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REPORT TO THE AEC REGULATORY STAFF

STRUCTURAL ADEQUACY

OF

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

Consolidated Edison Company of New York, Inc.

Docket No. 50-247

by

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and
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Urbana, Illinois

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INTRODUCTION

This report is concerned with the structural adequacy of the containment structures, piping, equipment and other critical components for the Indian Point Nuclear Generating Unit No. 2 for which application for a construction permit and an operating license has been made to the United States Atomic Energy Commission by the Consolidated Edison Company of New York, Inc. The facility is located on the east bank of the Hudson River at Indian Point, village of Buchanan, in upper Westchester County, New York. The site is about 24 miles N of the New York City boundary and 2.5 miles SW of Peeksill, New York.

This report is based on a review of the Final Facility Description and Safety Analysis Report (Ref. 1) and the containment design report (Ref. 2). The report also is based in part on the discussion and inspection resulting from the visit to the site on 2 May 1969 by N. M. Newmark and W. J. Hall in conjunction with Mr. K. Kniel and Mr. M. McCoy of AEC-DRL. A number of topics were discussed with the applicant and his consultants at the time of this visit, and subsequently additional information has become available through supplements to the FSAR and through discussions with the personnel of DRS, DRL, and the applicant and his consultants. A discussion of the adequacy of the structural criteria presented in the Preliminary Safety Analysis Report is contained in our report of August 1966 (Ref. 3), and unless otherwise noted no comment will be made in this report concerning points covered there.

The design criteria for the containment system and Class I components for this plant called for a design to withstand a Design Basis Earthquake of 0.15g maximum horizontal ground acceleration coupled with other appropriate loadings to provide for containment and safe shut down. The plant was also to be designed for an Operating Basis Earthquake of 0.1g maximum horizontal ground acceleration simultaneously with the other appropriate loads forming the basis of containment design.

COMMENTS ON ADEQUACY OF DESIGN

Dynamic Analyses

(a) Containment Building. The answer to Question 1.9 of the FSAR indicates that only the containment building, the primary auxiliary building, and the electric cable tunnel were designed with the use of semi-formal dynamic analyses. A description of the method of analysis employed is given briefly in Section 5.1.3.8 of the FSAR and in Section 3.1.5 of the containment design report. The procedure employed involved a calculation of the fundamental frequency and mode shape by use of a modified Rayleigh method. The base shear for the structure was computed from the period and the spectral response corresponding to the appropriate degree of damping. The base shear was then applied as a loading to the structure as an inverted triangular loading. The shears at the nodes were used to calculate the moments and displacements at various points in the structure. For the structures involved it is believed that the approach leads to a design which is reasonably adequate.

A similar approach was followed for the primary auxiliary building as described in the answer to Question 1.9. It is noted there that a one-third increase over working stress was allowed in the design of the bracing in the case of the Design Basis Earthquake. This stress is below yield, and it is believed that the design will prove to be satisfactory.

(b) Other Buildings and Equipment. The discussion presented in answer to Question 1.9 of the FSAR for other buildings and equipment such as the control building, fan house, intake structure, etc., indicate that a refined static approach was used, which involves employing the peak value from the appropriate response spectrum curve for a given value of damping and multiplying this by the appropriate mass to obtain the inertial loading. From the description given for the various buildings and items of equipment, and the modeling techniques employed, it is concluded that the inertial loadings used in design are reasonably close to those that might be obtained with a more sophisticated analysis and lead to reasonable design values.

The submission in Question 1.3 of Supplement 13 indicates that the Turbine Building, and Fuel Storage Building Structure above the Fuel Storage Pit were reanalyzed by a multi-degree-of-freedom modal dynamic analysis method to check their adequacy. As a result of this reanalysis, the applicant advises that certain structural modifications will be made to columns and cross bracing in the Turbine Building to insure that it can withstand the DBE. The superstructure of the fuel storage building was ascertained to be adequately designed, without modification to withstand the effects of the DBE. The applicant states that reanalysis of the strengthened turbine building and superheater building for Indian Point No. 1 does not significantly affect the responses calculated for the original structures.

