CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

Indian Point Station, Unit No. 2

Docket 50-247

INTR

Applicant's Responses to Round Two Questions Submitted by the Citizens' Committee for the Protection of the Environment (Set H) on March 9, 1971

Part I

March 29, 1971

Question No. H-1

Question: In answer C-1 you use the term probability. Define this term as it is used in the answer. Is there any possibility of an explosive rupture of an element of the primary loops?

Answer: In the answer referred to, it was stated that "The conservatism in the design and in the manufacturing process combined with careful operation, strict quality control and quality assurance during every facet of the design and manufacturing process, and a responsible in-service inspection program eliminates the probability of an explosive rupture of the reactor vessel or large elements of the primary loop.

Applicant is aware of no catatrophic failure of nuclear grade vessels."

The definition of the term probability as used in the answer is "likelihood".

In light of the factors listed in this answer, applicant does not foresee any possibility of an explosive rupture.

Question No. II-2

Question: Explain why a double ended pipe break in the hot leg could not involve a rupture in which pieces of metal from the pipe could be propelled against the inside of the containment as a result of the rapid release of pressurized water from the loop.

Answer: The stainless steel used in the reactor coolant piping is a highly ductile material.

Question No. H-3

Question: If the answer to question H-1 is yes, what would be the force in psig with which the largest, the median and the smallest piece (specify size) would strike the containment. In this answer use conservative values at least with respect to the following elements:

a. age of the pipe
b. location of the rupture at a welded joint
c. proximity of the pipe to the containment wall.

Answer: Not applicable. See Answer to Question H-1.

Question No. H-4

Question: If the answer to question H-1 is yes, provide the following information:

a. How many individual fragments would result from this rupture:

b. Have you analyzed the force of these fragments (in psig) and if so what is that force?

c. Have you analyzed the probable route of fragments and if so, how many will come in contact with other equipment or pipes within the containment?

d. With respect to c., have you analyzed the effect of these fragments on the objects they could strike and the result of that collision on the ability of the post-accident function of equipment or pipes within the containment? If so, please provide the analysis in detail.

In this question also use conservative values for the factors specified in question H-3.

Answer: Not applicable. See answer to Question H-1.

Question No. H-5

Question: Answer question H-4 (regardless of the answer to H-1) with respect to the water released by the rupture and also with respect to the broken ends of the ruptured pipe assuming they remain attached to the remainder of the pipe.

Answer: As stated in the FSAR (see Page 4.1-4), the dynamic effects during blowdown following a loss-of-coolant accident were evaluated in the detailed layout and design of the high pressure equipment and barriers which afford missile protection. Support structures are designed with consideration given to fluid and mechanical thrust loadings. The steam generators are supported, guided and restrained in a manner which prevents rupture of the steam side of a generator, the steam lines, and feedwater piping as a result of forces created by a Reactor Coolant System pipe rupture.

These supports, guides and restraints also prevent rupture of the primary side of a steam generator as a result of forces created by a steam or feedwater line rupture.

The mechanical consequences of a pipe rupture are restricted by design such that the functional capability of the engineered safety features is not impaired.

FSAR Appendix 4B gives a brief description of the Reactor Coolant System component support structures.

FSAR Section 5.1.5 describes the design of the supports with loading conditions which include pipe rupture loads. FSAR Q1.5 discusses the effects of combining the pipe rupture loads with seismic loads although this was not a condition for the design.

FSAR Q5.11a, Q5.11b, Q5.11c and Q5.8 provide details of the design of the interior structure of the containment used to support and enclose the primary system including considerations of pipe rupture loads and jet forces.

Question No. II-6

Question: Were the steel plates used in the reactor vessel and the welds for the vessel subjected to ultrasonic testing? Radiograph or X-Ray testing? With respect to all such tests of the plates and the welds provide the following information (Please do not merely refer to the information provided in pages Q4.1.1-1 to Q4.12-1 of the FSAR):

a. At what stage(s) of the manufacturing (including ingot stage) and installation of the plates and the manufacturing and installation of the vessel were the tests conducted and by whom?

b. How much of each plate was tested with the instrumentation perpendicular to the plate, and how much was tested with oblique (shear wave angle beam) shots (note that Tech. Spec. p. 3, 1-5 suggests that only certain plates received 100% testing of both perpendicular and oblique beams)?

c. When were the tests conducted?

d. Were flaws (regardless of whether they were within Code specifications) of any size permitted in the plates and if so, what was the largest size permitted for each kind of plate or weld used in the reactor vessel?

e. Were maps made of the flaws and can their exact location be shown on the reactor vessel as it is now installed? If so, please provide the map.

f. How many flaws and of what sizes exist in the reactor vessel plates and welds?

g. Define the term "indications" in answer D-41.

Answer: All steel plates used in the reactor vessel were subjected to ultrasonic examination and all welds for the vessel were subjected to radiographic or X-ray examination. See FSAR Section 4.5.1 and Table 4.5-1 in particular.

a. Ultrasonic examination of the plate material was performed in:

1) The as-rolled condition (material sub-vendor)

2) After forming into cylindrical shell (vessel Mfr.)

3) After hydrotesting of vessel (vessel mfr.)

b. As stated on Pages 4.2-13 and 4.5-1 of the FSAR, 100% of all plates were required to be ultrasonically examined using both the longitudinal and angle beam techniques.

c. See answer to a. for stages of manufacture, when the tests were conducted.

d. Only flaws smaller than the ASME Boiler and Pressure Vessel Code, Section III acceptable limits were permitted. The size of the largest flaw can be obtained by examining the records which are in the safekeeping of Combustion Engineering in Chattanooga, Tennessee.

e. Maps were made of all discernable flaws; the exact location of these flaws can be shown on the reactor vessel as it is now installed. A copy of the map is available for review.

f. This information can be obtained by examining the records in the safekeeping of Combustion Engineering in Chattanooga, Tennessee.

g. The term "indications" is defined as any defect or discontinuity disclosed by the various nondestructive testing techniques.

In the answer to Question D-41, it was stated "There are no faults in the Indian Point Unit No. 2 reactor pressure vessel since no indications exceeded code requirements. The test results of ultrasonic and other techniques demonstrate compliance with the code. Underlying documents which are voluminous, bulky and in part, non-reproducible with respect to such results are in the safekeeping of the vessel manufacturer Combustion Engineering at Chattanooga, Tennessee."

Question No. II-7

Question: Will ultrasonic testing of plates or welds which are perpendicular to the plate or weld detect all or any cracks that are parallel to the beam of the equipment? If the answer is yes, please explain in detail.

Answer: No.

Question No. H-8

Question: Will radiograph or X-ray testing which is perpendicular to the plate or weld detect a vertical crack if it is less than 2% of the thickness? If the answer is yes, please explain in detail.

Answer: No.

Question No. II-9

Question: If the welds are tested by radiographs or X-ray, what standards are used for approving the weld? For instance, in the 1968 Section 3, ASME Boiler and Pressure Vessel Code for Nuclear Vessels, pages 172-178, it explains that visual comparison of the picture is made with the gauge charts (pp. 174-178) and the gauges show what size and how many flaws can remain. Were these or similar gauges used and if so how many of which size holes in the weld were permitted?

Answer: The acceptance standards for all radiographic examinations were in accordance with paragraph N-624.8 of the Boiler and Pressure Vessel Code, 1965 Edition of Section III. The number of "flaws" or "holes" of the various sizes which were permitted can be obtained by examining the records which are in the safekeeping of Combustion Engineering in Chattanooga, Tennessee.

