William J. Cahill, Jr.
Vice President

Regulatory File Cy.

Consolidated Edison Company of New York, Inc.
4 Irving Place, New York, N Y 10003
Telephone (212) 460-3819

July 9, 1976

Re: Indian Point Unit No. 3
Docket No. 50-286

Director of Nuclear Reactor Regulation
ATTN: Mr. Robert W. Reid, Chief
Operating Reactors Branch No. 4
Division of Operating Reactors
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Reid:

Your letter dated June 9, 1976 has advised us that a reassessment of the reactor vessel support system is required, taking into consideration various conditions and effects described in the enclosure to that letter.

A program of analysis designed to provide the required reassessment for Indian Point Unit No. 3 has been in progress since January 1976, and is scheduled to be completed by December 1, 1976. The principal features of that program, including assumptions, analytical methods, and the acceptance criteria, are given in the attachment to this letter.

While the scope of the analysis now in progress generally conforms to the analytical requirements of the enclosure to your letter, certain assumptions and considerations contained therein differ from our present scope. Accordingly, we wish to arrange a meeting with the Commission during the week of July 26, 1976 to review our ongoing program in light of the requirements in your letter.

Depending on the results of such a meeting we will notify you whether any change in our schedule is necessary.

Very truly yours,

William J. Cahill, Jr.

William J. Cahill, Jr.
Vice President

Attachment
RPR/mw

Copy to: Mr. George T. Berry
General Manager and Chief Engineer
Power Authority of the State of New York
10 Columbus Circle
New York, N.Y. 10009

6994

CRITERIA FOR ANALYSIS OF REACTOR VESSEL SUPPORTS FOR INDIAN POINT UNIT NO. 3

JUNE 1976

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1.0 Purpose and Objective of Analysis

The analysis being performed on the reactor coolant system of Indian Point Unit #3 is to verify the safety of the system for postulated pipe breaks at the reactor vessel nozzles. Pipe breaks at these locations will be postulated and the dynamic response of the system calculated. The calculated system response will be evaluated to determine that the reactor can be safely shut down, and the fuel adequately cooled following shutdown. This document will outline the criteria that will be used in the performance of this analysis and the evaluation of the results.

2.0 Postulated Accident

The plant accidents which will be independently postulated to occur are pipe ruptures at a reactor vessel inlet nozzle and at a reactor vessel outlet nozzles. Load contributions from asymmetric reactor vessel cavity pressurization, asymmetric internals depressurization, loop depressurization and release of forces in pipe during steady state operation will be included. The ruptures will be postulated to occur when the plant is at full power.

2.1 Postulated Break Characteristics

The break will be postulated to occur in the field welds at the reactor vessel inlet and outlet nozzle safe-ends. For the purposes of analysis, the breaks will be postulated to occur in Loop 31. The postulated breaks will be circumferential and the crack will be assumed to run around the circumference of the pipe in 10 milliseconds.

The time required for the pipes to seperate will also be included in the analysis. This time will be determined by evaluation of the rate of motion of the pipe after the pipe severance and by estimates of the magnitude and rates of motion of the reactor vessel and of either the reactor coolant pump or steam generator, depending on whether the inlet or outlet nozzle pipe break is being evaluated. The time relationship of the break opening area will be determined by this motion. The broken reactor coolant pipe will be assumed to move until it reaches the concrete in the pipe annulus of the primary shield wall. Credit will be taken for the capability of the insulation on the reactor coolant piping to resist crushing. The crush characteristics of the insulation will be determined by test.

Both the break opening area and the break opening time must be estimated prior to performance of the complete dynamic analysis. The estimate made for break opening area will be such that it should be equal to or greater than that which will be determined from the dynamic analysis. The estimate for break opening time from pipe end separation will be such that it will be equal to or smaller than that which will be determined from the dynamic analysis. Iterations using successively more accurate analytical determinations of break opening area and the time may be required to obtain a realistic value of system response.

2.2 Reactor Vessel Internals and Reactor Coolant Loop Applied Hydraulic Loads

The applied hydraulic blowdown loads on the reactor vessel internals are caused by the propagation of acoustic decompression waves into the vessel region. These waves originate at the break in the postulated loss-of-coolant accident.

