Report by the AEC Regulatory Staff

In the Matter Of

Consolidated Edison Company of New York, Inc.

Indian Point Station, Unit No. 2

In Response to

ASLB Questions Concerning Reactor Vessel Integrity

and

"Additional Testimony of Applicant

Concerning Reactor Vessel Integrity (September 17, 1971)"

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U.S. Atomic Energy Commission

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INDIAN POINT UNIT 2 REACTOR VESSEL

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I. SUMMARY

The regulatory staff concurs with the applicant's conclusion that the Indian Point 2 reactor vessel (1) has been designed, fabricated, and tested so as to provide a high level of initial quality and structural integrity, (2) will be subjected to carefully controlled operating conditions during its service lifetime so as to prevent any loadings beyond the specified design conditions for safe operation, and (3) will be monitored periodically during operation by inservice inspections so as to provide continued assurance of its quality and integrity during its service lifetime. This combination of an initial high level of quality, controlled operating conditions with conservative margins of safety, and continued surveillance by an inservice inspection program provides assurance that the Indian Point 2 reactor vessel can be operated over its service lifetime with a negligible risk of failure. For the purpose of this evaluation, failure is defined as a vessel rupture of such an extent that the capability of emergency core cooling systems to adequately cool the core may be impaired.

The bases for this judgment are the many elements of conservatism and quality which have been incorporated in the design, construction and planned operation of nuclear reactor pressure vessels. The principal elements for our conclusion include the following:

A. Nuclear reactor pressure vessels are required to be designed, fabricated, constructed and tested to exceptionally high quality standards.

B. The controlled selection of acceptable materials and demonstration of their properties, the application of advanced methods of design and stress analysis, the specification of numerous and exacting quality control measures during fabrication, and the requirements for extensive inspection and testing programs are the basic elements of Section III of the ASME Boiler and Pressure Vessel Code which was specifically developed for the nuclear power industry and which was applied to the Indian Point 2 reactor pressure vessel.

C. The manufacturers of nuclear reactor vessels are limited in number because of the extensive specialized fabrication facilities and the many years of fabrication experience needed to satisfy the quality requirements of the ASME Section III Code rules.

D. The extensive quality assurance programs required in the shops of the reactor vessel manufacturer by the Section III Code rules and the AEC are subject to continuing reviews and audits of performance by the ASME and the AEC Division of Compliance as a means of verifying the maintenance of suitable quality levels.

E. Where supplementary safety requirements are considered necessary, the AEC imposes additional requirements that may not be covered fully by the Section III requirements for reactor vessels. Examples of such requirements are fracture toughness properties for reactor vessel materials, and material surveillance programs to monitor the behavior of these materials under radiation during service.

F. Nuclear pressure vessels are required to be carefully examined periodically during their entire service lifetime by the application of inservice inspection rules of ASME Section XI Code in order to detect any structural degradation which might affect their integrity.

G. Auxiliary systems, safety controls, alarms, and safety trips as well as operating limitations are provided for the specific purpose of assuring with large margins that the reactor vessel design conditions are not exceeded during normal reactor operation or highly unlikely postulated accidents.

H. Service and operator experience to date has provided confirmation of the quality and reliability expected of nuclear reactor pressure vessels. From data available to date, 95 nuclear pressure vessels of commercial pressurized and boiling water reactor plants have successfully completed over 3,500,000 operating hours without any structural failure and without evidence of any unanticipated problems which could be related to potential vessel failure. This experience, which represents 400 vessel-years of reliable and safe operation, includes nuclear pressure vessels which have seen as much as 10 years of operating service.

An identification of the elements of conservatism and quality of nuclear reactor pressure vessels which formed the bases for the regulatory staff's conclusions has been amplified in the following discussion in response to questions posed by The Atomic Safety Licensing Board during the July 16, 1971 session of the Indian Point Station Unit 2 public hearing.

II. SIGNIFICANCE TO SAFETY OF ASME SECTION III CODE RULES

A. Design Requirements

1. Design Rules for Class A Vessels

The Indian Point 2 reactor vessel was designed in accordance with the Class A rules of Section III of the ASME Boiler and Pressure Vessel Code developed under the sponsorship of the American Society of Mechanical Engineers for specific applications to pressure vessels intended for nuclear power plants. Since the initial publication of this code in 1963, the AEC has closely followed the development of the code rules by active participation in the ASME Subcommittee on Nuclear Power which is responsible for the formulation of safety rules governing design and construction of nuclear power plant components. As a consequence, we have had ample opportunity to familiarize ourselves with the code design bases as well as to evaluate the inherent conservatisms of the design rules.

