ATTACHMENT B

MARKED UP PAGES FOR
PROPOSED CHANGES TO APPENDIX A
TECHNICAL SPECIFICATIONS OF
FACILITY OPERATING LICENSES
NPF-72, AND NPF-77

BYRON STATION UNITS 1 & 2
REVISED PAGES:

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* This page has no changes but is included for continuity.

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 mircoCurie 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, reactor coolant DOSE EQUIVALENT I-131 will be limited to ~~0.35~~ microCuries per gram.

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 $100/\bar{E}$ 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, reactor coolant DOSE EQUIVALENT I-131 will be limited to ~~0.35~~ microCuries per gram.

0.20

See Insert A

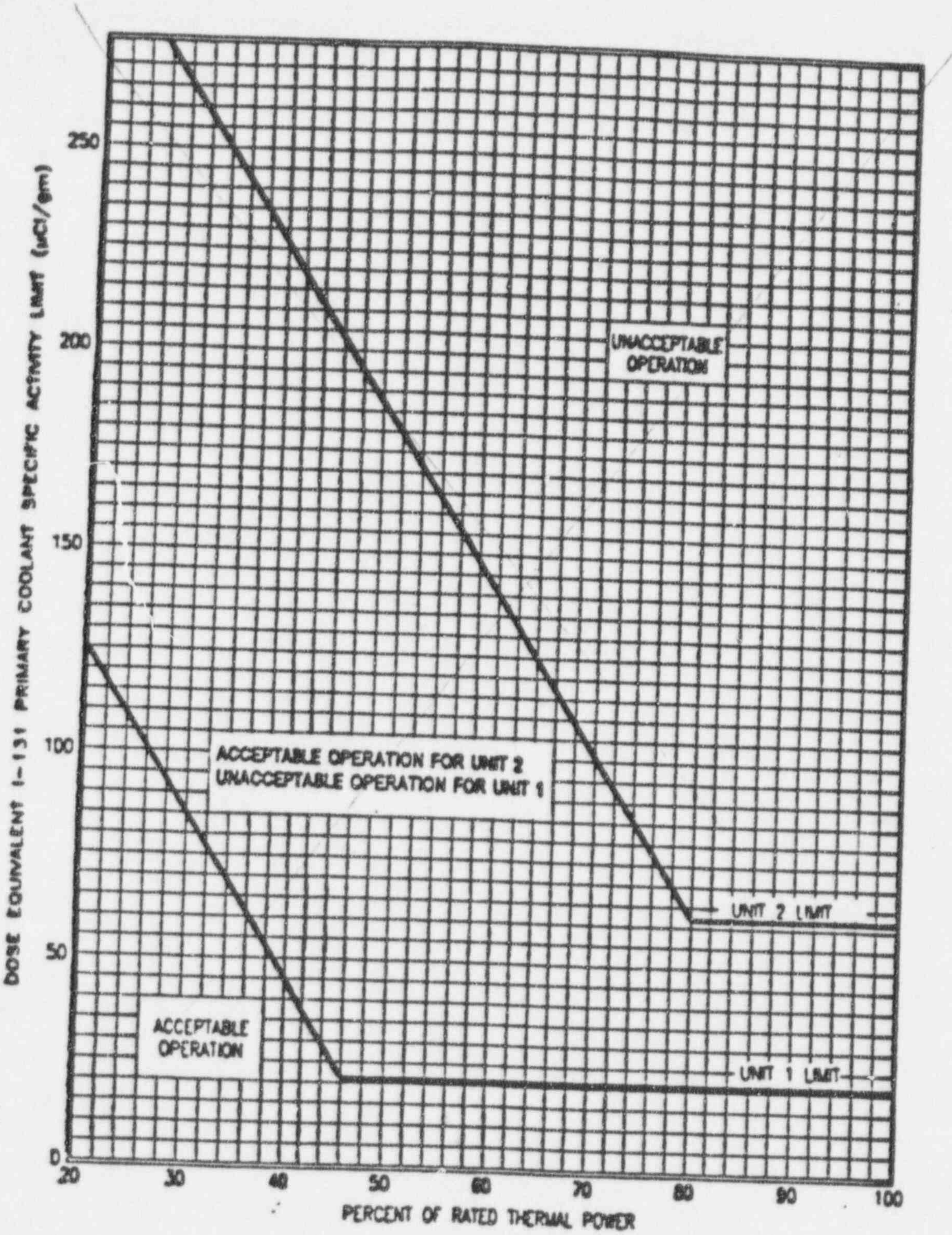

FIGURE 3.4-1
DOSE EQUIVALENT I-131 REACTOR COOLANT SPECIFIC ACTIVITY
LIMIT VERSUS PERCENT OF RATED THERMAL POWER WITH THE REACTOR
COOLANT SPECIFIC ACTIVITY >1µCI/GRAM DOSE EQUIVALENT I-131*

*For Unit 1, Reactor Coolant Specific Activity ~~>0.35~~ µCi/Gram DOSE EQUIVALENT I-131

REACTOR COOLANT SPECIF. 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 percent of the RATED THERMAL POWER within a 1-hour period.	1, 2, 3

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 percent 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 percent confidence level.

****For Unit 1, reactor coolant DOSE EQUIVALENT I-131 will be limited to ~~0.35~~ 0.46 microCuries per gram.

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. 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 Byron Station, such as SITE BOUNDARY location and meteorological conditions, were not considered in this evaluation.

