Docket No. 50-286

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March 28, 1984

Mr. J. P. Bayne, Executive Vice
President - Nuclear Generation
Power Authority of the State of New York
123 Main Street
White Plains, New York 10601

Dear Mr. Bayne:

Enclosed is our Safety Evaluation which concludes that, with the modifications described in the June 15, 1977 submittal (WCAP-9117), there is reasonable assurance that the reactor coolant system for Indian Point 3 could withstand the effects of the asymmetric LOCA loads.

Note that the enclosed SER is valid as long as the assumptions made regarding the acceleration of the blow out shield plugs are met. Therefore, the staff requires that you verify the shield plug assumptions and determined the effects of the plugs as missiles. Also we require that any structural components that inhibit the shield plug displacement be removed.

Pending confirmation for the above, this completes our review of the asymmetric LOCA loads issue for Indian Point 3.

Sincerely,

Original signed by
Steven A. Varga

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Enclosure:
As stated

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I. INTRODUCTION

On May 7, 1975, the NRC was informed by Virginia Electric and Power Company that asymmetric loading on the reactor vessel supports resulting from a postulated reactor coolant pipe rupture at a specific location (e.g., the vessel nozzle) had not been considered in the original design of the reactor vessel support system for North Anna, Units 1 and 2. It has been identified that in the event of a postulated, instantaneous, double-ended offset shear pipe break at the vessel nozzle, asymmetric loading could result from forces induced on the reactor internals by transient differential pressures across the core barrel and by forces on the vessel due to transient differential pressures in the reactor cavity. With the advent of more sophisticated computer codes and the development of more detailed analytical models, it became apparent that such differential pressures, although of short duration, could place a significant load on the reactor vessel supports and on other components, thereby possibly affecting their integrity. Although this potential safety concern was first identified during the review of the North Anna facilities, it was determined to have generic implications for all pressurized water reactors (PWRs).

Upon closer examination of this situation, it was determined that postulated breaks in a reactor coolant pipe at reactor pressure vessel (RPV) nozzles were not the only area of concern; but, rather that other pipe breaks in the reactor coolant system could cause internal and external transient loads to act upon the reactor vessel and other components.

Although the NRC staff's original emphasis and concern were focused primarily on the integrity of the reactor vessel support system with respect to postulated breaks inside the reactor cavity (i.e., at a nozzle), it became apparent that significant asymmetric forces could also be generated by postulated pipe breaks outside the cavity and that the scope of the problem was not limited to the vessel support system, itself. The staff, after reviewing this problem, determined that a re-evaluation of the primary system integrity of all PWR plants to withstand these loads was necessary.

By letters dated July 22, 1975 and June 9, 1976 the Nuclear Reactor Regulation Staff requested from Consolidated Edison Company of New York, Inc. various information concerning the Indian Point 3 Nuclear Power Plant reactor vessel supports. Partial responses to those requests were forwarded to the Staff by letters dated August 15, 1975, September 4, 1975, November 14, 1975, and July 9, 1976.

On June 15, 1977, Consolidated Edison submitted a Proprietary Class 2 Westinghouse Report WCAP-9117, "Analysis of Reactor Coolant System for Postulated Loss-of-Coolant Accident: Indian Point 3 Nuclear Power Plant" and in that way completed licensee responses to the July 22, 1975 and June 9, 1976 information requests.

The report presents the evaluation of the reactor coolant system (RCS) for the loads induced by a LOCA which results from the unlikely event of a pipe rupture within that system. The objective of the evaluation is to verify the capability of the plant to reach and maintain a safe shutdown condition following the event. The analyses include all loads in the system among which are the asymmetric loads in the reactor internals and the reactor cavity pressurization loads for the RPV nozzle break locations, and the effect of any inelastic structural reponse.

After the initial review of the above report, the staff sent a letter to Consolidated Edison in January 1978 requesting additional information. The Power Authority of the State of New York (PASNY) submitted a response to the above inquiries. Materials contained in the response are also being reviewed herein. In addition, the Indian Point, Unit #2 licensee, Consolidated Edition Company, submitted a letter dated June 15, 1978 in which the applicability of the Indian Point, Unit 3 analysis (WCAP-9117) to Indian Point, Unit #2 was demonstrated. We reviewed and approved the above study. Therefore, this Safety Evaluation Report is applicable to both Units 2 and 3.