The ASME Subcommittee on Nuclear Power has a balanced representation of members from nuclear components manufacturers and designers, nuclear power plant architect-engineers, insurance underwriters, state inspectors, National Board of Boiler and Pressure Vessel Inspectors, nuclear power utilities, as well as from the Atomic Energy Commission. In order to carry on its functions and responsibilities, the Subcommittee on Nuclear Power is supported directly by numerous subgroups and working groups (e.g., Subgroup on Materials, Subgroup on Design, Working Group on Vessels, etc.) whose recommendations are subject to formal approval by Subcommittee on Nuclear Power. The Subgroups and Working Groups whose membership approaches several hundred, are composed of experts who have recognized competence and direct experiences in their respective fields and disciplines.

All recommendations and actions voted upon by the Subcommittee on Nuclear Power are printed periodically in the ASME "Mechanical Engineering" publication to invite public comment. When formally approved by the ASME Council, the proposed rules are incorporated into the ASME Section III Code in the form of Addenda.

The rules of ASME Section III - Nuclear Vessel Code introduced for the first time a design approach that recognized the need for the special design consideration associated with the service conditions under which reactor vessels must operate. Unlike the rules of other ASME Codes applicable to power boilers of fossil-fueled plants (ASME Section I) and unfired pressure vessels (ASME Section VIII), the nuclear vessel code (ASME Section III) contains rules which provide safety margins for protection against potential vessel failures which could be caused by metal fatigue, metal embrittlement by irradiation (at the reactor beltline region), and metal overstress at points of major stress concentrations (e.g., vessel nozzles). The ASME Section III Code Rules take into account the fact that different modes of vessel failure may potentially jeopardize the structural integrity of a reactor vessel.

The regulatory staff, has evaluated the adequacy of the design rules of the 1965 Edition of ASME Section III Code which has been applied to the Indian Point 2 reactor vessel. We have investigated the design bases, stress criteria, and methods of analyses employed to determine their applicability and conservatism not only in terms of the margins of safety, but also with respect to design control measures which were applied. Based on this review, we are confident that the resulting design of the Indian Point 2 vessel will provide the degree of safety we considered to be necessary for nuclear reactor pressure vessels.

2. Design Control Measures

Typical design control measures are the requirements imposed by the code rules on both the owner and manufacturer of the reactor vessel. Before the manufacturer could proceed with the design of the vessel, the owner, through his design agent, (in the case of Indian Point 2 vessel, the Westinghouse Electric Corporation) was required to prepare a document identified as the Design Specification. The specification included (1) the specific functions and operating conditions of the reactor vessel, (2) the mechanical and operational loadings which the vessel would be expected to withstand during service, (3) the predicted environmental conditions, such as radiation, to which the vessel material would be exposed, (4) the range of transient conditions expected during reactor heatup and cooldown, as well as during operating periods of plant loading and unloading or step changes in power, (5) the anticipated loadings imposed by upset conditions such as reactor trips, loss of power to operating system components (i.e., recirculating reactor coolant pumps), etc. (6) the dynamic loadings on the vessel which would result in the event of an earthquake occuring in the vicinity of the plant site and (7) loadings from the postulated failure of reactor coolant piping. Such a design specification had not heretofore been required by any other pressure vessel code.

To assure that such design requirements are correctly stated and complete in providing an adequate basis for design, the Code rules further require that the Design Specifications must be reviewed and certified by one or more registered professional engineers competent in the field of design of pressure vessels and related nuclear energy system requirements. Such requirements were fully met in the case of the Indian Point 2 reactor vessel by the Westinghouse Electric Corporation which was responsible for the preparation of the specifications.

With such Design Specifications in hand, the vessel manufacturer (Combustion Engineering, Incorporated) was required by code rules to make a complete stress analysis establishing that the vessel design details developed and used in construction complied with the requirements of the Design Specification as well as with the design rules of the Code. Such analyses were performed for the Indian Point 2 reactor vessel, and compiled as the vessel Stress Report. The report was further reviewed and certified by the manufacturer's registered professional engineer competent in the field of pressure vessel design, after it had been properly and completely reconciled with the design rules of the Code.

The code requirements pertaining to the preparation of the vessel Design Specifications, and Stress Report are recognized by the regulatory staff as fulfillment, in part, of the design control measures specified in the AEC Quality Assurance Requirement of 10 CFR Part 50, Appendix B. These measures of design control provide assurance that the system conditions to which the Indian Point 2 reactor vessel will be exposed in service have been properly communicated to the vessel designer, and, in turn, that no design oversight is committed whose consequences could cause failure of the vessel in service.