Question No. H-10

Question: Technical Specifications 4.2 set forth the pre-operational and in-service structural surveillance of the reactor vessel and primary system boundary. With respect to this specification, please answer the following questions (References are to the Tech. Spec.):

a. Will baseline data come exclusively from ultrasonic, visual and surface (please describe) techniques conducted after the reactor vessel is installed? What will be done with the data from earlier tests (see answer to H-6) and will there be any radiograph or X-ray testing for baseline data? (4.2(a)).

b. Define the term "defects" and explain the role of the AEC in evaluating and investigating these defects. (4.2(c)).

c. Describe in detail every difference between the inspection Code referred to in 4.2.3 of the Tech. Specs. and the Code referred to at Q. 4.1.1-1 by the AEC. Attach a copy of each Code.

d. Describe in detail the basis for the claim that ultrasonic testing is an acceptable alternative for radiographic examination. In particular, what kind of flaws (defects, indications, etc.) will be detected by radiograph which cannot be detected or cannot be detected as well by ultrasonic and if there are none, justify your conclusion. (4.2.3(b)).

e. At 4.2-3 of the Tech. Specs. and elsewhere in 4.2 (see 4.2-12; Table 4.2-1; and Notes (1) (4.2-17), you indicate that radiation levels in the reactor vessel, among other factors, present special problems which prevent certain in-service inspections until new equipment is developed. With respect to this, answer, the following:

1. Explain the meaning of answers A-11 and A-24.

2. Describe in detail the present level of development of these testing techniques (and the techniques themselves) including who is now developing them, how far along has development come (design, prototype, full tests of equipment, etc.), any firm commitments that you have on delivery date of these techniques, how design and manufacture procedures have been prepared for these developments, anticipated cost of the new techniques.

3. Justify in detail the delay in in-service testing referred to in Items 1.1, 1.2, 1.3 and 1.7, and what is the outer limit of that delay.

f. Justify the delay in inspection referred to in Item 1.4.

g. Explain in detail how the visual examination referred to in Items 1.5 and 1.6 will be able to detect any internal growth in flaws (defects, indications, porosity) in the welds.

h. Provide a copy of the Code Section referred to in Item 1.15.

i. Justify your refusal to conduct tests referred to in the first paragraph of Item 4.2 both in terms of the impossibility of conducting the test and your belief that such tests are necessary.

Answer: a. A complete baseline inspection is being performed for primary coolant system components which are to be in-service inspected with the exception of the reactor vessel shell and the upper half of the pressurizer which are not accessible because

of plant construction. The techniques being employed are the same as those described in Section 4.2.3 of the Technical Specifications. Earlier test data (shop inspections, etc.) will be used as supplementary reference material. No radiographic or X-ray techniques are planned for the pre-operational examinations.

b. Defect is defined to mean any discontinuity in the weld or base metal which is reportable under the procedures established for pre-service and in-service inspection. The AEC will be duly notified of any unusual indications and will be free to inspect and assess the procedures and results.

c. The applicant has not prepared a detailed listing identifying the differences between the original draft code N-45 and the final issued code ASME Section XI. Our policy is to utilize the latest official published versions of existing codes with regard to In-service Inspections consistent with plant design limitations.

d. Ultrasonic testing is an acceptable alternate for radiograph because it is a volumetric examination covering the complete volume of material scanned. The sound waves may be aimed and received at a number of desired angles to disclose flaws lying in various planes. Ultrasonic examination can penetrate heavy sections more easily and with less equipment than X-radiation. In radioactive areas, film for radiographs would be fogged by the background activity. The types of defects that cannot be detected or be detected as well by UT are cracks or indications parallel to the path of the sound. However, all UT examinations use more than one sound direction. Radiographics on the other hand can only detect laminar flaws which are parallel to the rays. The use of ultrasonic tests for in-service inspections is consistent with the applicable code ASME Section XI.

e. 1. The answer to Question A-11 stated "The proposed Technical Specifications for Indian Point Unit No. 2 require periodic testing of

safety systems (Section 4.0, Surveillence requirements). Plant operation will not interfere with testing of safety systems as required by the Technical Specifications. Con Edison will shutdown Indian Point Unit No. 2 if this is necessary in order to conduct required testing."

The answer to Question A-24 stated "There are no parts of the reactor which may not be entered for inspection following any testing or operation because of induced radioactivity in components. However, at the center section of the reactor vessel, remote inspection may be preferable due to levels of induced radioactivity."

All safety tests will be performed as required by the plant Technical Specifications. If a particular test requires that the plant be shutdown and it cannot be scheduled during routine plant outages, the plant will be shutdown to perform the test in compliance with the Technical Specifications.

All parts of the reactor containment building will be accessible for inspection following ordinary operation and testing. However, in order to reduce personnel radiation exposure, certain internal sections of the reactor pressure vessel and its associated piping will usually be inspected using remote methods.

2. The remote testing methods are under development by Combustion Engineering, Babcock and Wilcox, and Southwest Research Institute. Techniques include the remote ultrasonic testing of reactor vessel shell welds from the inside surface of the vessel, the remote examination of the reactor vessel nozzle welds and nozzle safe-end welds from the inside surface. Southwest Research Institute has performed remote inspections at two nuclear facilities, one foreign and one domestic. The results, while noteworthy, were less than satisfactory. Difficulties with equipment fit up and operations were encountered. Personnel in some cases were exposed to their maximum limits of radiation.

Until a more productive demonstration of the remote inspection has been established, Con Edison has no firm commitments for purchase of such services. We cannot at this time estimate the cost of such an inspection program. However, we are committed to utilizing some form of remote inspection within the time interval allowed by the Code.

3. Items 1.1, 1.2, 1.3 and 1.7 are delayed because the means to perform the inspections do not now exist commercially as explained in e.2. above. We anticipate and are committed to conduct such inspection within the ten-year inspection interval.

f. Item 1.4 is delayed because the means to perform the examination do not exist commercially as explained in e.2. above.

g. Code Category E-1; Item 1.5 is not applicable to Indian Point Unit No. 2.

Category E-1 includes control rod drive and instrumentation penetrations. These items are clusters of tubes utilizing partial penetration welds. Because of the design of the vessel penetrations and the pressure boundary weld, no meaningful volumetric examinations can be performed.

Code Category E-2; Item 1.6 requires only visual inspection. It is our intent to utilize this method in these areas, because of the limitations on volumetric inspections as outlined in E.3.g Category E-1 above.

h. See Code sections attached; Page 15, Category N and Page 16, Item 1.15.

i. Welds which are physically not accessible due to high radiation or structural impossibility will not be examined. Indian Point Unit No. 2 was designed and constructed before the ASME XI Code was developed. At the time of code inception, the plant design was completed. The intent of the code for later plant designs is to take these code requirements into account during the design stage.

Socket welded, sampling and instrument piping and thermowells are all of small size and are not as relevant to plant safety as is the larger piping socket welded, piping UT tests are precluded because of joint geometry.

RULES FOR INSERVICE INSPECTION OF NUCLEAR REACTOR COOLANT SYSTEMS

1970 EDITION

January 1, 1970

FORMULATED BY THE BOILER AND PRESSURE VESSEL COMMITTEE
SUBCOMMITTEE ON NUCLEAR POWER
SUBGROUP ON INSERVICE INSPECTION (SG-IHI)

THE AMERICAN SOCIETY OF MECHANICAL ENGINEERS
UNITED ENGINEERING CENTER
345 EAST FORTY-SEVENTH STREET, NEW YORK, N.Y. 10017

TABLE IS-251 (Cont.)
EXAMINATION CATEGORIES

AREAS SUBJECT TO EXAMINATIONS	EXTENT AND FREQUENCY OF EXAMINATIONS
L-2 PUMP CASINGS	
The areas subject to examination on pump casings shall include the internal pressure boundary surfaces.	The internal surfaces of one disassembled pump, (with or without pressure-containing welds) in each of the group of pumps performing similar functions in the system shall be visually examined during each inspection interval. The internal examinations may be performed on the same pump selected for the volumetric examination of pressure-containing welds. The examinations of pump casings may be performed at or near the end of the inspection interval.
M-1 PRESSURE-CONTAINING WELDS IN VALVE BODIES	
The areas subject to examination shall include the pressure-containing welds in valve bodies, on valves three inches and over in nominal pipe size. The area to be examined shall include the weld metal and the base metal for one wall thickness beyond the edge of the weld.	The examinations performed during each inspection interval shall include 100 percent of the pressure-containing welds in, at least, one valve (with pressure-containing welds), in each group of valves of the same constructional design, (e.g., globe, or gate or check valve) manufacturing method and manufacturer, performing similar functions in the system, (e.g., containment isolation, system overpressure protection, etc.). The examination on valve bodies may be performed at or near the end of each inspection interval.
M-2 VALVE BODIES	
The areas subject to examination shall include the internal pressure boundary surfaces, on valves three inches and over in nominal pipe size.	The internal surfaces of one disassembled valve (with or without pressure-containing welds) in each of these groups of valves of the same constructional design, manufacturing method, manufacturer and performing similar functions in the system, shall be visually examined during each inspection interval. The internal examination may be performed on the same valve selected for the volumetric examination of the pressure-containing welds. The examination on valve bodies may be performed at or near the end of each inspection interval.
N INTERIOR SURFACES AND INTERNAL COMPONENTS OF REACTOR VESSELS	
The areas subject to examination shall include those interior surfaces of the reactor vessel, the internal components, the internal support attachments welded to the vessel wall and the space below the reactor core and above the bottom head, which are made accessible for examination by the removal of components during normal refueling outages.	The examinations of the interior surfaces of the vessel, the internal components and the space below the reactor core shall be performed at the first refueling outage and during subsequent refueling outages at approximately three year intervals. The examination of internal support attachments welded to the vessel wall whose failure may adversely affect core integrity shall be examined at least once during each inspection interval. Visual examination shall cover additional selected points throughout the vessel to provide a reasonably representative sampling of the condition of the cladding.

