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U.S. NUCLEAR REGULATORY COMMISSION

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Docket No. 50-247

Mr. John O'Toole
Vice President
Nuclear Engineering and Quality Assurance
Consolidated Edison Company of
New York, Inc.
4 Irving Place
New York, New York 10003

Dear Mr. O'Toole:

Enclosed for your information is a copy of documents that we used to support our conclusion that operation could resume at Indian Point 2 following the accumulation of water in containment.

Enclosure 1 is our Safety Evaluation of Attachments A and C, and Items 2, 15, 18 and 19 of your December 22, 1980 letter which includes the reactor vessel stress analysis. Enclosure 2 is our evaluation of the discussion of Potential Unreviewed Safety Questions in your January 5, 1981 letter. Enclosure 3 is a report of the independent nondestructive examination of the reactor vessel that was performed under our direction.

These documents were also forwarded to the Commission after the April 7, 1981 briefing. During this briefing, the Commission determined that its July 15, 1980 decision regarding continued operation remains valid.

Sincerely,

Original signed by:
S. A. Varga
Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Enclosures: As stated

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See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

May 8, 1981

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Division of Licensing

Enclosures: As stated

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Response to Reference 2

(1) Attachment A and C, Reactor Vessel Stress Analysis

The Indian Point-2 reactor vessel has mirror insulation on its external surface. This type of insulation is an effective heat barrier when in an air atmosphere. However, the insulation is not leak tight to water and therefore, the actual cooling mechanism that occurred when the hot reactor vessel was subjected to external flooding with relatively cold water cannot be determined with certainty. Judgments have to be made as to how rapidly the water penetrated the insulation and the degree of preheating that occurred as it approached the reactor vessel surface. The consequence is an uncertainty regarding the magnitude of the heat transfer coefficient at this surface.

Shortly after the flooding event occurred at Indian Point-2, the NRC staff performed one-dimensional heat transfer and thermal stress analyses. Included was a conservative or bounding analysis for which it was assumed that the heat transfer coefficients at both the internal and external surfaces of the vessel were infinite. Although it is recognized that this assumption is not realistic and will result in an absolutely worst possible case, the resulting calculated thermal stresses of this analytical model are still tolerable, provided that the event is not repeated often enough to violate fatigue limits and that relatively large flaws do not pre-exist in the vessel wall.

The NRC staff also contracted with EG&G, Idaho, to perform more sophisticated finite element analyses to determine thermal and pressure stresses during the Indian Point-2 event. There are two regions of the reactor vessel where geometric discontinuities cause higher local stresses than elsewhere in the vessel wall. These are (1) the interface between the cylindrical portion of the vessel and the hemispherical lower shell and (2) the partial-penetration welds joining the instrument tubes to the lower shell. Although the water level external to the vessel rose rather slowly during the event because of the relatively large volume of the reactor cavity so that quasi-steady state analyses could be performed rather than transient analyses, the magnitude of the stresses at various vessel wall locations is dependent on water level. Near the water level, heat flow in the vessel wall is both axial and radial. Thus multidimensional finite element analyses are appropriate and were performed by EG&G for various assumed water levels. Their results and conclusions are presented in a January 29, 1981 EG&G memorandum (Enclosure 3).

With this background, the NRC staff has reviewed the thermal and stress analyses reports submitted by the licensee; specifically Attachments A and C to the licensee's letter of December 22, 1980. Although somewhat different heat transfer and flood water temperature assumptions were used by the various parties who performed thermal and stress analyses, there is agreement that the incremental usage factor attributable to this event is no greater than 0.01 and probably much less and that the total usage factor to date is significantly less than unity. In addition, specific investigations by the staff, our contractor EG&G and Westinghouse (on behalf of the licensee) of the penetration weld regions in the lower shell lead to the conclusion that the integrity of these welds has not been compromised by the event.

Thermal ratchetting and fatigue limits are met regardless of the fact that in certain conditions of the analyses, the primary plus secondary stress intensity ranges at the transition region exceeded $3S_m$. However, Section III of the ASME B&PV Code (NB-3228.2) provides for such cases with rules that prevent thermal ratchetting (incremental collapse) and rules that conservatively increase the calculated fatigue usage factor. In the worst case assumption of vessel O.D. response to 65°F water at the vessel-head junction (i.e., the highest stressed area analyzed) thermal ratchetting rules are met and the partial usage factor calculated is 0.01. Considering an arbitrary 5 occurrences of the event added to the initial usage factor of 0.003 results in a total usage factor of 0.053 which is clearly insignificant when compared with the Section III allowable value of 1.0.

Following the determination that the reactor vessel lower head and incore instrument conduits were submerged in service water while at normal operating temperature, the licensee performed a magnetic particle and liquid penetrant inspection respectively, on the affected ferrous and nonferrous welds. No relevant indications were reported.

To verify the licensee's nondestructive examination results the NRC contracted an outside inspection firm to perform an independent surface examination of all the affected welds including the circumferential weld at the transition region and one foot of the longitudinal shell welds intersecting the circumferential shell to lower head weld. No relevant indications were detected by either magnetic particle or liquid penetrant examination.

Thus, based on analyses and inspections, we conclude that the incremental fatigue usage caused by the October, 1980 flooding of the exterior of the Indian Point 2 vessel was no greator than 0.01; that the total usage to date is significantly less than unity; that the vessel surfaces that experienced the higher tensile stresses during the event are free of flaws; and that the vessel can be returned to normal operation.

(2) Reactor Vessel Insulation

The mirror insulation of the lower reactor vessel shell was found to be in good physical condition except that a deposit from the river water adhered to its surfaces as a consequence of the October 1980 flooding event. The licensee proposes to re-use this insulation.

One panel of the insulation was disassembled to permit examination of the inner liners and the cutting of a liner sample for thermal testing. The tests were conducted by Mirror Insulation, a Unit of Diamond Power (the insulation manufacturer).

Results of the tests indicated that the emittance of the liner sample increased by about a factor of five. An increase in emittance reduces the resistance of the insulation panels to radiant heat loss. Heat is also transmitted through the panels by convection and conduction. The licensee estimates that the net result will be an overall increase in insulation conductance of from 63% to 107% depending on whether the insulation is mounted vertically or horizontally and that, as a result, the increase in overall heat loss from the vessel is approximately one tenth of one percent of the normal containment cooling load and thus tolerable. The licensee also concludes that the degradation of the insulation sustained as a result of exposure to river water will not result in temperature changes that will adversely affect the reactor vessel or surrounding structures and equipment.

The staff has reviewed the licensee's documentation and concurs with his findings. We conclude that the degraded insulation may be re-used but that it would be prudent to verify the predicted results of operation with degraded insulation after the return to normal operating conditions. The licensee has verbally agreed to do this. Of possible concern is the temperature of the surface of the biological shield facing the degraded insulation. We also recommend that the physical condition of the insulation on the lower vessel shell be visually examined at each of several refueling outages to assure that mechanical degradation does not occur because of the adhering crud.

(3) Reactor Vessel Paint Chloride Retention

The reactor vessel was painted with an alkyd based aluminum-silicone coating. On being heated to normal operating temperatures, the organic component volatized leaving a metallic aluminum dispersed in silica oxide polymer remaining. This surface is a tenacious, inert, non-ionic coating that would not bond to an ionic species such as chloride. Any chlorides that may have been deposited during external flooding with river water have been removed by washing with demineralized water leaving no residue retained by the paint. Thus, we conclude that chlorides on the vessel surface are not a concern during future operation.

(4) Incore Instrument Stub-Tube-to-Reactor-Vessel Weld Failure Conseouences

Even though water containing chlorides contacted the bottom head of the vessel, this water could not penetrate the clearance between the stub-tubes and the shell because the metal temperature, especially near the welds, was much above the boiling temperature of the water and prevented the deposition of chloride at the weld location. Thus, contaminants in the water are not a cause for concern regarding corrosion and/or crack initiation in the weld region.

Stress analyses for this region performed by EG&G and by Westinghouse indicate that these welds were not jeopardized by the October, 1980 flooding event. A fracture mechanics analysis performed by the staff led to the same conclusion. Nevertheless, in the unlikely event that cracks were to initiate in this region in the future, we conclude that it is very unlikely that they would unite and propogate as a cylindrical crack of the same diameter as the stub-tubes because of the stress field at these locations. Thus, detectable leaks would result rather than tube ejection.

(5) Modifications

The licensee is planning a number of plant modifications following the flooding of Indian Point Unit 2. These modifications address previous failure modes and they enhance the overall system design. Included among the modifications are:

A. Replacement of the fan coolers with units of an improved design. Connecting piping to the units will allow for improved accessibility for tube repairs and tube plugging, if needed.

B. Replacement of the existing isolation valves with ones designed to eliminate the leakage problems experienced with the current rubber liner.

C. Improvements for the containment sump pumps include:

- Magnetrols to control pump start and stop levels, protect against loss of suction head, and equalize running hours. (Eliminated exposed mechanical float type switch control.)
- Remote control station (on, auto, and off) in Central Control Room.
- Separate Power Supplies
- Eliminate piggy-back connection of reactor cavity pumps to load side of power supplies.

D. Instrumentation added to assist in identifying leakage into the containment sump and its source:

- Continuous level indication, alarm and record. (Revision to TMI Instrumentation.)
- Discharge Integrating Flow Meter
- Discharge Water Chemical Hardness
- Provision for Chromate Grab Sample
- Discharge Temperature

- Flood level switch in the RHR suction line compartment of the containment sump.
- TV camera focused to monitor initiation of water accumulation on containment floor.

E. Improvements for the recirculation sump include:

- Sump to be normally dry during future operation. This is an improvement because now the plant operators will receive information that both the containment and emergency recirculation sumps fill with water before the reactor cavity begins to flood.
- Alarm two lowest level lights (mod. of existing float instruments).
- Continuous level indication (TMI Instrument).

F. Improvements in the Reactor Cavity include:

- TV camera focused on low point of curb.
- Alarm lowest switch position on one of the two pump control magnetrols.
- Continuous level indication (TMI Instrument).

G. Improvements for the reactor cavity pumps include:

- Open Pipe/Funnel anti-siphoning device in discharge line at containment sump.
- Separate Power Supplies (each pump from other and from containment sump pumps).
- Install submersible level control switches (including loss of suction head protection and equalized running hours.)

We have reviewed the modifications proposed by the licensee and conclude that the modifications will improve the present design at Indian Point Unit 2 and will assist the plant operators in identifying leakage into the containment sumps. The modifications in the licensee's proposal satisfies the requirements made in IE Bulletin 80-24, "Prevention of Damage Due to Water leakage Inside Containment (October 17, 1980 Indian Point 2 Event)."

Potential Unreviewed Safety Questions

(1) Impact of cold water on the submerged vessel.

(2) Impact of cold brackish water on the submerged stainless steel conduits.