Since the identification of the asymmetric load problem in May 1975, our contractor, EG&G Idaho, Inc. has performed a number of independent audit analyses to verify licensee submittals on this problem. For the Indian Point #3 plant, EG&G analyzed the entire primary coolant system with a non-linear, fully coupled, three dimenstional, inelastic finite element structural model. EG&G's audit analysis addressed one of the breaks considered in the submittal, that being a cold leg RPV inlet nozzle break. Also, the NRC staff provided audit calculations of the thermal hydraulic and cavity pressure loads for use in EG&G's structural analysis. Thus, there is a firm technical basis within the NRC staff and its consultants to evaluate the licensee's submittal.

In addition to the analysis reported in WCAP 9117, which considered only the worst case break locations relative to the effect on the reactor vessel and unbroken reactor coolant loops, the licensee has analyzed the reactor coolant system for the postulated break locations outside the reactor cavity. That analysis considered all the applicable transient blowdown loads. All piping systems and system supports were shown to have acceptable stress levels when subjected to these loads. The consideration of pipe ruptures in the cross-over leg steam generator primary nozzles safe ends, and reactor coolant pump primary nozzle safe ends provide assurance that the structural integrity of the loops is maintained. Those analyses were previously approved by the staff in the original plant FSAR. Cavity pressure loads are additive to the loads used in the original analysis for breaks outside the reactor cavity. However, in a letter from PASNY on August 24, 1978, the licensee has stated that the region immediately surrounding the steam generator is not conducive to assymmetric pressurization because of the openness of the design. There are no secondary shield walls surrounding the steam generators and thus assymmetric loads would not be generated for primary coolant system pipe rupture in the steam generator and pump compartment. The steamline runs above the biological shield wall in an open area and, consequently, a rupture in the vertical drop would cause no significant asymmetric pressurization. Therefore, the analysis and evaluation presented in WCAP 9117 by the licensee represents the limiting cases for the asymmetric LOCA evaluation.

The effect of computing the combined responses of SSE and LOCA by combining the respective strain components absolutely is given in Table 1 of WCAP-9117. The ultimate strain for the support materials falls in the range of 20%-30%. The onset of strain hardening is at a strain level of approximately 2%. Initial yield occurs at approximately 0.2% strain. Since the maximum total strain from the combined SSE and LOCA responses is approximately 0.5%, the system is at a level of strain well within its capability of maintaining its function of supporting the components. (Reference d)

The proposed primary shield wall restraints are not presented in the table, because the restraint was designed with a gap and would not be loaded during an SSE. The maximum load in a primary shield wall whip restraint for a 0.568 inch reactor vessel LOCA motion is 2450 kips. This load is well below 4860 kip capacity of this restraint. Therefore, the shield wall restraints are acceptable for SSE + LOCA loads.

Subsequent sections of this safety evaluation report summarize the evaluations performed by the licensee for subcooled blowdown loads, cavity pressure analysis, and structural analysis and evaluations. Following this is the staff evaluation of these same analyses which includes our evaluation of both licensee's compliance with acceptance criteria.

II. LICENSEE EVALUATIONS

A. Subcooled Loads Analysis

The MULTIFLEX computer program, WCAP-8708, "MULTIFLEX, a FORTRAN-IV Computer Program for Analyzing Thermal-Hydraulic Structure System Dynamics," was used to predict the transient hydraulic response of the entire reactor primary coolant system for three postulated break locations. The breaks considered are 110 square inch at the reactor vessel inlet nozzle of a full area break at the outlet nozzle, and a full pipe area break at the reactor coolant pump outlet. A beam structural model of the core support barrel is included in the representation of the Indian Point 3 plant input to the MULTIFLEX program. The purpose of this is to consider the coupled fluid to structure interaction effects of core barrel motion on the downcomer pressure transients. The hydraulic model input into MULTIFLEX is discussed qualitatively and in a very synoptic fashion. The vertical loads in the internals region are computed by the FORCE 2 computer program using the output from the MULTIFLEX program. The conversion of downcomer pressure to horizontal vessel and core barrel forces is described qualitatively.