TABLE IS-261
COMPONENTS, PARTS AND METHODS OF EXAMINATION

Item No.	Examination Category Table IS-251	Components and Parts to be Examined	Method
		Reactor Vessel and Closure Head	
1.1	A	Longitudinal and circumferential shell welds in core region	Volumetric
1.2	B	Longitudinal and circumferential welds in shell (other than those of Category A and C) and meridional and circumferential seam welds in bottom head and closure head (other than those of Category C)	Volumetric
1.3	C	Vessel-to-flange and head-to-flange circumferential welds	Volumetric
1.4	D	Primary nozzle-to-vessel welds and nozzle-to-vessel inside radiused section	Volumetric
1.5	E-1	Vessel penetrations, including control rod drive penetrations and control rod housing pressure boundary welds	Volumetric
1.6	E-2	Vessel penetrations, including control rod drive penetrations and control rod housing pressure boundary welds	Visual
1.7	F	Primary nozzles to safe-end welds	Visual and Surface and Volumetric
1.8	G-1	Closure studs and nuts	Volumetric and Visual or Surface
1.9	G-1	Ligaments between threaded stud holes	Volumetric
1.10	G-1	Closure washers, bushings	Visual
1.11	G-2	Pressure retaining bolting	Visual
1.12	H	Integrally welded vessel supports	Volumetric
1.13	I-1	Closure head cladding	Visual and Surface or Volumetric
1.14	I-1	Vessel cladding	Visual
1.15	N	Interior surfaces and internals and integrally welded internal supports	Visual
		Pressurizer (PWR Plants)	
2.1	B	Longitudinal and circumferential welds	Visual and Volumetric

Question No. II-11

Question: Explain in detail the manner in which the following factors taken together and separately can affect the growth of flaws (indications, defects) in the reactor vessel including its welds and the primary piping system (if there is no effect, justify the conclusion; if there is, in your opinion, an insignificant effect, justify the conclusion regarding the extent of the effect and the insignificance of the effect).

a. Long term (10, 20, 30 years) exposure to the 550-650 degree temperatures of the primary coolant;

b. Long term (10, 20, 30 years) exposure to the radioactivity in the primary coolant - Supply a copy of the report referred to in Answer C-5;

c. The impact of emergency core cooling water on the reactor internal and external walls in the case of double ended cold leg break. For this answer provide also an analysis using the formulae in D-50 (In Answer D-50 (page 2) to what does "stainless steel cladding" refer) as well as the following formula from Reference 2 to that answer:

$$\text{sigma theta} = E \frac{(A.\ T)}{2} (1-hu)$$

E = Young's modules of elasticity
A = coefficient of thermal expansion
T = temperature
nu = poisons ratio

In this case provide the following analysis:

1) temperature of the interior of the reactor walls for each second following the break;

2) level of the water in the reactor for each second following the break (or confirmation of the relevance of FSAR Figures 14.3.2-1 and 14.3.2-5);

3) temperature of the emergency cooling water (both accumulator and the main supply) at the earliest possible moment of contact with any uncovered (with water) portion of the reactor wall and time at which contact will be made;

4) total stress on the reactor wall at the point of contact as well as analysis of the total effect (in terms of pressure created) within the reactor of the cooling water contacting the reactor walls (assume the contact occurs at a point on a plate where the maximum permissible flaw (defect, indication) exists for a reactor in operation for 25 years - make the same assumption for contact with a weld;

5) all other relevant factors which will demonstrate the maximum possible stress at the weakest, possible point; and

6) answer the question with respect to the simultaneous impact of cooling water on the exterior of the reactor vessel as a result of the pipe break and the containment spray.

Answer: a. There is an insignificant effect on the growth of flaws from long-term (40 years) exposure to primary coolant at 550°F.-650°F. This is based on data obtained from a Westinghouse Nuclear Energy Systems experimental program. Preliminary results of the program indicate that flaws of the size allowed under Code acceptance criteria, i.e., less than 2 percent of the wall (plate) thickness, will grow less than 4 mils during the life of the reactor vessel.

b. The only radioactivity that effects flaw growth in the reactor vessel and primary piping material is by neutron bombardment of the material. As there is no neutron irradiation from the primary coolant, there is no effect on the growth of flaws due to long-term exposure to the radioactivity in the primary coolant. The latest report on the Heavy Section Steel Technology Program (WCAP-7561) is being furnished separately.

c. See response to Question D-50 which is attached.

The analysis described in answer to Question D-50 is the appropriate analysis for the evaluation of the effects of cooling the reactor vessel with emergency core cooling water.

Figure 1, attached, gives the temperature distributions through the vessel wall obtained by that analysis as a function of time after initiation of cooling.

The analysis assumes that the inside surface of the vessel wall is instantaneously exposed to cold water at a temperature stated in D-50.

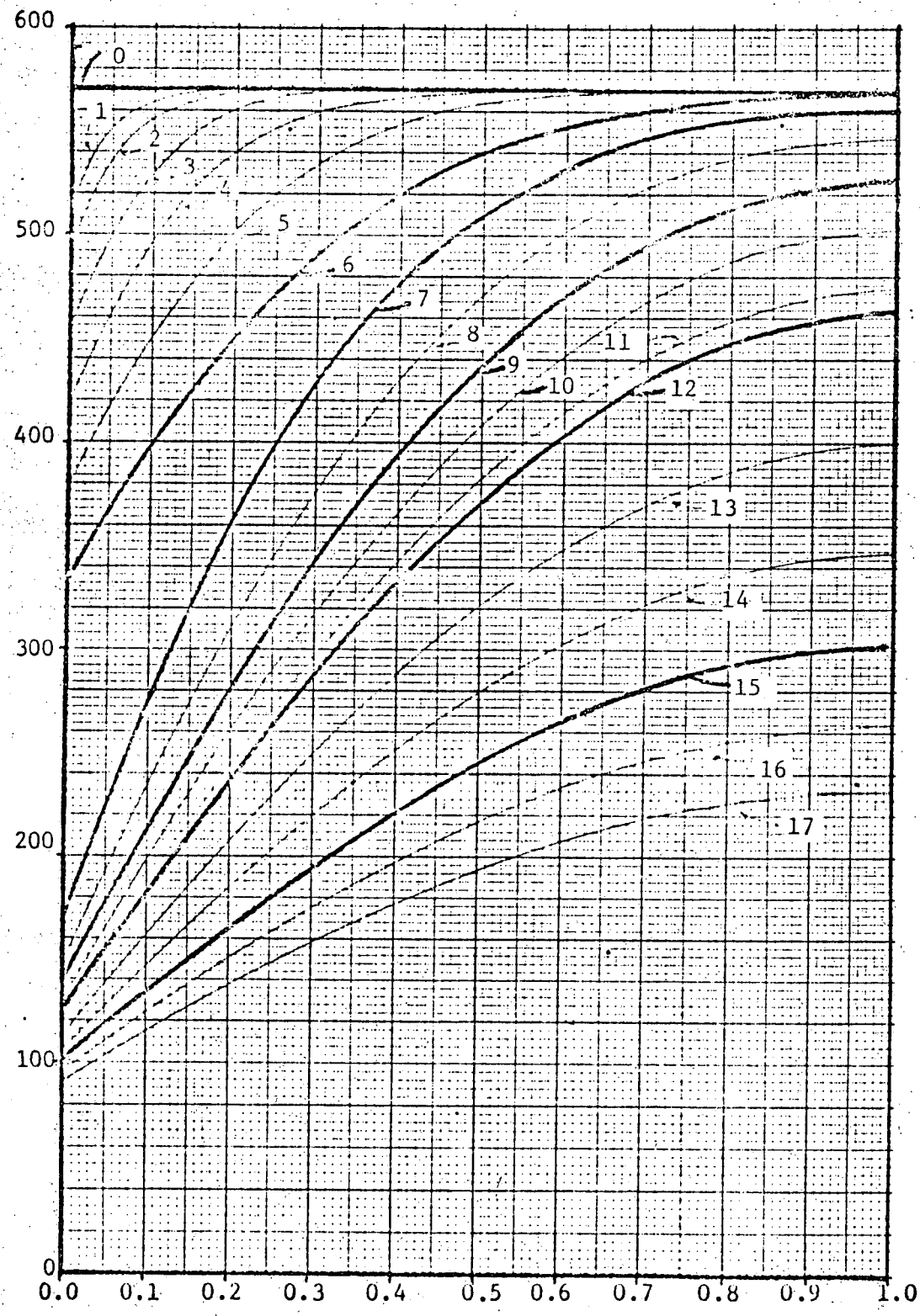
Applicant assumes the intervenor is referring to Equation No. 250 in Reference 2 of the answer to Question D-50. That expression is derived from the simultaneous equations used in the analysis described in answer to Question D-50 by making a number of simplifying assumptions. The stress curves given in answer to Question D-50 give the stresses calculated without the simplifying assumptions. Also, Equation No. 250 gives only the stresses at the inner and outer surface of the vessel wall. It does not give the stresses at other points in the wall which are needed for the evaluation as did the analysis described in answer to Question D-50.

