September 3, 1998

Mr. Oliver D. Kingsley, President
Nuclear Generation Group
Commonwealth Edison Company
Executive Towers West III
1400 Opus Place, Suite 500
Downers Grove, IL 60515

SUBJECT: ISSUANCE OF AMENDMENTS - BRAIDWOOD STATION, UNITS 1 AND 2 (TAC NOS. MA0696 AND MA0697)

Dear Mr. Kingsley:

The U.S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 95 to Facility Operating License No. NPF-72 and Amendment No. 95 to Facility Operating License No. NPF-77 for the Braidwood Station, Unit Nos. 1 and 2, respectively. The amendments are in response to your application dated January 14, 1998, as supplemented by letter dated July 17, 1998.

The amendments change the Braidwood, Unit 1, Technical Specification limits on Reactor Coolant System (RCS) Dose Equivalent Iodine-131 (DEI) from 0.35 microcuries/gram to 0.05 microcuries/gram for the remainder of Cycle 7. While there are no changes to the Technical Specifications for Braidwood, Unit 2, both units are being amended to maintain consistency.

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

ORIGINAL SIGNED BY:

Stewart N. Bailey, Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Docket Nos. STN 50-456 and STN 50-457

Enclosures:
1. Amendment No. 95 to NPF-72
2. Amendment No. 95 to NPF-77
3. Safety Evaluation

cc w/encl: See next page

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NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

September 3, 1998

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A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Stewart N. Bailey, Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
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Docket Nos. STN 50-456 and STN 50-457

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cc w/encl: See next page

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-456

BRAIDWOOD STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 95
License No. NPF-72

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Commonwealth Edison Company (the licensee) dated January 14, 1998, as supplemented by letter dated July 17, 1998, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-72 is hereby amended to read as follows:

7. Steam Released to the Environment

Faulted SG (0 - 2 hours)	96,000 lbs
Faulted SG (2 - 40 hours)	33,945 lbs
Intact SGs (0 - 2 hours)	406,716 lbs
Intact SGs (2 - 8 hours)	939,604 lbs

8. Letdown Flow Rate (gpm) 75

9. Release Rate for 0.05 μCi/gm of Dose Equivalent ^{131}I

Release Rate (Ci/hr)	500X Release Rate (Ci/hr)
^{131}I = 0.422	211
^{132}I = 3.09	1550
^{133}I = 1.06	528
^{134}I = 1.58	790
^{135}I = 1.07	535

10. Atmospheric Dispersion Factors

	sec/m³
EAB (0-2 hours)	*7.7 $\times 10^{-4}$
LPZ (0-8 hours)	*7.9 $\times 10^{-5}$
(8-24 hours)	*5.2 $\times 10^{-5}$
(24-96 hours)	*2.1 $\times 10^{-5}$
Control Room (0-8 hours)	6.24 $\times 10^{-3}$
(8-24 hours)	3.16 $\times 10^{-3}$
(24-96 hours)	8.42 $\times 10^{-4}$

11. Control Room Parameters

Filter Efficiency (%)	
Air intake filter	99
Air recirculation filter	90
Volume (ft^3)	70,275
Makeup flow (cfm)	5400
Recirculation Flow (cfm)	44,550
Unfiltered Inleakage (cfm)	15
Occupancy Factors	
0-1 day	1.0
1-4 days	0.6
4-30 days	0.4

*NRC staff calculated values

TABLE 1

INPUT PARAMETERS FOR BRAIDWOOD, UNIT 1, EVALUATION OF MAIN STEAMLINE BREAK ACCIDENT

1. Primary Coolant Concentration of 3 μCi/gm of Dose Equivalent ^{131}I

<u>Pre-existing Spike Value (μCi/gm)</u>

^{131}I = 1.98
^{132}I = 2.22
^{133}I = 3.17
^{134}I = 0.48
^{135}I = 1.75

2. Volume of Primary Coolant and Secondary Coolant

Primary Coolant Volume (ft^3)	12,062
Primary Coolant Temperature (°F)	586.2
Secondary Coolant Steam Volume (ft^3)	3,889
Secondary Coolant Liquid Volume (ft^3)	2,070
Secondary Coolant Steam Temperature (°F)	544
Secondary Coolant Feedwater Temperature (°F)	440

3. TS Limits for DE ^{131}I in the Primary and Secondary Coolant

Maximum Instantaneous DE ^{131}I Concentration (μCi/gm)	3.0
Primary Coolant DE ^{131}I Concentration (μCi/gm)	0.05
Secondary Coolant DE ^{131}I Concentration (μCi/gm)	0.1

4. TS Value for the Primary to Secondary Leak Rate

Primary to secondary leak rate, maximum any SG (gpd)	150
Primary to secondary leak rate, total all 4 SGs (gpd)	600