Insert B

INSERT A

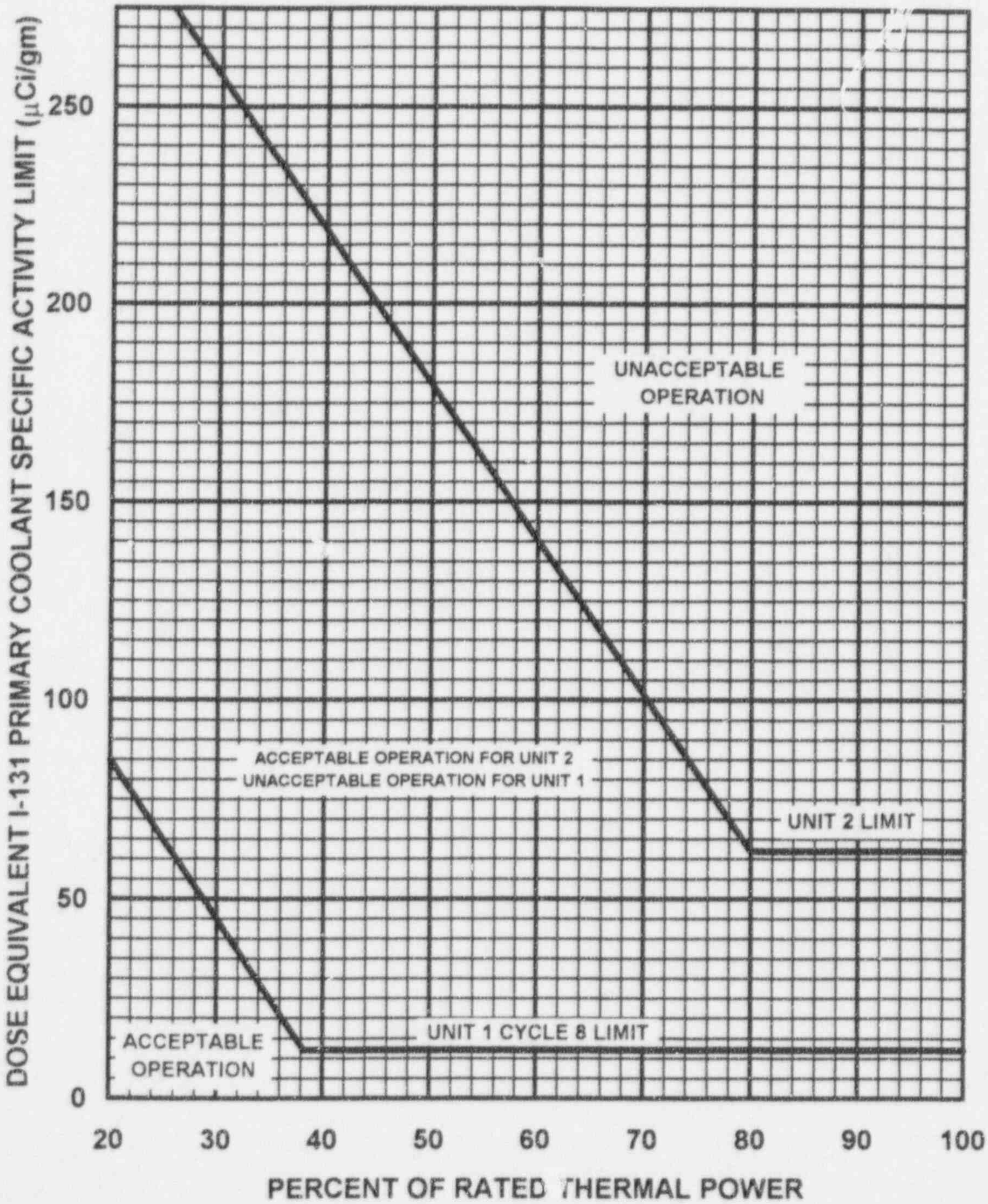

FIGURE 3.4-1

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 8, Reactor Coolant Specific Activity > 0.2 μCi/Gram DOSE EQUIVALENT I-131.

Insert B

For Unit 1 through Cycle 8, 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 unfaulted steam generator and maximum site allowable primary-to-secondary leakage from the faulted steam generator.

ATTACHMENT C

EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATIONS FOR
PROPOSED CHANGES TO APPENDIX A
TECHNICAL SPECIFICATIONS OF
FACILITY OPERATING LICENSES
NPF-72 AND NPF-77

Commonwealth Edison (ComEd) has evaluated this proposed amendment and determined that it involves no significant hazards considerations. According to Title 10 Code of Federal Regulations Section 50 Subsection 92 Paragraph c (10 CFR 50.92 (c)), a proposed amendment to an operating license involves no significant hazards considerations if operation of the facility in accordance with the proposed amendment would not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated; or

2. Create the possibility of a new or different kind of accident from any accident previously evaluated; or

3. Involve a significant reduction in a margin of safety.

A. INTRODUCTION

ComEd proposes to revise Byron Technical Specification (TS) 3.4.8, "Specific Activity," and the associated Bases. This revision will lower the Unit 1 Reactor Coolant System (RCS) Dose Equivalent (DE) Iodine 131 (I-131) level from 0.35 microCuries per gram (μCi/gm) to 0.20 μCi/gm and correct a typographical error. This revision will also lower the RCS DE I-131 limit in TS Figure 3.4-1. These revisions will be in effect for the remainder of Unit 1 Cycle 8. At the completion of Byron Unit 1 Cycle 8, ComEd will be replacing the original Westinghouse Model D-4 Steam Generators.

