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ComEd

November 24, 1997

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U. S. Nuclear Regulatory Commission
Washington, DC 20555

Attention: Document Control Desk

Subject: Byron Nuclear Power Station Unit 1
Response to Notice of Violation
Inspection Report No. 50-454/97013
NRC Docket Number 50-454

Reference: John A. Grobe letter to Mr. Graesser dated October 30, 1997, transmitting NRC Inspection Report 50-454/97013

Enclosed is Commonwealth Edison Company's response to the Notice of Violation (NOV) which was transmitted with the referenced letter and Inspection Report. The NOV cited two (2) Severity Level IV violations requiring a written response. ComEd's response is provided in the attachments.

This letter contains the following commitments:

1) As a result of the identification of two items with inadequate safety evaluations, additional reviews have been initiated to focus on possible unidentified impacts of RSG changes on Safety Systems. These additional reviews will ensure that, as a minimum, the bases for conclusions of no impact are adequately documented.

2) Revise UFSAR Section 5.4.7, "Residual Heat Removal System," to discuss the quantitative evaluation on RHR performed as part of the SGR project UFSAR update.

3) Revise UFSAR Section 6.1.3.2 to reflect the sump pH response with the RSGs as part of the SGR project UFSAR update.

4) Revise UFSAR Figure 6.1-1 to reflect the containment sump water volumes with the RSGs as part of the SGR project UFSAR update.

5) Review all calculations that utilized RCS volume as a design input and revise calculations, as necessary, to ensure acceptable results due to the RCS volume increase.

6) Include as part of the UFSAR update, a detailed table of RCS total and component volumes, hot and cold, for both units.

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If your staff has any questions or comments concerning this letter, please refer them to Don Brindle, Regulatory Assurance Supervisor, at (815)234-5441 ext. 2280.

Respectfully,

K. C. Kofron
for K. L. Graesser
Site Vice President
Byron Nuclear Power Station

KLG/DB/rp

Attachment(s)

cc: A. B. Beach, NRC Regional Administrator - RIII
G. F. Dick Jr., Byron Project Manager - NRR
Senior Resident Inspector, Byron
M. J. Jordan, Reactor Projects Chief - RIII
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bcc: D. L. Farrar, Nuclear Regulatory Services Manager, Downers grove
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ATTACHMENT I

VIOLATION (50-454/97013-01a,b)

10 CFR 50.59(a)(1) states, in part, that a licensee may make changes to the facility as described in the safety analysis report without prior Commission approval unless the proposed change involves an unreviewed safety question.

10 CFR 50.59(b)(1) states, in part, that the licensee shall maintain records of changes in the facility as described in the safety analysis report and that records must include a written safety evaluation which provides the basis for the determination that the change does not involve an unreviewed safety question.

The Byron Updated Final Safety Analysis Report (UFSAR) Section 5.4.7.1 "Design Basis" stated that "..., the RHRS [Residual Heat Removal System] is designed to reduce the temperature of the reactor coolant from 350°F to 140°F within 36 hours."

The Byron UFSAR Section 6.1.3 "Post Accident Chemistry," Section 6.1.3.1 "Steamline Break Inside Containment" and Section 6.1.3.2 "Main Feedwater Line Break Inside Containment" described the effect of a main steamline break (MSLB) and main feedline break (MFLB) on containment sump level and pH.

a. Contrary to the above, as of September 9, 1997, the licensee had not performed an adequate safety evaluation to determine whether the impact on the design basis of the RHR system for the replacement steam generator (RSG) modification, constituted an unreviewed safety question. Specifically, the evaluation was deficient because it failed to consider the effect of the increased heat load (associated with the increased reactor volume for the RSG modification) on the RHR system performance. (50-454/97013-01a(DRS))

b. Contrary to the above, as of September 9, 1997, the licensee had not performed an adequate safety evaluation to determine whether the impact on the containment sump level and pH for the RSG modification, constituted an unreviewed safety question. Specifically, the evaluation was deficient because it failed to consider the RSG increased secondary side mass inventory and larger feedwater break area on containment sump and pH level under a MSLB or MFLB. (50-454/97013-01b(DRS))

This is a Severity Level IV Violation (Supplement I)