The "stainless steel cladding" refers to the layer of stainless steel which is weld-deposited on the interior surfaces of the reactor vessel. Figure 1 of the answer to Question D-50 gives

the total stresses not only for the point of contact with the cooling water, but for all other points in the vessel wall. The stainless steel cladding precludes contact of the cooling water with the reactor vessel, base metal and associated welds, during vessel operational lifetime.

Calculation of stresses for contact of water on the inside only is conservative. Contact of cooling water on both sides of the vessel wall produces more uniform cooling and reduces the overall temperature gradient across the vessel wall thus reducing thermal stresses.

0 - 0 SECONDS
1 - 10 SECONDS
2 - 20 SECONDS
3 - 50 SECONDS
4 - 100 SECONDS
5 - 200 SECONDS
6 - 400 SECONDS
7 - 600 SECONDS
8 - 800 SECONDS
9 - 1000 SECONDS
10 - 1200 SECONDS
11 - 1400 SECONDS
12 - 1500 SECONDS
13 - 2000 SECONDS
14 - 2500 SECONDS
15 - 3000 SECONDS
16 - 3500 SECONDS
17 - 4000 SECONDS



FRACTIONAL DISTANCE THROUGH THE WALL

Fig 1 Hile

Question No. D-50

Furnish a stress analysis (thermal) for the DBA conditions where all the water has been expelled and the vessel is at 600°F and 80°F water is introduced in the flooding mode.

Answer: THERMAL STRESSES

The stresses in the reactor vessel due to a radial temperature gradient are somewhere between the plane stress lower bound (thin disc) and the plane strain upper bound (long cylinder). In this calculation plane strain is assumed for conservatism.

The material properties will be assumed uniform, at any given time, having the values specified for the mean temperature of the wall at that time. These values are taken form Reference (1).

Reference (2) gives the following expressions for thermal stresses in a long cylinder due to a radial temperature gradient:

$$\left.\begin{aligned}
\sigma_r &= \text{radial stress} = \frac{\alpha E}{(1-\nu)}\frac{1}{r^2}\left[\frac{r^2-a^2}{b^2-a^2}\int_a^b Trdr - \int_a^r Trdr\right] \\
{}^*\sigma_\theta &= \text{circumferential stress} = \frac{\alpha E}{(1-\nu)}\frac{1}{r^2}\left[\frac{r^2+a^2}{b^2-a^2}\int_a^b Trdr + \int_a^b Trdr - Tr^2\right] \\
\sigma_z &= \text{axial stress} = \frac{\alpha E}{(1-\nu)}\left[\frac{2}{b^2-a^2}\int_a^b Trdr - T\right]
\end{aligned}\right\} \quad (1)$$

The temperatures were calculated using equation 2, and the stress profiles were then calculated by equation 1. The results are shown in figure 1.

2.0 TEMPERATURE

Assumptions:

1) Thermal properties remain constant with temperature and are uniform throughout the vessel wall,

* Typographical errors in original equations submitted have been corrected.

2) Vessel wall is perfectly insulated at the back face ($r = b$),

3) At time zero, the vessel wall is at uniform operating temperature. At this time the inside surface is exposed to the safety injection water inducing a convective environment at a constant temperature of 70°F.

The heat transfer probelm reduces to the following one dimensional, transient conduction problem. (reference 3)

$$\frac{\partial T}{\partial t} = \left(\frac{k}{\rho c_p}\right)\left(\frac{\partial^2 T}{\partial r^2} + \frac{1}{R}\frac{\partial T}{\partial r}\right) \quad ; \quad T = T(r,t) \qquad (2)$$

Initial Condition i) $T(r,o) = T_o$

Boundary Conditions ii) $K \frac{\partial T(a,t)}{\partial r} = h\,[T(a,t) - T_w]$

iii) $\frac{\partial T(b,t)}{\partial r} = 0$

Where the temperature ($\bar{T}$) is a function of radial position (r) and time (t) only, and the thermal diffusivity ($k/\rho c_p$) for SA-302 B steel is 0.45 ft^2/hr. T_o is the initial operating temperature (570°F), T_w is the safety injection water temperature (70°F) and k is the thermal conductivity for SA-302 B steel (26.4 Btu/hr-ft-°F).

The following variation of heat transfer coefficient (h) is used to take into account the three separate convection regimes.

The effect of the stainless steel cladding is taken into account by means of an equivalent overall heat transfer coeffieient (h_{tot}).

$$h_{tot} = \frac{1}{\delta/k_1 + 1/h} \qquad (3)$$

where δ is the cladding thickness (.21875 in), k_1 is the thermal conductivity of the cladding (10.0 Btu/hr-ft-°F), and h is the heat transfer coefficients shown in Figure 2-2. Substitution of these numerical values into equation (3) yields the following overall heat transfer coefficients for the three convection regimes:

Regime I ($h = 1000$ Btu/hr-ft^2-^{o}F)

$h_{tot} = 354$ Btu/hr-ft^2-^{o}F

*Regime II ($h = 10{,}000$ Btu/hr-ft^2-^{o}F)

$h_{tot} = 520$ Btu/hr-ft^2-^{o}F

*Regime III ($h = 100$ Btu/hr-ft^2-^{o}F)

$h_{tot} = 84.6$ Btu/hr-ft^2-^{o}F

An implicit finite difference scheme (Ref. 4) has been used to integrate equation (2) for temperature as a function of radial position and time ($T(r,t)$).

The whole process has been computerized to increase the speed of calculation Computerization also allows much smaller increments to be used in both the finite difference scheme for getting temperatures and the trapezoid rule integration for getting stresses, thus improving the accuracy as well.