We conclude that the above impacts do not constitute an unreviewed safety question. The basis for this conclusion is presented in Enclosure 1.

(3) "Potential post loss-of-coolant-accident (LOCA) water levels in containment in excess of the assumptions used in the Safety Analysis Report (SAR)."

The licensee has examined the consequences of a LOCA along with the initial water inventory due to (1) plant conditions discovered on October 17, 1980 and (2) conditions which could have developed, had the plant again been returned to power without discovery of the leakage and the flooding problems.

Had a LOCA occurred with plant conditions as found as found on October 17, 1980 (125,000 gallons of water on the floor), the licensee has calculated that the flood level would have risen to elevation 51' 7 1/2". This is due to the initial 125,000 gallons being added to the 423,000 gallons which comes from the LOCA (primary content plus injected water from the Refueling Water Storage Tank). The initial water volume for this case raises the flood level 1' 6 1/2" above that previously analyzed. The licensee states that the additional flooding would submerge safety injection valves 856A, B, C, E and F along with the second tier of electrical penetrations.

The cold leg safety injection valves (856A, C and E) are normally open, receive a confirmatory safety injection signal to open, and are designed to fail "as is". We agree with the licensee that submergence of these valves should not prevent them from being in the open position.

The hot leg safety injection valves (856 B and F) may be needed approximately 24 hours following a LOCA to prevent boron precipitation. Although the valves de-energize to the closed position, there are two other hot leg injection paths available if needed.

All safety related electrical cables inside containment have been designed to withstand submergence in borated water, extremes of temperature, humidity and pressures as well as radiation. However, in order to verify their operability against the short and long term effects of chloride exposure, the licensee is conducting tests of representative cables and splices.

Trays and conduits would not be affected by submergence due to the physical make-up of the material (non-porous and no failure mode).

The second case examined by the licensee assumed that a maximum of 150,000 gallons of water could accumulate in containment before the operators would notice flood level indicating lights in the control room and manually terminate service water flow. The resultant post-LOCA water level, assuming 573,000 gallons of water accumulate inside containment, would be 51" 11". The licensee states that no additional equipment beyond that discussed above would be submerged.

We conclude that the post-LOCA water level in containment, as a result of the Indian Point Unit 2 flooding, does not constitute an unreviewed safety question unless the results of the field testing of the electrical cables and splices currently being performed fail to confirm the licensee's position that the cables and splices are qualified for submergence.

(4) "Potential Post-LOCA Water boron concentrations less than the assumptions in the SAR."

When the recirculation phase of a loss-of-coolant-accident begins, all fluids in the containment sump are mixed and pumped back to the reactor vessel. The unborated water from the Hudson River initially in the containment sump would dilute the borated water from the RWST. If sufficient dilution exists, there is a possibility that the core could return to criticality.

The licensee has examined the consequences of a LOCA along with the initial water inventory due to (1) plant conditions discovered on October 17, 1980, and (2) plant conditions which could have developed, had the plant again been returned to power without discovery of the leakage and the flooding problems.

The licensee's analysis minimized the boron inventory in both the boron injection tank and the refueling water storage while maximizing the water inventory in the spray additive tank. With these boron sources in containment, the licensee calculates that approximately 950,000 gallons of unborated water must be added to containment before the reactor returns critical. Since this is far in excess of both the 125,000 gallons assumed in case 1 and the 150,000 gallons assumed in case 2, we concur with the licensee that a return to criticality would not occur following a LOCA in conjunction with the containment flooding.

The Indian Point Technical Specifications state that boron concentration in the reactor coolant system should be sufficient to maintain a minimum shutdown margin of one percent reactivity. The staff has asked the licensee to determine whether the Post-LOCA boron concentration in the sump (with the dilution factor) would provide the shutdown margin required in the plant's Technical Specification. The licensee has examined this case and concludes that a sufficient amount of boron would exist in the sump to meet this requirement.

The staff, therefore, concludes that Post-LOCA boron concentrations in the sump would not be lower than that previously assumed in the SAR, and, that this does not constitute a potential unreviewed safety question.

EG&G Idaho, Inc.
P.O. BOX 1625, IDAHO FALLS, IDAHO 83415

January 29, 1981

Mr. R. E. Tiller, Director
Reactor Operations and Programs Division
Idaho Operations Office - DOE
Idaho Falls, ID 83401

THERMAL ANALYSES AND STRESS ANALYSES OF THE INDIAN POINT UNIT 2 REACTOR VESSEL BOTTOM HEAD (A6432) - BFS-4-81

Ref: R. E. Vollmer ltr to C. E. Williams, INEL Technical Assistance to the Division of Engineering, NRR, NRC - "Component Integrity Evaluation Program," (FIN A6432), November 25, 1980

Dear Mr. Tiller:

Following the October 17, 1980 accident which exposed the Indian Point Unit 2 reactor vessel bottom head to cold river water, EG&G Idaho was asked by the Nuclear Regulatory Commission (NRC) to assist them in evaluating the severity of the stresses induced by the imposed thermal gradients. Specifically, EG&G has performed a number of thermal, stress, and fatigue analyses to quantify the problem for the worst case scenario.

Attachments 1 and 2 document the thermal and stress analyses performed. Attachment 2 also reports the reduction in fatigue life caused by the accident. These attachments satisfy Project I of the referenced letter.

Very truly yours,

B. F. Saffell, Manager
Code Assessment and
Applications Division

BLB:bb

Attachments:
As stated - 2

cc: R. M. Gamble, NRC-DE
R. W. Kiehn, EG&G Idaho

Attachment 1
January 29, 1981
BFS-4-81

A Final Report on the Thermal Analysis of the Indian Point Unit 2 Reactor Vessel

A thermal analysis has been performed on the Indian Point Unit 2 reactor vessel. This analysis was needed to access the condition of the vessel following the containment flood event discovered October 17, 1980.

The areas of interest in this study included a steady-state analysis of the reactor vessel, a steady-state and transient analysis of a typical bottom head instrument penetration, and an evaluation of the reduction in effectiveness of the reactor vessel insulation. These areas are discussed in the following paragraphs.

A two-dimensional finite element axisymmetric representation was used in the steady-state analysis of the reactor vessel. The model consisted of 795 elements with 960 nodal points.

In the steady-state analysis of the reactor vessel, the stainless steel cladding was neglected. It was also assumed that the inside surface temperature of the vessel was maintained at 550°F. Actually, the inside surface of the stainless steel cladding will be at a temperature less than 550°F as related by some convective heat transfer coefficient. Then, a radial temperature drop through the cladding will occur before reaching the inside surface of the reactor vessel. Therefore, these assumptions are conservative since they will yield a higher inside vessel temperature and a correspondingly higher radial temperature gradient.

The steady-state analysis of the reactor vessel examined four different water elevations. Elevation 1 corresponded to an elevation of 43 ft or a point 9 ft above the bottom of the hemispherical head. This point was the maximum water elevation which occurred during the flood event. Elevation 2 corresponded to an elevation of approximately 40 ft or a point 6 ft above the bottom of the hemispherical head. This point is the junction between the cylindrical shell and the bottom hemispherical head. Elevation 3 corresponded to an elevation of approximately 37.5 ft or a point 3.5 ft above the bottom of the hemispherical head. And finally, Elevation 4 corresponded an elevation of approximately 35 ft or a point 1 ft above the bottom of the hemispherical head.

A steady-state temperature distribution was computed for each of the water elevations just described. In all cases, the effect of the reactor vessel insulation was neglected for all submerged portions of the vessel. At all elevations above the water levels, an adiabatic boundary was assumed. Actually, with the insulation in place, the water contacting the outside surface of the reactor vessel will be at approximately saturation temperature and, on initial contact, probably in a film boiling regime. As the outside surface of the vessel cools, a transition to a nucleate boiling regime will occur lowering the outside surface temperature to a point slightly above the saturation temperature. With the insulation in place, the influx of cool water to the surface will be retarded which will stabilize the outside surface temperature of the reactor vessel at or near the saturation temperature. By neglecting the vessel insulation for all submerged surfaces, free convection[a] to 40°F water was allowed which yielded outside surface temperatures in the range of 129 - 152°F. These surface temperatures are considerably below the saturation temperature thereby providing a correspondingly larger thermal gradient. In addition, the adiabatic condition above the water levels provides a "worst" case for axial temperature gradients. Therefore, these assumptions are also conservative.

The steady-state temperature distributions for the four different water elevations have been computed and used in a stress analysis of the reactor vessel. Figures 1a through 1d illustrate representative temperatures for each of these cases. In each figure, the vessel temperature at the water level and the vessel temperatures one node above and one node below the water level have been included. The nodes are approximately 1.6 in. apart.

A transient thermal analysis of the reactor vessel which included the effect of the flood water rising along the outside surface of the vessel

(a) J. P. Holman, Heat Transfer, 4th edition, New York: McGraw-Hill Book Company, Inc., 1976, pp. 235 - 255.