B. Cavity Pressurization Analysis

The licensee used the Westinghouse Electric Corporation TMD computer code, with the compressibility factor and the unaugmented homogeneous critical flow correlation to perform the reactor cavity pressure analysis. The staff has previously reviewed and approved the TMD code as part of the NRC topical report evaluation program. The licensee performed the analysis considering the postulated ruptures of the reactor coolant system hot and cold leg pipes at the reactor vessel outlet and inlet nozzle welds, respectively. The break size used for the analysis was a 110 square-inch break for both the hot and cold leg break.

The reactor cavity annulus, the volume between the reactor vessel and the shield wall, was modeled in the TMD code as a multinode region accounting for the geometric discontinuities or area changes. In order to reduce the effects of asymmetric pressurization the licenee has assumed that shielding material, i.e., shield plugs located over the nozzles, will blow out and provide increased flow area away from the reactor cavity. The licensee has committed to redesign and replace the current shield plugs with a design that is more readily displaced by the pressure resulting from a pipe rupture.

C. Structural Evaluation

C.1 Introduction

The licensee's structural analysis methods to predict the RPV's dynamic response to the LOCA loads were based on mass decoupling of the primary coolant loops and a simplified representation of the fuel with a planar vessel and internals model. The LOCA induced loads in the various subsystems comprising the primary coolant system were then computed using the vessel displacements as input to detailed models of these subsystems. Descriptions of the various system and subsystem models and how they were used to predict the loads and response throughout the primary coolant system are presented in the following paragraphs. Also presented is a description of assumptions inherent in the licensee's analysis methods.

C.2 Primary Coolant System Analysis

A static analysis was performed on the reactor coolant loop piping and supports to develop load deflection curves or stiffness matrices to be used in the dynamic reactor pressure vessel analysis and to evaluate loads and stresses in the piping, supports, and nozzles. To perform this analysis a nonlinear, elastic-plastic finite element structural model of all four loops including the primary piping, steam generator, coolant pumps, and supports in each loop was formulated for use in the WECAN computer program. The piping, steam generator, and coolant pumps were represented by three dimensional, elastic-plastic pipe and elbow elements while the steam generator supports and pump supports were modeled by three dimensional assemblages of linear elastic, nonlinear elastic, and elastic-plastic plate, beam, and spar or truss elements. Also, included in the loop model were linear elastic, elastic-plastic, and gapped elements to represent steam generator and reactor pressure vessel shell stiffnesses, hot stops and snubbers on the steam generator supports, concrete embeddment stiffnesses, and primary piping restraint stiffnesses. The loop load deflection curves were then determined by applying to the model in separate computer runs incremental horizontal translational deflections in the direction of the assumed break, vertical translational deflections, and rotations about an axis perpendicular to the break and the reactor centerline.

The RPV and internals LOCA dynamic response was determined using a planar centerline structural model of the vessel and internals in the DARIWOSTAS computer code. The model included the core barrel, lower support columns, bottom nozzles, skeletons, fuel rods, top nozzles, upper support columns, upper support structure, water mass, and reactor vessel. These components were modeled in the horizontal direction by beam elements and concentrated masses connected by rigid links, translational impact springs with dashpots, or rotational springs. In the vertical direction the components were represented by concentrated masses, springs, dashpots, gaps, and frictional elements. The RPV supports were included as horizontal and vertical nonlinear stiffnesses which grounded the model. The horizontal RPV support stiffness was determined by the licensee's load deflection test of this component. The vertical stiffness acted only in the downward direction to reflect the noncaptive nature of the RPV supports. As previously indicated the piping loops in the vessel and internals model were represented by nonlinear load deflection curves or stiffness matrices applied at the nozzle elevation of the RPV. The vessel response due to the LOCA was calculated by the DARIWOSTAS code using applied loads consisting of reactor internal hydraulic forces, reactor cavity pressurization forces, and loop mechanical loads caused by the release of normal operation static equilibrium forces at the postulated break.

