cities ISI or IST programs. ComEd commits to develop an appropriate test program to ensure reliability of this valve.

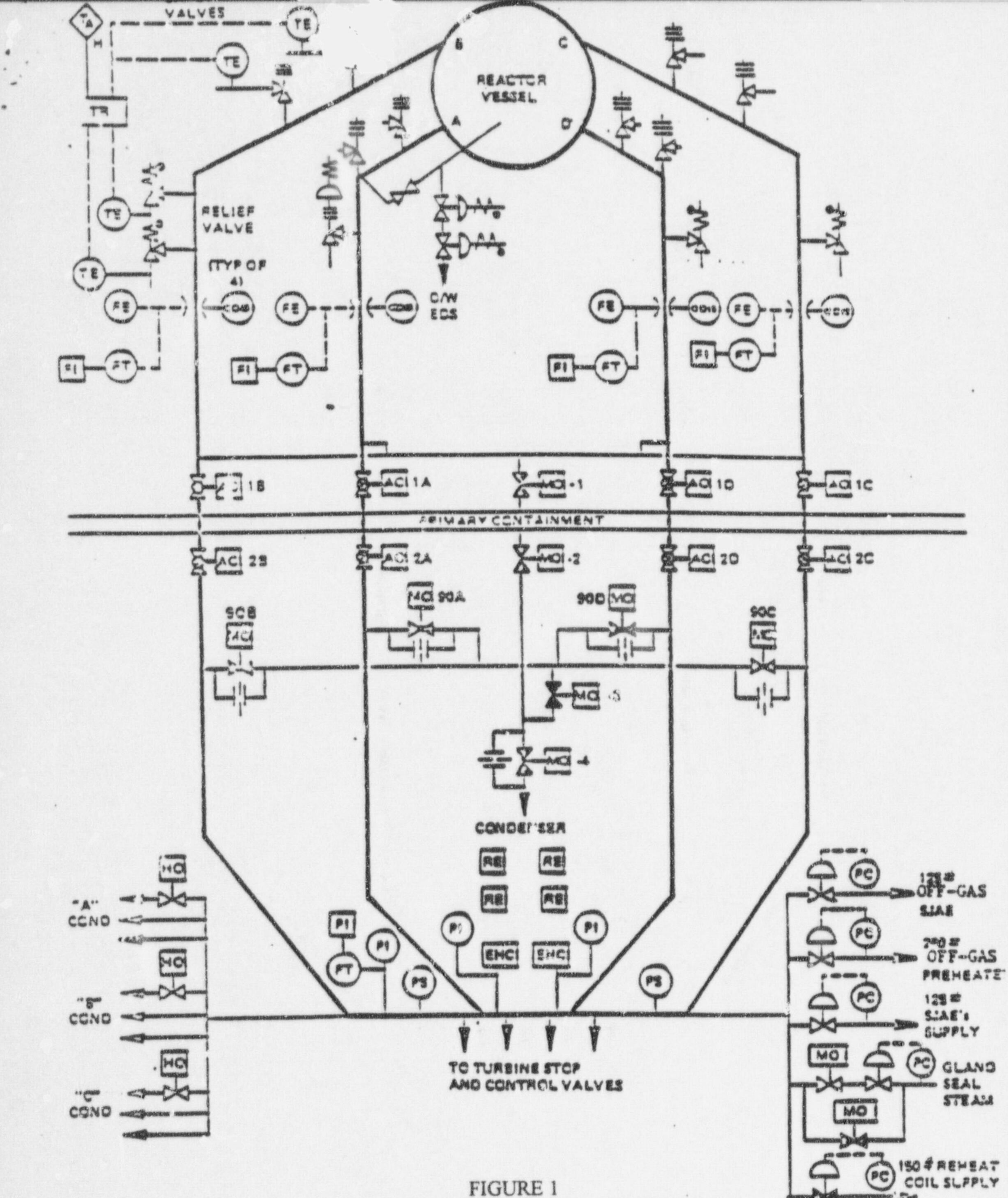

FIGURE 1
"MAIN STEAM PIPING SYSTEM"
SVP-98-345
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