References:

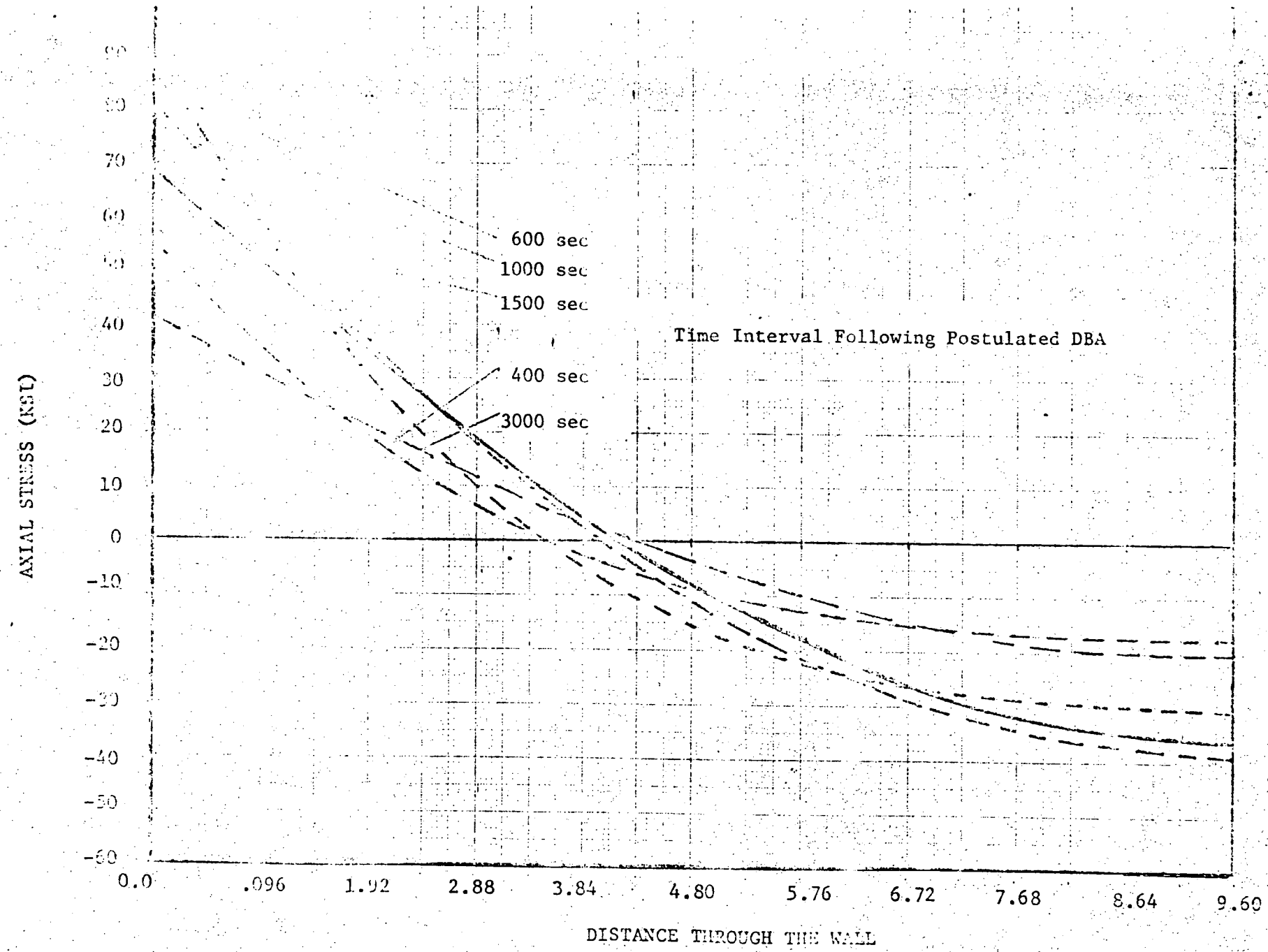
1. Tentative Structural Design Basis for Reactor Pressure Vessels and Directly Associated Components PB-151987

2. Timoshenko and Goodier, Theory of Elasticity, McGraw-Hill Book Co., 1951

3. U.S. Arpaci, Conduction Heat Transfer, Addison-Wesley Publishing Co., 1966

4. G.W. Crandall, Engineering Analysis, McGraw Hill Book Co. 1958

* Typographical errors in original equations submitted have been corrected.


AXIAL STRESS (KSI)
90
80
70
60
50
40
30
20
10
0
-10
-20
-30
-40
-50
-60
600 sec
1000 sec
1500 sec
400 sec
3000 sec
Time Interval Following Postulated DBA
0.0
.096
1.92
2.88
3.84
4.80
5.76
6.72
7.68
8.64
9.60
DISTANCE THROUGH THE WALL

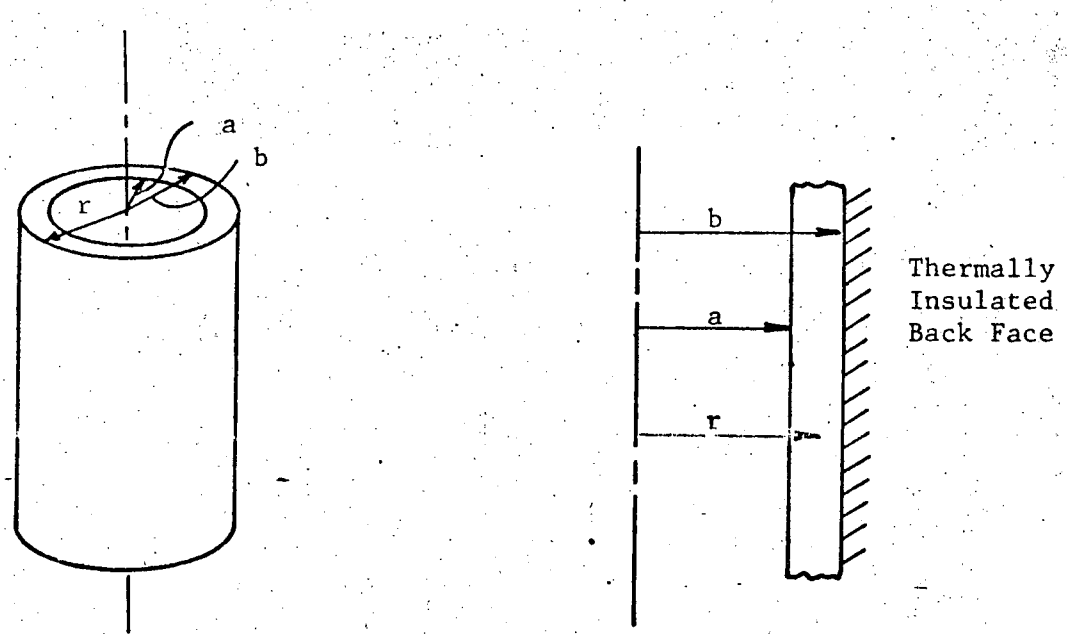


Figure 2-1

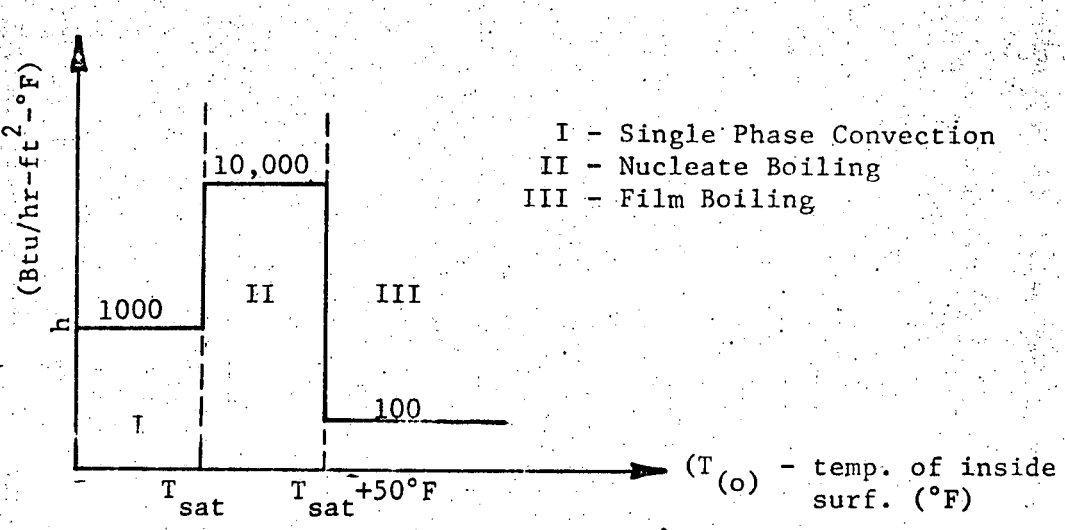

(Btu/hr-ft^2-°F)
h
1000
10,000
100
I
II
III
I - Single Phase Convection
II - Nucleate Boiling
III - Film Boiling
T_{sat}
T_{sat}+50°F
($T_{(o)}$ - temp. of inside surf. (°F)

Figure 2-2

Question No. II-12

Question: Justify the substantial time lag between the examination of the irradiation samples and relatively few samples used for purposes of adequately keeping track of the shift in NDTT. See Tech. Spec. 3.1-6 and 4.2-16. Explain in particular, inter alia how the samples will adequately detect the presence of unusually high radiation leakage from a specific area of the reactor near a specific section of the vessel wall. Also explain the manner in which answer to 14.3.1-1 is relevant to this. Why does that question mention 8 samples and the Tech. Specs. (4.2-16) refer to 6 samples?