This change is required in order to provide additional margin to the maximum site allowable primary-to-secondary leakage limit. The total potential leakage includes primary-to-secondary leakage from circumferential indications which may exist in the faulted steam generator, leakage from indications remaining in service in the faulted steam generator due to application of the approved Interim Plugging Criteria and F* criteria, and 150 gpd leakage from each of the three unfaulted steam generators.

B. NO SIGNIFICANT HAZARDS ANALYSIS

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Generic Letter 95-05, "Voltage Based Repair Criteria For Westinghouse Steam Generator Tubes Affected By Outside Diameter Stress Corrosion Cracking," allows lowering of the RCS DE I-131 activity as a means for accepting higher projected leak rates if justification for equivalent I-131 below 0.35 μCi/gm is provided. Four methods for determining the impact of a release of activity to the public were reviewed to provide the justification. They are as follows:

Method 1: NRC NUREG 0800, Standard Review Plan (SRP) Methodology

Method 2: Methodology described in a report by J.P. Adams and C.L. Atwood, "The Iodine Spike Release Rate During a Steam Generator Tube Rupture," Nuclear Technology, Vol. 94 p. 361 (1991), using Byron Station reactor trip data.

Method 3: Methodology described in Adams and Atwood report, using normalized industry reactor trip data.

Method 4: Methodology described in draft EPRI Report TR-103680, Revision 1, November 1995, "Empirical Study of Iodine Spiking in PWR Plants".