The MULTIFLEX computer code calculates the hydraulic transients within the entire primary coolant system. This hydraulic program considers a coupled fluid-structure interaction by accounting for the deflection of the core support barrel, which is represented by a beam oscillator. A beam model of the core support barrel is considered, whereby the cylindrical barrel is divided into fifteen segments and the pressure as well as the wall motion are projected onto the plane parallel to the broken inlet nozzle. Vertically, the barrel is divided into 5 flexible plates, each plate consisting of 3 separate walls. The spatial pressure variation at each time step is transformed into 5 horizontal forces, which act on the 5 mass points of the beam model.

The ability to treat multiple flow branches and a large number of mesh points gives the MULTIFLEX code the required flexiblity to represent the various flow passages within the primary RCS. Time-history values of the pressure, mass velocity, density, and other thermodynamic properties within the RCS, which are computed by the MULTIFLEX program, are utilized in the determination of both the reactor vessel and reactor coolant loops reaction forces.

The FORCE-2 computer code determines the vertical hydraulic loads on the RPV internals during blowdown by utilizing a detailed geometric description of the vessel components and transient pressures and mass velocities computed by the MULTIFLEX code. In the evaluation of the vertical hydraulic loads on the internals, the following types of transient forces are considered: (1) pressure differential acting across the component, (2) flow stagnation on and unrecovered orifice losses across the component, and (3) friction losses along the component.

Variations of the fluid pressure distribution in the downcomer annulus region during the subcooled portion of the blowdown transient produces pressure loadings on the reactor vessel internals. Utilizing the transient pressures computed by the MULTIFLEX code, the horizontal hydraulic loads on the reactor vessel wall, core barrel, and thermal shield are calculated. The annular region between the reactor vessel wall and the core barrel is modeled as cylindrical segments formed by dividing this region into circumferential and axial zones. At each elevation the resultant x- and y-direction hydraulic forces are computed.

The blowdown forces on the loop piping and components result from the transient flow and pressure histories in the coolant piping. Utilizing the computed fluid pressures and mass velocities, together with the appropriate plant layout information, e.g., areas and elevations, the THRUST computer program calculates the time-dependent loads exerted by the fluid on the loops. After proper coordinate transformation these forces are applied as external loadings on the RCL dynamic model.

2.3 Reactor Cavity Pressure Applied Loads

The applied reactor cavity pressure forces on the outside surface of the reactor vessel is caused by the pressurization of the break compartment and of the asymmetric pressurization of the reactor cavity. The calculation of these loads are developed in two phases: (1) the calculation of the mass and energy release rate and (2) the resulting pressures and forces on the reactor vessel.

The design basis mass and energy release calculations are performed in the SATAN-V Code. The SATAN-V Code is used to predict release data during the early blowdown transient. This code uses a control volume approach to model the behavior of the Reactor Coolant System resulting from a large break in a main coolant pipe. Release rate transients are determined by the SATAN-V break model which includes a critical flow calculation and an implicit representation of pressure wave propagation.

The SATAN-V critical flow calculation employs appropriately defined critical flow correlations applied for fluid conditions at the break element. For the early portion of blowdown, subcooled, saturated and two phase critical flow regimes are encountered. For this particular application, available critical flow data and other critical flow correlations are presently being reviewed and compared to the flow correlations used in current design basis calculations. Emphasis will be placed on the range of fluid conditions applicable to the Indian Point Unit #3 plant. Based on these studies, a critical flow correlation will be chosen for use in the final analysis.

The pressure distribution around the break and around the vessel is determined using the TMD computer code. The design basis TMD Code uses an unaugmented homogeneous critical flow correlation and the isentropic compressible subsonic flow correlation.

The critical mass flow rate correlation utilized assumes a homogeneous mixture of air, steam and water. Data which show comparisons between measured critical mass flow rates and predictions using the homogeneous critical flow model at low pressures have been analyzed. Specifically, an equation which augments the homogeneous model critical mass flow rates has been developed which provides a conservative lower bound to experimental critical mass flow rates. The augmentation factor applied to homogeneous model flow rates is (1.2 - .2X), where X is the quality of the upstream compartment.