5. Maximum Primary to Secondary Leak Rate to the Faulted and Intact SGs

Faulted SG (gpm @ room T/P)	132.8
Intact SGs (gpm/SG)	0.1

6. Iodine Partition Factor

Faulted SG	1.0
Intact SG	0.1

The staff is aware that when primary coolant activity levels of dose equivalent ^{131}I approach values of 0.1 µCi/gm or lower, data show that the probability of iodine spiking factors greater than 500 increases. The staff believes, however, that the probability of accident doses exceeding 10 CFR Part 100 limits as a result of limiting primary coolant activity to 0.05 µCi/gm is highly unlikely.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Illinois State official was notified of the proposed issuance of the amendments. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts and no significant change in the types, of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration and there has been no public comment on such finding (63 FR 11914). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Attachments:
1. Table 1 - Input Parameters for Braidwood, Unit 1, Evaluation of Main Steamline Break Accident
2. Table 2 - Thyroid Doses from Braidwood, Unit 1, Main Steamline Break Accident (Rem) (Values Calculated by NRC Staff)

Principal Contributor: C. Hinsen
J. Hayes

Date: September 3, 1998

Review of Radiological Analysis

The staff reviewed the licensee's calculations and performed confirmatory calculations to check the acceptability of the licensee's methodology and the resulting doses. The staff calculated the doses resulting from a MSLB accident using the methodology associated with SRP 15.1.5, Appendix A. The staff performed two separate assessments. One was based upon a pre-existing iodine spike, with an RCS activity of 3 µCi/g, and the other was based upon an accident initiated iodine spike. For the accident initiated spike case, the staff assumed that the DEI was 0.05 µCi/gm. The accident initiated an increase in the release rate of iodine from the fuel by a factor of 500 over the normal release rate that would maintain a DEI of 0.05 µCi/gm.

For these two cases, the staff calculated the thyroid doses for individuals located at the exclusion area boundary (EAB) and at the low-population zone (LPZ). The staff also calculated the thyroid dose to the control room operator. The parameters which were utilized in the staff's assessment are presented in Table 1 (attached). The staff used the lower range (nominal flow rate value less ten percent) of values for control room makeup and recirculation flow rates, as allowed by the TS. The EAB, LPZ, and control room doses calculated by the staff are presented in Table 2 (attached). The staff's calculations confirmed that the doses from a postulated MSLB meet the acceptance criteria and that the licensee's calculations are acceptable.

The results of both the licensee's and staff's calculations showed that the offsite doses met the guidelines of SRP 15.1.5, Appendix A (i.e., 10 CFR Part 100 limits, or a fraction thereof) and that the control room doses met the guidelines of SRP 6.4 (i.e., 10 CFR Part 50, Appendix A, General Design Criteria 19). On this basis, the staff finds the requested reduction in the allowable DEI from 0.35 to 0.05 µCi/gm, along with the increase in the allowable maximum primary to secondary leakage, to be acceptable.

3.0 SUMMARY

The staff reviewed the licensee's prediction of the primary-to-secondary leakage following a main steam line break accident, which was submitted in the 90 day report for the interim plugging criteria used at Braidwood Unit 1. The licensee predicted 122 gpm using a bounding binning strategy in their model of voltage-dependent growth rates for ODSCC indications at the tube support plates. The staff concludes this leakage estimate is bounding and conservative. However, the staff believes leakage at this level can contribute to the risk from events that are not specifically modeled as part of the plant's licensing basis (e.g. core damage accidents). The staff considered the risk implications and concluded that, based on the short time interval (e.g. limited to the duration of Braidwood Unit 1 cycle 7) coupled with the conservatism of the leakage calculations, the risk implications of the proposed license amendment are acceptable.

The staff performed a radiological dose assessment to determine the consequences of the increased primary-to-secondary leakage and reduced DEI. The staff calculated the potential consequences based upon two conditions: a pre-existing iodine spike and an accident initiated iodine spike. The evaluation assumed that the activity level of dose equivalent ^{131}I in the primary coolant was 0.05 µCi/gm for the accident-initiated spike and 3 µCi/gm for the pre-existing spike. The staff concludes that the proposed change meets the 10 CFR Part 100 limits for offsite doses and 10 CFR Part 50, Appendix A, General Design Criterion 19 for control room doses and is acceptable.

The licensee also utilized the results of the draft EPRI Report TR-103680, Rev. 1, November 1995, "Empirical Study of Iodine Spiking in PWR Power Plants," in its assessment of spiking factors for primary coolant at 0.05 μCi/gm. The EPRI report had spiking factors in the range of 45-150. This supports the conclusion that the NRC spiking factor of 500 is conservative.