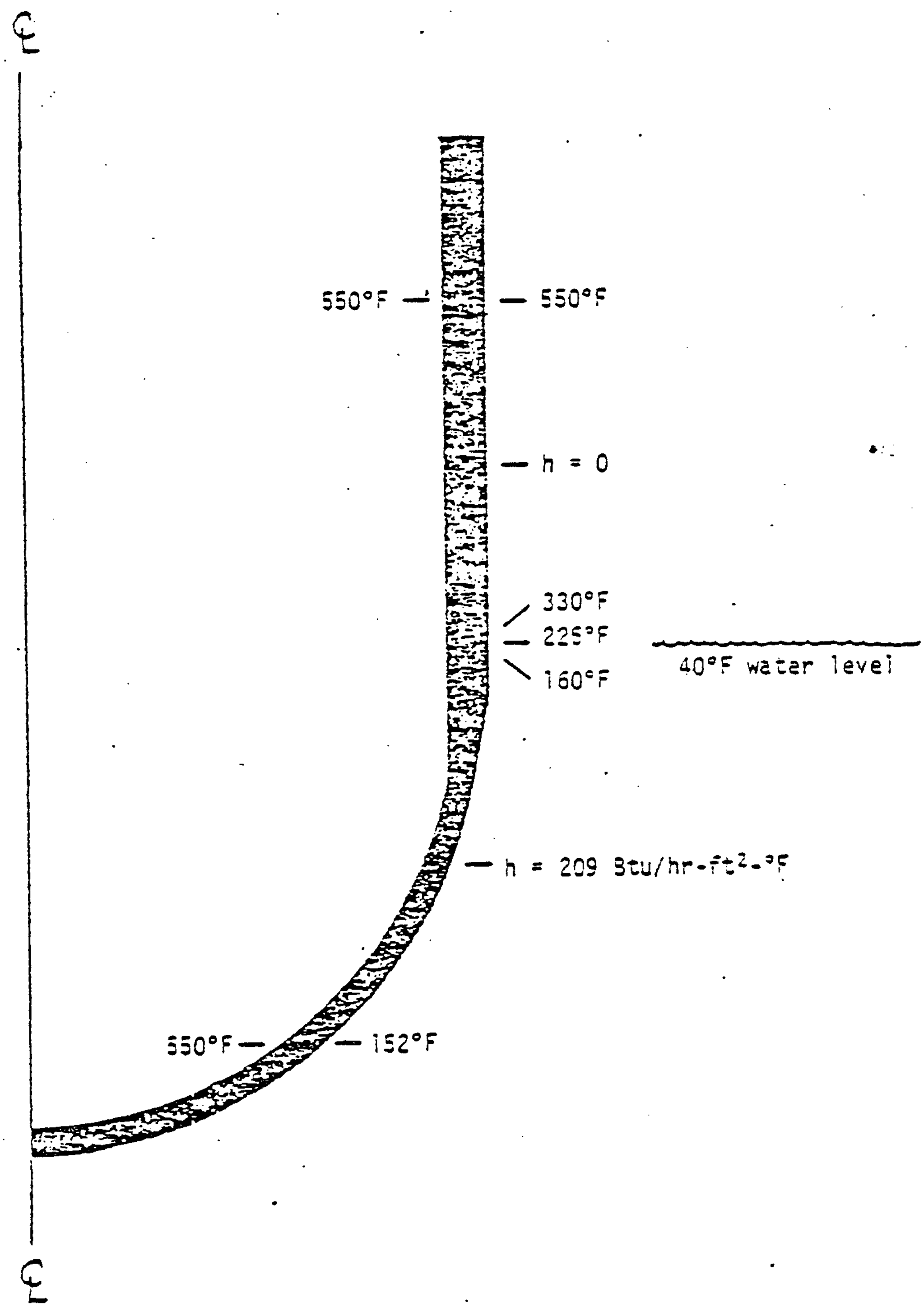

Figure 1a. Steady-state reactor vessel temperatures assuming 40°F water at the 43 ft elevation point.

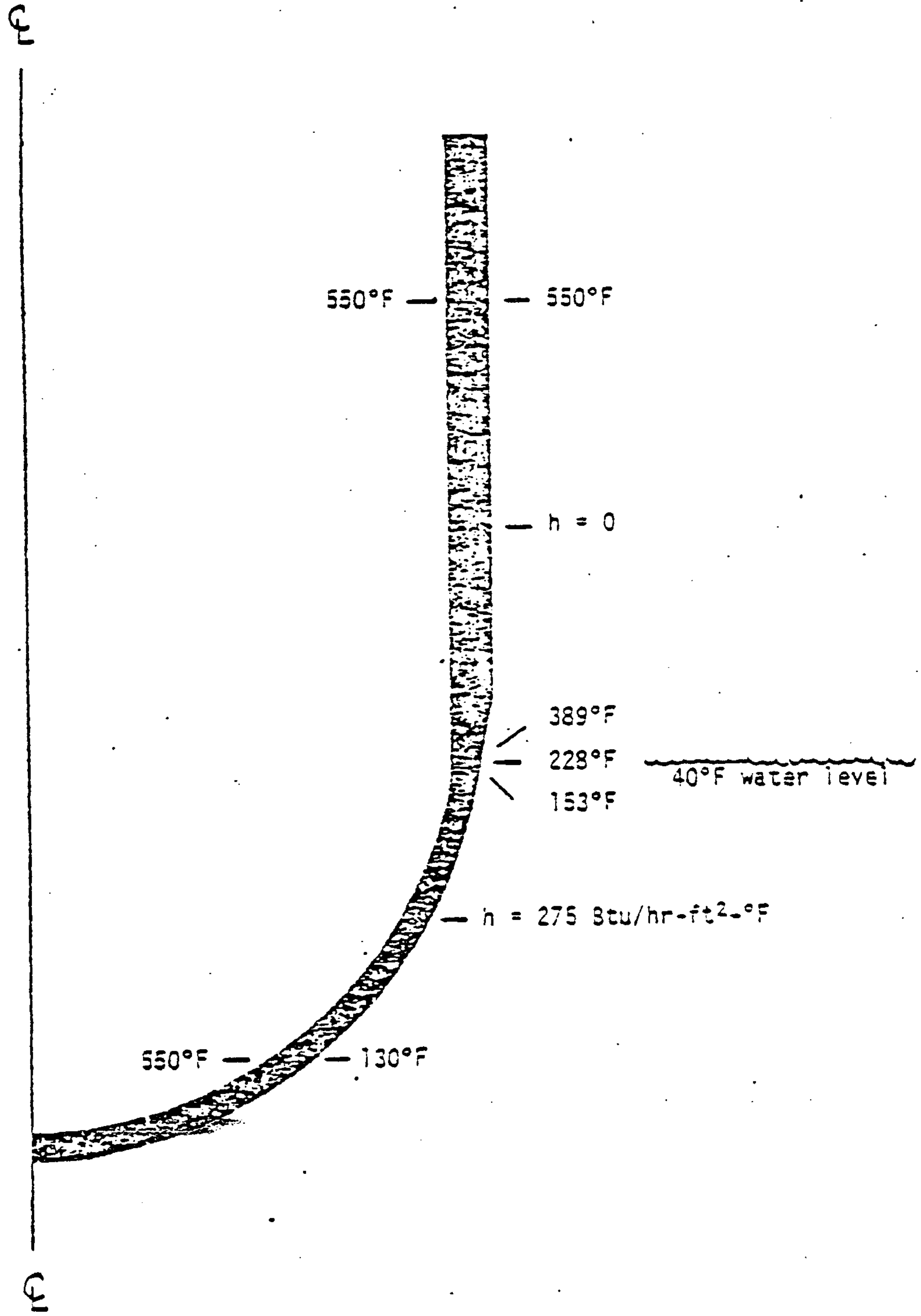

Figure 1b. Steady-state reactor vessel temperatures assuming 40°F water at the 40 ft elevation point.

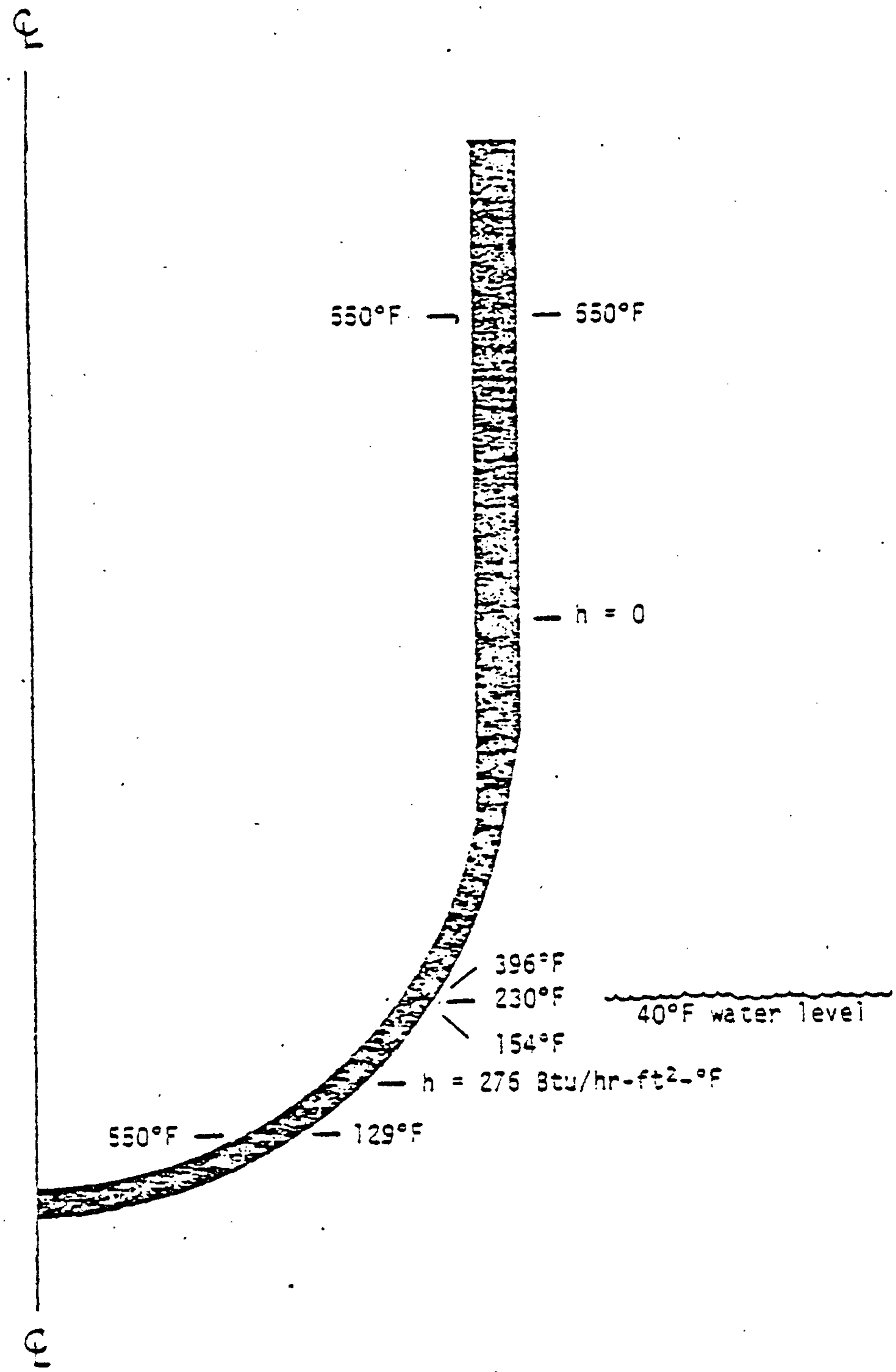

Figure 1c. Steady-state reactor vessel temperatures assuming 40°F water at the 37.5 ft elevation point.

