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August 21, 1987

Mr. A. Bert Davis
Regional Administrator
U.S. Nuclear Regulatory Commission
Region III
799 Roosevelt Road
Glen Ellyn, IL 60137

Subject: Byron Station Units 1 and 2
NRC Inspection Report No.
50-454/87-007 and 50-455/87-024
NRC Docket Nos. 50-454 and 50-455

Reference (a): July 9, 1987 letter from
J.J. Harrison to Cordell Reed

Dear Mr. Davis:

Reference (a) provided the results of an inspection at Byron Station and Nutech Engineers by Messrs. Liu and Gavula from February 9 through May 21, 1987. As a result of this inspection, certain activities were found in violation of NRC requirements. Attachment A of this letter contains Commonwealth Edison's response to the Notice of Violation enclosed with reference (a). On August 4, 1987, Commonwealth Edison was granted a two week extension on the due date for the response to the Notice of Violation.

Our review of the items discussed in the Notice of Violation has not concluded that each example is a violation of NRC requirements. In light of the information presented in Attachment A, we request the NRC to reconsider whether items 1a, 1b, 2a, and 2c are examples of violation of 10 CFR 50, Appendix B. Additionally, we believe the other examples are of minimal safety significance and would more appropriately be classified as Severity Level V.

Commonwealth Edison would appreciate the opportunity to meet with the NRC staff to discuss our response to the Notice of Violation. Please direct any questions regarding this matter to this office.

Very truly yours,

L. D. Butterfield
Nuclear Licensing Manager

Attachment

cc: Byron Resident Inspector

3464K

1a Continued

This issue was addressed in two distinct phases of the snubber reduction project. First, Nutech identified snubbers to be deleted or replaced on individual subsystems. They performed stress analyses and pipe support qualification calculations to assure that the revised configuration met ASME Code allowables. Pipe displacements at all locations were computed as part of this process. This phase culminated in the issuance of design packages, referred to as PECN's, to the piping contractor.

Upon completion of the piping modifications, field walkdowns (the second phase) were performed to address the seismic interaction concern. These walkdowns were performed by Nutech field engineers under the direction of Commonwealth Edison personnel. The walkdown program was premised on the fact that much of the piping operates well below allowable stresses and with very small displacements during seismic events. Inspections were focused on areas having either potential high stress (low margin to allowable stress limits) or displacements which were either large or significantly increased over the original displacement. A key source of information was the displacement data obtained from the analysis phase.

The procedure which controlled the scope of the walkdowns was COM-P1-BYR21. This procedure had instructions for identifying portions of piping to be inspected as well as methods to resolve potential interferences. The following portions of piping were covered:

1) Previously identified locations with radial clearance of 3" or less (referred to as "rattle points") where the new displacements exceeded both the clearance requirement and the original displacement computed in the initial Westinghouse pipe analysis.

2) Areas of potentially high stress including elbows, branch connections, equipment nozzles, and postulated break locations. This included enough adjacent piping to account for all directions of movement.

3) Locations at which newly computed displacements exceeded the original displacements by more than one inch. This was added to account for localized increases in displacements due to snubber removal, even though the overall subsystem motion was not significantly changed. Procedure COM-PI-BYR19 (mentioned in the violation) simply defined the methods for determining these locations.

4) Locations at which new total computed displacement (thermal + seismic) exceeded the limits of the original design criteria (3 inches). The three inch criteria was arbitrarily selected early in the plant design process based upon preliminary assessments of congestion in the auxiliary building and containments. Three inch separation was a goal for both the original design and installation. In the limited cases where 3 inch separation could not be achieved during snubber reduction design, the PECN'S identified these segments for preinstallation inspection which led to field change requests and subsequent walkdowns and resolutions. The piping installers also identified all other points where the three inch separation could not be achieved so those areas could also be analyzed.

1a Continued

The seismic interaction walkdown requirements described above exceeded those implemented during initial plant startup, as well as those established for Byron Station Unit 2. The walkdowns were augmented with requirements that addressed significant increases in displacements, or pre-identified tightly congested areas where displacements increased. This walkdown program, as presented, was judged satisfactory by the NRC during the inspection exit meeting.