Answer: The design of the reactor vessel irradiation surveillance program provides eight irradiation capsules containing a total of 261 specimens of Indian Point Unit No. 2 reactor vessel material. Four of these capsules, containing a total of 130 specimens of Indian Point Unit No. 2 reactor vessel material, are located between the reactor and the vessel wall, one at each of the points where the reactor is closest to the vessel wall; hence, where the integrated flux on the vessel wall is the highest. Because these capsules are closer to the core than is the vessel wall, they experience radiation exposure that leads the vessel wall, making it possible to determine the effects of irradiation of vessel material before the vessel wall experiences the same amount of irradiation. As stated in the

answer to FSAR Q4.1.1, those specimens lead the reactor vessel by a factor of 2.6, and these four capsules would experience the equivalent of 2.6 times as many years of exposure as that actually experienced by the vessel. The withdrawal program is such that the measured shift in NDTT on the irradiated test specimens will confirm the conservatism of the predicted shifts in NDTT over the life of the reactor vessel, as shown in FSAR Figure 4.2-9. Three separate test capsules are considered adequate for this purpose in accordance with criteria set forth in ASTM E-185 "Surveillance Tests for Nuclear Reactor Vessels." However, as noted in the Technical Specifications (4.2-16), six capsules are withdrawn and evaluated as part of the Surveillance Program.

Question No. H-13

Question: With respect to the answer to Question 4.8 and the reactor vessel stress analysis explain in detail whether:

a. the calculations were made with respect to the particular reactor vessel involved in Indian Point No. 2 or only with respect to that type of vessel. If the latter justify this decision.

b. the calculations take account of the presence of flaws (defects, indications) in the vessel and their growth as a result of the factors discussed in H-11. If not, justify the validity of the analysis and the answer.

c. how the fact that actual shift in NDTT has to await periodic examinatin of test samples (Tech. Spec. 3.1-6) affects the validity of the analysis and the answer.

Answer: a. The calculations were made with respect to the as-built Indian Point Unit No. 2 reactor vessel.

b. The ASME Boiler and Pressure Vessel Code, Section III, which is the controlling document for the calculations, takes into account the presence of flaws in the vessel. Criteria of Section III consider crack growth.

c. The predicted shift in NDT is computed by techniques described in FSAR Appendix 4A and Figure 4.2-9; the actual NDTT shift is evaluated

by examination of test coupons as described in FSAR Section 4.5.1 (refer to Question H-12 for additional discussion of the test coupons). The operating limitation on the vessel during heatup and cooldown (i.e., pressure and temperature relationships) affected by the shift in NDTT are periodically revised to stay within the stress limits as stated in FSAR Section 4.3.1. The operating temperature of the reactor vessel as used in the stress analysis always exceeds the highest anticipated DTT during the life of the plant. Neither the validity of nor the answers to the vessel stress analysis are affected by the NDTT shifts.

Question No. H-14

Question: Discuss in detail the data which supports the conclusions which comprise the answer to Question 4.9.1. On the basis of the answer it will be determined whether a request will be made to see WCAP-7332.

Answer: Attached is a copy of Section 4 of the report WCAP-7332 referenced in the response to the answer to Question 4.9.1. Also attached is Reference 12 referred to on Page 4-2 which was taken from:

"Effects of Radiation on Structural Metals"
ASTM Special Publication, No. 426,
Philadelphia, 1967

SECTION 4

THERMAL STRESSES

In the event of a double-ended break and subsequent operation of the emergency core-cooling system, cold water is injected from the accumulators, through the inlet nozzles and downcomer, to the core. For approximately the first ten seconds, the water from the accumulators will not reach the reactor, and the temperature remains unchanged. Consequently, for the period of time in which the internals are excited by the wave propagation due to the break, no thermal stresses or thermal distortion occur. After ten seconds the accumulators are injecting cold water into the reactor and differences in temperatures could result.

a. Cold-leg break: The water is injected from the three unbroken inlet nozzles and cools the external surface of the barrel, which is hot. Conservatively assuming the barrel surface is instantaneously cooled to the water temperature, the stresses are still acceptable, as shown below.

$$\sigma_{max} = \frac{\alpha E}{1-\nu} (T_{barrel} - T_{water}) = \frac{10 \times 10^{-6} \times 25 \times 10^{6}}{.7} (550° - 120°) =$$

$$= 154{,}000 \text{ psi}$$

$$\varepsilon_{max} = \alpha (T_{barrel} - T_{water}) = 10 \times 10^{-6} \times (550° - 120°) = 0.43\%$$

This maximum stress is below the allowable given in Section III of the ASME code, which is a very conservative limit for this type of accident. Following the code criteria for design conditions the barrel could be subjected to a stress amplitude of 650,000 psi during ten cycles; consequently, the peak value of 154,000 psi obtained previously gives a large margin of safety.

Barrel distortion due to nonuniform cooling of the walls around the circumference will not affect the injection of the cold water. If there is any asymmetry due to localized cooling it will distort the barrel in a favorable direction (the distance between barrel and vessel in front of the inlet nozzles will increase). If the accident occurs at the end of the reactor life, when the barrel has been irradiated for a long period of time, the analysis performed previously is still valid in view of the stainless steel's residual ductility after irradiation (material still shows approximately 50-60 percent reduction in area per Reference 12 p. 495). In comparison, the lower support structure temperature differences will be much smaller because the water reaching the lower structure has heated to 270°F and, in addition, cools the support casting and lower core plate more uniformly because of the large and numerous flow holes. Consequently thermal stresses will be lower.

b. Hot-leg break. This case is similar to that of the cold-leg break as regards barrel thermal stresses. It will be less severe because the water cooling the outside barrel wall will be warmer.

Experimental Results

Impact Tests

Results from tests on both the subsize Izod specimens and Charpy V-notch specimens from Capsule BM1-24-14 irradiated to 5.75 × 10^{21} n/cm² were reported previously [5]. Four additional impact tests, two on unirradiated and two on irradiated Charpy V-notch specimens were

TABLE 4 *Tensile properties of irradiated and unirradiated Type 347 stainless steel tested at room temperature, 316 C, and 750 C.*

Total Fast Flux (>1 Mev), 10^{21} n/cm²	Condition	0.2 Per Cent Offset Yield Strength, 1000 psi	Ultimate Tensile Strength, 1000 psi	Fracture Strength, 1000 psi	Elongation, Per Cent		Reduction in Area, Per Cent
					Uniform	Total	
		Tested at Room Temperature					
0	unirradiated	34.5 to 39.3	87.5 to 92.8	...	>40	61.8 to 66.3	72 to 76
5.5	irradiated	109	111.5	252	23.0	32.7	70
16	irradiated	114	114	257	25.0	33.2	69.8
16	irradiated	112	112	251	26.0	33.8	79.1
21	irradiated	113	113	236	24.3	34.5	67.2
16	annealed 1 hr at 982 C	32.3	84.8	194	50.9	56.6	66.3
		Tested at 316 C					
0	unirradiated	26.8	64.2	...	31.9	36.0	...
5.5	irradiated	82.5	84.0	149	11.2	16.3	60
16	irradiated	86.2	86.2	139	8.0	13.7	58.6
16	irradiated	83.6	84.3	148	8.0	14.0	60.2
21	irradiated	86.0	86.0	142	5.0	10.2	56.8
16	annealed 1 hr at 982 C	20.2	58.2	104	27.8	32.6	55.5
		Tested at 750 C					
0	unirradiated	22.0	33.0		16	26	...
21	irradiated	35.0	35.0		0.5	0.5	0
21	annealed 1 hr at 982 C	32.0	32.2		[illegible]	[illegible]	[illegible]
21	annealed 1 hr at 1350 C	19.0	20.2		0.5	0.5	0

made at 316 C. Results of these tests along with the previous results are given in Fig. 2. It can be seen that the impact properties of the irradiated material at 316 C are about the same as they were at 25 C. At both test temperatures, the appearance of the fracture is 100 per cent fibrous which is characteristic of a ductile failure. The most pronounced effects of irradiation are the predominately brittle behavior at −196 C and the lower values of absorbed energy obtained from irradiated specimens at all test temperatures. A somewhat less pronounced effect is the smaller lateral contraction values noted for the irradiated material when compared with those for the unirradiated material.