The licensee used the EPRI empirical model to predict the rate of iodine release from the fuel rods in postulated MSLB/SGTR accident sequences. Since the RCS mass and the clean-up system constant for Braidwood are similar to that used in the EPRI model, the licensee used the SRP methodology with a DEI of 1.0 μCi/gm and a spiking value of 500. The resulting post-trip RCS activity 2 hours after the event is approximately 38 μCi/gm. At a DEI of 0.05 μCi/gm, it would require a spiking factor of nearly 10,000 to obtain a post-trip RCS activity near 38 μCi/gm. With a post-trip RCS activity of 38 μCi/gm, an increase in the allowable leak rate could impact the 10 CFR Part 100 limits. To accommodate an increase in the allowable leak rate by a factor of 20 (1.0 μCi/gm / 0.05 μCi/gm), the resultant activity would need to be below 1.9 μCi/gm. Although two of the seventeen post-trip data points from Braidwood exceeded 1.9 μCi/gm, both occurred during cycles with fuel defects. Braidwood, Unit 1, is currently operating with no fuel defects.

The staff has recently evaluated the issue of iodine spikes greater than 500 for primary coolant activity levels less than 1.0 μCi/gm. The staff examined the possibility of spiking factors increasing as a function of decreasing reactor coolant activity levels. One of the problems in addressing this issue is the lack of spiking data associated with MSLB events. All spiking data are associated with events typifying a SGTR. There is a certain amount of uncertainty when trying to project the magnitude of an iodine spike resulting from a MSLB using data which are more representative of a SGTR. One could estimate a spike from a MSLB to be within a factor of 10 of the spike from a SGTR based upon pressure drops and pressure drop rates of a MSLB to be within a factor of three of those of a SGTR. Adams has indicated that the spiking factor for a SGTR may be conservative by an order of magnitude. Given the order of magnitude conservatism in the SGTR spiking factor and estimating that the MSLB iodine spike is an order of magnitude higher than the SGTR spike, the staff determined that the two factors tend to offset each other.

The staff believes that the uncertainty associated with the iodine spiking factor was recognized in the development of SRPs 15.1.5 and 15.6.3 when the acceptance criterion for the accident-initiated spike case was established as a small fraction of 10 CFR Part 100. Establishing the acceptance criterion for this case at 10 percent of 10 CFR Part 100 limits allows a certain degree of error in the spiking assessment. If the dose consequences for the accident-initiated spike case were a small fraction (i.e., 10 percent) of 10 CFR Part 100, then a spiking factor of 5,000 (10 x spiking factor of 500) would need to occur before doses would exceed 10 CFR Part 100. The existing spiking data, even with consideration of an increase factor of 10 to account for a MSLB event, do not tend to support the suggestion that spiking factors in the 5,000 range are a likely occurrence. Therefore, the staff has concluded that the dose criteria are sufficient to address potential uncertainties in spiking factors associated with reactor coolant activity levels less than 1 μCi/gm.

The staff concludes that the use of a spiking factor of 500 is acceptable for analysis of this application.

Evaluation of Reducing Reactor Coolant System Dose Equivalent ^{131}I Limits below 0.35 μCi/gm

Generic Letter 95-05 states that licensees who wish to take credit in the radiological dose calculation for reducing DEI below 0.35 μCi/gm should provide a justification that evaluates the release rate data described in a report by J.P. Adams and C.L. Atwood entitled, "The Iodine Spike Release Rate During a Steam Generator Tube Rupture" (Nuclear Technology, Volume 94, page 361 (1991)). The requested information was contained in ComEd's application.

In the evaluation, ComEd indicated that Braidwood Station has experienced six occasions where the iodine spiking factor was greater than 500, which is the value assumed by the licensee in its dose assessment. A spiking factor of 500 is recommended for use in analyzing the consequences of MSLB and Steam Generator Tube Rupture (SGTR) accidents and is discussed in Standard Review Plan (SRP) Sections 15.1.5 and 15.6.3 (NUREG 0800). Because spiking data have shown occasions where spikes greater than 500 have occurred for DEI lower than 0.3 μCi/gm, the licensee addressed the implications of lowering the primary coolant activity to a value of 0.05 μCi/gm.

The licensee evaluated the Braidwood data against the SRP methodology. For a steady-state reactor coolant activity level of 1.0 μCi/gm, the licensee calculated a pre-trip release rate of iodine from the fuel of 27.5 curie/hour (Ci/hr). Based upon the SRP spiking factor of 500, the licensee determined the maximum release rate to be 13,733 Ci/hr. The highest release rate from the Braidwood trip data was 1,335 Ci/hr. The licensee stated that previously the doses were determined to be acceptable at 1.0 μCi/gm. A reduction in reactor coolant activity level to 0.05 μCi/gm would offset a 20 fold increase in primary to secondary leak rate and would result in a release rate of 686.7 Ci/hr. Two of the Braidwood data points showed release rates exceeding 686.7 Ci/hr, while 15 of the data points in an article in Nuclear Technology by Adams and Atwood in June 1991 exceeded this value. Of the 17 data points, only two had a pre-trip reactor coolant activity level below 0.05 μCi/gm.