(c) Piping Analysis. The method used by the applicant for analysis of the piping, as described in the answer to Question 1.6 of the FSAR, is the same as was used in Ginna. The peak ground response spectrum value for 0.5 percent damping was used, applied as static accelerations in each direction separately, and the resulting stresses superposed. It was assumed by the applicant that the piping was supported along rigid systems and therefore not subjected to amplified ground motion at points of support. The system was analyzed with the anchors and supports as actually used, according to the discussion presented to us during the time of our visit in May 1969. It was the view of the applicant that the thermal motions were greater than any differential ground displacements and the latter therefore are not critical items in the design. In answer to Question 1.13 (Suppl. 13) the applicant advises that relative seismic displacement was considered for the main steam lines, where the largest relative displacements are expected; stress differentials of less than 10% resulted. Also, seismic supports installed to date are those specified in the design and employed in the analyses; where deviations in supports must occur, reanalysis will be carried out. These results and approaches appear satisfactory to us.

Since this plant was designed before recent developments and changes in piping design specifications, the 1968 ASME Addenda were not applied. Blow-down and earthquake were considered as separate items and not combined in this design. We are advised that the response to Question 1.9 of Supplement 12 states that a review of the Indian Point 3 reactor coolant system which is identical to Indian Point 2, for combined earthquake and blow-down indicates that the design is adequate.

It is stated in the answer to Question 1.6 of the FSAR that the approach resulted in a seismic design load approximately equal to 0.60W horizontally and 0.40W vertically taken simultaneously. It is further stated that for the Design Basis Earthquake the sum of the resulting additional stress plus the normal stresses was limited to 1.2 times the B31.1 code allowable stresses. In a similar manner the stresses in the pipe supports and hangers were limited to 1.2 times code allowable stresses.

The applicant originally made use of the maximum spectrum value only and no modal analyses were made; in other words only a static analysis with uniform accelerations was made. Consideration was not given to modified distribution of the inertial loading to take account of the combination of modal effects.

The response to Question 1.9 of Supplement 8, describing more detailed analyses of the reactor coolant system, feedwater lines, surge lines and typical steam lines by more formal methods as carried out later lends confirmation to the adequacy of the design. On this basis, there is reason to believe that the design is adequate.

Backfill Surrounding Containment Vessel

Nine feet of crushed rock backfill was placed between the external wall of the reinforced concrete containment vessel and the retaining wall holding back the rock on the uphill side. This crushed rock backfill is drained at the bottom to avoid water pressure against the containment structure. The fill is approximately 60 to 70 feet higher on one side of the structure than on the other because of the slope of the rock surface. The design, as discussed in Section 3.1.5 of the containment design report, considered local inertial forces of loose rock as an added loading against the containment pressure vessel, and also considered passive pressures caused by failure of the rock along the surface behind the retaining wall. The localized loadings from these forces were considered in the design of the containment structure and the discussion presented in the containment design report provides reasonable assurance that the containment vessel is capable of resisting these localized forces.

Class I Equipment in Structures other than Class I

The turbine building is Class III and not designed for earthquake loadings. The answer to Question 1.3 of the FSAR indicates that the only Class I structures and components which are so located that they could be endangered by failure of Class III structures are the control building, main steam piping and feedwater piping, all of which could possibly be endangered by the Class III turbine building. It is further indicated there that no special provisions have been provided for protection except in the case of the main steam and feedwater lines up to the isolation valves, which are protected by the shield wall and the structural frame at the north end of the shield wall. Since these are located near the braced end of the turbine building, it is not anticipated by the applicant that there will be any structural failure in this area. Our judgment as to the adequacy of this aspect of the design is based on the statement given in the application. And, in this respect, the answer to Question 1.3 (Supplement 13) which describes the analysis and strengthening of the Turbine Building and Superheater Building for Indian Point Unit No. 1, and their ability to withstand the DBE, should give additional protection for the control room.

