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Nuclear Power Plant
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NewYorkPower Authority

July 16, 1984
IP-WDH-2852

Docket No. 50-286
License No. DPR-64

Mr. Thomas T. Martin, Director
Division of Engineering and Technical Programs
U. S. Nuclear Regulatory Commission
Region I
631 Park Avenue
King of Prussia, Pennsylvania 19406

Subject: Inspection No. 50-286/84-04

Dear Mr. Martin:

This letter provides the Authority's response to your Inspection Report No. 50-286/84-04 dated June 15, 1984 and received at this office on June 21, 1984.

Attachment I to this letter addresses the specific concerns cited in Appendix A ("Notice of Violation") and Attachment II addresses the more general concerns cited in the cover letter of your June 15, 1984 letter.

Very truly yours,

John C. Brons

John C. Brons
Resident Manager
Indian Point Unit 3
Nuclear Power Plant

Attachment

cc: IP-3 Resident Inspector's Office

Violation

Criterion V of Appendix B to 10CFR50, Instructions, Procedures, and Drawings requires that "activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures or drawings. Instructions, procedures, or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished." The Authority's quality assurance (QA) program requires that the Authority and/or delegated organizations shall have established measures for the control and implementation of instructions, procedures, and drawings for quality-related activities applicable to the scope of their responsibilities.

Contrary to the above, as stated in Appendix A of Inspection Report No. 50-286/84-04, the architect-engineer had no documented instructions, procedures, or drawings addressing the seismic analysis performed in their response to NRC IE Bulletin No. 79-07 which was delegated to the architect-engineer by the Authority. The uncontrolled data used in this effort was the same as that used in 1972.

Response to Inspection Report No. 50-286/84-04 Notice of Violation

The Authority does not agree with the Notice of Violation as stated in Appendix A to Inspection Report No. 50-286/84-04.

As a result of the NRC's issuance of IE Bulletin No. 79-07 on April 14, 1979, and other related IE Bulletins, the Authority retained UE&C to seismically reanalyze the safety related piping systems at IP-3 that were impacted by the specific concerns cited in these bulletins. This entire effort was conducted in accordance with both the Authority's and UE&C's QA programs. As such, UE&C developed Project Procedure No. 9 ("Pipe Stress Analysis") which provided, among other things, the instructions for performing and controlling the IP-3 piping reanalysis required by IE Bulletin No. 79-07. This procedure was approved by UE&C in accordance with their approved QA program on August 21, 1979. Amendment No. 5 to QA Topical Report No. UEC-TR-001, as approved by the NRC on March 21, 1978, provides a description of the UE&C QA program in effect at the time the 1979 safety related piping reanalysis effort was conducted.

The 1979 safety related piping reanalysis effort therefore was not "uncontrolled," as indicated in Inspection Report No. 50-286/84-04. Moreover, the Authority's QA Department conducted audits to assure that Project Procedure No. 9 was followed. (A copy of this procedure as well as the QA audit reports are available for review by members of your staff).

In view of the documented procedure governing the 1979 safety related piping reanalysis effort, as required by Criterion V of Appendix B to 10CFR50, the Authority respectfully requests that the Notice of Violation be rescinded.

Discussion of General Concerns

This attachment addresses the adequacy of the modeling and analysis of the IP-3 safety related piping systems as requested in the cover letter to Inspection Report No. 50-286/84-04.

The Authority submitted the analytic methodology to be utilized in the reanalysis effort required by IE Bulletin No. 79-07 for NRC review and approval in a letter dated May 24, 1979 (IPN-79-27). The Authority received NRC approval of the proposed methodology in a letter from A. Schwencer (then Operating Reactors Branch No. 1, Chief) to G. T. Berry (then Executive Director of the Power Authority of the State of New York) dated June 22, 1979. At no time during this period were there any expressed concerns regarding the adequacy of the original 1972 modeling and analysis techniques, which were to be modified to the extent necessary to address the specific concerns regarding inter- and intra-modal seismic response combinations as required by IE Bulletin No. 79-07, and to address the concerns of other related IE Bulletins. In fact, as indicated in the NRC's March 24, 1980 interim safety evaluation report on the IE Bulletin No. 79-07 IP-3 piping reanalysis, the Authority provided the NRC with two problems for confirmatory analysis which retained the original piping models of 1972. These confirmatory analyses were independently conducted by Brookhaven National Laboratory (BNL), a contractor to the NRC at that time.

Conversations with NRC staff members have indicated that there are basically two open areas of concern; namely, the determination of mass point spacing and cut-off frequency. While the modeling and analysis techniques of 1979 had evolved to a more sophisticated level than those of 1972, the Authority contends that what was done in 1972 was nevertheless adequate, especially in light of the inherent conservatisms associated with the original seismic piping analyses.

Further, for reasons specified in Attachment I to this letter, the 1979 reanalysis effort was not "uncontrolled." Certain data utilized during the 1979 reanalysis effort was originally developed in 1972 as input to the original piping analyses that confirmed the seismic capabilities of safety related piping systems as part of the IP-3 licensing process. While the QA standards in effect in 1979 had evolved to a more sophisticated level than those in effect in 1972, it is maintained that the 1979 reanalysis effort adequately addressed the concerns as stated in IE Bulletin No. 79-07.