Under these circumstances, Commonwealth Edison does not understand how the seismic intraction walkdowns at issue could be in violation of 10 CFR 50 Appendix B, Criterion III when they met or exceeded the piping walkdown requirements which were found to be appropriate at the time the plant was licensed.

VIOLATION 1b

10 CFR 50, Appendix B, Criterion V, as implemented by CECo Topical Report CE-1-A, "Quality Assurance Program for Nuclear Generating Stations," and CECo Corporate Quality Assurance Manual, Nuclear Generating Stations, "Quality Requirements," requires that activities affecting quality shall be prescribed by documented instructions of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions. Instructions shall include appropriate quantitative or qualitative acceptance criteria for determining that activities affecting quality have been satisfactorily accomplished.

Contrary to the above, during the course of reviewing safety related design efforts performed by Nutech Engineers, the NRC inspectors observed that activities affecting quality were not prescribed by documented instructions in that:

b. Instructions did not include seismic anchor movement (SAM) effects during seismic walkdowns.

Response

The seismic walkdowns at issue here were of a type for which the consideration of seismic anchor motion was not necessarily appropriate. Only for piping anchored to structures which are expected to flex significantly during an earthquakes is it appropriate to account for forces imposed by differential seismic anchor motion in the piping analysis. By contrast, in this case, the flexural effect from structures is generally not a significant factor in nuclear power plant piping systems.

Seismic walkdowns were performed to resolve instances where predicted pipe movement exceeded rattle point dimensions. Rattle points are defined as any location along a piping run where the gap between the pipe and any other object is less than 3". The procedure used to resolve rattle points during the piping walkdowns prior to plant startup did not require consideration of seismic anchor motion (SAM). It only considered seismic inertial displacement of the piping. This walkdown procedure was considered appropriate when the plant received an operating license.

Under these circumstances, Nutech correctly determined that this same procedure was appropriate for resolving rattle points. Although Nutech used different damping values in their piping analysis (from ASME Code Case N-411), NRC approval to use the revised damping values did not stipulate any changes to the method used to resolve rattlepoints.

In addition, even though Commonwealth Edison believes it was not necessary, Nutech performed a detailed review of SAM displacements for all subsystems within the scope of the snubber reduction project. This review showed that all piping except those connected to reactor coolant pumps and decoupled lines had a SAM displacement of less than 1/8 inch. For the few instances where the SAM displacements exceeded 1/8 inch, the potential impacts were satisfactorily resolved without modifications. This review confirmed that the original design basis was correct and that SAM had no adverse effect on the piping systems. This conclusion remains unchanged after applying snubber reduction to these piping systems.

1b Continued

Consistency with the original design and walkdowns was maintained during implementation of the snubber reduction project. The review discussed above has shown that SAM has no safety effect on the Byron piping systems. Based on the foregoing, we believe instructions do not need to include seismic anchor movement effects during seismic walkdowns. As a result, we request the NRC to reconsider whether this issue represents a violation of 10 CFR 50 Appendix B, Criterion V.

VIOLATION 1c

10 CFR 50, Appendix B, Criterion V, as implemented by CECo Topical Report CE-1-A, "Quality Assurance Program for Nuclear Generating Stations," and CECo Corporate Quality Assurance Manual, Nuclear Generating Stations, "Quality Requirements," requires that activities affecting quality shall be prescribed by documented instructions of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions. Instructions shall include appropriate quantitative or qualitative acceptance criteria for determining that activities affecting quality have been satisfactorily accomplished.

Contrary to the above, during the course of reviewing safety related design efforts performed by Nutech Engineers, the NRC inspectors observed that activities affecting quality were not prescribed by documented instructions in that:

c. Instructions were insufficient in the OPTPIPE User's Manual. Seismic zero period acceleration (ZPA) was listed on Pages 3.6, 4.2, 4.61, 4.65, and 4.66; however, no instructions were given as to when and how the ZPA was to be utilized in the computer application.