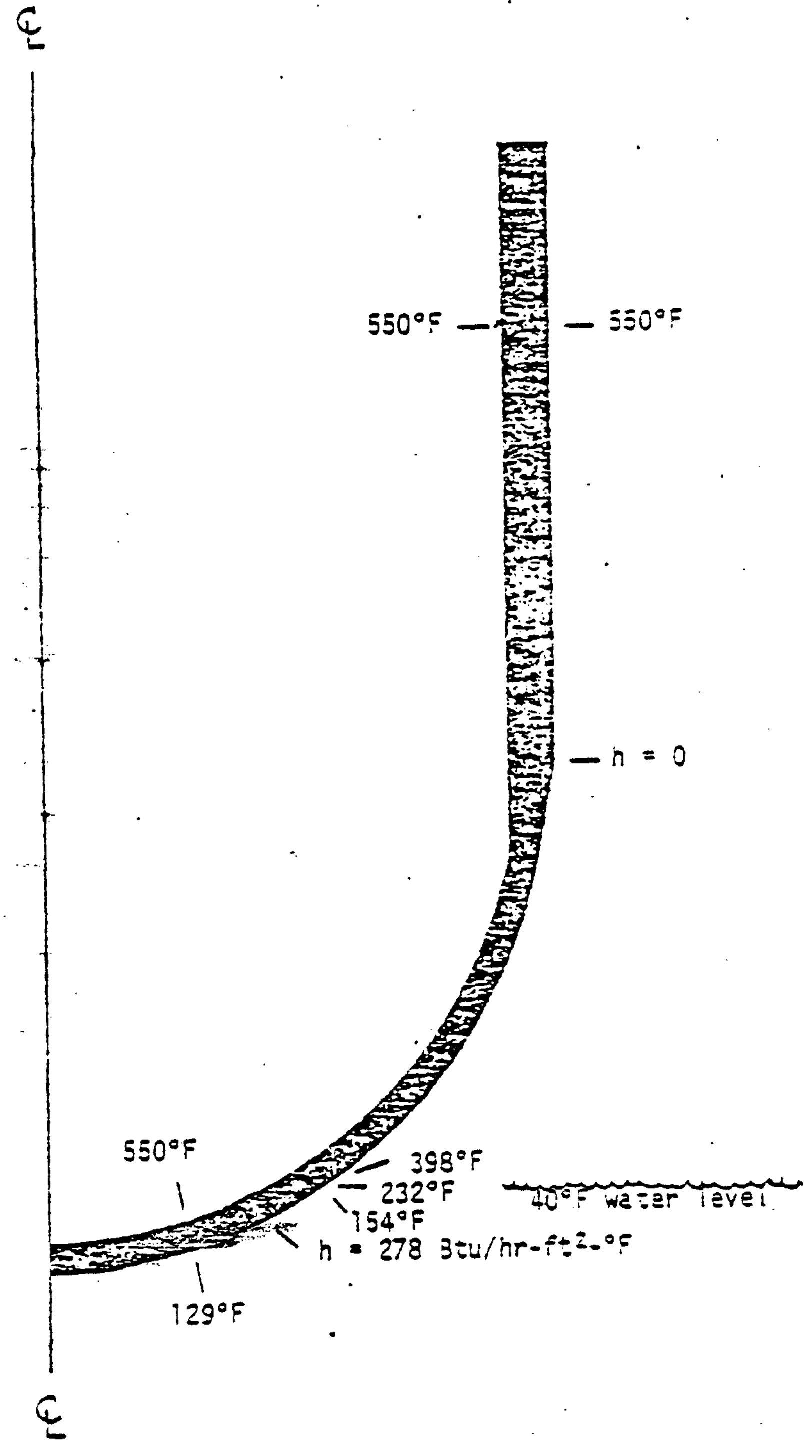

Figure 1d. Steady-state reactor vessel temperatures assuming 40°F water at the 35 ft elevation point.

was not performed. From the steady-state analysis just described, the maximum thermal gradients and, therefore, the maximum thermal stresses occurred well below the air-water interface. The maximum rate of rise of the air-water interface was found to be 22 in./hr (based on a water leak rate of 17.3 gpm) which was sufficiently rapid to induce nucleate boiling heat transfer and approximately saturation temperatures at the interface. (It should be stated that saturation temperatures on the outside surface of the vessel do not produce the largest thermal gradients.) This rate of rise was not sufficient, however, to maintain boiling below the water surface. Once the interface passes any particular point, steady-state conditions will be approached. Therefore, a transient analysis of the reactor vessel would only reproduce the maximum thermal gradients and the resulting thermal stresses seen in the steady-state analysis.

A two-dimensional finite element axisymmetric representation was used in the steady-state and transient analysis of a typical bottom head instrument penetration. The model consisted of 392 elements with 455 nodal points. The mesh used is shown in Figure 2. For the purpose of presenting results, two sections have been identified on the figure.

Two different steady-state cases were considered in the analysis of a typical bottom head instrument penetration. Both steady-state temperature distributions were computed assuming that the interior surface of the reactor vessel was maintained at 550°F while portions of the exterior surface were cooled by free convection to 40°F water. (The free convection correlations and the computed heat transfer coefficients were comparable to those used in the steady-state analysis of the reactor vessel.) In addition, the effects of the stainless steel cladding and the reactor vessel insulation were neglected. These assumptions are conservative for the reasons discussed earlier in this report.

The first steady-state case assumed that the penetration tube was cooled by free convection to 40°F water while the exterior surface was unaffected. This condition could have occurred as the flood water moved up the vessel, cooling the protruding tube before contacting the reactor vessel wall. The predicted temperature distribution was then used in a

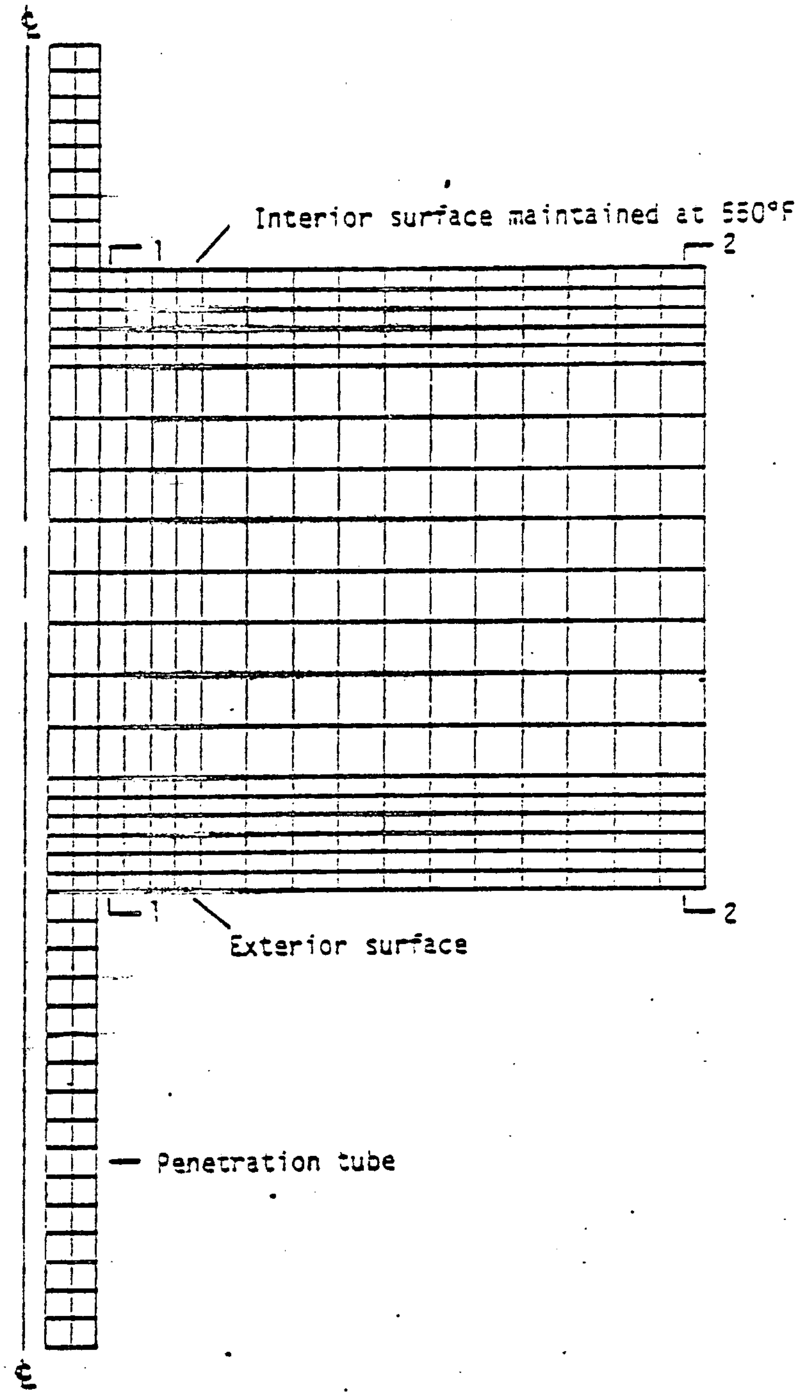

Figure 2. Bottom head instrument penetration model.

stress analysis of the bottom head instrument penetration. The resulting thermal stresses were found to be less than those based on either the second steady-state case or the transient case.

The second steady-state case assumed that the penetration tube and the exterior surface of the reactor vessel surrounding the tube were simultaneously cooled by free convection to 40°F water. Although this condition cannot actually occur (without a very sudden increase in the flood water level), it was felt that it provided a "worst" case approach. Of all cases considered, this predicted temperature distribution indicated the largest thermal stresses in the INCONEL weld used to attach the penetration tube to the reactor vessel. The steady-state temperature profiles for Sections 1-1 and 2-2 are given in Figure 3.

A transient thermal analysis of a typical bottom head instrument penetration was performed assuming that the penetration tube and the reactor vessel surrounding the tube were at an initial uniform temperature of 550°F. The penetration tube and the exterior surface of the reactor vessel surrounding the tube were then simultaneously exposed to 40°F water. Temperature distributions as a function of time were computed as the vessel cooled. Again, the interior surface of the vessel was maintained at 550°F. In addition, the effects of the stainless steel cladding and the reactor vessel insulation were neglected.

On initial contact with 40°F water, film boiling was established on the exterior surface of the reactor vessel. Heat transfer coefficients are relatively small in the film boiling regime due to the large conductive resistance across the film. The modified Bromley equation[b] was used to calculate these heat transfer coefficients.

At a surface temperature of 524°F (the Leidenfrost temperature), it was assumed that film boiling gave way to a transition boiling regime. Transition boiling on the vessel surface was maintained until the surface temperature dropped to 307°F. (A surface temperature of 307°F corresponds to the point of departure from nucleate boiling (DNB). This point was

(b) L. J. Siefken, "FRAP-T4: A Computer Code for the Transient Analysis of Oxide Fuel Rods", CDAP-TR-78-027, July 1978, p. 135.