Question No. II-15

Question: The answer to Question 4.10 indicates that Class I plant components are designed to the ASME Code prior to 1968. In ORNL-NSIC-21 (Technolofy of Steel Pressure Vessels for Water-Cooled Nuclear Reactors) the following comments appear with respect to these ASME Codes:

1. p. 150

   The maximum temperature at which light water reactor pressure vessels are designed to operate is 650 degrees Fahrenheit. No problems attributable solely to the loss of tensils properties due to temperature are anticipated for the materials being used in the construction of nuclear pressure vessels provided the steels possess at least the minimum tensible properties stated in Table N-424, Section 3 of the ASME Code. Adherence to these properties can be assured by imposing supplementary requirements on the materials suppliers such as those given in 5-7, high temperature tension tests of ASTM spec. 8533.

   At least one pressure vessel customer currently requires that tensile data be obtained at 550 and 650 degrees Fahrenheit for the shell plate material as part of the fabrication test program. (Emphasis added).

2. p. 51

   Another area of concern is the relevancy of present requirements of authorized inspectors, as established by the National Board of Boiler and Pressure Vessel Inspectors, with regard to nuclear pressure vessels. The existing requirements are heavily weighted toward the needs of Sections 1 and 8, rather than 3. Consequently, presently qualified inspectors may not have sufficient understanding of the design requirements and non-destructive test methods required for nuclear vessels. We therefore recommend the upgrading of qualifications of code inspectors of nuclear pressure vessels to a level of competency achieved by professionally educated and experienced personnel. (Emphasis added).

3. p. 410

In order to assure that an adequate stress analysis of the vessel has been made, the Section 3 rules stipulate that a stress report be prepared, certified by a professional engineer and filed with the proper authorities at the point of installation. The rules also provide that experimental stress analysis methods, either strain gauge or photoelastic, may be used to verify specific design areas, when theoretical calculations are unavailable, or for determining fatigue reduction factors for cyclic operation. The results of such tests are to be included in the design report. The code specifies only that a complete set of stress analysis calculations shall bymade and reported. It deos not specify that the calculational methods used must yield correct or conservative results as verified by experimental data, or that such evidence shall be offered in support of the calculations. The code does require that the stress report be certified by a registered, professional engineer experienced in pressure vessel testing. The Code does not specifically say that the professional engineer must be experienced or qualified in stress analysis. The inspector who affixed the code stamp is specifically not responsible for the completeness or correctness of the design calculations as set forth in this stress report. (emphasis added).

With respect to the underlined material, indicate whether the additional requirement suggested has been applied to Indian Point No. 2. If so, how and if not, why not.

Answer: 1. The Subcommittee on Properties of Materials of the ASME reviewed material property data to set design stress intensity values (S_m). These values are based on a study of trend

curves (to 700° F) for pressure vessel materials listed under Section II of the ASME Code. This Subcommittee has, in making their review, established that below 700° F for ferritic materials, the properties are based on room temperature properties and do not require elevated temperature tests and in particular, have identified this approach in the Summer 1970 Addenda Table 424, Note 1, which reads "Tabulated values at the specified temperatures are for design calculations and are not measured values." Accordingly, the additional requirements suggested are not appropriate.

22. The code Inspector is schooled in non-destructive tests and fabricating techniques in order to administer properly his function throughout vessel fabrication as required by ASME Code rules. The inspection organizations are staffed with technically competent people to cover all aspects of fabrication and examination. Steps have been taken to upgrade continually the code inspection organizations

by requiring technical backup within the particular inspection organization for utilization, as required by the local Inspector. Accordingly, the Code Inspectors have the level of competency and understanding appropriate for the proper performance of their job.

3. The complete stress analysis done on the Indian Point Unit No. 2 reactor vessel was accomplished utilizing theoretical calculations and stress indices as provided in ASME Code Section III. The analytical techniques used are based on classical strength of materials theory. The stress analysis performed on the Indian Point Unit No. 2 reactor vessel was accomplished by a group of specialists whose education is in the field of stress analysis. The Professional Engineer certifying this report supervises this analytical group, and is highly qualified in this area. The Code Inspector verifies that the design specifications and stress reports are on file and have been properly certified.

Question No. H-16

Question: Do the ASME Codes have different requirements today than the ones used and referred to in the answer to Question 4.10? What about the Draft ASME Codes or the AEC Reactor Development and Technology program standards dated July 31, 1970? To the extent that any of these are more stringent than the Code used for the Class I Components explain in detail the difference and why the more stringent requirement is not needed for greater safety. If the answer requires more than you are prepared to provide at this time then give the answer only with respect to the reactor vessel.

Answer: The ASME Code today and the draft ASME code have requirements which vary somewhat from those referred to in answer to Question 4.10 with respect to the reactor vessel. However, the requirements of the current and proposed codes are consistent with the requirements of codes which are applicable to the Indian Point Unit No. 2 reactor vessel with regard to safety. The updated code reflects interpretations of the code, code cases, clarifications and new changes. With respect to the AEC Reactor Development and Technology Program Standards referred to in the question, these standards are not applicable to the Indian Point Unit No. 2 reactor vessel.

Question No. H-17

Question: Describe in detail the tests of pipe line vibration for pipes penetrating the containment which will be conducted after plant operation begins. Give inter alia, frequence of tests, extent of piping tested, and what kind of corrective measures will be taken.

Answer: The possibility of significant pipe vibration for pipes penetrating containment is effectively eliminated by design near the penetrations. The pipes and supports are designed to minimize vibration.

During preliminary plant operation, these pipes will be checked for vibration and, if necessary, vibration dampers will be fitted. Therefore, pipe vibration will not be a problem after plant operation begins.

Question No. II-18

Question: Justify the failure to consider jet forces and tornado loads in the design of the large openings of the Containment.

Answer: The 3'-0 thick crane wall, the 4'-0 to 6'-0 thick Refueling Canal and the 2'-0 thick operating floor are capable of resisting jet force loads from Primary Coolant Piping (see Question 5.11 (a)-1 Supplement 9 to the Unit No. 2 FSAR). Thus, jet force loads from the Primary Coolant System cannot impinge on the Containment Structure outer walls; consequently, these jet force loads are not considered in design of the large openings. The large openings are adequately shielded or are far enough away to preclude impingement from Main Steam and Feedwater pipe break loads.

Tornado loads are not a design criterion for Indian Point Unit No. 2; however, they are small compared to the seismic loadings (see Question 5.4(c), Supplement 1 to the Unit No. 2 FSAR). The tornado shear loads from torsion and translational wind force and the overturning moments caused by wind load have a minimum factor of safety of approximately 2.5 with earthquake shears and moments which

were used to size the seismic reinforcing bars. The tornado moment and shears are in fact smaller than the minimum earthquake moments and shears considered in design. On the basis of the above, the seismic bars provide more than an adequate mechanism for resisting tornado loads (see Question 5.1(b), Supplement 1 to the Unit No. 2 FSAR). In addition, tornado loads act independently of other severe loads; therefore, the Equipment Hatch and Personnel Lock reinforced concrete bosses, which were designed for simultaneous design basis accident and earthquake loads, which were larger than tornado loads, are of more than adequate strength to resist tornado loads.

Question No. II-19

Question: With respect to the answers to Questions 5.14(a), 5.14(b), 5.14(d), 5.14(e), 5.15, please provide copies of the relevant pages of the Indian Point Unit No. 3 PSAR.

Answer: The relevant pages from the Indian Point Unit No. 3 PSAR will be furnished separately.