C.3 Subsystem Analyses

The remaining subsystems analyzed consisted of the most highly stressed auxiliary piping lines (the accumulator line in loop 33 and the RHR line in loop 32), the control rid drive mechanisms (CRDM's), the reactor core, the reactor core barrel, and the primary shield wall. In contrast to the primary piping loop model, detailed models and analyses of these subsystems were used only to evaluate loads and stresses in the above mentioned components resulting from the RPV's motion and not to represent their effects on the RPV's response. It is noted that a simplified representation of the core barrel, fuel, and CRDM's was included in the DARIWOSTAS model of the RPV and internals.

The accumulator line in loop 33 and the RHR line in loop 32 were modeled with the WECAN code using three dimensional, elastic-plastic pipe and elbow element to represent the auxiliary piping. Restraints in the lines were represented with spring gap element and slider or elastic plastic truss elements. The stiffness effect of the primary piping was represented with a generalized stiffness matrix. The loading for the analysis consisted of time history vessel motions and hydraulic loads due to loop depressurization.

A SCRAM time analysis involving the CRDM's was performed to determine the ability of the control rods to drop properly in the event of the postulated LOCA. This analysis was performed using the planar horizontal and rotational portion of the DARIWOSTAS program. The model which employed many of the element types used to predict the dynamic response of the RPV consisted of the reactor vessel, internals, center row of CRDM's, four drive rod assemblies, and the seismic support platform. A static displacement input at the reactor vessel was used to determine friction forces and the resulting adverse effects on SCRAM time caused by permanent vessel motion. Additionally, the dynamic time history motion of the reactor vessel was imposed on the model and a dynamic analysis was performed to evaluate loads and stresses in the CRDM's.

To determine the adverse effects on core cooling caused by fuel assembly spacer grid impacting during the LOCA a lateral core model consisting of one row of fifteen fuel assemblies and the reactor baffle was formulated. The fuel assemblies were represented by special elements modeling the in-grid and through grid stiffnesses of the grids and the stiffness and mass of the fuel rods and guide tubes. The baffle was represented by a single beam element. The fuel assemblies and baffle were interconnected at the spacer grids by gap elements to permit impact forces to be transmitted from the baffle to the fuel assemblies. Time history motions of the upper and lower core plates and the core barrel at the upper core plate elevation from the DARIWOSTAS RPV and internals analysis were input at the top and bottom of the fuel and at the top and bottom of the baffle to determine spacer grid impact forces and fuel deflections.

To completely determine stresses in the core barrel from the asymmetric downcomer depressurization, a separate dynamic shell analysis of the core barrel was performed. Stress results from this analysis were then combined with the beam bending and axial stresses from the DARIWOSTAS vessel and internals analysis to obtain the total stresses in the barrel.

The ability of the primary shield wall to sustain the worst case pipe rupture loads was determined from a three dimensional continuum model of the wall using the MARC-CDC computer code. The concrete slab and shield wall which were assumed uncracked and without were modeled using 20 node isoparametric brick elements. The embedded steel ring girder at the top of the shield wall was modeled with 4 and 8 node isoparametric membrane elements. The loads on the model consisted of RPV support reaction loads, tie rods loads, and reactor cavity pressurization loads.

C.4 Summary of Assumptions

Although not explicitly stated a number of assumptions are evident from the licensee's description of the structural analysis methods. The most important of those is that the inertia and hydraulic load effects of the primary coolant loops on the RPV's response is negligible. This assumption is implied since only the stiffness of the piping was included in the DARIWOSTAS RPV and internals blowdown model. Since the loop stiffness matrices are nonlinear and inelastic, vertical, horizontal, and rotational decoupling of the piping loops is assumed. Additionally, planar vessel motion is assumed for the vessel and piping loop's analyses. Also, the dynamic effect of the auxiliary lines and the shield wall are neglected in the dynamic analysis of the RPV and internals.