When the Braidwood data were combined with the data by Adams and Atwood, the licensee concluded that the Braidwood data would not change the conclusions which were reached by Adams and Atwood. Although the Braidwood data included spiking factors greater than 500 when DEI was less than 0.3 μCi/gm, none of the data approached the proposed TS value of 0.05 μCi/gm. The licensee reviewed the combined Adams and Atwood data and Braidwood data in the range of 0.01 μCi/gm through 0.1 μCi/gm. In this range of reactor coolant activity level, the licensee determined that 76 percent of the data points had spiking factors which were below 500. The Adams and Atwood data showed that the highest spiking factor at the Pre-Trip activity of 0.05 μCi/gm was 773. The licensee indicated that the corresponding release rate of 368 Ci/hr at this value was less than the maximum calculated Braidwood release rate of 686.7 Ci/hr which was calculated for a primary coolant activity level of 0.05 μCi/gm and a spiking factor of 500.

The licensee also indicated that Adams and Atwood had derived a 95 percent/85 percent iodine release rate of 0.608 Ci/hr per megawatt electric (MWe). Application of this value at Braidwood would result in a calculated release rate of less than 714 Ci/hr. Of 17 reactor trips at Braidwood, two had exceeded 686.7 Ci/hr. Both occurred during cycles with fuel defects. Braidwood, Unit 1, is currently operating with no fuel defects. The licensee concluded the probability of iodine release rate exceeding 686.7 Ci/hr is small. The licensee further concluded that, due to the conservatisms in the dose analysis, there is a low probability of exceeding a small fraction of the 10 CFR Part 100 limits should a fuel release rate greater than 686.7 Ci/hr occur.

the staff considered the potential defense-in-depth and risk implications of the SG tube leakage predictions that support this license amendment.

The staff has considered the following:

1. ComEd employed a very conservative voltage binning strategy to support its estimate of the potential MSLB leak rate at EOC-7. The licensee reviewed a substantial amount of foreign and domestic plant data and performed several benchmarking studies on those data. The 90 day report provided strong support that a realistic and conservative estimate of the accident leak rate for the EOC-7 conditions would be approximately 60 gpm (based on 6.1 gpm for each indication restrained from burst).

2. Except as noted for indications restrained from burst (IRBs), the empirical model used to calculate MSLB leakage is intended to produce an 95% upper quantile estimate evaluated at the 95% confidence level based on the available field and laboratory data for freespan cracks. Best estimate predictions for non-IRB defects are typically an order of magnitude lower. However, a significant fraction of the estimated total leak rate at Braidwood Unit 1 is contributed by IRBs. The MSLB leakage model assumes 6.1 gpm leakage for each IRB, which bounds available laboratory specimen leakage data. A recent test of a pulled tube specimen under IRB conditions produced a leak rate of 0.9 gpm, corroborating the conservatism of the assumed IRB leak rates.

3. The support plates and associated crevice deposits have proven effective in minimizing the occurrence of operational leakage under normal operating pressures and temperatures in spite of the widespread presence of through wall cracks at these intersections at a large number of plants. No instances of operational leakage leading to forced shutdowns due to ODSCC at the support plates have been reported in the U.S. Had these through wall cracks been located in the freespan, the cracks would be expected to cause frequent forced shutdowns due to leakage.

4. Laboratory tests of tube to support plate intersections with included crevice deposits, in France, demonstrates a one to two order of magnitude reduction in leak rate associated with the resistance of the crevice deposits. This test was performed under cold conditions with simulated MSLB differential pressure. The tube specimens each contained drilled through wall holes. Thus, the test did not include the containing effect of the packed crevice deposits against crack opening displacement, which is thought to be an important effect and would further reduce leakage. However, similar tests under hot conditions have not been performed. Furthermore, any tendency for flashing steam or hot gases (under high temperature/high pressure severe accident conditions) to cut through the crevice deposits has not been evaluated.

The staff notes that operation in the above condition is limited until the end of the current operating cycle, since the SGs are scheduled to be replaced. Based on this short time interval coupled with the conservatisms of the leakage calculations, the staff concludes that the proposed license amendment is acceptable.

ComEd addressed this issue by revising the predictive methodology to account for voltage-dependent growth rates. ComEd applied the revised methodology to predict the EOC-7 conditions. The staff reviewed the revised methodology and questioned the adequacy of the binning strategy and the benchmarking. ComEd performed sensitivity studies on the binning and conducted additional benchmarking and used this information to develop a conservative upper bound to the leakage. This information is contained in a supplement to the "Braidwood Unit 1 Cycle 7 Interim Plugging Criteria Report" (also called the 90-day report), submitted on January 14, 1998. Using a bounding voltage binning strategy, ComEd predicted a primary-to-secondary leak rate of 122 gpm at the EOC-7. Based on this leak rate, ComEd determined that the DEI should be lowered to 0.05 μCi/gm.