3. Protection Against Ductile Failure

To assess the conservatism associated with the design strength incorporated in the construction of the Indian Point 2 reactor vessel, the regulatory staff reviewed the design criteria of the ASME Section III Code with respect to the fraction of the ultimate strength of vessel materials relied upon to sustain the service loadings. It is recognized that one potential mode of vessel failure, namely, ductile yielding, is associated with overstress of vessel material beyond permissible design stress limits to the level of the ultimate strength properties of vessel components.

In order to prevent unacceptable plastic deformation of the reactor vessel and to provide a nominal factor of safety on the ductile burst pressure of the vessel, the ASME Section III design criteria permit the vessel designer to utilize no more than one-third of the ultimate strength of the vessel material when the reactor vessel is subjected to the operating loads such as pressure and other mechanical loads. As an example, the cylindrical shell sections of the Indian Point 2 reactor vessel have been designed to limit the stress to a value not in excess of 26,700 pounds per square inch at design temperature, which compares with an ultimate strength value of approximately 80,000 pounds per square inch for such materials at the corresponding temperature.

Other categories of stress loadings, such as bending, are recognized by the ASME Section III Code design criteria to be additive to the stresses due to pressure, and slightly higher design stress limits are permitted for such combinations.

However, these limits on combined stresses, which derive from proven principles of limit design theory, are not permitted to exceed the yield strength of the vessel material under normal reactor operating conditions in order to prevent undue permanent distortions in localized areas of the vessel. The margins of safety between the design limits and ultimate strength in terms of "collapse" of the vessel section remain substantially similar to those for sections of the vessel associated with only pressure loadings.

The conservatism inherent in these allowable design stress limits for primary loads expected during normal reactor operation can be expressed in terms of the vessel's ductile burst pressure and compared to the operating pressure of the Indian Point 2 reactor vessel. The conservatism may also be expressed in terms of the overstrain required to dilate the vessel to the limits of its ductility before failure might be expected.

In terms of nominal strains imposed on the reactor vessel, a factor of approximately 50 is available between the strains sustained by the vessel during normal reactor operation, and the strains corresponding to the ultimate ductility of the reactor vessel materials. This factor when related to the reactor vessel shell ductility defines the extent of deformation of the metal which would be required to achieve ductile tearing of the vessel. In more practical terms, the energy required to rupture the vessel is approximately 500 times greater than the strain energy contained in the vessel material under normal operating stress. These comparisons of margins in terms of strain and strain energy provide added assurance of adequate design conservatism in design stress limits.

Experimental tests of pressure vessels under static loadings as well as analytical studies indicate that a factor of approximately 2.8 is closely representative of the margin between design pressure and burst pressure. However, tests have also demonstrated that, under dynamic loading pressure (i.e., very rapid pressure rise within the vessel), a factor of 3.0 or greater on pressure is not unreasonable to attain, as the vessel undergoes plastic dilation.

Based on the foregoing considerations, the burst pressure of the Indian Point 2 reactor vessel under static pressure loading would be estimated as 2.8 times the design pressure of 2485 psig, or 6958 psig, and under a dynamic pressure pulse, the burst pressure could rise to 3.0 times the design pressure of 7455 psig.

Such pressure increases in the reactor vessel are, however, not realizable in practice because of the overpressure protection systmes provided to protect the reactor vessel, as well as all components in the reactor coolant pressure system will prevent a rise in pressure in excess of 2735 psig (as required by the ASME Section III Code rules on overpressure protection), by fully opening and discharging the reactor coolant until such time as the pressure drops to the operating level, and by the operation of the Reactor Control and Protection System which functions concurrently to terminate pressure transients.

The regulatory staff believes that the identified design conservatism and margins provided by the ASME Code design stress limits for primary loads with respect to the ultimate strength of the Indian Point 2 reactor vessel are adequate to assure that vessel rupture by ductile yielding is exceedingly unlikely.

4. <u>Protection Against Failure by Metal Fatigue</u>

Another potential mode of vessel failure recognized by the rules of the ASME Section III Code is related to the high localized strains imposed on vessel components (primarily at vessel geometric discontinuities as a consequence of transient conditions which result in cyclic loadings. Temperature changes of the reactor coolant induce cyclic thermal stresses and concomitant strains) which have the potential to initiate and propagate flaws in the vessel. Repetitive thermal cycling, such as may be expected during reactor heatup and cooldown, contributes to metal fatigue. Specific design rules which require a fatigue analysis are an important provision of the ASME Section III Code that takes into account metal fatigue as a potential mode of vessel failure.

The system design transients and their expected number of occurrences (i.e., number of fatigue cycles considered in the design of the vessel) have been specified in the Design Specification for the Indian Point 2 reactor vessel. The vessel designer, by making use of the design fatigue curves contained in the ASME Section III Code and the calculated stress amplitudes which each design transient imposes upon the vessel components, determines the permissible safe number of cycles for vessel operation.