D. Summary of Licensee's Structural Evaluation and Conclusions

D1. Introduction

The basic criteria for acceptability of the plant for the postulated LOCA was that the reactor could be safely shutdown and the fuel assembly adequately cooled. To demonstrate acceptability the following components and structures were evaluated: reactor core, reactor internals, piping, steam generators, reactor coolant pumps, reactor vessel, CRDM's, reactor vessel supports, reactor coolant pump supports, steam generator supports, and shield wall concrete. The basic stress and load criteria were consistent with the ASME Code, Section III, Appendix F and Appendix XVII for the LOCA loading.

D2. Primary Coolant System

The reactor coolant loop piping evaluation was performed on the basis of the static loop analysis previously described. To account for the dynamic motion of the vessel and the hydraulic loads in the loops during a LOCA a dynamic load factor of 2.0 was applied to the vessel peak displacement to determine stresses and loads in the piping loops. The maximum primary piping stress found to occur at the reactor safe ends. This stress was within the corresponding Appendix F allowable of 0.7 Su.

The reactor vessel support integrity was verified using the results from the DARIWOSTAS RPV's and internals dynamic analysis. The peak horizontal vessel displacement was the displacement required to produce support failure as determined by the licensee's reactor support tests. The maximum vertical support load was less than the vertical load capability.

D3. Subsystems

The steam generator and reactor coolant pump supports were also evaluated with the resulting stresses and loads compared to criteria of Appendix XVII. Most members were within allowables; however, a few members in loop 31, 32, and 34 experienced local yielding. Acceptability of the components was based on their relatively small strains.

Evaluation of the auxiliary lines was performed using the time history vessel motions and the loop hydraulic loads in the dynamic models of the most highly stressed lines (the accumulator line in Loop 33 and the RHR line in Loop 32). In the accumulator line the maximum stress occurred in the branch connection. This stress was within the Appendix F limit for inelastic system analysis and elastic component analysis. Similarily in the RHR line the maximum stress occurred at the branch nozzle. This stress was also within Appendix F limits.

The evaluation of the reactor coolant system components consisted of reviewing previous Indian Point Unit 3 or similar plant analyses for the components. The new loads from the piping loop analysis were then compared to the previously computed loads which were within equivalent ASME Appendix F allowables (Appendix F didn't exist at the time the original analyses were performed). The components reviewed were steam generator primary inlet and outlet nozzles, steam generator support feet, reactor coolant pump inlet and outlet nozzles, and reactor coolant pump support feet. In all cases the loads were less than the loads used to qualify the equipment in the previous analyses. A separate analysis of the reactor vessel primary inlet and outlet nozzles were performed to demonstrate adequacy of these components. The most highly stressed region was the outlet nozzle shell juncture. The elastically computed stress was within the Appendix F allowable.

The CRDM's were evaluated with the SCRAM time analysis. SCRAM time increased by less than 5%. Additionally, from the dynamic analysis the maximum CRDM bending moment was within the allowable.

Evaluation of the reactor core was performed using the dynamic core model with time history upper and lower core plate and baffle motion from the vessel dynamic analysis. The maximum fuel assembly motion resulted in fuel and guide tubes stresses considerably below allowables. Grid impact forces were also determined. One grid experienced an impact load greater than that at which permanent deformation begins, (partial). Several other grids showed loads less than the above but greater than practical. The effect of distorted grids on ECCS performance was evaluated using the 1975 version of the Westinghouse evaluation model with permanent grid deformation postulated in the limiting fuel location. Assuming a full double offset guillotine break (600 in^2) break (the limiting large break LOCA) rather than the 110 in^2 break an increase of less than 20°F in peak clad temperature (PCT) was predicted. Thus, the effect of grid crushing on core coolability was concluded to be insignificant.

The core barrel shell analysis combined with its beam response was used to evaluate this structure for the asymmetric blowdown pressures.

The maximum membrane and bending stress intensities were below the Appendix F allowables of 2.4 Sm and 3.6 Sm respectively.

The shield wall was evaluated with the three dimensional continuum model using reactor vessel reaction loads from the blowdown analysis, the tie rod loads at yield, and cavity pressurization for the full break (600 in^2). The resulting membrane and bending loads in the hoop direction were within the shield wall capacity assuming the rebar carries the entire load. Shear from pressure and the tie rods reaction were checked and found to be within the capacity of the concrete.