The staff reviewed the 90-day report and concluded that the revised methodology provided a conservative prediction of the EOC-7 voltage distribution and leak rate. The staff's review of the 90-day report is documented in a letter to ComEd dated September 3, 1998.

2.0 EVALUATION

ComEd proposed to reduce the DEI limits from 0.35 to 0.05 μCi/gm for the 48 hour limit, and from 21 to 3 μCi/gm for the instantaneous limit, for the remainder of Braidwood, Unit 1, Cycle 7. These limits are based on increasing the total MSLB leak rate to 132.8 gpm, which encompasses ComEd's prediction of 122 gpm in the affected SG for EOC-7. Leakage in the unaffected SGs remains at 150 gpd, the TS limit for normal operation.

The licensee performed an assessment of the radiological dose consequences of a MSLB in support of its amendment request. Braidwood, Unit 1, is currently approved to operate based upon 26.8 gpm primary-to-secondary leakage with a DEI limit of 0.35 μCi/gm. The licensee found the dose consequences of increasing the total leakage to 132.8 gpm to be acceptable, with DEI reduced to 0.05 μCi/gm, based on the NRC acceptance criteria for doses at the Exclusion Area Boundary (EAB), the Low-Population Zone (LPZ), and the control room. ComEd provided additional information by letter dated July 17, 1998, to confirm that MSLB is the bounding design basis accident that is affected by accident-induced SG tube leakage.

Assessment of Higher Steam Generator Tube Leakage

As mentioned above, ComEd predicted 122 gpm primary-to-secondary leakage in the affected SG during a MSLB. This prediction was made for the purpose of the dose calculations that were performed in accordance with the guidance in Generic Letter 95-05. The staff has reviewed this prediction and concludes that it is conservative and acceptable for that purpose.

Nevertheless, the staff considers a leakage value of 122 gpm to be high. The SG tubes serve as a containment barrier as well as the reactor coolant pressure boundary. Tubes which may leak at 122 gpm during a design basis MSLB may leak significantly and provide a containment bypass mechanism during other events that are not specifically analyzed as part of the plant's licensing basis (e.g. core damage accidents with high primary pressure, including station blackout (SBO) sequences leading to core melt and high primary system temperature and pressure while the secondary side is depressurized). Decreasing the allowable limit on DEI does not effectively mitigate the offsite dose from these events. For this reason, significant steam generator tube leakage could have risk implications for the health and safety of the public. In keeping with the Commission's policy for incorporating risk insights into the regulatory decision-making process,

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 95 TO FACILITY OPERATING LICENSE NO. NPF-72

AND AMENDMENT NO. 95 TO FACILITY OPERATING LICENSE NO. NPF-77

COMMONWEALTH EDISON COMPANY

BRAIDWOOD STATION, UNIT NOS. 1 AND 2

DOCKET NOS. STN 50-456 AND STN 50-457

1.0 INTRODUCTION

By letter dated January 14, 1998, as supplemented by letter dated July 17, 1998, Commonwealth Edison Company (ComEd, the licensee) requested a change to the Technical Specification (TS) limit on Reactor Coolant System (RCS) Dose-Equivalent Iodine-131 (DEI) for Braidwood Station, Unit 1. The change would reduce the DEI from the current limit of 0.35 microcuries/gram (μCi/gm) to 0.05 μCi/gm for the remainder of Cycle 7. The July 17, 1998, submittal provided additional clarifying information that did not change the initial proposed no significant hazards consideration determination.

Braidwood, Unit 1, is currently operating under an operability assessment and has established an administrative control which limits DEI to 0.05 μCi/gm. The change to the DEI limit is necessary due a higher prediction of the primary-to-secondary leakage in the affected steam generator (SG) during a main steamline break (MSLB) accident. The leakage is associated with the interim plugging criteria (IPC) used for the Braidwood, Unit 1, SG tubes that were approved by the NRC by letter dated May 14, 1997 (amendment number 82). Consistent with the guidance in NRC Generic Letter 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking," ComEd evaluated a MSLB and reduced the DEI in order to ensure that (1) the 10 CFR Part 100 limits, or some fraction thereof, are met for offsite doses, and (2) 10 CFR Part 50, Appendix A, General Design Criterion 19 is met for control room doses. This issue is only applicable until the end of Cycle 7, at which time the SGs will be replaced.

