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April 10, 1980  
IPN-80-38

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Director of Nuclear Reactor Regulation  
U. S. Nuclear Regulatory Commission  
Washington, D. C. 20555

Attention: Mr. Albert Schwencer, Chief  
Operating Reactors Branch No. 1  
Division of Operating Reactors

Subject: Indian Point 3 Nuclear Power Plant  
Docket No. 50-286  
I. E. Bulletin 79-07, Final Report

Dear Sir:

This letter provides the final report of work performed under I. E. Bulletin 79-07 in accordance with our letter IPN-80-30, dated March 13, 1980.

The safety related systems (or portions thereof), as identified in Table I of our letter of April 24, 1979, were re-evaluated. All safety class and category I piping systems that were originally analyzed for seismic loadings using computer codes as identified in our letter of April 24, 1979, were reanalyzed in accordance with the procedure agreed upon during the NRC-Power Authority meeting of May 22-23, 1979 as delineated in Attachment 1 to our letter of May 24, 1979. The "As Built" verification of the piping systems was performed in accordance with the "Field Check Plan" submitted with our letter of May 31, 1979. The results of the "As Built" verification were incorporated in the re-analysis of the piping lines as well as the re-evaluation of the pipe supports, equipment nozzles and containment piping penetrations.

A total of 181 lines were re-analyzed and the results of the re-analysis show that for all of the 181 lines the total stresses, for both upset and emergency plant operating conditions, are within their respective applicable allowable limits.

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Pipe supports, hangers, snubbers and pipe whip restraint components including the base plate and anchor bolts were re-evaluated for the new applied piping loads for both upset and emergency conditions. Of the 1059 pipe supports associated with the re-analyzed lines, 821 were found to be capable of performing their safety function within their respective applicable allowable limits. The remaining 238 supports were determined to require modification. Of the 238 supports requiring modification, 83 are located inside containment and 155 outside containment. The 83 supports inside containment have been modified. Of the 155 supports outside containment, 140 have been modified. The 140 modified supports include all those supports whose evaluation showed that the structural integrity of the support components could be affected when the new piping loads were applied. The remaining 15 supports outside containment all of which have safety factors greater than 2 but less than 4 are scheduled to be modified by June 29, 1980.

Additionally, 112 equipment nozzles (53 inside and 59 outside containment) and 32 containment piping penetrations were re-evaluated. The results of the evaluation confirmed that for all of the above 112 equipment nozzles and 32 containment piping penetrations the new applied piping loads, both for upset and emergency plant operating conditions, are within their respective applicable allowable limits.

The Authority has also investigated the concerns of the I.E. Information Notice 79-36, "Computer Code Defect in Stress Analysis of Piping Elbow", and has been informed by its Architect Engineer and NSSS supplier that the NUPIPE code was not used. WESTDYN, the ADLPIPE computer code version used by Westinghouse before WESTDYN development, and the general Westinghouse finite element computer code WECAN were used. However, Westinghouse concludes that the NUPIPE code defect as described in I.E. 79-26 does not exist in these three codes.

The Authority concludes that the results of the above re-evaluation and modifications to date provide adequate justification and assurance that the plant may continue to operate without undue risk to the public health and safety.

Very truly yours,

*George M. Wilverding*

*for* Paul J. Early  
Vice President and  
Assistant Chief Engineer-Projects

cc: Attached

cc: Mr. T. Rebelowski  
Resident Inspector  
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