The effect of reducing the RCS DE I-131 limit on the amount of activity released to the environment remains unchanged when the maximum site allowable primary-to-secondary leakage limit is proportionately increased. With a DE I-131 limit of 1.0 μCi/gm, the maximum site allowable leakage limit was calculated in accordance with the NRC SRP methodology to be 12.8 gpm. The corresponding calculated activity released during a MSLB is 15.8 Ci. ComEd has evaluated the reduction of the DE I-131 to 0.20 μCi/gm along with the increase of the allowable leakage to 64 gpm and has concluded:

-the maximum activity released is not changed, and

-the offsite dose including the iodine spiking factor is bounded by method 1.

Therefore, the offsite dose assessment and conclusions previously reached remain valid and continue to meet the requirements of 10CFR100.

An evaluation of Control Room dose attributed to a MSLB concurrent with steam generator primary-to-secondary leakage at the site allowable leakage limit was performed in support of a license amendment request for application of 1.0 volt Interim Plugging Criteria. This evaluation concluded that Control Room dose due to the MSLB scenario is bounded by the existing loss of coolant accident analysis. Therefore, the maximum site allowable primary-to-secondary leakage

limit continues to be based on offsite dose at the Exclusion Area boundary due to MSLB leakage. This conclusion was previously submitted to the Staff in a September 22, 1994, transmittal in support of the 1.0 volt Interim Plugging Criteria license amendment request.

Based on the NRC SRP methodology for dose assessments, the Control Room dose, the Low Population Zone dose, and the dose at the Exclusion Area Boundary continue to satisfy the appropriate fraction of the 10CFR100 dose limits.

The Adams and Atwood report concluded that the NRC SRP methodology, which specifies a release rate spike factor of 500 for iodine activity from the fuel rod to the RCS, is conservative. In order to justify that a release rate spike factor of 500 is conservative, actual operating data from the previous reactor trips of Byron Unit 1 and Unit 2, with and without fuel failures, were reviewed and analyzed using the methodology presented Section II.C of the Adams and Atwood report (Method 2). The same five data screening criteria described in the Adams and Atwood report were applied to the Byron data to ensure consistency and validity when comparing the Byron results to the data in the Adams and Atwood report. Of the twenty-eight (28) reactor trip events at Byron Units 1 and 2, twelve(12) met the five data screening criteria.

Three of the Byron trips occurred during cycles with no failed fuel. In all three of these instances, the calculated spike factor was less than the spike factor of 500 assumed in the NRC SRP methodology. Byron Unit 1 Cycle 8 is currently operating with no failed fuel and a DE I-131 activity of approximately 6E-4 μ Ci/gm. The three previous trips with no fuel failures had steady-state iodine values that are relatively close to current operating conditions. It is therefore reasonable to conclude that the calculated spike factors from those trips would reflect the spike factor expected from an actual trip during the current cycle.

Based on the data in the Adams and Atwood report, the NRC SRP release rate spike factor of 500 may seem non-conservative since the Adams and Atwood factor was typically greater than 500 when initial concentrations were less than 0.3 μCi/gm. The primary reason for these high ratios (up to 12,000) is not because the absolute post-trip release rate is high (factor numerator), but rather because the steady-state release rate (factor denominator) is low. The Byron specific data only resulted in one trip with a calculated release rate spike factor greater than 500, a value of 603.9. The trip occurred during the first operating cycle of Unit 2 which experienced failed fuel and a very low steady-state release rate. It is not expected based upon the current fuel cycle conditions that a spiking factor of greater than 500 would occur.

In order to compare the Byron specific data to the NRC SRP methodology, the release rate for a steady-state RCS DE I-131 activity of 1.0 μCi/gm was calculated. Using the Byron specific data, the steady-state release rate is 17.6 Ci/hr. Using a release rate factor of 500 for the accident initiated spike, the post-trip maximum release rate would be 8797 Ci/hr. This is significantly higher than the largest iodine release rate of 127 Ci/hr from the Byron data. This demonstrates that, although a data point shows an iodine spike factor greater than 500, the resulting post-trip RCS DE I-131 fuel rod iodine release rate is less than the fuel rod iodine release rate from the NRC SRP methodology.