During the spring 1997 refueling outage at the end-of-cycle (EOC) 6, ComEd compared the actual EOC-6 voltage distribution of indications attributed to outside-diameter stress corrosion cracking (ODSCC) with the predictions that were made at the beginning of the cycle. The licensee found the predicted voltage distribution was significantly nonconservative with respect to the actual voltage distribution. ComEd found that the major contributor to the nonconservative prediction was the more frequent occurrence of large growth rates associated with larger ODSCC indications (i.e., greater than 2.0 volts). Because the predictive methodology assumed a voltage-independent growth distribution, the apparent voltage-dependent behavior of ODSCC indications at Braidwood, Unit 1, was not accounted for and led to the nonconservative prediction of the EOC-6 conditions. The nonconservative prediction of the EOC-6 voltage distribution resulted in a nonconservative calculation of the leak rate associated with a MSLB.

SPECIFIC ACTIVITY (Continued)

take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomenon. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

The temperature and pressure changes during heatup and cooldown are limited to be consistent with the requirements given in the ASME Boiler and Pressure Vessel Code, Section III, Appendix G:

1. The reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figures 3.4-2a (3.4-2b) and 3.4-3a (3.4-3b) for the service period specified thereon:

    a. Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation; and

    b. Figures 3.4-2a (3.4-2b) and 3.4-3a (3.4-3b) define limits to assure prevention of non-ductile failure only. For normal operation, other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity, may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

2. These limit lines shall be calculated periodically using methods provided below,

3. The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the steam generator is below 70°F,

4. The pressurizer heatup and cooldown rates shall not exceed 100°F/hr and 200° F/hr respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F, and

5. System preservice hydrotests and in-service leak and hydrotests shall be performed at pressures in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI.

The fracture toughness properties of the ferritic materials in the reactor vessel are determined in accordance with the 1973 Summer Addenda to Section III of the ASME Boiler and Pressure Vessel and Code.

SPECIFIC ACTIVITY (Continued)

The sample analysis for determining the gross specific activity and $\bar{E}$ can exclude the radioiodines because of the low reactor coolant limit of 1 microCurie/gram DOSE EQUIVALENT I-131, and because, if the limit is exceeded, the radioiodine level is to be determined every 4 hours. If the gross specific activity level and radioiodine level in the reactor coolant were at their limits, the radioiodine contribution would be approximately 1%. In a release of reactor coolant with a typical mixture of radioactivity, the actual radioiodine contribution would be about 20%. The exclusion of radionuclides with half-lives less than 10 minutes from these determinations has been made for several reasons. The first consideration is the difficulty to identify short-lived radionuclides in a sample that requires a significant time to collect, transport, and analyze. The second consideration is the predictable delay time between the postulated release of radioactivity from the reactor coolant to its release to the environment and transport to the SITE BOUNDARY, which is relatable to at least 30 minutes decay time. The choice of 10 minutes for the half-life cutoff was made because of the nuclear characteristics of the typical reactor coolant radioactivity. The radionuclides in the typical reactor coolant have half-lives of less than 4 minutes or half-lives of greater than 14 minutes, which allows a distinct window for determination of the radionuclides above and below a half-life of 10 minutes. For these reasons the radionuclides that are excluded from consideration are expected to decay to very low levels before they could be transported from the reactor coolant to the SITE BOUNDARY under any accident condition.

Based upon the above considerations for excluding certain radionuclides from the sample analysis, the allowable time of 2 hours between sample taking and completing the initial analysis is based upon a typical time necessary to perform the sampling, transport the sample, and perform the analysis of about 90 minutes. After 90 minutes, the gross count should be made in a reproducible geometry of sample and counter having reproducible beta or gamma self-shielding properties. The counter should be reset to a reproducible efficiency versus energy. It is not necessary to identify specific nuclides. The radiochemical determination of nuclides should be based on multiple counting of the sample with typical counting basis following sampling of less than 1 hour, about 2 hours, about 1 day, about 1 week, and about 1 month.

Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the reactor coolant is below the lift pressure of the atmospheric steam relief valves. The Surveillance Requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to

OPERATIONAL LEAKAGE (Continued)

The Surveillance Requirements for RCS pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS pressure isolation valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

3/4.4.7 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduces the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady-State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride, and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady-State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady-State Limits.

The Surveillance Requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the reactor coolant ensure that the resulting 2-hour doses at the SITE BOUNDARY will not exceed an appropriately small fraction of 10 CFR Part 100 dose guideline values following a steam generator tube rupture accident in conjunction with an assumed steady-state reactor-to-secondary steam generator leakage rate of 1 gpm. For Unit 1 through Cycle 7, the limitations on the specific activity of the reactor coolant ensure that the resulting 2-hour off-site doses will not exceed an appropriately small fraction of the 10 CFR Part 100 dose guideline values following a Main Steam Line Break accident in conjunction with an assumed steady-state primary-to-secondary steam generator leakage rate of 150 gpd from each of the unfaulted steam generators and maximum site allowable primary-to-secondary leakage from the faulted steam generator. The values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the Braidwood Station, such as SITE BOUNDARY location and meteorological conditions, were not considered in this evaluation.