In evaluating the design conservatism included in such fatigue analyses, the regulatory staff examined the basis for the ASME Section III fatigue design curves. Such curves were derived from experimental data based on fatigue tests conducted with specimens of materials representative of those used in the Indian Point 2 reactor pressure vessel. These data were published in 1964 by the American Society of Mechanical Engineers in the "Criteria of the ASME Boiler and Pressure Vessel Code for Design by Analysis in Section III." The data showed that these design fatigue curves were established by applying a safety factor on the representative trend curve for crack initiation and failure in the tested specimens. The number of fatigue cycles to be allowed in designing the vessel was reduced by a factor of not less than 20. The permissible cyclic stress amplitude which could be imposed on the vessel component reduced to not less than 50 percent of the stress amplitude which resulted in crack initiation in the test specimens.

Although these safety factors were applied to take into account environmental factors, such as the reactor coolant chemistry, unanticipated differences in fatigue characteristics with size and geometry of vessel components, and experimental data scatter, additional tests conducted by the Pressure Vessel Research Committee and other organizations have verified the margins in the ASME Section III fatigue design curves.

A more meaningful measure of the design conservatism associated with the design of the reactor pressure vessel to withstand the cyclic service loads can be obtained from the analysis of the extent of cumulative effects of fatigue during its service lifetime. It is recognized that the reactor vessel will, in fact, be subjected in service to a variety of system transients. Each stress cycle has an additive effect on the fatigue damage that the vessel material may experience, provided the strains are sufficient to cause damage.

The cumulative effects of such stress cycles on the vessel's permissible number of cycles are evaluated, as required by the ASME Section III Code rules for fatigue analyses, by comparing the expected number of cycles (n) for each transient with its respective permissible safe number of cycles (N) as determined from the fatigue design curves. The summation of these fractions (n/N) for all transients yields a "cumulative usage fraction" which represents the fraction used of the available safe fatigue life for the reactor vessel materials.

In the case of the Indian Point 2 reactor vessel, the "cumulative usage fractions" for the majority of the vessel components are significantly less than 1.0, which means that the vessel could safely sustain a significantly greater number of fatigue cycles during service than those expected to occur without exceeding the safe limit permitted by the design fatigue curves of the Code.

In consequence of such conservative values, the regulatory staff believes the Indian Point 2 reactor vessel has been designed in accordance with the fatigue design rules of the ASME Section III Code with sufficient margin to assure that the expected number of cyclic loads imposed over its service lifetime will result in negligible damage by metal fatigue.

B. Material Requirements

1. Control of Materials for Reactor Vessels

The ASME Section III Code contains rules which recognize the importance of controlling the quality of all materials which are used to construct Class A vessels (such as the Indian Point 2 reactor vessel). It is the policy of the ASME Boiler and Pressure Vessel Committee to approve only materials whose properties meet the most stringent metallurgical requirements with respect to physical properties, microstructure, weldability, structural stability, the influence of fabrication processes, thermal treatment effects on strength properties and ductility, and the degree of retention of fracture toughness with exposure to operating temperatures and radiation.

In the majority of cases, these materials are identified by detailed specifications issued by the American Society for Testing and Materials (ASTM), and contained in the ASME Boiler and Pressure Vessel Code - Section II - Material Specifications. Weldability of materials must be established by the application of detailed procedures and qualification tests prescribed in the ASME Boiler and Pressure Vessel Code - Section IX - Welding Qualifications. Materials which fail to meet these requirements are not acceptable for construction of pressure vessels within the scope of ASME Section III Code - Nuclear Vessels.

In order to assure that the materials used in the construction of reactor vessels conform to the prescribed specifications, the Code rules require each material manufacturer to certify that all materials furnished meet the requirements of the material specifications. The certification (Mill Test Report) requires a report of the results of physical properties tests (e.g., tensile strength, yield strength, fracture toughness, etc.) and chemical analyses actually conducted for each plate or forging as well as special tests required by the Code rules. In addition, the manufacturer must report the results of nondestructive examinations and weld repairs (if performed) on the materials.

These test, analyses, and examinations are intended to provide assurance to the reactor vessel designer that the strength and quality of the materials furnished to the manufacturer are not below the specified limits upon which his stress analyses are based, and that the materials are free of unacceptable defects. These requirements were met, in all respects, for the Indian Point 2 reactor vessel. In addition, Westinghouse chose to augment the Code requirements for examination of plate materials of the reactor vessel, such as by requiring 100 percent volumetric examination instead of the approximately 40 percent required by the ASME Section III Code, under which rules the Indian Point 2 reactor vessel was built.