Corrective Action Taken and Results Achieved

Nutech conducted formal training sessions with project personnel on February 19, 1987. This training encompassed the use of OPTPIPE program features related to ZPA. Nutech also performed an evaluation of all subsystems within the scope of the snubber reduction program to specifically address ZPA effects. The results of this evaluation are discussed in the response to item 2a.

Corrective Action to Avoid Further Violation

The OPTPIPE User's Manual is being revised to clarify the use and limitations of some of the input parameters. In particular, the input and use of ZPA values have been addressed. Additionally, Nutech modified the OPTPIPE program so that ZPA effects are now calculated by the program and included in the response spectrum results for future analysis.

Date When Full Compliance was Achieved

The OPTPIPE User's Manual will be revised and appropriate OPTPIPE program changes will be verified by September 30, 1987.

VIOLATION 2a

10 CFR 50, Appendix B, Criterion III, as implemented by CECo Topical Report CE-1-A, "Quality Assurance Program for Nuclear Generating Stations," and CECo Corporate Quality Assurance Manual, Nuclear Generating Stations, "Quality Requirements," requires that measures shall be established to assure that regulatory requirements and the design basis for those structures and systems are correctly translated into specifications, drawings and instructions. These measures shall include provisions to assure that appropriate quality standards are specified and included in design documents and deviations from such standards are controlled.

Contrary to the above, during the course of reviewing safety-related design activities performed by Nutech Engineers, the NRC inspectors observed that the above criterion was not met in the following areas:

a. Zero period acceleration effects were not considered in the evaluation of safety-related piping systems. As a result, 106 supports and nine valves required requalification and one support required modification.

Response

There is no requirement to explicitly address zero period acceleration effects in the design of Byron's piping. Consideration of zero period acceleration (ZPA) is not specified in the Byron FSAR, the NRC Safety Evaluation Report or the NRC letter approving the use of ASME Code Case N-411 on Byron. The analysis performed for the snubber reduction project met the original licensing basis of the plant and the NRC requirements for implementing ASME Code Case N-411. Moreover, a study of all valves and remaining supports in the 67 subsystems of the snubber reduction project demonstrated that the omission of ZPA effects in the response spectra analysis for seismic loadings is not a safety concern. Therefore, we believe that regulatory requirements and the design basis were correctly translated into instructions for seismic analysis in the snubber reduction project. Accordingly, we request the NRC to reconsider whether this issue constitutes a violation of 10 CFR 50, Appendix B, Criterion III.

Background Information

The zero period acceleration (ZPA) refers to low participation, high frequency modes associated with high frequency accelerations which occur infrequently in a vibration spectrum. Inclusion of these modes ensures that the full mass of the piping system will be calculated in the system response. But neglect of the ZPA effects results in using only slightly less than the full mass of the piping system in calculating the system response. This "missing mass" or ZPA effect can cause the piping system response quantities such as accelerations and support reactions to be somewhat less than would be obtained by including the ZPA.

The methodology used to analyze the Byron piping systems involved response spectra for three independent directions of earthquake motion input and included modes up to at least 33 hertz. In addition, at least 30 modes were included in the analysis. This methodology is consistent with FSAR requirements for 33 hertz (Sections 3.7.3.1.1, 3.7.3.3, and 3.7.3.3.1)

although it does not always provide responses which include the entire ZPA effect. This is particularly true in regions of piping where the predominant frequencies of the piping are greater than 33 hertz and, therefore, the responses are underestimated. However, the resulting lower valve accelerations and support loads are not significantly different from those that would have been determined had ZPA been considered, and are adequately compensated for by other conservatisms in the seismic analysis.

The methodology used for the snubber reduction project treated ZPA in a manner similar to that used in the original Byron piping analysis. It is typical of most plants of the Byron vintage. However, recognizing the technical merit of including the ZPA effect, Nutech performed a study of all 67 subsystems analyzed during the snubber reduction project. This study involved a uniform static acceleration analysis in each of the three spatial directions. The acceleration amplitudes corresponded to the response spectra acceleration values at the modal cut-off frequency (33 Hz or greater). Responses from the three static acceleration cases were then combined to obtain a resultant response due to the ZPA analysis by the square-root-sum-of-the-square (SRSS) technique.