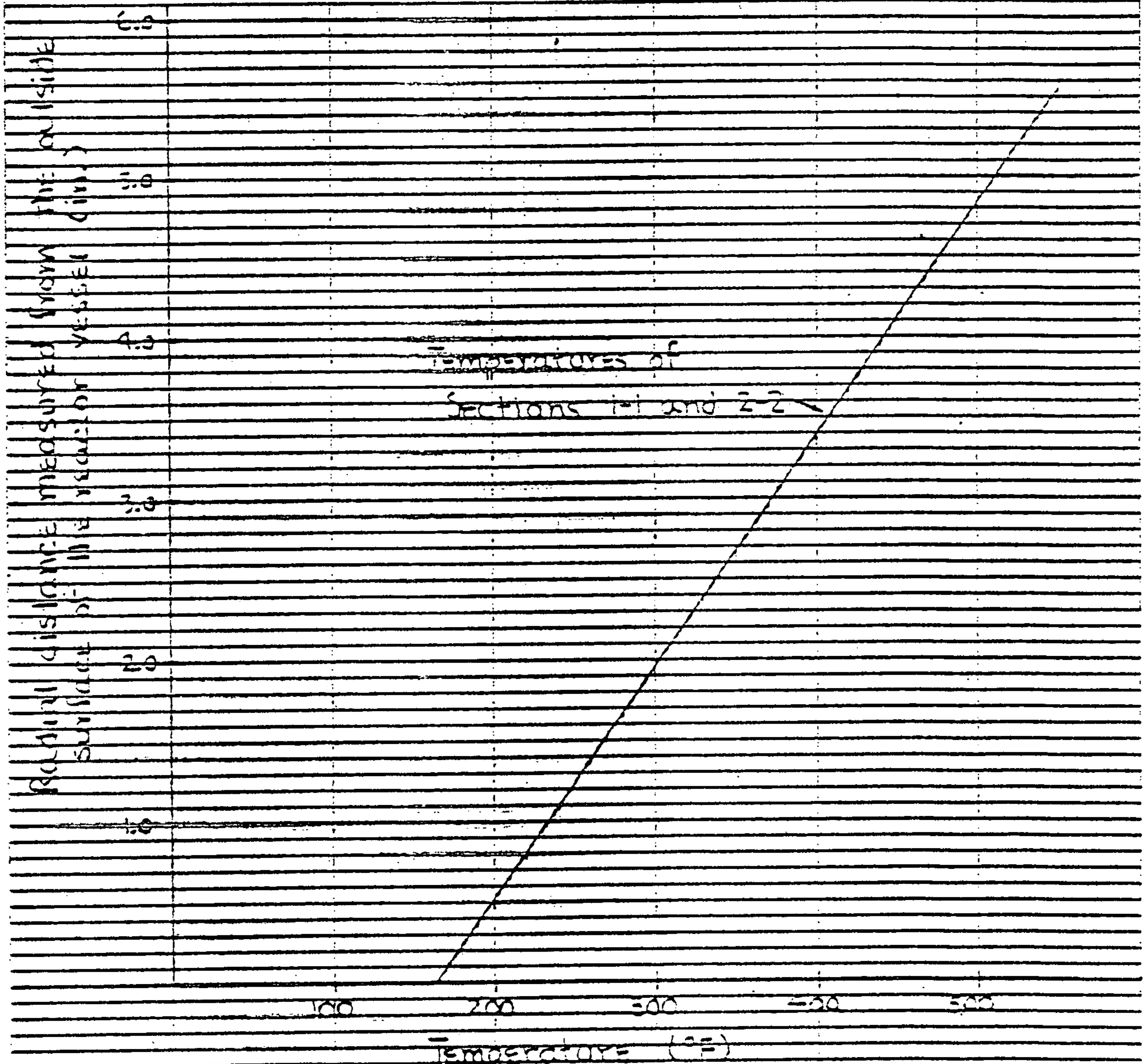

Figure 3. Steady-state temperature profiles for Sections 1-1 and 2-2 (see Figure 2).

established by calculating the peak heat flux[c].) The transition heat transfer coefficient, between these temperatures, was then established through a linear interpolation on a log-log plot of the surface heat flux versus the temperature difference between the surface and the saturation temperature[d].

At surface temperatures between 222 and 307°F, nucleate boiling heat transfer occurred. Relatively large heat transfer coefficients characterize this portion of the boiling process. The correlation used to calculate the appropriate heat transfer coefficients was the Rohsenow equation[e]. (At surface temperatures below 222°F, free convection to 40°F water occurred. This heat transfer was computed as discussed earlier in this report.)

Figures 4a and 4b show transient temperature profiles as a function of time for Sections 1-1 and 2-2, respectively. These curves, of course, represent points throughout the boiling process. All transient temperature distributions have been provided for use in a stress analysis of the bottom head instrument penetration.

The final aspect of this study was an assessment of the condition of the reactor vessel insulation following the flood event. It was assumed that as the flood water receded and the insulation dried, some scale and deposits remained on the insulation. Since the insulation is basically a series of radiation shields separated by air gaps, these deposits could increase the effective emissivity of the insulation, thereby increasing the amount of heat transferred through the insulation.

Two different types of reactor vessel insulation were considered; namely, insulation types HH1THR014 and HB1THR024. Both types consisted of a series of alternating stainless steel and aluminum plates separated by air gaps. The overall thickness of the insulation was 3 in. The primary difference between the two types of insulation was the thickness of the plates.

(c) F. Kreith, Principles of Heat Transfer, 3rd edition, New York: Harper and Row, Publishers, Inc., 1973, pp. 512-513.
(d) T. E. Rahl and G. A. Tatar, "Design Report for JAERI Slab Core Hot Leg Spool", EGG-EA-5226, August 1980, p. 4.
(e) J. P. Holman, Heat Transfer, 4th edition, New York: McGraw-Hill Book Company, Inc., 1976, p. 365.

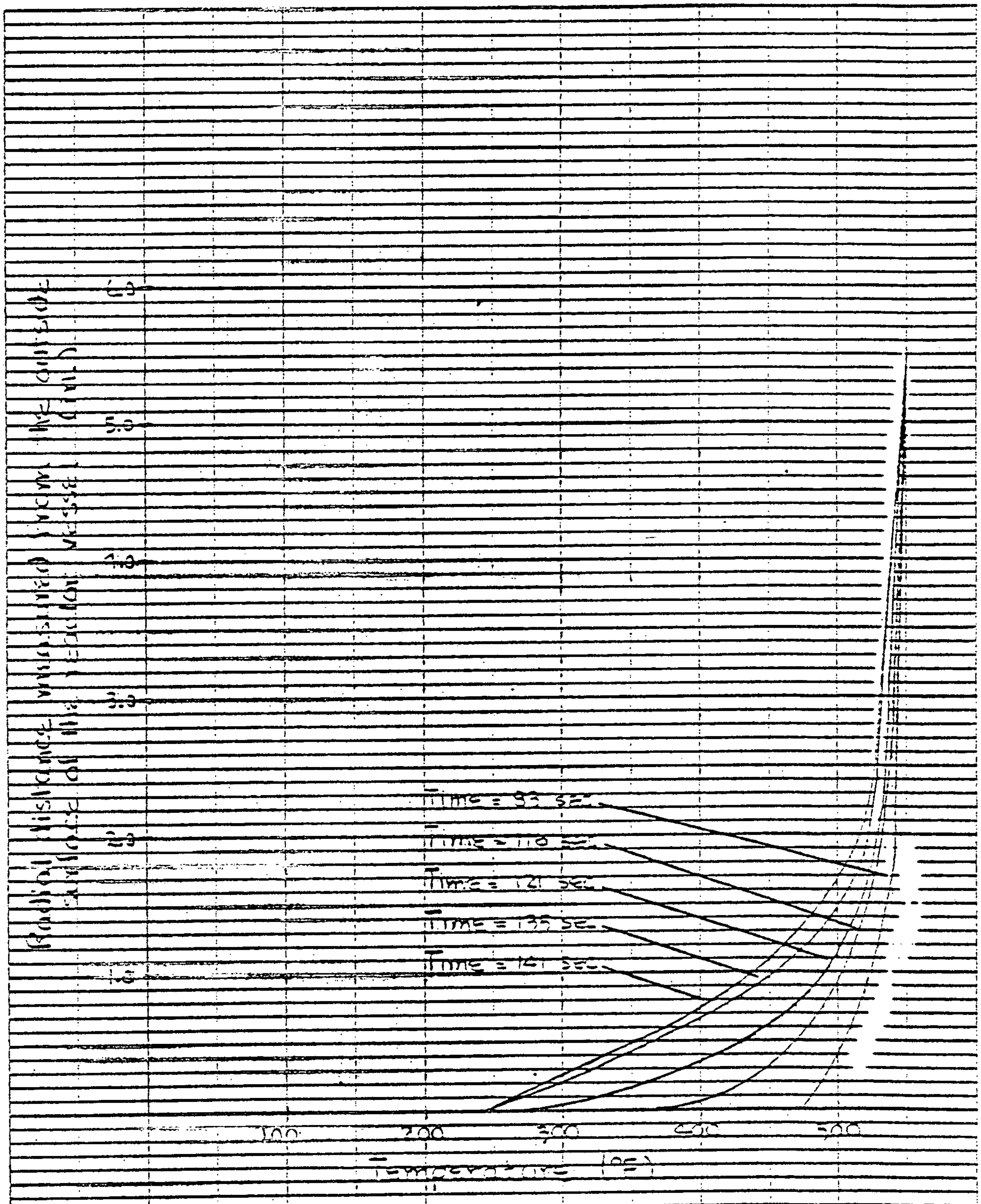

Figure 4a. Transient temperature profiles for Section 1-1 (see Figure 2).