2. Minimum Strength Properties of Materials

An added conservatism which has been incorporated in the materials specifications as part of ASME Code rules is the specification of the minimum values of tensile strength and yield strength. Based on a statistical treatment of test data collected for the materials produced by the various materials manufacturers, the extent of variability in the strength properties of each material was determined. From such studies, minimum values were selected in defining the specifications for tensile strength and yield strength. Examination of the actual physical properties reported by the material manufacturers (in the Mill

Test Reports) generally shows strength properties averages 10 percent higher than the minimum considered acceptable for design purposes. Although the vessel designer is not permitted to utilize this higher available margin of strength in designing the reactor vessel, it does contribute an additional conservatism in vessel strength. The regulatory staff believes the application of these material control measures to the Indian Point 2 reactor vessel provides assurance that the materials used in the construction of the vessel possess adequate strength for the intended reactor service.

C. Fabrication Requirements

1. Quality Control in Reactor Vessel Fabrication

To assure that the intended quality level is attained during the fabrication processes applied in manufacturing a reactor vessel such as the Indian Point 2 vessel, the ASME Section III Code rules imposed controls on each stage of fabrication.

Materials used in the construction of components of the reactor vessel are required to carry identification markings to assure that only the specified materials are applied in fabrication. The intent is to prevent the use of improper materials, since material testing after fabrication to verify quality is not practicable.

Vessel parts such as shells, heads, and nozzles, are heated to permit rolling, forming and forging operations, in order to produce the geometrical shapes required for a reactor vessel. Since such heating processes may induce metallurgical changes which could affect the physical properties of the materials, the manufacturer is required to qualify such processes by tests on representative materials. Where the strength of the materials is impaired by the heating processes, special heat treatments are required to restore acceptable properties.

Materials which are discovered during the process of fabrication to contain defects that developed as a consequence of cutting and working are unacceptable, unless the defects are completely removed and repairs and reexaminations are performed in accordance with procedures specified in the ASME Section III Code. All edges of material which are to be joined by welding to other parts are first nondestructively examined to detect flaws, laminations, and inclusions, since these edges become the weld heat affected zones that are most susceptible to crack development.

Since welding operations are recognized as one of the critical fabrication processes in joining parts of the reactor vessel, only qualified welding procedures are permitted to be utilized. In addition, each welder and welding machine operator must demonstrate his capability to perform welds which, upon tests conducted in accordance with the procedure of the ASME Section IX Code, demonstrate strength, ductility, and fracture toughness properties equivalent or superior to the materials joined. The vessel manufacturer is not permitted to proceed with production welding until after welding procedures to be used on the reactor vessel are qualified, and welding personnel have successfully demonstrated acceptable performance.

Despite the controls placed on permissible welding procedures and the performance qualifications of welders, the ASME Section III Code Rules further require that all completed welds in the reactor vessel be volumetrically examined to verify the soundness of each joint. Such requirements are intended to provide assurance of continued maintenance of weld quality during production welding on the vessel. Where unacceptable defects are detected, defect removal, reweld and reexamination are required by the Code.

The welding processes applied in the construction of a reactor vessel generally require preheating of the ferritic materials to be joined, followed by a heat treatment of the weldment after completion of welding. The underlying basis for such thermal treatments is not only to attain welds as free of defects as practical but also to assure welds with the most favorable metallurgical characteristics and with a minimum of residual stresses in the weld metal. For these reasons, the ASME Section III Code prescribes mandatory procedures which must be followed and controlled by the vessel manufacturer in the conduct of heat treatments of weld joints. Since the strength, ductility, and fracture toughness properties of the completed welds in a reactor vessel cannot be subsequently verified by any practical tests (without destroying the completed weld joint by removal of a section of the weld seam for testing purposes), assurance of acceptable weld joint quality is provided by stringent quality control measures exercised during the course of vessel fabrication as required by the Code.

The regulatory staff believes that the application of fabrication rules of the ASME Section III Code to the Indian Point 2 reactor vessel provides adequate assurance that the vessel, as fabricated, possesses both physical and metallurgical properties not significantly different from those verified by tests conducted on representative material and weld metal of the reactor vessel.

D. Inspection and Testing Requirements

1. Inspection Practices

The measure of soundness and quality achieved in the manufacture of a reactor vessel is established directly by the performance of nondestructive examinations, the number and extent of examinations conducted during each stage of fabrication, and the sensitivity of the examination methods employed in detecting flaws in metal.

The regulatory staff has examined the nondestructive examinations requirements as specified by the ASME Section III Code and as applied specifically to the Indian Point 2 reactor vessel. Essentially two categories of examination methods are specified by the codes; namely surface examinations, such as liquid penetrant and magnetic particle methods which are capable of detecting cracks originating on the surfaces of material, and to a limited extent, sub-surface flaws in close proximity to the surfaces, and volumetric examinations, such as radiography and ultrasonic techniques, which serve principally to locate subsurface flaws through the entire volume of metal.