For valves, accelerations obtained from the response spectra analysis were combined with the ZPA values by SRSS. In all cases (approximately 210 valves), the combined accelerations were determined to be within acceptable limits. For supports, the reactions obtained from the ZPA analysis were compared to those from the response spectra analysis. The maximum reaction of these two cases were checked against the design load for the supports. Of the 1700 supports on these subsystems, only 113 supports were requalified and only 1 of the supports, a strut, required replacement with a larger size. Although this strut change was not needed to meet the design basis requirements, the change was made to provide extra margin and avoid a possible reactor restart delay.

Since ZPA effects were not specifically included in the piping analysis of record at the time the plant was licensed, and since consideration of ZPA effects is not a regulatory requirement, Commonwealth Edison does not understand why a violation of 10 CFR 50 Appendix B was issued for this item.

VIOLATION 2b

10 CFR 50, Appendix B, Criterion III, as implemented by CECo Topical Report CE-1-A, "Quality Assurance Program for Nuclear Generating Stations," and CECo Corporate Quality Assurance Manual, Nuclear Generating Stations, "Quality Requirements," requires that measures shall be establised to assure that regulatory requirements and the design basis for those structures and systems are correctly translated into specifications, drawings and instructions. These measures shall include provisions to assure that appropriate quality standards are specified and included in design documents and deviations from such standards are controlled.

Contrary to the above, during the course of reviewing safety-related design activities performed by Nutech Engineers, the NRC inspectors observed that the above criterion was not met in the following areas:

b. No justification was made for the 3 inch criterion used for seismic interaction walkdowns and rattle point evaluations prior to the start of the snubber reduction program.

Corrective Action Taken and Results Achieved

Nutech performed a study to compute a mean pipe displacement for a representative sample of piping subsystems within the scope of the snubber reduction project. The methodology used was identical to that used by Sargent & Lundy in their November 16, 1984 study. The results of the Nutech study demonstrated that the new mean pipe displacements, after snubber reduction, were significantly less than three inches and were only 15% higher than the original values. The study demonstrates that use of the 3 inch criterion is appropriate. This study is documented in Nutech Letter CEC-97-005, J.W. Cummings to T.R. Tramm of CECo., dated May 20, 1987.

Corrective Action to Avoid Further Violation

The Nutech study discussed above justifies use of the 3 inch criterion and is sufficient to avoid further violation.

Date When Full Compliance was Achieved

The Nutech study justifying use of the 3 inch criterion was completed May 20, 1987.

VIOLATION 2c

10 CFR 50, Appendix B, Criterion III, as implemented by CECo Topical Report CE-1-A, "Quality Assurance Program for Nuclear Generating Stations," and CECo Corporate Quality Assurance Manual, Nuclear Generating Stations, "Quality Requirements," requires that measures shall be establised to assure that regulatory requirements and the design basis for those structures and systems are correctly translated into specifications, drawings and instructions. These measures shall include provisions to assure that appropriate quality standards are specified and included in design documents and deviations from such standards are controlled.

Contrary to the above, during the course of reviewing safety-related design activities performed by Nutech Engineers, the NRC inspectors observed that the above criterion was not met in the following areas:

c. Other than engineering judgements, there were no technical bases for the 90% stress or insulation criteria used in the rattle point walkdowns.

Response

The 90% stress criteria and insulation criteria were used as a screening process during only the initial analysis and design phase of the project. These criteria did not substitute for any walkdowns. These criteria simply allowed Nutech to identify the rattle points which would not likely exceed design criteria after snubbers were removed from the lines containing those rattle points. But in all cases, after the snubbers were removed, each rattle point was inspected for potential interference. Therefore, contrary to the violation, engineering judgment was not the only technical basis for the rattle point walkdowns. In fact, the 90% criteria had nothing to do with the walkdowns. Therefore, since these criteria were not used in resolving or evaluating potential impacts, and more rigorous analysis techniques were implemented for final resolution during walkdowns, this item should not be considered a violation. Accordingly, we request the NRC to reconsider whether this issue is a violation of 10 CFR 50, Appendix B, Criteria III.