TABLE 4.4-4 (Continued)

TABLE NOTATIONS

\# Until the specific activity of the Reactor Coolant System is restored within its limits.

* Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since reactor was last subcritical for 48 hours or longer.

** A gross radioactivity analysis shall consist of the quantitative measurement of the total specific activity of the reactor coolant except for radionuclides with half-lives less than 10 minutes and all radioiodines. The total specific activity shall be the sum of the degassed beta-gamma activity and the total of all identified gaseous activities in the sample within 2 hours after the sample is taken and extrapolated back to when the sample was taken. Determination of the contributors to the gross specific activity shall be based upon those energy peaks identifiable with a 95% confidence level. The latest available data may be used for pure beta-emitting radionuclides.

*** A radiochemical analysis for $\bar{E}$ shall consist of the quantitative measurement of the specific activity for each radionuclide, except for radionuclides with half-lives less than 10 minutes and all radioiodines, which is identified in the reactor coolant. The specific activities for these individual radionuclides shall be used in the determination of $\bar{E}$ for the reactor coolant sample. Determination of the contributors to $\bar{E}$ shall be based upon these energy peaks identifiable with a 95% confidence level.

**** For Unit 1 through Cycle 7, reactor coolant DOSE EQUIVALENT I-131 will be limited to 0.05 microCuries per gram.

TABLE 4.4-4

REACTOR COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY	MODES IN WHICH SAMPLE AND ANALYSIS REQUIRED
1. Gross Radioactivity Determination**	At least once per 72 hours	1, 2, 3, 4
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	Once per 14 days	1
3. Radiochemical for $\bar{E}$ Determination***	Once per 6 months*	1
4. Isotopic Analysis for Iodine Including I-131, I-133, and I-135	a) Once per 4 hours, whenever the specific activity exceeds 1 μCi/gram DOSE EQUIVALENT I-131**** or 100/$\bar{E}$ μCi/gram of gross radioactivity, and	1#, 2#, 3#, 4#, 5#
	b) One sample between 2 and 6 hours following a THERMAL POWER change exceeding 15% of the RATED THERMAL POWER within a 1-hour period.	1, 2, 3