Question No. H-20

Question: Justify the reliability of the Equipment Hatch during design basis accident and earthquake loads when the liner shows deformations which can be tested only for pressure (tensile stresses) and not for accident loadings (compressive stresses). (see Question 5.14 (c)-1). Explain how ductile behavior under tensile stress can adequately represent ductile behavior under compressive stress.

Answer: The subject of Equipment Hatch reliability during design basis accident and earthquake loads is extensively and completely discussed in Section 3.4.0 of the Containment Design Report in Supplement 6 to the Unit No. 2 FSAR. As can be seen from the referenced analysis, justification of the Equipment Hatch is not based exclusively on the pressure test but also on a rather involved finite element analysis approach. There would be no adverse affect on the load-carrying capacity of the Containment wall should local yielding of the liner occur (local liner yielding is hypothesized not expected). The case where the liner is not assumed to carry the load is discussed in Section 3.4.0. Results of the analysis indicate that the rebar is sufficient to carry the pressure

load. The sense of stresses (tension or compression) in the rebar during the pressure test will be the same as experienced in the event of a design basis accident.

The test pressure will demonstrate ductility of the liner under tensile stresses. During accident conditions, the liner is in tension in areas around the Equipment Hatch and Personnel Lock; therefore, ductility demonstrated during the pressure test will have relevance to conditions at accident conditions. For portions of the liner in compression due to temperature increases during the accident condition, the primary consideration is ductility of the liner.

Question No. II-21

Question: Explain in detail the operation changes with respect to the reactor when Indian Point No. 2 is connected to the Con Ed load frequency control system. When will this occur? Indicate to what extent the control of the reactor power level will be determined automatically by load demands from Con Ed's customers and the effect on the reactor power output of a sudden drop in power demand or a sudden increase in power demand on the system. Explain how these variations in nuclear power output of the reactor will affect the various safety features of the plant.

Answer: Indian Point Unit No. 2 will not initially be connected to the load frequency control system (LFC). The decision whether to connect Indian Point Unit No. 2 to the LFS will not be made until after at least a year or two of plant operation, at which time the desirability of this mode of operation can be evaluated in terms of plant performance and system requirements. The operation changes that would result from connection to LFC are essentially that power output would be directed automatically from Con Edison's Energy Control Center rather than being directed by the control room operation. This does not mean that the operator would no longer have control. The operator would still monitor all information as

he would when he was manually operating the plant and could immediately disconnect the LFC if he thought it desirable. There is a station high and low limit setting in the LFC that is used, to set the high and low generation limits beyond which control impulses to the station will be suspended. This device prevents the automatic LFC system from increasing or decreasing the level of generation at a particular station beyond predetermined limits.

As in manual operation, control signals go to the turbine-generator and the reactor can be described as following. The reactor, however, can only follow within limits set by the protection systems. Protection and control systems on Indian Point Unit No. 2 are completely separate and operation of a LFC system would affect the control system only. There would, therefore, be no effect on the safety features of the plant due to operation of a LFC system.

Question No. 22

Question: In answer B-10 you indicate that pressure in excess of 5 psig will not affect the function of the redundant flame recombiner unit. On FSAR, Question 6.8(a)-2 you state that the unit is designed to operate in pressures of 0-5 psig and indicate that it will not be operated until pressure reaches that level. See also pages Q6-8(b)4 and 5-2 and Q6.8(b)4 and 5-3. If pressure is in excess of 5 psig up to 40 psig and if the amount of hydrogen in the containment atmosphere exceeded 2% could the recombiner unit be used at that time. Explain fully a yes answer in light of the design of the unit. If the answer is no, what system would be used?

Answer: The flame recombiner is designed to withstand the design basis accident which includes a pressure of up to 47 psig without damage that would prevent its functioning when needed. However, it is not designed to operate at containment pressure above 5 psig.

The conservative design basis accident pressure transient is such that the containment pressure will be reduced from the maximum 47 psig to less than 5 psig during the first day following the accident. The containment pressure will remain below 5 psig for the duration of the post-accident period. Sufficient hydrogen to require the operation

of the recombiner does not accumulate until after 13 days have elapsed. Thus, there is no requirement that the unit be designed for operation at pressures in excess of 5 psig.

If one postulates the containment pressure to be in excess of 5 psig up to 40 psig, and if the amount of hydrogen in the containment atmosphere were to exceed 2%, the recombiner unit will not be used. Under such circumstances, the hydrogen would be removed by venting the containment using the backup vent system.

Question No. II-23

Question: The recombiner unit uses containment air to cool its exhaust which is allegedly below 300°F. Discuss the impact of the heat addition to the containment caused by the recombiner unit in the context of double-ended hot leg and cold-leg pipe ruptures. In particular how will operation of the recombiner affect the predicted post-accident pressure level in the containment and how will this affect the conservative estimates of radioactive leakage to the atmosphere and the control room.

Answer: The heat addition from operation of the recombiner constitutes less than 5% of the total energy capable of being removed by the containment and core cooling systems. Such addition comes at a time when the heat removal capability of those systems far exceeds the amount of heat required to be removed. Thus, there will be no significant effect on the containment pressure nor on radioactive leakage to the atmosphere and control room.

Question No. H-24

Question: Describe the situation in which oxygen will be added to the containment atmosphere for operation of the recombiner unit discussing when (in terms of hours after the worst accident) the oxygen will be needed and the method for injecting this oxygen into the containment. At the time when oxygen concentration is less than 12% what will be the likely chemical composition of the containment air, its temperature, its pressure and its moisture level.

Answer: Operation of the recombiner system consumes about 30 scfm of oxygen when operated under design conditions. The initial containment oxygen content is about 21% by volume. Since the minimum operating oxygen content is 12% by volume, approximately 235,000 scf of oxygen can be consumed from the containment air before the minimum oxygen level is reached. If the recombiner is started at 2% by volume hydrogen in the containment (~13 days post-accident), operation can continue for slightly over five days before oxygen would have to be added. When needed, oxygen will be added to the containment using piping and controls provided as an integral part of the recombiner system. Added gas will be injected directly into the containment gas space (not into the recombiner)

and mixed by the containment air circulation system.

At the time oxygen is added, the chemical composition of containment air will be approximately as follows (on a dry basis):

Nitrogen	86%
Hydrogen	1%
Oxygen	12%
Others	1%

The containment air pressure will be less than 5 psig, the air temperature less than 152°F, and the moisture content at or near saturation.

Question No. II-25

Question: Will use of the recombiner units require a decision to be made within the control room or will the units be started automatically when required. Specify the highest level of hydrogen which will be permitted to accumulate before the units are put in use and how many hours after the accident this will occur.

Answer: See FSAR Question 6.8.b.(1). The recombiner operation will be initiated manually and, therefore, will require a decision. Hydrogen concentration for recombiner startup is 2% by volume which occurs approximately thirteen days after the accident.

Question No. H-26

Question: Explain in detail the nature of the uncertainty associated with the catalylic recombination system for hydrogen removal. See Question 6.10-1. In particular does this uncertainty stem from uncertainty regarding the composition of the post-accident containment air or is it only uncertainty regarding operation of the catalytic recombiner itself under reasonably predictable conditions.

Answer: The reason why the catalytic recombiner system was not chosen at the time when the design decision had to be made regarding the type of recombiner to be used, was the uncertainty regarding operation of the catalytic recombiner under reasonably predicable conditions.

Question No. II-27

Question: Assuming 3/4 of the on-site spent fuel storage capacity is filled and assuming Indian Point No. 1 and 2 have been operating at full power level for 300 days, how much plutonium will be present at the Indian Point site in the:

a. Spent fuel storage
b. Reactor core for each reactor separately

As a basis of comparison relate this to the amount of plutonium released (best estimate) in fallout from the above-ground explosion of the largest plutonium nuclear weapon of the United States.

Answer: a. Spent fuel storage (for equilibrium cycles of Indian Point Units 1 and 2)

IP-1	198 Kg Pu
IP-2	762 Kg Pu
Total -	960 Kg Pu

b. Reactor core for each reactor separately after 300 full-power days (for equilibrium cycles of Indian Point Units 1 and 2)

IP-1	140 Kg Pu
IP-2	606 Kg Pu
Total -	746 Kg Pu

Applicant does not have access to information which would be necessary to make the requested comparison.