During the inspection, there was confusion regarding the use of the screening criteria described above and the criteria used to define the scope of the walkdowns. The screening criteria allowed for resolution of rattle points as long as the pipe stresses were less than 90%. That is, though the pipe deflection may have exceeded the rattle point dimension, the fact that the pipe stresses were less than 90% of code allowables enabled a PECN to be issued. The possibility of a major modification during the walkdown was considered improbable. The walkdown criteria defined boundaries of seismic interaction walkdowns to include potentially high stress locations, to include all piping where stresses exceed 90%. However, final resolution of all potential impacts within 3 inches of the piping (e.g., rattle point) was accomplished through rigorous analysis or modification. In no instance was 90% stress used to resolve these impacts.

Background Information

The 90% stress criteria and insulation criteria are based on engineering judgement, rather than rigorous analysis or study which can be applied to Byron Station Unit 1. Had these criteria been the basis for resolving potential interferences identified during the walkdowns, impacts between safety related subsystems and components would not have been evaluated to a level of detail commensurate with the original analysis.

However, the 90% criteria were used only for the initial design phase of the snubber reduction project. This phase used as-built isometrics to ensure that the piping models accurately depicted the current piping and support configuration. Included in these isometrics were rattle points (radial clearance of 3 inches or less) which identified the presence of piping, component(s), or structure(s) within three inches of the subsystem. The actual object and its precise location was not known, so any technique used to account for this information would necessarily assume any deflection to be in the direction of the specific object.

Westinghouse had used a rattle point procedure in the original analysis, and Nutech adopted this same procedure to address these rattle points. Recognizing that plant walkdowns for seismic interaction and thermal monitoring were planned during subsequent phases of the project, Nutech used this procedure as a screening process to decrease the likelihood or extent of modifications resulting from potential interferences identified during post installation testing/walkdowns. Since all potential interferences would be resolved considering precise field survey data and substantiated rigorous analysis techniques, the fact that a screening criteria based on engineering judgement would be used was not contrary to Appendix B, Criterion III. That criterion was satified during the next phase by the walkdowns and the subsequent resolutions of rattle point issues as described in the response to item 1a. The methods used to resolve all potential interferences are contained in procedure COM-PI-BYR21, and there are no instances where either the 90% or insulation criteria described in COM-PI-BYR15 are used to resolve interferences. Therefore, Commonwealth Edison believes there is no basis for a violation.

VIOLATION 2d

10 CFR 50, Appendix B, Criterion III, as implemented by CECo Topical Report CE-1-A, "Quality Assurance Program for Nuclear Generating Stations," and CECo Corporate Quality Assurance Manual, Nuclear Generating Stations, "Quality Requirements," requires that measures shall be establised to assure that regulatory requirements and the design basis for those structures and systems are correctly translated into specifications, drawings and instructions. These measures shall include provisions to assure that appropriate quality standards are specified and included in design documents and deviations from such standards are controlled.

Contrary to the above, during the course of reviewing safety-related design activities performed by Nutech Engineers, the NRC inspectors observed that the above criterion was not met in the following areas:

d. Design verification was not adequately performed. This resulted in an analytical error in an impact analysis, inappropriate use of seismic response spectrum, and omission of node points for rattle point evaluations.

Corrective Action Taken and Results Achieved

Impact Analysis

The cause of this error was a misinterpretation of the procedure for performing impact calculations documented in Nutech Project Instruction COM-PI-BYR-21, Revision 3. This project instruction was revised to clarify the impact calculation method. Training on the revised procedure was held for Nutech personnel involved in resolution of seismic interation effects. Other impact calculations were reviewed to assure they did not contain the same error. The impact calculation found to be in error was repeated and the error was corrected. The design adequacy of the subsystem involved was not adversely affected by this error.

Seismic Response Spectrum

The seismic response spectrum curve used in the instance cited is appropriate and meets the Byron FSAR commitments. This issue arose when the NRC inspector noted that one of the supports for subsystem 1FW05 attaches to the building structures at elevation 424 feet. The envelope response spectra used in the analysis of 1FW05 were developed from spectra at elevations of 412 feet and less. The inspector was concerned that the spectra at the higher support elevation may exceed the envelope spectra used.