From the above evaluations the licensee concluded that overall acceptability of the plant for the postulated LOCA was met.

III. STAFF EVALUATION

A. Introduction

The staff evaluation contained in this report considers only the effect of asymmetric loss of coolant accident loads due to postulated pipe breaks in the primary coolant loop piping. The licensee's analysis procedure including analytical models, computer methods, and acceptance criteria have been evaluated by the staff for asymmetric LOCA loads. The staff evaluation was accomplished by reviewing the licensee's submittal and using the independent audit calculations performed by the staff or their consultants. In general, the staff has concluded that the licensee's assessment of the asymmetric LOCA loads problem is acceptable. The staff evaluation of each specific analysis phase is addressed in subsequent paragraphs.

B. Subcooled Blowdown Loads

The subcooled blowdown calculation portion of the Indian Point Unit 3 asymmetric LOCA load submittal has been reviewed and is considered to be acceptable to the staff.

The basis for this acceptance is that the Indian Point Unit 3 plant was utilized in the staff's audit review. Loads calculated from the staff's subcooled blowdown analysis were used in an independent structural analysis performed by the staff's consultant which indirectly provided an audit of the MULTIFLEX loads calculation.

Only breaks which affect the asymmetric LOCA load issue were included in the licensee's submittal and therefore, addressed by this review. The licensee has noted (PASNY letter of August 24, 1978) that breaks at other locations were considered as part of the original analysis and are described in WCAP-8172-A. The Indian Point Unit 3 analysis used a 5 mass representation of the core barrel in MULTIFLEX. The staff's SER requires the core barrel be represented by 10 mass points. However, the staff has concluded that the specific Indian Point Unit 3 representation is acceptable as the differences between the 5 mass and 10 mass models are small.

C. Cavity Pressurization Analysis

The cavity pressurization analysis of the Indian Point Unit 3 plant for postulated breaks at the RPV inlet and outlet nozzles has been reviewed and is considered to be acceptable to the staff. The basis for acceptability is that the TMD computer code has been reviewed and approved by the staff, the input model was reviewed by the staff consultants, and that the calculated cavity pressure loads were input to a structural analysis performed by the staff's consultant, EG&G Idaho. This final item provided an indirect audit of the code used to calculate the cavity pressure transient.

Only breaks which affect the asymmetric LOCA load issue were included in the licensee's submittals and therefore, addressed by this review. The licensee has noted (PASNY letter of August 24, 1978) that breaks at other locations were considered as part of the original analysis and are described in WCAP-8172-A.

The staff will require that the licensee verify the assumptions regarding shield plug acceleration and determine the effects of the plugs as missiles after the redesign is complete. Presently the plant design incorporates a cover plate over the shielding material. We will require that the licensee remove all structural components such as the cover plate or seal ring which may inhibit the shield plug displacement or incorporate these effects in a re-analysis of the reactor cavity pressure transient.

D. Structural Evaluation

D1. Evaluation of Methods and Models

The structural computer codes cited in the licensee report are considered to be acceptable to the staff for the present application. Each of the codes used have either (a) widely used with sufficient history of success to justify its applicability and validity, (b) compared through use of audit calculation to solutions obtained using a code meeting criteria, or (c) been benchmarked against hand calculation, experimental test or results published in technical literature.

The mass decoupled analysis of the reactor pressure vessel and internals performed to determine the LOCA mechanical response of the system is acceptable to the staff. The relatively flexible piping connecting the RPV with the loop components permit this decoupling, and has been verified by EG&G Idaho's audit analysis.[a,b,c] The planar vessel and internals motions assumption in this analysis is acceptable on the basis of the arrangement and stiffness of the RPV supports and the generally planar nature of the applied loading.

Analysis of the ECCS and auxiliary lines is acceptable based on the bounding analysis performed by the licensee. This analysis consisted of a dynamic analysis of the most highly stressed ECCS and auxiliary lines for hydraulic loop depressurization loads and RPV motion as determined from the RPV and internals dynamic analysis.