Both examination categories yield results which have safety significance with respect to the influence of flaws upon the structural reliability and integrity of reactor vessels in service. Surface flaws, particularly on the interior surfaces of the vessel are more susceptible to growth by fatigue and stress-assisted corrosion mechanisms than subsurface flaws within the metal thickness because of the generally higher stresses sustained at the interior surfaces, and the exposure to the reactor coolant environment. Subsurface flaws may form the nucleus for crack growth which, if undetected, could enlarge in service and propagate to the surface. Both types of flaws, if permitted to grow to critical size, may contribute to local reduction in vessel strength, and introduce the potential for failure.

2. Nondestructive Examination Sensitivities

In recognition of the importance to safety of minimizing the presence of any flaws in reactor vessels, the ASME Section III Code rules have established stringent examination procedures and acceptance standards. Radiographic examination techniques have demonstrated capabilities to detect flaws in excess of 2 percent of the wall thickness, and less than 2 percent when the flaw irregularities are favorably oriented with respect to the radiation source. Ultrasonic examination techniques possess sensitivities which permit detection of flaws in heavier sections, such as commonly used in reactor vessels, in excess of 3 percent of the wall thickness, and less than 3 percent when the flaws are explored from several angles and direction.

Liquid penetrant and magnetic particle examinations techniques can generally detect extremely shallow surface flaws, which correspond to a much lesser percentage of the wall thickness than detected by the volumetric examination methods. Flaws less than 1/16 inch in depth are considered as nonrelevant and are within the acceptance standards established for these techniques. Experimental tests, as well as analytical evaluations based on the principles of fracture mechanics, have demonstrated that flaws less than 1/16 inch in depth have negligible influence upon the strength and fatigue resistance of materials.

These limits of examination sensitivities form the basis for the acceptance standards established in the ASME Section III Code. The Indian Point 2 reactor vessel has been examined in accordance with such acceptance standards, and records compiled by the vessel manufacturer are maintained as evidence of compliance with the Code rules.

3. Testing Practices

As a final measure of verification of the structural adequacy of the completed reactor vessel, the ASME Section III Code rules require that the vessel be subjected to a hydrostatic test by sealing all nozzle openings in the vessel and pressurizing with water to a value considerably higher than the vessel will experience in service. This test provides a direct means to confirm not only the design adequacy of the individual components to withstand an overload without unacceptable deformation, but also to assure that the leak tight integrity of the vessel is established.

In the case of the Indian Point 2 reactor vessel, the hydrostatic test pressure imposed was 3125 psi which, when compared with the operating pressure of 2235 psi during normal reactor operation, represents an overload of approximately 40 percent. Although this overload may appear substantial, the materials for the major portions of the vessel are not subject to overstrain (and consequent unacceptable distortion) during the test since the test strains remain essentially below the yield strength of the materials.

The adequacy of the hydrostatic test conducted on the Indian Point 2 reactor vessel as proof of its structural integrity for service may be judged by recognition of the fact that the test stress attained at ambient test temperature in the vessel shell, for example, was 67 percent of the minimum specified yield strength of the material which compares with approximately 56 percent of the design yield strength of the material expected during normal reactor operation.

The regulatory staff believes the combination of (a) nondestructive examinations performed on both the materials and welds of the Indian Point 2 reactor vessel during the course of fabrication, (b) the stringent acceptance standards applied which control the final quality of completed vessel, and (c) the conduct of a successful hydrostatic test, provides assurance that no flaws exist which, under service conditions, might influence its safe operation.

III. BRITTLE BEHAVIOR AND RADIATION DAMAGE

A. Protection Against Brittle Fracture

Ferritic steels that are commonly used in the construction of pressure vessels exhibit properties which, under a specific combination of stress, temperature, and the presence of flaws in the metal, may lead to brittle fracture. The ASME Section III Code rules have recognized this potential mode of failure in pressure vessels by including requirements that ferritic materials meet certain levels of fracture toughness.

Brittle fracture is generally associated with those temperature conditions where the materials exhibit a marked reduction of fracture toughness properties. The temperature range for such brittle behavior is usually below 100°F for those steels used in the construction of reactor vessels. To identify the appropriate range for each material of the vessel, the vessel manufacturer conducts impact tests on specimens of the materials in order to establish the nil-ductility transition (NDT) temperature, at which, brittle fracture may generally be expected if the material is subjected to significant loads in the presence of flaws. (The maximum NDT temperature of all materials in the Indian Point 2 reactor vessel is in the unirradiated condition 20°F.)