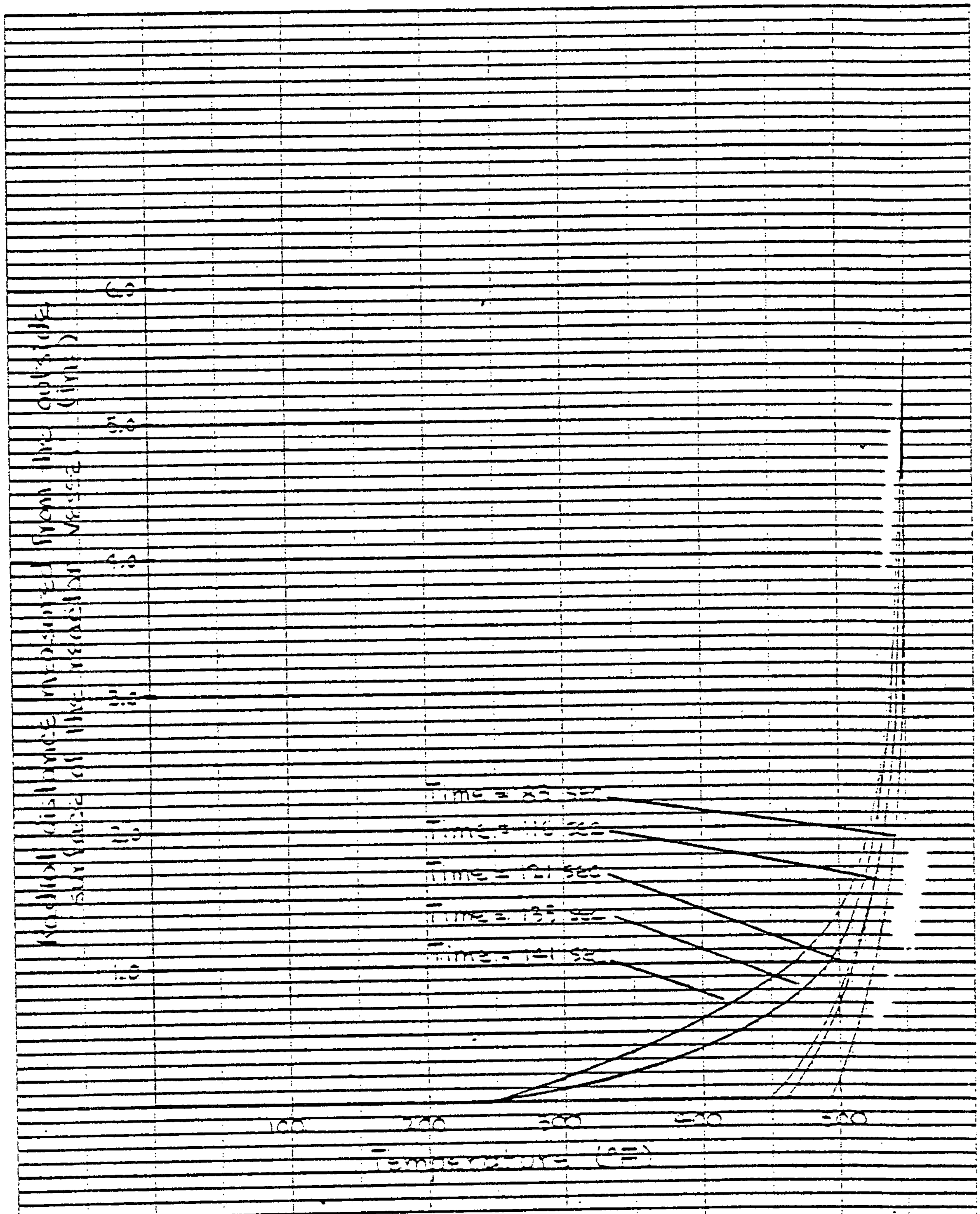

Figure 4b. Transient temperature profiles for Section 2-2 (see Figure 2).

This analysis assumed that the inside surface of both insulation types was maintained at 550^{o}F while the outer surface was cooled by free convection to air at 150^{o}F. These boundary conditions should approximate normal operating conditions. It should be noted, however, that this analysis is concerned with the relative change in heat transfer which accompanies a change in emissivity and not with prediction of the actual thermal performance of the insulation. Therefore, any set of reasonable boundary conditions could be established while varying the emissivity to accomplish this objective.

The results of this analysis are given in Figures 5a and 5b for insulation types HH1THR014 and H81THR024, respectively. Each figure displays the magnitude of the heat transferred as well as the percentage increase in heat transferred from a reference condition as a function of emissivity. The reference condition for both insulation types assumed an aluminum emissivity of 0.10 and a stainless steel emissivity of 0.15. These values assume that the respective materials are clean and smooth[f].

After the insulation was wetted by the flood water and allowed to dry, some change in emissivity will occur. Based on limited literature information, it could be assumed that the emissivity of the aluminum would increase to a value between 0.50 and 0.60 while the emissivity of the stainless steel increases to a value between 0.55 and 0.65[f]. From Figures 5a and 5b, these changes in emissivity would result in a 100 to 135% increase in the amount of heat transferred by both insulation types. A more accurate estimate of the actual effect on the reactor vessel insulation would require some experimental analysis of the affected materials.

(f) Y. S. Touloukian and D. P. Dewitt, Thermal Radiative Properties, Metallic Elements and Alloys, New York: IFI/Plenum Data Corp., 1970.

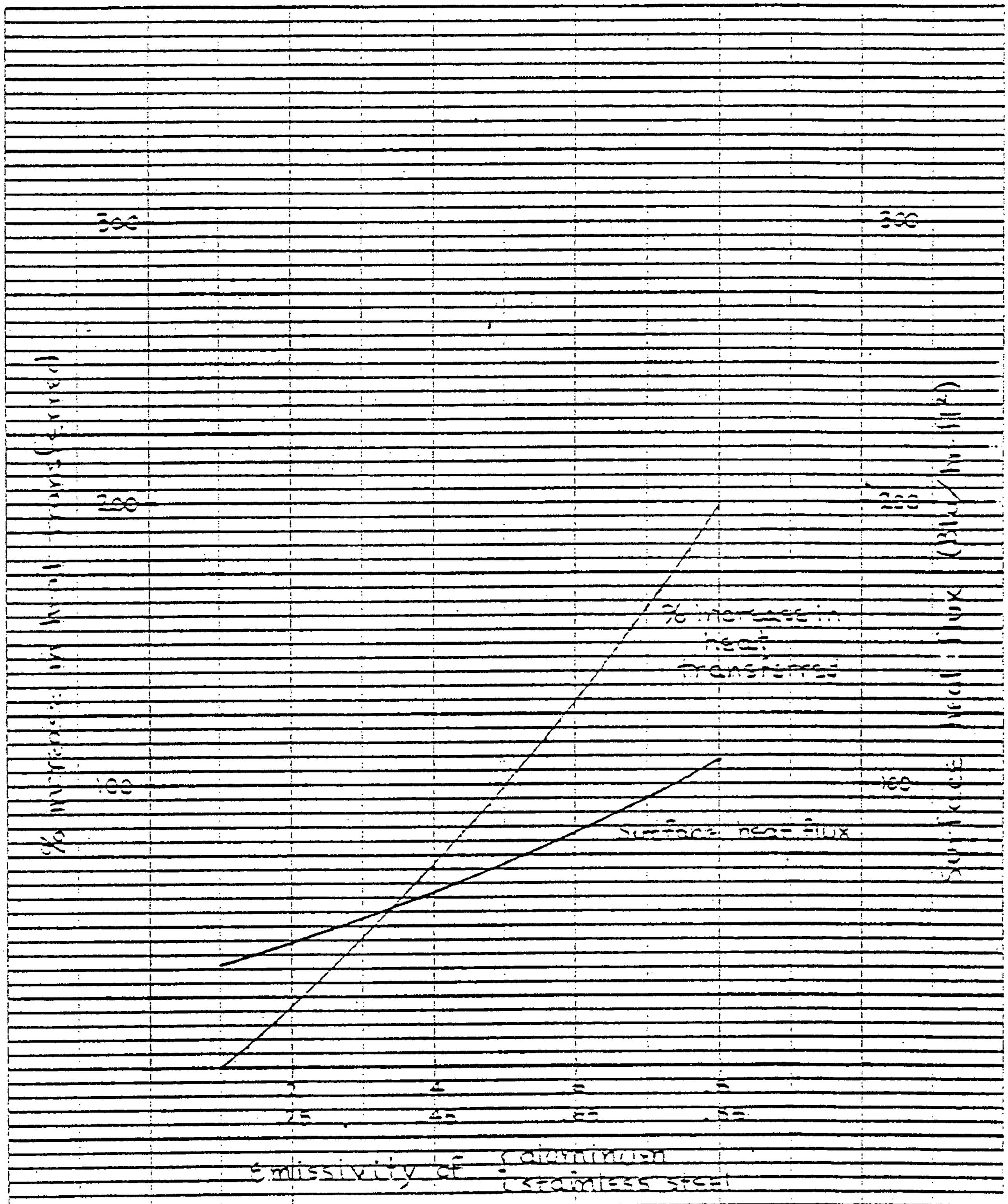

Figure 5a. Surface heat flux and percentage increase in heat transferred by reactor vessel insulation HHITHR014 as a function of emissivity.

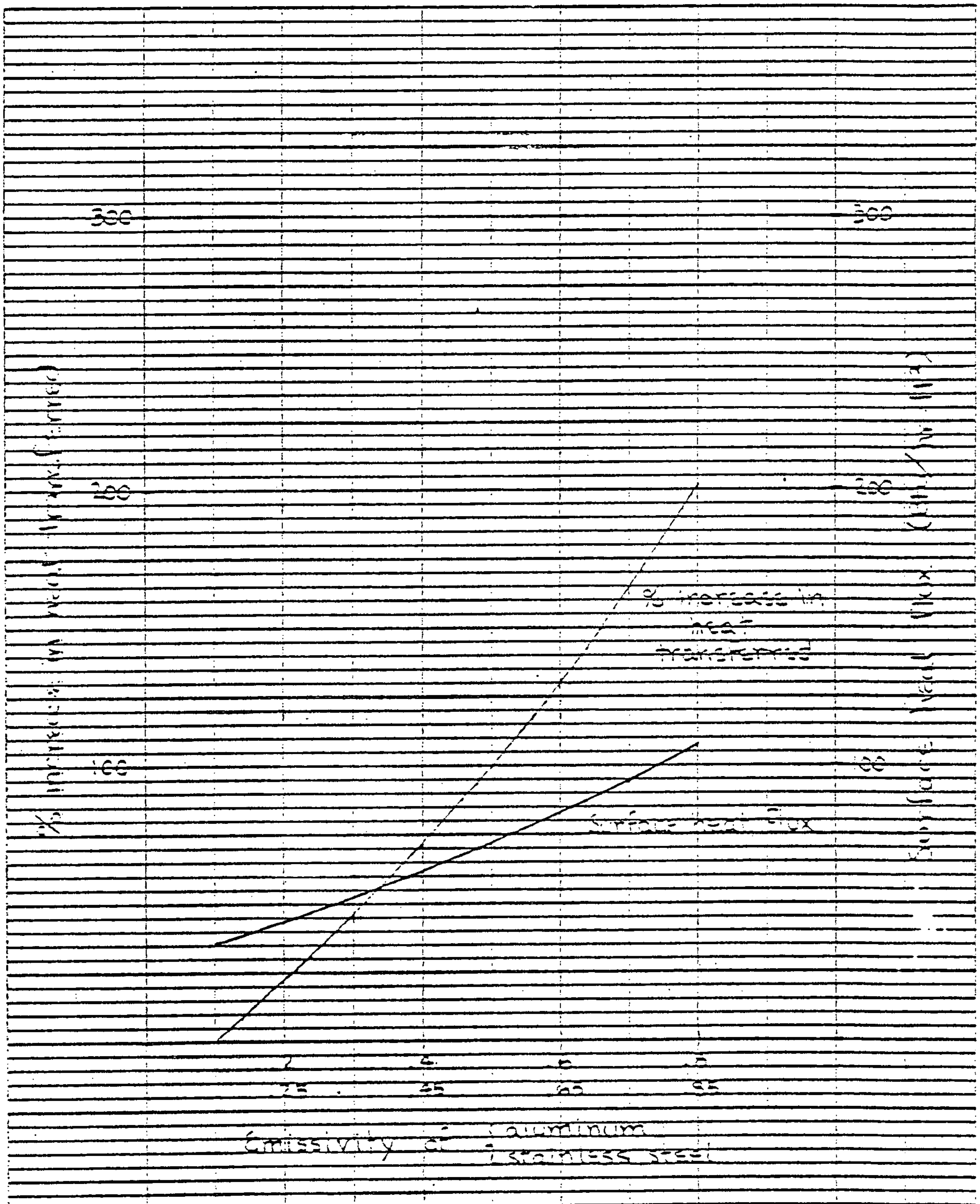

Figure 5b. Surface heat flux and percentage increase in heat transferred by reactor vessel insulation HB1THR024 as a function of emissivity.