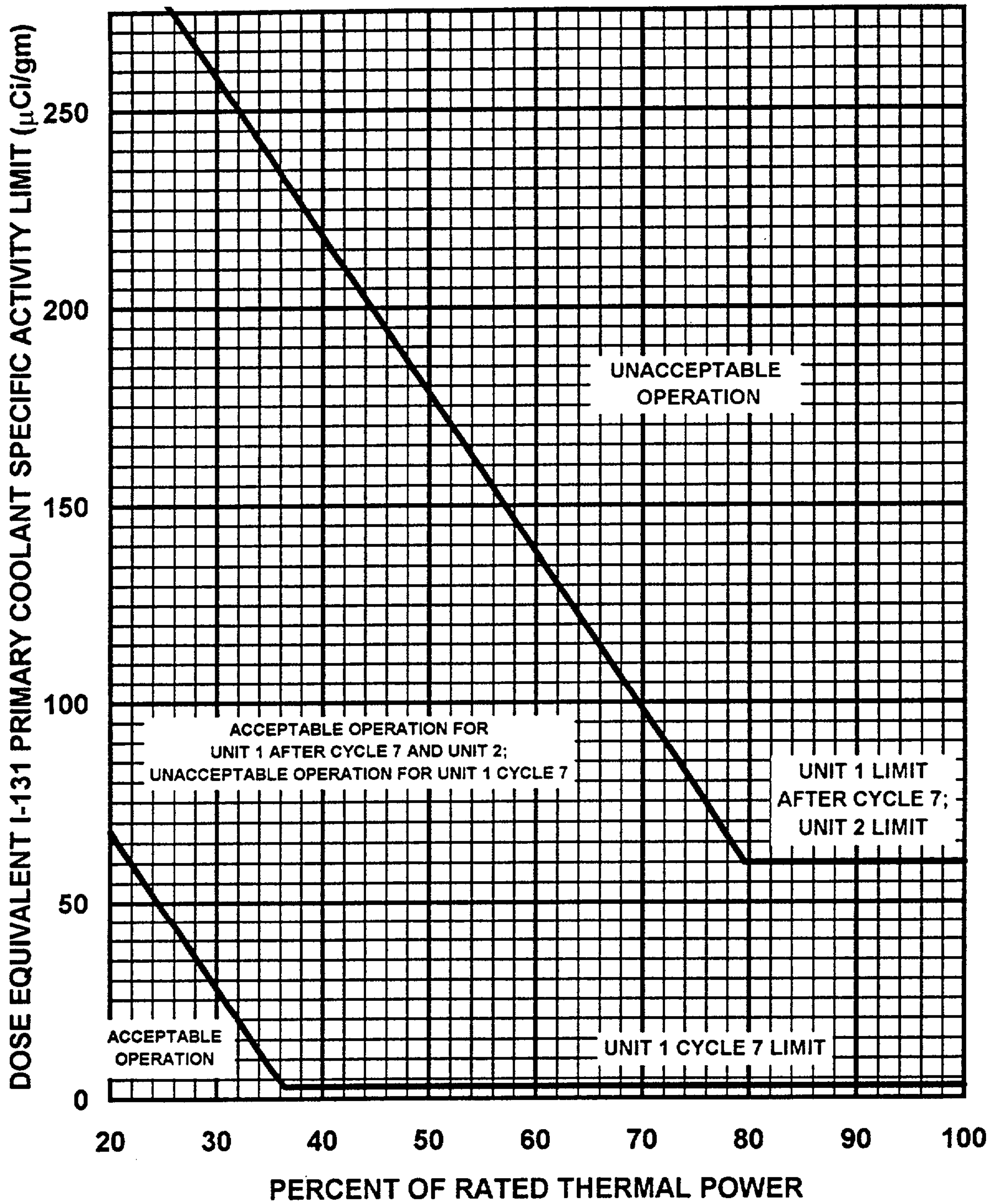

FIGURE 3.4-1

DOSE EQUIVALENT I-131 REACTOR COOLANT SPECIFIC ACTIVITY LIMIT VERSUS PERCENT OF RATED THERMAL POWER WITH THE REACTOR COOLANT SPECIFIC ACTIVITY > 1μCi/GRAM DOSE EQUIVALENT I-131*

* For Unit 1 through Cycle 7, Reactor Coolant Specific Activity > 0.05 μCi/Gram DOSE EQUIVALENT I-131.

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

MODES 1, 2, 3, 4, and 5:

With the specific activity of the reactor coolant greater than 1 microCurie per gram DOSE EQUIVALENT I-131* or greater than 100/E microCuries per gram, perform the sampling and analysis requirements of Item 4.a) of Table 4.4-4 until the specific activity of the reactor coolant is restored to within its limits.

SURVEILLANCE REQUIREMENTS

4.4.8 The specific activity of the reactor coolant shall be determined to be within the limits by performance of the sampling and analysis program of Table 4.4-4.

*For Unit 1 through Cycle 7, reactor coolant DOSE EQUIVALENT I-131 will be limited to 0.05 microCuries per gram.

REACTOR COOLANT SYSTEM

3/4.4.8 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.8 The specific activity of the reactor coolant shall be limited to:

a. Less than or equal to 1 microCurie per gram DOSE EQUIVALENT I-131**, and

b. Less than or equal to $100/\bar{E}$ microCuries per gram of gross radioactivity.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTION:

MODES 1, 2 and 3*:

a. With the specific activity of the reactor coolant greater than 1 microCurie per gram DOSE EQUIVALENT I-131** for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure 3.4-1, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours; and

b. With the specific activity of the reactor coolant greater than $100/\bar{E}$ microCuries per gram, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours.

*With T_{avg} greater than or equal to 500°F.

**For Unit 1 through Cycle 7, reactor coolant DOSE EQUIVALENT I-131 will be limited to 0.05 microCuries per gram.

ATTACHMENT TO LICENSE AMENDMENT NOS. 95 AND 95

FACILITY OPERATING LICENSE NOS. NPF-72 AND NPF-77

DOCKET NOS. STN 50-456 AND STN 50-457

Replace the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change. Pages marked with an asterisk are provided for convenience.

Remove Pages	Insert Pages
3/4 4-27	3/4 4-27
3/4 4-28	3/4 4-28
3/4 4-29	3/4 4-29
3/4 4-30*	3/4 4-30*
3/4 4-31	3/4 4-31
B 3/4 4-5	B 3/4 4-5
B 3/4 4-6*	B 3/4 4-6*
B 3/4 4-7*	B 3/4 4-7*

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 95 and the Environmental Protection Plan contained in Appendix B, both of which were attached to License No. NPF-72, dated July 2, 1987, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date if its issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Stewart N. Bailey, Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: September 3, 1998

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-457

BRAIDWOOD STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 95
License No. NPF-77

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Commonwealth Edison Company (the licensee) dated January 14, 1998, as supplemented by letter dated July 17, 1998, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-77 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 95 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Stewart N. Bailey, Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: September 3, 1998

TABLE 2

THYROID DOSES FROM BRAIDWOOD, UNIT 1, MAIN STEAMLINE BREAK ACCIDENT (REM) (VALUES CALCULATED BY NRC STAFF)

	DOSE	
LOCATION	**Pre-Existing Spike**	**Accident-Initiated Spike****
EAB	39.6*	23.3
LPZ	4.18*	2.57
Control Room **	4.50	0.30

* Acceptance Criterion = 300 rem thyroid
** Acceptance Criterion = 30 rem thyroid