Since reactor vessels are exposed to temperatures and stresses during initial stages of heatup and final stages of cooldown where the materials of the vessel are approaching the NDT temperatures, (the range of brittle fracture potential), operational limitations of pressure and temperature must be imposed to protect the vessel against brittle fracture.

In its review of the Indian Point 2 reactor vessel, the regulatory staff established that the vessel manufacturer had fully complied not only with the fracture toughness requirements as specified in the ASME Section III Code but also the additional test requirements specified by Westinghouse Electric Corporation. Notwithstanding compliance with these requirements, the regulatory staff examined in detail the fracture toughness measurements taken for each component part of the reactor vessel, including the weld metal which joined the parts, in order to assess the quantititive measure of conservatism and safety margins in establishing operating limitations.

In recognition of quantitative theoretical methods of analysis made available by the application of the principles of fracture mechanics, and the increased knowledge of material properties of ferritic steels derived from both industry and AEC directed research programs, the regulatory staff developed quantitative criteria for fracture toughness requirements for nuclear power reactors. These requirements, which were published in July, 1971 by the AEC, in 10 CFR 50, Appendix G are more stringent than the current ASME requirements and have been fully applied to the Indian Point 2 reactor vessel in establishing the operating pressure and temperature limitations.

The measure of added conservatism resulting from the application of the AEC fracture toughness criteria to the Indian Point 2 reactor vessel may be gained from a comparison of the operating temperature limitations imposed prior to reaching full pressurization of the reactor vessel during reactor heatup. The initial temperature limit derived from the current rules of ASME Section III Code was 136°F, while the AEC criteria required a temperature of not less than 220°F. The AEC temperature limit, which has been accepted by Consolidated Edison, is currently specified in the the licensee's Technical Specification concerning reactor vessel operation.

Operation of the Indian Point 2 reactor vessel at significant pressures only when above the specified temperature of 220°F assures that all vessel materials have conservative values of fracture toughness sufficiently above those values where the potential for brittle fracture may exist.

B. Radiation Effects

Despite the conservative approach taken to establish safe operating limits, an additional requirement was imposed by the AEC regulatory staff to take into account the expected degradation in fracture toughness properties of the beltline region material of the reactor as a result of the effect of radiation from the reactor core during service. Estimation of these radiation effects involves calculations of the predicted neutron fluence to which the vessel material will be exposed over the 40 year service lifetime of the reactor. Such calculations which were performed for the Indian Point 2 vessel using the modified PIMG one-dimensional 55 group diffusion computer code resulted in a neutron fluence value of 2.4×10^{19} n/cm^2 (E > 1 Mev).

Irradiation tests on specimens of the reactor vessel materials demonstrate that, for such neutron fluence, the initial nil-ductility transition shifts to a higher temperature as the neutron fluence increases. In other terms, this effect means the fracture toughness properties of the materials, at the initial specified operational limit of 220°F, will be significantly reduced. Accordingly, with increasing periods of service, the operational limit of the reactor must be adjusted to a higher temperature where the materials will continue to exhibit adequate fracture toughness even after radiation.

Because of the uncertainties associated with the calculations of the neutron fluence, the variability in radiation-induced changes in fracture toughness among the reactor vessel materials, and other indeterminate long-term effects on material properties the AEC requires capsules of specimens of the actual materials used in the construction of the vessel to be placed within the reactor vessel. The Indian Point 2 reactor vessel contains such capsules as part of its material irradiation surveillance program. Withdrawal of these capsules at periodic intervals during service and testing of the irradiated specimens provides a direct means to monitor the changes in materials fracture toughness properties.

To assure a timely adjustment of the operating limitations for the Indian Point 2 reactor vessel, the Technical Specifications will require Consolidated Edison to withdraw the first capsule at the first refueling outage (approximately two years of operation) and report the test results to the Commission. At that time, an adjusted operating limit will be specified for continued service to reflect the measured changes in material properties.

By establishing conservative operating limits initially and periodically during service, the regulatory staff believes the Indian Point 2 reactor vessel can be safely operated under conditions which assure that adequate material fracture toughness properties are always available to prevent brittle fracture.

IV. MONITORING CHANGES IN VESSEL QUALITY BY INSERVICE INSPECTIONS

A. Development of Inservice Inspection Code

The AEC has long recognized that the enhanced quality standards applied in the construction of reactor vessels in accordance with the rules of ASME Section III Code could best be maintained during service if a planned program of inservice inspections was implemented. The AEC regulatory staff accordingly initiated a program to develop requirements for the inservice inspection of nuclear reactor pressure vessels. A comparable effort on the part of industry was also established at the request of the AEC.