STRESS ANALYSIS OF INDIAN POINT 2 REACTOR VESSEL FOR RIVER QUENCH CONDITION

INTRODUCTION

The Indian Point reactor vessel was inadvertently exposed to river water to a height of about nine feet above the bottom of the vessel. A stress analysis is contained in this report for this quenching condition.

Two separate axisymmetric finite element models were used in the analysis. A model was made of the lower section of the vessel including the lower head and a section of the cylindrical portion of the vessel. A second model was made of an individual penetration tube and a portion of the vessel wall around the penetration. This model was used to investigate stress concentration effects near the penetrations.

In the following sections of the report each model will be discussed and the results of the analysis will be presented. In the final section, a summary will be given and conclusions will be presented.

REACTOR VESSEL ANALYSIS

The stress analysis model for the Indian Point pressure vessel is axisymmetric and includes the lower hemispherical head and a portion of the cylindrical section above the maximum water level location. The height of the cylindrical section above the maximum water level is greater than the characteristic length defined by the ASME Code, Section III, so end effects are eliminated. The model is shown in Figure 1. The steady state temperature of the vessel is 550°F before the quench occurs so the upper end of the model is fixed in the axial direction at all nodes through the thickness with a stress free reference temperature of 550°F. The upper end of the model is left free to translate in the radial direction.

The thermal analysis consisted of steady state runs for various water levels. The qualitative stress analysis results indicated that the maximum surface stress is seen just below the water level on the outside surface and a little smaller surface stress continues down around the bottom head. The surface stress drops rapidly above the water level.

The highest stress for the various steady state conditions was found with the water level at the top of the hemispherical section. The maximum stress occurred two elements below the water surface and the stress distribution is shown in Figure 2. The stress intensity on the outside surface was calculated using the stress components extrapolated to the surface, as shown on Figure 2. The resulting stress intensity was 68,000 psi. This stress is less than 3 S_m so there is no reduction in fatigue life caused by plasticity. The resulting number of allowable cycles is about 15,000. The fatigue usage for one cycle of the river quench is 6.7×10^{-5}. The cumulative usage factor for all previously defined reactor transients is 3×10^{-3}. So the river quench causes very little additional usage.

PENETRATION ANALYSIS

An axisymmetric model was also made for the penetration tube assembly. A section of the shell out to a radius of 6.75 in from the penetration tube center line was modeled as an axisymmetric flat plate. Boundary conditions were applied to the outside edge of the model as will be discussed later in this section of the report. The model is shown in Figure 3.

The thermal analysis included two steady state conditions and a transient condition. A steady state thermal analysis was made with the water touching the penetration tube up to just below the vessel surface. This condition could possibly give high stresses in the weld between the penetration tube and the shell. A second steady state analysis was made with the water touching both the penetration tube and the outside surface of the vessel. This condition would produce the maximum gradient through the vessel wall. A thermal transient analysis was run with the water rising up the tube and onto the vessel surface to investigate the possibility of large peak thermal gradients on the surface caused by changing boiling regimes.

The stress analyses for the steady state conditions showed that for the case with water touching the vessel surface, the stresses in the weld are larger than for the other steady state condition mentioned above. The stresses during the transient are lower in the weld region than for the steady state condition.

Since the penetration model is only a section of the whole vessel, a radial stress distribution was applied to this model to represent the effect of the rest of the vessel. This radial stress distribution was obtained from the vessel model and applied to the penetration as a pressure.

The stress intensity at the top of the weld (point 1 of Figure 3) is about 72,000 psi and at the bottom of the weld it is about 53,000 psi. By approximation, the linearized surface stress intensity is less than 3 S_m so K_e will be assumed to be 1.0. Using 72,000 psi as S_n and a concentration factor of 4.0, the alternating stress is 144,000 psi which yields about 150 allowable cycles. One cycle of quenching would, therefore, produce a usage factor of 6.7×10^{-3}. (The stresses in the weld region are shown in Figure 4.)

Stress analysis computer runs were made at several times during the thermal transient. A radial stress distribution was applied to the model at each time step to represent the effect of the rest of the reactor vessel. This radial stress distribution was found by calculating the stresses which would be present in a sphere of radius equal to the radius of the hemispherical head of the vessel, caused by the thermal gradient and by pressure.

At the center of the element nearest to the outside surface of the vessel and also to the penetration, the highest component of stress was found to be 136,650 psi at 140.8 seconds into the transient. The steady state value of this stress component was 108,148 psi. Therefore, there is additional stress due to a peak thermal gradient.

The fatigue reduction factor, K_e, was found to be 1.0 based on the linearized stress distribution shown in Figure 5. The stress distribution shown is at a location sufficiently far away from the penetration so that no stress concentration effects are present. The 3 S_m value for the vessel at 550°F is 80,100 psi.

The stress components on the outside surface of the vessel at the periphery of the hole (point 2 of Figure 3) were found by extrapolating in the radial direction first and then in the axial directions. No stress concentration factor was applied at the hole because of this two-way extrapolation and the use of a relatively fine mesh. The resulting stress intensity is about 161,000 psi. Combining this stress with zero stress for the other end of the cycle yields a stress range of 161,000 psi; a fatigue life of 850 cycles. The additional usage factor for one cycle of quenching is 1.18×10^{-3}. The total usage factor for previously defined transients at the penetration is 0.142. Again, as in the vessel analysis section of this report, little additional fatigue usage is caused by one cycle of quenching with river water.

SUMMARY AND CONCLUSIONS

Two analyses have been presented in this report. The first investigates the overall effect of the river quenching on the reactor vessel and the second examines the local effects near the vessel penetrations. In both cases, the results have shown that little additional fatigue usage has been caused by the quenching condition. When the additional fatigue is combined with the fatigue usage for defined reactor transients given in the FSAR, the resulting usage is still much less than the allowable 1.0.

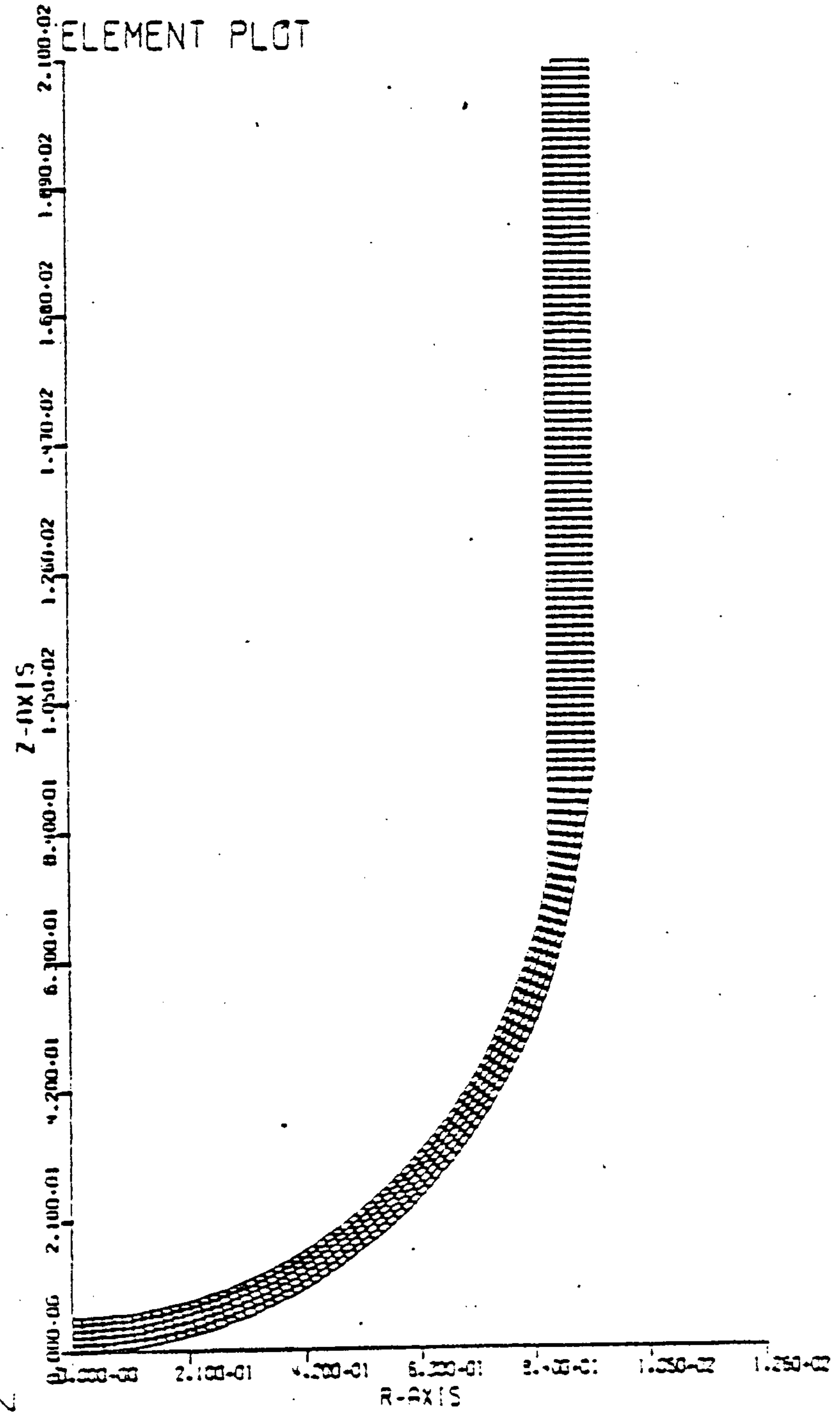

Figure 1. Finite element model of lower section of reactor vessel.