REASON FOR THE VIOLATION

a. Inadequate Safety Evaluation - RHR Performance

The RHR system is capable of reducing the temperature of the reactor coolant per UFSAR Figures 5.4-6 and 5.4-7 for dual and single train operation, respectively (UFSAR Section 5.4.7). The initial review of system impacts did not identify impact on RHR performance as requiring quantitative analysis because: 1) the integrated decay heat is much larger than the added heat due to the increased RSG volume and metal mass, and 2) qualitatively the RSG impacts are small compared to the existing margin between

UFSAR Section 5.4.7.1 "Design Basis" and the calculated system performance. There was a failure to document engineering judgement, and therefore, a documented quantitative analysis was not performed to support the conclusion of no impact.

b. Inadequate Safety Evaluation - Containment Sump Level & Sump pH

Calculations were performed to assess the impact of the RSG design on UFSAR Section 6.1.3, "Post-Accident Chemistry". Both the minimum and maximum pH calculations were performed for the appropriate limiting accident conditions. The results were verified to be within the acceptable pH band as specified in the plant Technical Specifications. However, UFSAR Sections 6.1.3.1 and 6.1.3.2 discuss the safety system response to the MSLB and MFLB, respectively. Changes in the MFLB transient response (Containment Spray (CS) actuation) were not specifically identified and documented in the safety evaluation regarding containment sump pH values.

Calculations were also performed to determine the maximum volume of water in the containment following an accident. However, only the limiting case for containment maximum flood level following a Large Break Loss of Coolant Accident (LB LOCA) was determined. UFSAR Sections 6.1.3.1 and 6.1.3.2 discuss the safety system response to the MSLB and MFLB, respectively. The specific containment water volumes for these transients were not determined. Changes in the MFLB transient response (CS actuation) were not specifically identified and documented in the safety evaluation regarding containment sump water volumes.

The initial review of these transients did not identify the potential impact as requiring quantitative analysis for containment sump level and pH because qualitatively the RSG impacts are bounded by the analysis performed for the LB LOCA transient. Therefore, due to a lack of attention to detail, a documented quantitative analysis was not performed to support the conclusion of no impact.

CORRECTIVE STEPS TAKEN AND RESULTS ACHIEVED

a. Inadequate Safety Evaluation - RHR Performance

A quantitative analysis was performed to document the conclusion of no impact. The RSGs contain more primary liquid mass and more metal mass than the original Steam Generators (OSGs). There are also differences in the relative amounts of water and steam on the secondary side. A calculation that accounts for these differences was performed based on the assumptions and methodology used to generate the UFSAR curves. Single train and dual train RHR cooldown were analyzed to evaluate the effects on RHR system performance. For RHR operations with only a single operable RHR train, the measure of RHR performance is based on the duration required to cool the RCS from 350°F to 200°F (UFSAR Figure 5.4-7). In this case the RSGs cause an increase of approximately 0.3 hours to the existing 39 hour single train RHR cooldown time.

For RHR operations with two operable RHR trains, the measure of RHR performance is based on the duration required to cool the RCS from 350°F to 140°F (UFSAR Figure 5.4-6). In this case the RSGs cause an increase of approximately 0.3 hours to the existing 30.3 hour dual train RHR cooldown time. This is within the design basis of less than 36 hours..

b. Inadequate Safety Evaluation - Containment Sump Level & Sump pH

A quantitative evaluation of the MFLB transient was performed to determine the impact of the revised Main Feedwater/Auxiliary Feedwater configuration on the containment sump pH. The current UFSAR evaluation (Section 6.1.3.2, "Main Feedwater Line Break") states that the MFLB transient does not initiate CS since containment pressure remains lower than the CS actuation setpoint. The sump pH, therefore, is considered to be the pH of the condensate fluid assumed in the UFSAR. The evaluation performed for the RSGs demonstrates that a MFLB could elevate containment pressure sufficiently to actuate CS, thus causing the event to respond similar to the MSLB transient. However, the mass and energy release for the MFLB with the RSGs is less than that of the MSLB transient, therefore, the MSLB remains the more limiting transient for the containment environmental conditions. The current UFSAR evaluation for MSLB (Section 6.1.3.1) indicates a constant pH for the CS fluid that is higher than in condensate fluid (Section 6.1.3.2). Therefore, the impact of the RSGs on the MFLB is that the containment sump pH could increase to a value consistent with t[illegible]MSLB transient due to the actuation of CS. The amount of additional RSG seconda[illegible]de mass is not sufficient to appreciably reduce the pH of the sump fluid. This char[illegible]erefore, has no safety significance since the sump pH remains within the limits acceptable for post-accident conditions specified in the Technical Specifications.