The static SCRAM time analysis is considered appropriate since SCRAM occurs after the time frame for which dynamic effects are predominant. The conservative vessel displacement applied insures a reasonable upper bound on SCRAM time. Additionally, the dynamic analysis of the CRDM'S provide acceptable estimates of critical loads.

The determination of fuel deformation and spacer grid impact loads is accepted as the appropriate internals motion (upper and lower core plate and baffle) is adequately incorporated as the fuel assembly forcing functions. The stiffness of the baffle permits decoupling barrel shell modes from the analysis.

Determination of the total stresses in the core barrel resulting from the asymmetric downcomer depressurization using decoupled beam and shell modes is acceptable since this procedure has been shown to be mathematically exact for linear analyses.

Acceptability of the shield wall analysis is based on the conservatism employed in the structural model and the applied loads.

As indicated in the last paragraph of Section II-C.3 the assumptions of planar vessel motion and negligible inertia and hydraulic load effects in the piping loops are acceptable. Since the vessel motion was mainly horizontal and vertical inelastic piping deformation small, the assumption of decoupled loop stiffness matrices is acceptable. Planar imposed motion in the piping loop analyses is accepted since planar vessel motion was shown to be acceptable. Neglecting the dynamic effects of the auxiliary lines and the shield wall on RPV response is acceptable based on the relative flexibility of these lines and the limited dynamic load path through the RPV supports to the concrete.

D2. Compliance with Acceptance Criteria

The licensee's stress and/or load evaluation of the core barrel, primary piping, auxiliary piping, and primary coolant system nozzle components is acceptable since the appropriate ASME Appendix F criteria are met.

Although significant plastic deformation of the RPV supports occurs during the postulated LOCA acceptance of the licensee's evaluation is based on the additional RPV support capability provided by the primary piping restraints.

It was previously indicated that some members in the steam generator and pump supports exceeded yield and Appendix XVII and Appendix F allowables. Acceptance of the support stress evaluations is based on the localized nature of the yielding, the conservatism in the applied vessel motion, and the relatively small strains in the members exceeding allowables.

Two principle acceptance criteria apply for the asymmetric LOCA: (1) fuel rod fragmentation must not occur as a direct result of the blowdown loads, and (2) the 10 CFR 50.46 temperature and oxidation limits must not be exceeded.

The first criterion is satisfied if the calculated loads on the fuel rods and components other than grids remain below designated allowable values. The second criterion is satisfied by an ECCS analysis.

Stresses are calculated in accordance with the previously approved methods documented in WCAP-8236 and WCAP-8236 Addendum No. 1. The maximum stress levels associated with the fuel rods and fuel assembly components other than grids are determined by the licensee to be below designated allowable values. Fuel rod fragmentation will therefore not occur.

Although a small number of spacer grids are predicted to experience some permanent deformation as a result of a LOCA, the effect of this grid distortion was conservatively incorporated into an appropriate ECCS analysis. The peak clad temperature predicted for this distorted geometry increased less than 20°F. A coolable geometry will, therefore, be maintained.

Control rod insertability is not required for a large break LOCA (See NUREG-0609). Based on the fuel system analysis, control rod insertion should not be significantly impaired.

Acceptance of the shield wall stress evaluation is based on compliance with industry standards for reinforcement concrete.

In conclusion, there is reasonable evidence that the Indian Point Unit 3 fuel systems would withstand the effects of an asymmetric

LOCA without impairing either coolable geometry or control rod insertion.

IV. References

a. R. W. Macek, "Nonlinear LOCA Dynamic Analysis of the Indian Point Unit 3 Primary Coolant System, "EG&G Interim Technical Report RE-A-77-106, October, 1977.

b. R. W. Macek, "Structural Sensitivity Studies Using the Indian Point Unit 3 Finite Element Model," EG&G Interim Technical Report RE-A-78-030, February, 1978.

c. R. L. Grubb, "Indian Point Unit 3 Fuel Assembly Mechanical Response Analysis," EG&G Idaho Interim Report, RE-A-77-144, January, 1978.

d. P. Zarakas, Letter to T. Novak dated September 3, 1980 regarding the assessment of combining LOCA and SSE response.