To address the concerns of members of your staff, the Authority, in conjunction with UE&C, remodeled the piping systems of two problems that members of your staff had audited during the special inspection on February 27 - March 2, 1984 at the IP-3 site (specifically, Problem Nos. 449 & 451) accounting for more up-to-date mass lumping techniques and allowing for higher frequency cut-off. Although the results of the new computer runs incorporating these two particular modeling and analysis aspects indicated piping load increases in some cases, in no case did the piping stresses or support/nozzle loads exceed allowable values. In fact, in many cases, decreases in the piping stresses and support/nozzle loads were evidenced.

As discussed in detail with members of your staff, it is essential to realize that the safety related piping systems at IP-3 (other than the reactor coolant loop, main steam and main feedwater piping inside containment which were analyzed by Westinghouse) are based on an initial static design and analysis, the validity of which was not in question by IE Bulletin No. 79-07. Seismic Class I piping six inches in diameter and larger together with the two inch diameter high head safety injection lines were further subjected to a more rigorous dynamic analysis for seismic response equivalent to the design basis earthquake (DBE) to confirm the initial static design. Thus, the 1979 reanalysis effort should be perceived as a reconfirmation of the original static design.

There are a number of additional points to take into account in this respect. The first is the basis for the analytic methodology utilized in the 1979 reanalysis program as detailed in the Authority's May 24, 1979 (IPN-79-27) letter. This methodology was agreed to by members of the NRC staff and the Authority at a May 22-23, 1979 meeting. This methodology, when approved, became the basis for the 1979 reanalysis effort and required worst case values for the piping seismic stresses, support loads, and component nozzle loads, as calculated using the two-dimensional response spectrum analysis approach, to be increased by a factor of 38% (i.e., to be multiplied by a factor of 1.38). The reason for the use of this multiplicative factor, which was not required by IE Bulletin No. 79-07, was a result of certain unclarified information presented in the answer to Question 5.22 of the original IP-3 FSAR. The answer to this Question regarding the seismic design of piping states that the "results of analyses for each of two orthogonal, horizontal directions of excitation were combined directly with the results for vertical excitation on the basis of absolute sums."

Representatives of the NRC staff at the May 22-23, 1979 meeting interpreted this to mean that absolute sums was the licensing basis for all dynamically analyzed seismic Class I piping designs and hence could not be deviated from. Thus, the factor of 1.38 was utilized to account for the difference in the proposed piping reanalysis methodology which was to utilize the square-root-sum-of-the squares (SRSS) approach versus that of an analysis based on absolute sums.

Available information based on the Authority's recent review of the 1.38 factor's evolution indicates that the response to Question 5.22 of the original FSAR was only meant to apply to the intra-modal response combinations for Westinghouse-analyzed piping systems (namely, the reactor coolant loop, main steam, and main feedwater piping inside containment). Thus, it is believed that piping seismic stresses, support loads, and component nozzle loads predicted by the 1979 reanalysis effort can justifiably be reduced by 38%. This reduction drastically reduces the effect of the mass point spacing technique and cut-off frequency utilized in the 1979 reanalysis, and is consistent with the response combination methods permitted by IE Bulletin No. 79-07.

The methodology utilized in the 1979 reanalysis effort as approved by the NRC also permitted reanalysis based on an equivalent analytical approach which would have included all three earthquake components and the SRSS method for both the inter- and intra-modal response combinations if a result calculated using the two-dimensional response spectrum analysis approach exceeded the applicable allowable limit. This three-dimensional reanalysis was not required to utilize the 1.38 factor. While this three-dimensional analysis approach was not utilized during the 1979 reanalysis effort, it is expected its use would have further reduced the seismic loads predicted by the reanalysis and hence also the effect of the mass point spacing technique and cut-off frequency.

To provide a basis for this contention, the Authority has independently run Problem 451 using the NRC-approved NUPIPE computer code based on a three-dimensional Regulatory Guide 1.92 (RG 1.92) analysis. The results of the above analysis were then compared to:

(1) The modified two dimensional analysis conducted recently and transmitted to NRC staff members for Problem 451. This NUPIPE analysis retained the 1.38 multiplicative factor and accounted for more up-to-date mass point spacing techniques in addition to allowing for higher frequency cut-off.

(2) The original 1979 two-dimensional analysis approach which, as indicated earlier, utilized the 1.38 factor but retained the mass point spacing and frequency cut-off originally developed in 1972.

The comparison of these results indicates that the pipe stresses predicted by the three-dimensional RG 1.92 analysis were in all cases either equivalent to or lower than those predicted by the analyses described in (1) and (2) above.

In addition, the comparison of these results indicates that the support load(s)/capacity ratios predicted by the three-dimensional RG 1.92 analysis were either equivalent to or lower than those predicted by the analysis described in (1) above, with the exception of one support. (For this support, the difference predicted by the two approaches was less than 1% of the allowable limit; both ratios predicted, however, were well below the allowable limit). The comparison further indicates that the support load(s)/capacity ratios predicted by the three-dimensional RG 1.92 analysis were either equivalent to or less than those predicted by the analysis described in (2) above, with the exception of four supports. (For these four supports, the increases observed were all less than 6% of the applicable allowable limit and in no case were above these limits).

With respect to Item 4.1 in the "Details" section of Inspection Report No. 50-286/84-04, please be advised that it is our understanding that this item has been satisfactorily addressed since the requested information was provided to the inspectors.

In view of the discussions presented above and the inherent conservatisms associated with the original seismic piping analyses, the Authority does not agree with the statements appearing in Item 4.2 in the "Details" section of Inspection Report No. 50-286/84-04 and maintains that the IP-3 safety related piping systems have all been adequately modeled and analyzed.