It is further stated that the only Class III crane whose failure could endanger any Class I function is the fuel storage building crane and that the failure of this crane will not impair a safe and orderly shutdown. The answer to Question 1.3 (Suppl. 13) indicates that the only potential for crane lift off will be in the unloaded condition with the trolley parked near the support; the applicant advises that the unloaded crane will not be parked over the pool, so no hazard exists. It is also noted in the answer to Question 1.1.3 that the manipulator crane in the containment building, a Class III crane, is restrained from overturning and will not endanger Class I structures.

Deformation Criteria

The general stress criteria applicable to the seismic design are summarized in Appendix A of the FSAR. The statement given on page A3 of Appendix A states that for all components, systems and structures classified as Class I, the primary steady state stresses, when combined with seismic stresses resulting from the response to the Design Basis Earthquake, are limited so that the function of the component system or structure shall not be impaired so as to prevent a safe and orderly shut-down of the plant.

We were advised at the time of our inspection of the plant in May 1969 that, for normal loadings plus the Operating Basis Earthquake, the intention was to use code allowables plus the 20 percent increase for transient conditions on Class I components and systems. For the Design Basis Earthquake and blow-down, basically the same criteria were used, although originally it had been planned to adopt higher allowables going into the plastic range using the code for faulted conditions. In actuality, as described in the answer to Question 1.7 of the FSAR, the allowable stresses in the case of the Design Basis Earthquake were limited to the yield point, or slightly below (see answer to Question 1.3 of Supplement 13).

The only references that we note where there was a calculation of stresses exceeding the yield point were at several places in the containment design report where it was mentioned that the calculations indicate that there could be possible local yielding of the liner under certain loading combinations, but that this would be limited and not be expected to be of a nature as to cause concern with regard to the integrity of the liner.

Reactor Internals

The mechanical design and evaluation of the reactor core and internals is described generally in Section 3.2.3 of the FSAR. From the discussion given it appears that the core support structure and core barrel have been designed with proper attention to support points and limitations of motions. The design criteria for the internals themselves, and specifically with reference to deflections under abnormal operation, are given in Table A.3-2 of the FSAR. These appear reasonable and should provide an adequate margin of safety.

Large Penetrations

A finite element analysis of the large penetrations in the containment vessel was made by the Franklin Institute and a description of the analysis and the results obtained is presented in the containment design report. Several analyses were made for different load combinations, and in addition a number of hand calculations were made to check the order of magnitude of the expected forces and stresses and to verify that the results were reasonable Our review of the material presented, to the extent possible, indicates that the penetration design is adequate.

Splices in Large Reinforcing of Bars

Cadweld splices were used in general in the construction of the containment vessel. We were advised that the early splices, about 10 percent of the total, were made with a bronze base, and the remaining 90 percent were made with ferritic base filler metal. Around the hatch opening, we observed that there was approximately a three foot stagger of adjacent splices, but in questioning we learned that there may not be such a stagger over other areas of the containment vessel. Lack of stagger of adjacent splices could lead to planes of weakness and cause cracking under conditions of over-loading. The pressure tests, however, will reveal any such cracking.

Approximately one in 200 splices was removed for test purposes. This is generally adequate.

Instrumentation and Controls

At the time of the May 1969 visit it was ascertained that the applicant considers the control room as a Class I structure and intends that the housing of it will also be subject to Class I requirements. However, the instrumentation for the control room as well as other instrumentation critical to containment and safe shutdown, has been purchased from the vendors according to applicant's specifications. The answer to Question 1.9 describes the vibration tests employed for selected items of essential equipment; the purpose of these tests is to help demonstrate that little or no difficulty will be expected in the operating characteristics thereof under seismic conditions. Although not absolute proof of acceptability, satisfactory test results certainly help to confirm the adequacy of such instrumentation and control items. Further information on the design and procurement approach for protection system equipment is given in the answer to Question 7.27 (Suppl. 13), and lends confirmation to the approach adopted.