The nodalization scheme used around the reactor vessel produces an accurate post-LOCA pressure profile because of its design. The number of nodes included is based on the results of a nodalization sensitivity study in which the total number of nodes was changed from 6 to 68. Elements near the break are made small to minimize internal element pressure gradients. Elements farther from the break are made larger because the pressure gradients are lower in those regions.

The inspection port sand plugs will be assumed to blow open at a time when the pressure is large enough to move the mass of the sand away from the break compartment. All insulation is assumed in place and uncrushed during the entire transient with the exception of the insulation on the broken pipe and broken nozzle which is assumed to be blown away. Less conservative assumptions concerning insulation will be used if justified by further studies.

The force on the vessel is obtained by integrating the pressure over the area of the vessel and nozzle. The component of force for each element in a given direction will be summed to obtain the force time history of the cavity load applied to the vessel.

2.4 Reactor Coolant Pipe Static Release Force

A pipe rupture force is exerted on the reactor vessel resulting from the release of the static forces which exist in the pipe during steady state operation.

By disturbing a load path in the pipe which is postulated to break, there is a force applied to the vessel with the magnitude of the normal operating forces in the pipe. The rate of release of this force is calculated based on the crack running time.

2.5 Load Combination

All of the above loads are a result of the postulated break and are calculated on a time history basis with time equal zero when the postulated crack in the pipe begins to run. These loads are all applied simultaneously to the reactor vessel, internals, and reactor coolant loops. The stresses and loads used in the evaluation of the system will be due to the above described blowdown loads and normal operating loads. Seismic loads will not be combined with the blowdown loads.

3.0 Structural Analysis

The loads resulting from the postulated pipe break described in Section 2.0 are simultaneously applied to a mathematical model of the reactor coolant loops, reactor vessel and internals. Two analyses are planned, an elastic analysis and, if necessary, an inelastic analysis. The structural analysis will be described in this section.

The mathematical model of the reactor coolant system includes the reactor vessel, vessel internals, fuel, loop piping, reactor coolant pump, steam generator and component supports. The model is three-dimensional and is constructed from beam elements, pipe elements, non-linear gap elements, friction elements, springs and dampers. Detail will be included in the system model or in seperate analyses in those regions which are highly loaded or where the inclusion of detail will lead to less conservatism and/or more realism in the calculated results. These regions include the core barrel flange region, lower radial core barrel supports, fuel, and the pump and steam generator supports.

The analysis will include the structural damping values tabulated in Table 1, impact damping at gap elements calculated from the coefficient of restitution of the materials of the impacting structure (approximately 12%) and friction damping at appropriate interfaces based on the properties of the materials in contact. Increased damping will be used locally in regions where plasticity is experienced.

The loads will be applied to the model in a time-history form. A dynamic solution will be obtained using the WECAN code. The solution will provide motion of the components and pipe, loads in supports, loads in the pipe and components, fuel core plate motion and values of the fuel grid impact force. These loads and displacements will be used to calculate stresses in the system.

The analysis will first be performed based on elasticity in all components. The results of the elastic analysis will indicate if plasticity of the system will occur and, if so, where plasticity must be included. The analyses will then be rerun to determine the inelastic response.

TABLE 1

Components	Structural Damping
Reactor Coolant Loop Piping	4%
Aux. Line Piping	4%
Components	4%
Fuel	10%

4.0 Criteria of Acceptability

The basic criteria of acceptability of the plant for the postulated pipe rupture will be that the reactor can be safely shut down and the fuel adequately cooled. Verification of this ability is the objective of the analysis. The criteria for acceptance will be outlined in this section.

The results of the elastic analysis, stresses in supports, components, piping, etc., will be initially compared to the guidelines outlined in the ASME Boiler and Pressure Vessel Code, Section III, Appendix F. Exceeding these elastic stress limits by a large margin will indicate the need for an inelastic analysis. These high stresses in isolated components may be acceptable dependent on the location of the component and its effect on the system response. The basic overall criteria to be used for each of the individual components will be outlined below.