Also, the impact of the RSGs on the containment sump water volumes was evaluated for both the MSLB and the MFLB. In the case of the MSLB, the difference for the RSG case is an increase in secondary side mass (approximately 22,000 lbs). Relative to the total volume of fluid pumped from containment spray in 30 minutes, this increase is negligible and the sump water volume with the RSG remains consistent the quantities indicated in UFSAR Figure 6.1-1. UFSAR Figure 6.1-2 provides the containment sump water volume following a MFLB accident. Since Unit 1 actuates CS with the RSGs, the containment water volume in Figure 6.1-2 no longer applies to Unit 1. The sump water volume shown in Figure 6.1-1 which account for CS actuation would apply for the Unit 1 MFLB. Both the MSLB and MFLB transients have containment sump water volumes below the limiting case for containment flooding (LB LOCA) documented in UFSAR Attachment D3.6. This change, therefore, has no safety significance since the sump water volumes remain within the maximum volume acceptable for post-LOCA conditions.

CORRECTIVE STEPS THAT WILL BE TAKEN TO AVOID FURTHER VIOLATION

a, b. Inadequate Safety Evaluation - RHR Performance and Containment Sump Level & Sump pH

As a result of the identification of two items with inadequate safety evaluations, additional reviews have been initiated. These re-reviews focus on possible unidentified impacts of RSG changes on Safety Systems. They include: 1) A re-review of select UFSAR and Technical Specification sections, and 2) A re-review of FTI's "NSS and BOP Systems Review" (document 51-1239285-03). These additional reviews will ensure that, as a minimum, the bases for conclusions of no impact are adequately documented. The SGR safety evaluation will be augmented, if required, to reflect the review results so that the bases of conclusions are readily accessible to the reviewer. This action will be tracked by NTS item# 454-100-97-01301-01.

UFSAR Section 5.4.7, "Residual Heat Removal System," will be revised to discuss the quantitative evaluation performed as part of the SGR project UFSAR update. Additionally, the SGR safety evaluation will be revised to reflect the results of this analysis. This action will be tracked by NTS item# 454-100-97-01301-02.

UFSAR Section 6.1.3.2 will be revised to reflect the sump pH response with the RSGs as part of the SGR project UFSAR update. The SGR safety evaluation will be revised to reflect the results of this evaluation. This action will be tracked by NTS item# 454-100-97-01301-03.

UFSAR Figure 6.1-1 will be revised to reflect the containment sump water volumes with the RSGs as part of the SGR project UFSAR update. The SGR safety evaluation will be revised to reflect the results of this evaluation. This action will be tracked by NTS item# 454-100-97-01301-04.

A thorough understanding of and strict adherence to the requirements of the 10CFR50.59 process is necessary to ensure an adequate safety evaluation. Initiatives underway by ComEd include advanced training to provide a depth of understanding for those performing and reviewing safety evaluations. This training focuses on the need to identify potential impacts associated with changes. This training stresses the requirement for adequate research and documentation to provide the bases of safety evaluation conclusions. Appropriate individuals from the SGR project engineering organization have successfully completed this training.

DATE WHEN FULL COMPLIANCE WILL BE ACHIEVED

Full compliance will be achieved on December 31, 1997 with the final issuance of the RSG safety evaluation.

ATTACHMENT II

VIOLATION (50-454/97013-06a,b)

10 CFR Part 50, Appendix B, Criterion III, "Design Control," requires in part, that design control measures shall provide for verifying or checking the adequacy of design.

a. Contrary to the above, as of September 3, 1997, licensee design change control measures for verifying the adequacy of the replacement steam generator modification had been inadequate for BWI Calculation 222-7720-A13 "Engineering Calculations – Byron/Braidwood RSG – Primary Fluid Volumes vs. Height," Revision 0, issued April 5, 1995 in which the new reactor coolant system volume had been incorrectly determined. (50-454/97013-06a(DRS))

b. Contrary to the above, as of September 18, 1997, licensee design control measures for verifying the adequacy of the replacement steam generator modification had been inadequate for FTI calculation 51-1266158-01, "RSG AFW (Auxiliary Feedwater) Cooldown Requirements," Revision 1, issued June 6, 1997, in which the licensee had failed to consider the specific heat capacity of the replacement steam generators and the heat load of the main feedwater system. (50-454/97013-06b(DRS))