Tornado Loadings

The information contained in Section 3.4 of the containment design report, and the answer to Question 5.7 of the FSAR indicates that the structure is designed for the usual wind loadings. The analyses described in Appendix B of Supplement 6, indicate that the containment building can resist the design tornado. What effect if any that a tornado could have on the control room or other critical facilities is not stated. However, the applicant states that the siding of the control room can resist wind velocities up to 162 mph, and the girts (supporting the panels) will fail at 0.62 psi negative pressure; the building is protected by other buildings on the south and west.

Steel Liner and Containment Vessel

The analyses that have been carried out with regard to the liner are summarized in the FSAR and some additional information is presented in the containment design report. It is our understanding that where bulges of the liners occurred during construction, of less than 2 in., nothing was done to correct the bulges. However, when bulges were 2 in. or greater the liner was pushed back into a position of not more than 2 in. away from its intended position, and additional studs were used to anchor the liner in place. Temporary bracing was employed to hold it in position until the concrete was cast. Because of the foregoing, and since the temperature rise in the lower part of the structure in the liner is reduced by the use of insulating material, it is not expected that the departures from the intended original surface will lead to any difficulties.

Proof Test Procedures and Instrumentation

It is our understanding that a detailed description of the proof test procedures is to be submitted at a later date. At the time of our visit in May 1969 it was proposed by the applicant that strain readings be taken only on the liner around the penetrations. We suggested that additional readings be made which would include diameter changes of the penetrations and other measurements that can be made conveniently and without excessive expense to provide evidence that the design meets the design criteria. Fig. 5.13-4 suggests that such readings will be made. In any event, an interpretative report on the measurements that are taken should be provided and should be correlated with the calculations to provide evidence of validity of the design calculations.

Protection of Pipe Lines for Service Water

We were advised that pipelines for service water are embedded in the ground without any special protection. However, there appear to be alternate lines, although they are generally in the same location and/or trenches. In view of the foundation conditions surrounding the plant, and since there is no indication of previous fault motion or potential faulting, this design approach appears to be adequate. If redundancy in critical water supply is desired, it would be preferable to have separate water lines following independent routes.

Seismograph Installation

The answer to Question 1-1 of Supplement 3 indicates that one seismograph will be installed in the yard area, to provide further evidence of the extent of seismic excitation to which the plant might be subjected if an earthquake occurs. This is acceptable to us.

Containment Design Report

The containment design report, prepared for the applicant by Westinghouse Nuclear Energy Systems and United Engineers and Constructors, has proven to be helpful in arriving at an evaluation of many of the factors inherent in the design. The tables presented are useful in helping to arrive at decisions as to the adequacy of the design; we commend those responsible for the preparation of this summary type material.

We should like to encourage this type of approach to studies of the containment, structures, piping, equipment and other Class I items. We should like to urge that attention be given also to summaries and tabulation of the most important information, in terms of stresses and deformations, including the sources of the various stress components, how they were combined, and related discussion and explanatory material (including figures) which would lend itself to a much better basis for judgment as to the adequacy of design of nuclear facilities in general.

CONCLUDING REMARKS

On the basis of the information made available to us concerning the Class I structures, piping, reactor internals, and other Class I items, it is our belief that the plant possesses a reasonable margin of safety to meet the original design requirements, including the imposed Design Basis Earthquake loading conditions.

REFERENCES

1. "Final Facility Description and Safety Analysis Report -- Vols. I through V including Supplements 1, 2, 4, 5, 6, 7, 8 and 13," Indian Point Nuclear Generating Unit No. 2, Consolidated Edison Company of New York, Inc., AEC Docket No. 50-247, 1969 and 1970.

2. "Containment Design Report," for Indian Point Nuclear Generating Unit No. 2, Consolidated Edison Company of New York, Inc., prepared by Westinghouse Nuclear Energy Systems and United Engineers and Constructors, March 1969. (Labeled Final Draft)

3. "Adequacy of the Structural Criteria for Consolidated Edison Company of New York, Inc., Indian Point Nuclear Generating Unit No. 2," by N. M. Newmark and W. J. Hall, August 1966.

W. J. Hall

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