In the fourth method, the results from Draft EPRI Report TR-103680, Rev. 1, November 1995, "Empirical Study of Iodine Spiking In PWR Power Plants" were applied. The objective of the EPRI study was to quantify the iodine spiking in postulated Main Steam Line Break / Steam Generator Tube Rupture (MSLB/SGTR) sequences. In the EPRI report, an iodine spike factor between 40 and 150 was determined to match data from existing plant trips. The maximum iodine spike factor value of 150 was applied to a steady-state equilibrium RCS DE I-131 activity of 0.33 μCi/gm. The resulting 2-hour average iodine concentration for a postulated MSLB/SGTR sequence was determined to be 3.1 μCi/gm. Since the EPRI report is based on industry data and the EPRI method predicted a post-accident iodine activity which is a small fraction of the activity predicted by the NRC SRP methodology, it can be expected that, for the proposed 0.2 μCi/gm limit under a MSLB/SGTR sequence, the post-accident iodine activity would be a small fraction of the RCS DE I-131 activity predicted by the NRC SRP methodology.

Lowering the Unit 1 RCS DE I-131 activity limit is conservative and remains bounded by the NRC SRP methodology. Thus, all offsite and control room dose assessment conclusions satisfy the appropriate limits of 10CFR100 and GDC 19. These proposed changes do not result in a significant increase in the consequences of an accident previously analyzed.

The RCS DE I-131 activity limit is not considered as a precursor to any accident. Therefore, this proposed change does not result in a significant increase in the probability of an accident previously analyzed.

The correction of the typographical error is administrative in nature and has no impact on either the probability or consequences of an accident previously analyzed.

2. **The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.**

The changes proposed in this amendment request conservatively reduce the Unit 1 DE I-131 limit at which action needs to be taken and correct a typographical error. The changes do not directly affect plant operation. These changes will not result in the installation of any new equipment or systems or the modification of any existing equipment or systems. No new operating procedures, conditions or modes will be created by this proposed amendment.

Thus, this proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **The proposed change does not involve a significant reduction in a margin of safety.**

NRC Generic Letter 95-05 allows lowering of the dose equivalent iodine as a means for accepting higher projected leakage rates provided justification for equivalent I-131 below 0.35 μCi/gm is provided. Four methods for determining the fuel rod iodine release rates and spike factors during an accident were reviewed. Each of these methods utilized actual industry data, including Byron Unit 1 and Byron Unit 2, for pre- and post-reactor trip DE I-131 activities. Each of the methods demonstrated that the actual fuel rod iodine release rates are a small fraction of the release rate as calculated using the NRC SRP methodology. All design basis and off-site dose calculation assumptions remain satisfied. This proposed change will not result in a reduction in a margin of safety.

Correction of the typographical error is administrative in nature and does not impact the margin of safety. Therefore, the proposed changes do not result in a significant reduction in a margin of safety.

Therefore, based on the above evaluation, ComEd has concluded that these changes involve no significant hazards considerations.

ATTACHMENT D

ENVIRONMENTAL ASSESSMENT FOR
PROPOSED CHANGES TO APPENDIX A
TECHNICAL SPECIFICATIONS OF
FACILITY OPERATING LICENSES
NPF-72 AND NPF-77

Commonwealth Edison Company (ComEd) has evaluated this proposed license amendment request against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with Title 10, Code of Federal Regulations, Part 51, Section 21 (10 CFR 51.21). ComEd has determined that this proposed license amendment request meets the criteria for a categorical exclusion set forth in 10 CFR 51.22(c)(9). This determination is based upon the following:

1. The proposed licensing action involves the issuance of an amendment to a license for a reactor pursuant to 10 CFR 50 which changes a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or which changes an inspection or a surveillance requirement. This revision will lower the Reactor Coolant System Dose Equivalent Iodine 131 limit from 0.35 microCuries per gram to 0.20 microCuries per gram. This revision will also lower the Reactor Coolant System Dose Equivalent Iodine 131 limit in Technical Specification Figure 3.4-1. These revisions will be in effect for the remainder of Unit 1 Cycle 8;

2. this proposed license amendment request involves no significant hazards considerations as demonstrated in Attachment C;

3. there is no significant change in the types or significant increase in the amounts of any effluent that may be released off-site; and

4. there is no significant increase in individual or cumulative occupational radiation exposure.

Therefore, pursuant to 10 CFR 51.22(b), neither an environmental impact statement nor an environmental assessment is necessary for this proposed license amendment request.