Nutech used response spectra at the same elevations as used by Westinghouse in the original analysis. The Westinghouse work was checked at the time it was performed and was used as design input for Nutech's work. In the instance cited, Nutech reviewed the envelope spectra used and found that they bounded the spectra for the support at elevation 424 feet. Nutech also reviewed the spectra used in the analyses of the four other subsystems examined in this inspection, including 1CV50, 1RY05, 1CC34, and 1SI10. In all instances, the spectra used bounded at the highest support elevation.

2d Continued

The review performed for these five subsystems encompassed a total of 71 supports. Section 3.7.3.8.4 of the Byron FSAR states that the most severe floor resonse spectrum corresponding to support locations is used in the evaluation of piping systems supported at different elevations. Nutech contacted Westinghouse and confirmed that this approach was implemented in the original analysis. The sample subsystems that were checked indicated that the analysis method used met the FSAR commitment.

Missing Node Points

On one page of the piping calculations for subsystem 1CV50, displacements at rattle point locations were tabulated, but the corresponding node numbers for two locations were omitted. For this page of the calculation, the check copy of the calculation package was reviewed. The check copy contained the missing node numbers and provides evidence that the displacements were checked. Due to the large number of corrections required as a result of the checking, the displacements tabulated on this page were copied over onto a new calculation sheet by the preparer and the two node numbers were omitted in the copying. This has been corrected on the original calculations (i.e., the missing node numbers have been added).

Corrective Action to Avoid Further Violation

We believe these items are isolated cases and the actions discussed above are sufficient to avoid further violation.

Date When Full Compliance was Achieved

These items were corrected prior to the end of the inspection on May 21, 1987.

ATTACHMENT A

VIOLATION 1a

10 CFR 50, Appendix B, Criterion V, as implemented by CECo Topical Report CE-1-A, "Quality Assurance Program for Nuclear Generating Stations," and CECo Corporate Quality Assurance Manual, Nuclear Generating Stations, "Quality Requirements," requires that activities affecting quality shall be prescribed by documented instructions of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions. Instructions shall include appropriate quantitative or qualitative acceptance criteria for determining that activities affecting quality have been satisfactorily accomplished.

Contrary to the above, during the course of reviewing safety related design efforts performed by Nutech Engineers, the NRC inspectors observed that activities affecting quality were not prescribed by documented instructions in that:

a. Instructions did not address seismic effects for all portions of the piping subsystems. Paragraph 4.1.1 of Instruction BYR 19, Revision 2, states that seismic interaction walkdown shall be specified for portions of subsystems where seismic displacements exceed the original design seismic displacements by more than one inch.

Response

Seismic effects were addressed for all portions of the piping subsystems. Computerized piping analyses were performed and these were used to identify portions of piping requiring physical inspection for seismic interactions. Where necessary, appropriate actions were taken to resolve potential interferences resulting from a seismic event. Walkdowns were conducted to examine clearances on selected portions of the piping including portions where seismic displacements increased by more than one inch.

The initial walkdown criteria developed by Nutech for this project was more extensive than the criteria used in the original design and installation of Byron Unit 1 piping systems and met the requirements of the previously approved Byron Unit 2 walkdown program. Though revisions to the governing walkdown procedure, COM-PI-BYR21, were made throughout the project, in no case were the requirements less than those previously committed to at Byron Station. All subsystems associated with this project were walked down per the final revision to COM-PI-BYR21.

Based on the foregoing, we believe activities affecting quality were properly prescribed by documented instructions, and that regulatory and design basis requirements were met. We request the NRC to reconsider whether this item is an example of violation of 10 CFR 50, Appendix B.

Background Information

Stress analyses of individual piping subsystems assure integrity of pipes and their supports on a stand-alone basis. Because this piping is located close to structures, components, and other piping, evaluations must be made to assure that potential pipe impacts with other objects will not jeopardize plant safety. This is a fundamental systems interaction issue which was resolved during initial plant construction. It had to be revisited during the Byron 1 snubber reduction program because supports were being removed, creating the potential for increased piping movements during seismic events.