These efforts led in late 1967 to a joint AEC-industry cooperative code development program under the auspices of the American National Standards Institute (ANSI) N-45 Committee with the sponsorship of the American Society of Mechanical Engineers. The combined efforts culminated in the publication of the 1970 Edition of the ASME Boiler and Pressure Vessel Code - Section XI. - "Inservice Inspection of Nuclear Reactor Coolant Systems". In recognition of the acceptability of the ASME Section XI Code in fulfilling the requirements of the AEC, the rules of this inservice inspection code were adopted by the AEC with the publication of 10 CFR 50.55a - "Codes and Standards for Nuclear Power Plants."

B. Preoperational Baseline Examination

The ASME Section XI Code requires that the reactor vessel pressure-containing welds be subjected to a nondestructive method of examination as a preoperational requirement prior to initial plant startup. The method of examination employed involves the use of an ultrasonic technique, which permits detection of any significant surface or sub-surface flaws by examining the entire volume of metal contained between the surfaces of plates, forgings, and bars from which the reactor vessel is constructed. Such methods of examination are therefore identified as "volumetric examinations."

The regulatory staff has recognized that these volumetric examinations, which have been applied to the Indian Point 2 reactor vessel, serve two important purposes. First, this preoperational examination, as required by the rules of ASME Section XI, provides a record of the location of any discontinuities in the metal (such as extremely small flaws) that may exist in the reactor vessel welds. The examinations are required despite the fact that the entire welds seams, as well as the base materials of the reactor vessel, were 100 percent volumetrically examined during the course of vessel fabrication and successfully met the acceptance standards.

The intent of the ASME Section XI Code rule is to assure the availability of a record of the initial condition of the vessel's integrity for comparison with the examination results of the planned future inservice inspections. Any discontinuities in the vessel materials will therefore be periodically monitored to detect any tendency of these flaws to grow in service.

Particular emphasis is given to the examination of pressure containing welds of the vessel, since service experiences with welded structures, in general, confirm that weld joints and weld heat-affected zones in base material are potential areas for flaws to initiate and grow under service loadings.

The second and more important purpose served by the preoperational examination is the confirmation and re-verification of the acceptable structural integrity of the reactor vessel following its installation in the plant. Although the reactor vessel has been fully examined during fabrication to meet the acceptance standards of ASME Section III construction code, the vessel may be subjected to loadings during the hydrostatic testing which could alter the vessel's structural conditions. The post-hydrotest preoperational examination of the

Indian Point 2 reactor vessel verified that no defects developed during hydro-testing and confirmed the vessel's quality level as acceptable for reactor operation.

The regulatory staff has assured itself that the inservice inspection program for the Indian Point 2 reactor vessel will comply with the examinations requirements of the ASME Section XI Code, and in most respects, with the inspection frequency required during each ten-year interval. In order to permit time for the development of specialized remote mechanical ultransonic examination devices which will be required to examine those areas of the vessel not readily accessible, the AEC has required Consolidated Edison (in accord with the Licensee's Technical Specifications) to submit its program of inservice inspection for such areas for review by the Commission prior to the expiration of five years of service. The regulatory staff has received assurance from industry that examination equipment for remote inspections can be made available on a timely basis and applied to satisfy the examination requirements of the ASME Section XI Code, within this five-year period.

C. Sensitivity of Inservice Examination Methods

To gain assurance that the sensitivity of inservice examination methods (ultransonic techniques) and the frequency of examinations which are planned for the Indian Point 2 reactor vessel will monitor on a timely basis the growth of a postulated flaw in the vessel before attaining critical size, the regulatory staff has investigated the experimental flaw growth rates data for reactor vessel materials, utilizing the principles of fracture mechanics.

Since ultrasonic examination techniques have demonstrated capabilities to detect flaws in excess of 3 percent of the metal thickness, inservice examinations may fail to locate flaws below this threshold of detectability. When subjected to the fatigue cycles expected in service such flaws may grow on the order of 1/10,000 inches per cycle. With such limited flaw growth per cycle, the number of fatigue cycles required to develop a through-wall flaw (97 percent of wall-thickness of 8-5/8 inches in the case of Indian Point 2 vessel) would be many orders of magnitude greater than the number of transients which the reactor vessel may be expected to experience during the periods between inservice inspections.

On the basis of the relatively insignificant growth rate at which flaws in a reactor vessel may enlarge during normal reactor operation, the regulatory staff believes that the program of inservice inspection developed for the Indian Point 2 reactor vessel will not only assure timely detection of any unanticipated structural degradation in the vessel, but also provide confidence that the probability of any flaw growing unknowingly during the service lifetime to a critical size and resulting in sudden failure is negligible.