This is a Severity Level IV Violation (Supplement I)

REASON FOR THE VIOLATION

a. Inadequate Design Control – Reactor Coolant System (RCS) Volume Calculation

BWI calculation 222-7720-A13, Revision 0, was non-conservative with regard to the calculation of the RSG primary volume. The calculation did not account for hydraulic expansion of the tubes into the tubesheet and also did not calculate or address the increase in volume due to thermal expansion. The reason for the violation can be attributed to lack of attention to detail.

b. Inadequate Design Control – AFW Cooldown Requirements

Framatome Technologies, Incorporated (FTI) Calculation 51-1266158-01, Revision 1, calculates the additional AFW required to meet Technical Specifications and UFSAR requirements for cooldown. Additional water is required since the RSGs have increased stored energy in: 1) additional primary coolant mass, 2) additional steam generator metal mass, and 3) additional feedwater piping metal mass. Also, the specific heat capacity value used in the FTI calculation was not adjusted for the materials and conditions in the RSG.

Because the calculation was performed using overall conservative assumptions, individual assumptions and non-conservatisms were not documented. The reason for the violation can be attributed to lack of attention to detail and failure to document with sufficient detail decisions/assumptions utilized in calculations. The new calculation demonstrates the overall conservative nature of the original calculation with the conclusion that the differential auxiliary feedwater requirement decreased.

CORRECTIVE STEPS TAKEN AND RESULTS ACHIEVED

a. Inadequate Design Control - RCS Volume Calculation

BWI calculation 222-7720-A13, Revision 0 was revised to address the violation concerns. The calculation was revised to account for three expansion factors not previously considered that are associated with the RSGs at operating conditions: 1) thermal expansion of the material (tubes and primary head), 2) pressure boundary dilation due to the differential pressure, and 3) hydraulic expansion of the tubes in the tube sheet. The calculation was acceptance reviewed by ComEd. The revised calculation has been transmitted for use as design input where applicable for other SGR related calculations.

b. Inadequate Design Control - AFW Cooldown Requirements

A new calculation, 32-1266253, has been prepared that rigorously addresses the impact of the RSG on the design and license basis for AFW cooldown. The new calculation demonstrates that the original calculation result was conservative. This calculation has been acceptance reviewed by ComEd. This result confirms the earlier evaluation that the total AFW requirements for the RSG remain below the design basis value of 200,000 gallons.

CORRECTIVE STEPS THAT WILL BE TAKEN TO AVOID FURTHER VIOLATION

a. Inadequate Design Control - RCS Volume Calculation

A review of all project documents was performed to identify all calculations that utilized RCS volume as a design input. This review covered RCS volume inputs from all sources not just BWI calculation 222-7720-A13, Revision 0. These calculations will be reviewed and revised, as necessary, to ensure acceptable results due to the RCS volume increase. This action will be tracked by NTS ite# 454-100-97-01306-01.

The revised calculation results also support a supplement to an existing Technical Specification amendment request. The amendment request primarily deals with the change in P_a, but also includes the change to the "Design Features" for the RCS, Technical Specification Section 5.4.2 which specifies the RCS volume. The supplement information corrects the value for the RSG RCS volume and documents the analysis of impacts.

ComEd also conducted an additional review of a sampling of B&W calculations to ensure technical accuracy. No deficiencies were identified that impact calculation conclusions.

As part of the UFSAR update, a detailed table of RCS total and component volumes, hot and cold, for both units will be included. This table will provide clear design basis parameters for utilization in future applications. This action will be tracked by NTS item# 454-100-97-01306-02.

b. Inadequate Design Control - AFW Cooldown Requirements

An additional review of UFSAR non-chapter 15 calculations was performed by FTI. The additional review did not identify any deficiencies that impacted calculation conclusions.

DATE WHEN FULL COMPLIANCE WILL BE ACHIEVED

a. Inadequate Design Control - RCS Volume Calculation

Full compliance was achieved on 10/21/97 when the BWI calculation 222-7720-A13, Revision 1 was reviewed and accepted by ComEd.

b. Inadequate Design Control - AFW Cooldown Requirements

Full compliance was achieved on 11/21/97 when the affected calculation had been replaced with the new calculation, reviewed and accepted by ComEd and the results demonstrated that the previous calculation was conservative.