4.1 Fuel

The fuel must be maintained in a coolable geometry. The fuel grids maintain the spacing of individual rods and the spacing of the fuel assemblies. The magnitude of the impact forces in these grids, therefore, are important in assuring the coolability of the fuel. The highest impact forces generally occur in the outer fuel assemblies due to impact into the core barrel baffle plates. The grid impact loads and the behavior of the fuel during the LOCA will be determined and satisfaction of the criteria of appendix K of 10CFR50 will be demonstrated. The stresses in the thimbles and fuel rods are limited to assure their integrity. Limits consistent with the guidelines set up in ASME Section III appendix F will be used.

4.2 Internals

The deformation of the core barrel and other core support structures are included in the evaluation so as to obtain accurate fuel core plate motions.

Local plastic deformation in these components is acceptable.

4.3 Piping

The ECCS piping in the unbroken loops must retain their integrity to assure delivery of coolant to the core. Plastic deformation is acceptable and integrity will be demonstrated by plastic evaluation of the most highly stressed lines and/or by comparison of loading experienced to available test results. The ECCS piping will be evaluated by statically applying the motion of the reactor coolant loop at the appropriate location to a mathematical model of the branch line piping. The stress calculated in the piping must correspond to a strain of less than 50% of the uniform ultimate strain.

The reactor coolant piping in the unbroken loops must be evaluated to assure its integrity. The analysis will determine the magnitude of the stresses in the pipe and must correspond to a strain level in the pipe of 50% of the uniform ultimate strain of the material.

4.4 Components

The pressure boundaries of the steam generator, reactor coolant pump, reactor vessel, and CRDM's must retain their integrity. Integrity can be assured by demonstrating that the stresses in these components correspond to strains which are less than 50% of the uniform ultimate strain of the material.

4.5 Component Supports

The supports of the reactor vessel, reactor coolant pump, and steam generator may sustain plastic deformations up to and including, in isolated cases, failure. Stresses and strains in local support members may equal or exceed the ultimate stress or strain as long as this inelastic behavior is included in the analysis for an accurate determination of the motion of the reactor vessel and components.

4.6 Concrete

The concrete must retain its integrity in areas where its integrity is required to assure the safety of the plant. For instance, the concrete of the reactor cavity must be able to withstand the combination of pressure and applied load through the ring girder to assure adequate support of the vessel. Analyses may be performed which would allow cracking of concrete and redistribution. Deformation of concrete at embedments of the component supports may be acceptable if the effect is included in the structural analysis.

William J. Cahill, Jr.
Vice President

REGULATORY DOCKET FILE COPY

Consolidated Edison Company of New York, Inc.
4 Irving Place, New York, N Y 10003
Telephone (212) 460-3819

July 6, 1976

Re: Indian Point Units 1,2, and 3
Docket Nos. 50-3, 50-247, & 50-286

Director of Nuclear Reactor Regulation
ATTN: Mr. Robert W. Reid, Chief
Operating Reactors Branch No. 4
Division of Reactor Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

RECEIVED
7-12-76
U.S. NUCLEAR REGULATORY COMMISSION
Mail Section

Dear Mr. Reid:

Your letter of February 18, 1976 (Indian Point Unit Nos. 1 & 2) and Mr. De Young's letter of February 20, 1976 (Indian Point Unit No. 3) provided Consolidated Edison and the Power Authority of the State of New York with guidance regarding the filing of information required by Appendix I to Part 50 of the Commission's regulations. The enclosures to these letters requested considerable information on the Indian Point site and facilities. Con Edison is in the process of obtaining the requested information and has made a commitment to forward this material to the Staff as it becomes available.

The present submittal provides responses to items 5,7, and 8 of Enclosure 2 to the February 1976 letters. These responses are contained within the Attachment and the two accompanying boxes of computer cards.

Very truly yours,

William J. Cahill, Jr.

William J. Cahill, Jr.
Vice President

cc: Mr. George T. Berry
General Manager and Chief Engineer
Power Authority of the State of New York
10 Columbus Circle
New York, N.Y. 10019