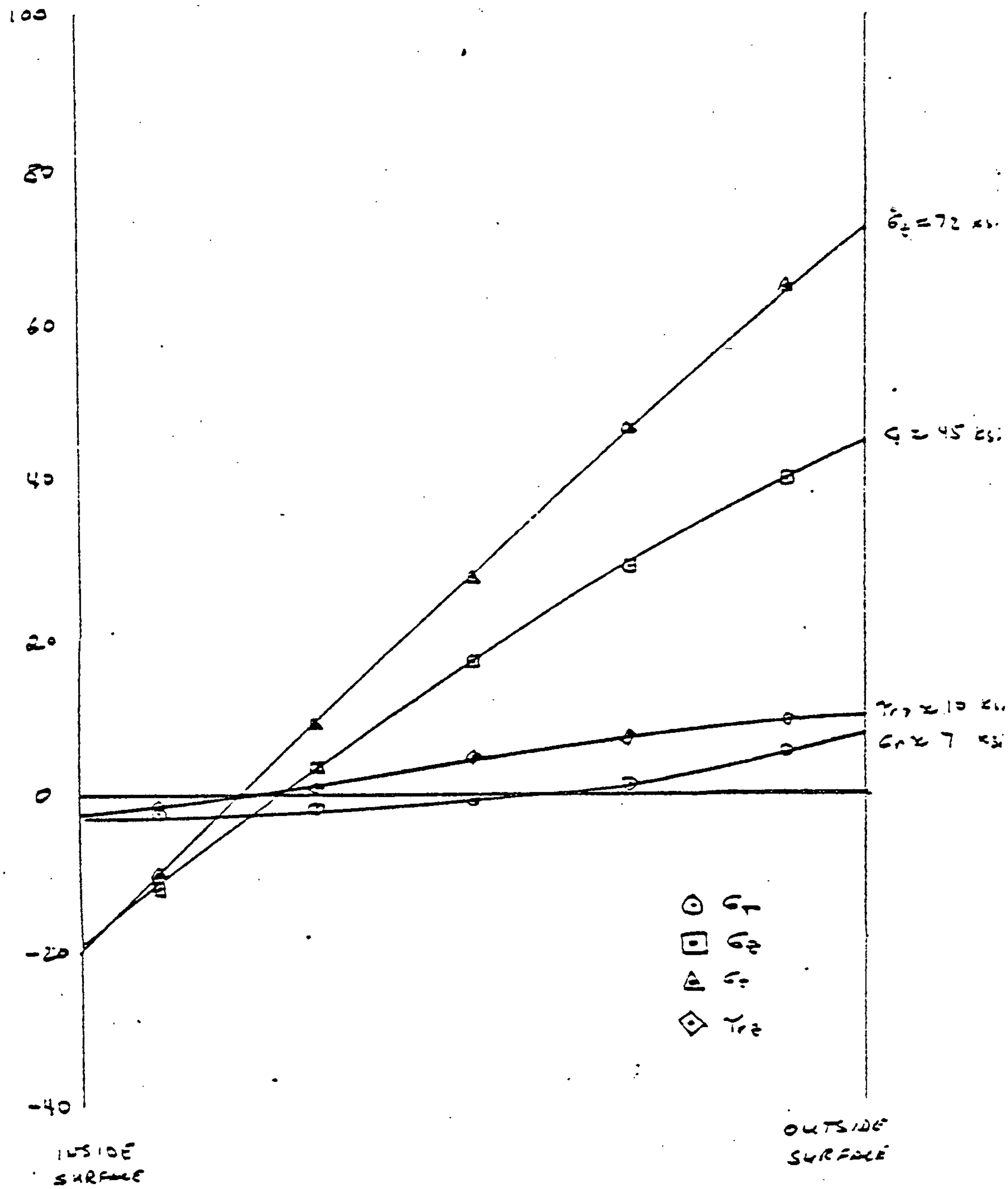

Figure 2. Stress distribution through reactor vessel wall.

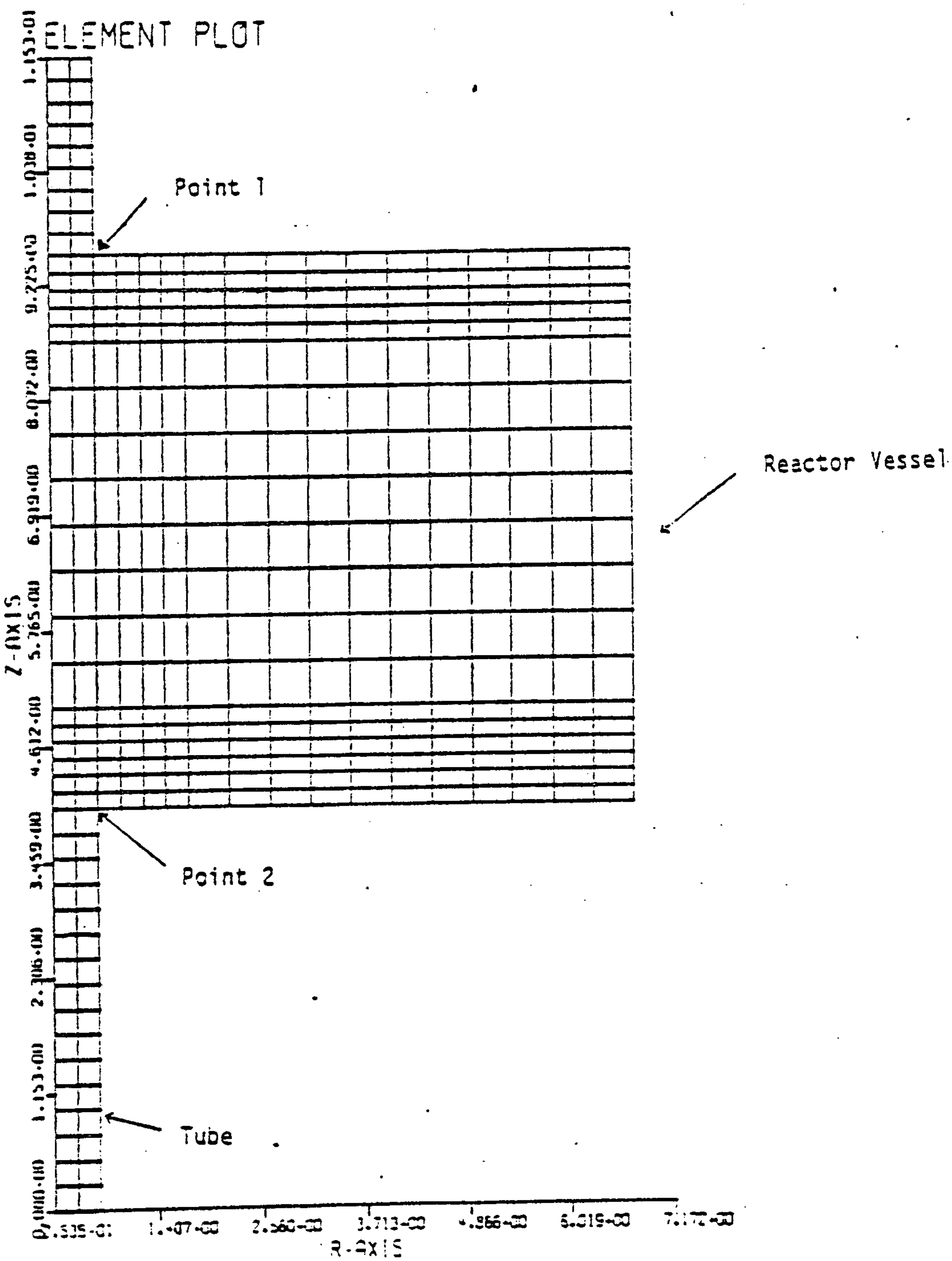

Figure 3. Penetration region finite element model.

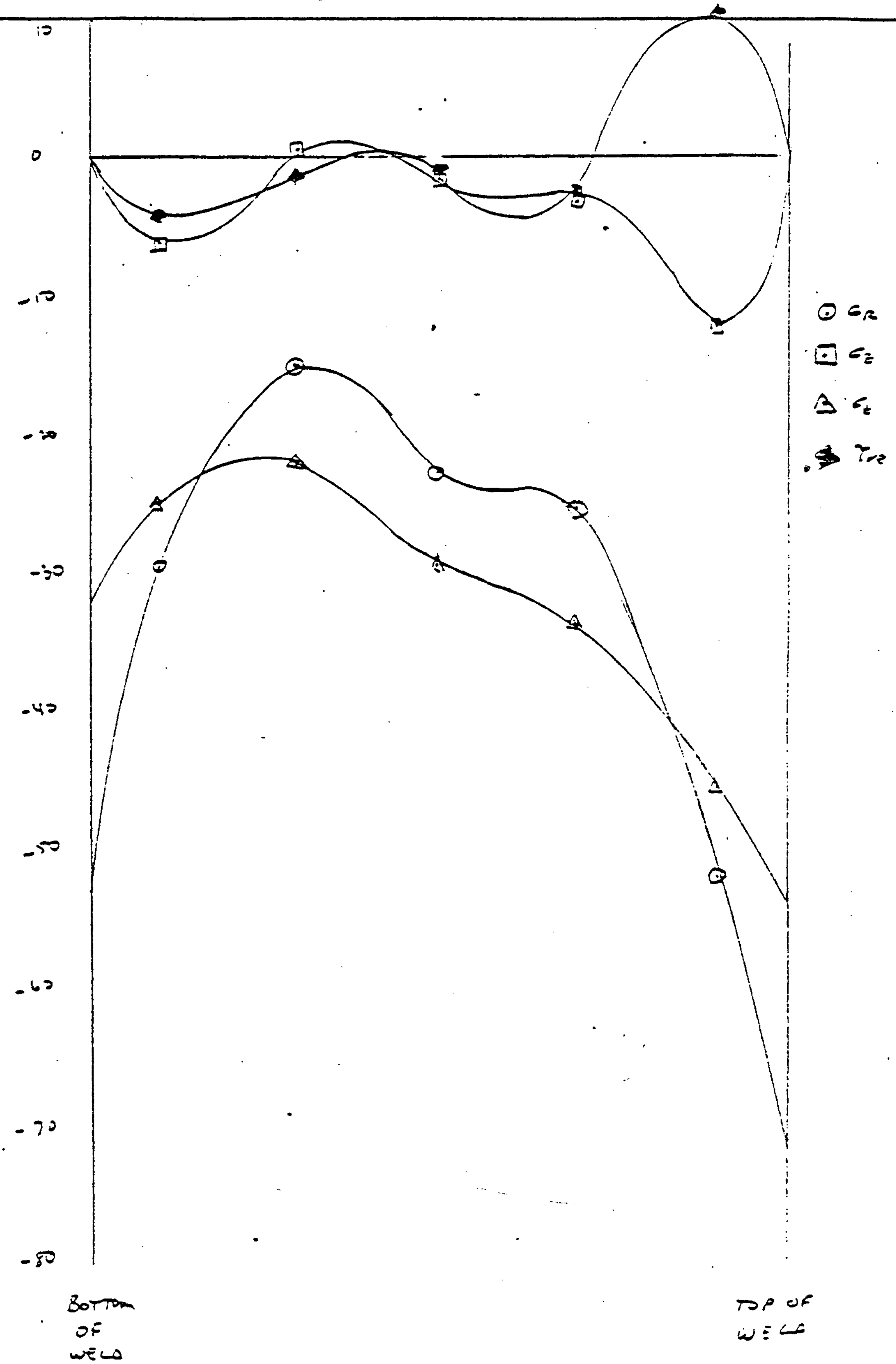

Figure 4. Stresses through depth of weld (steady state).

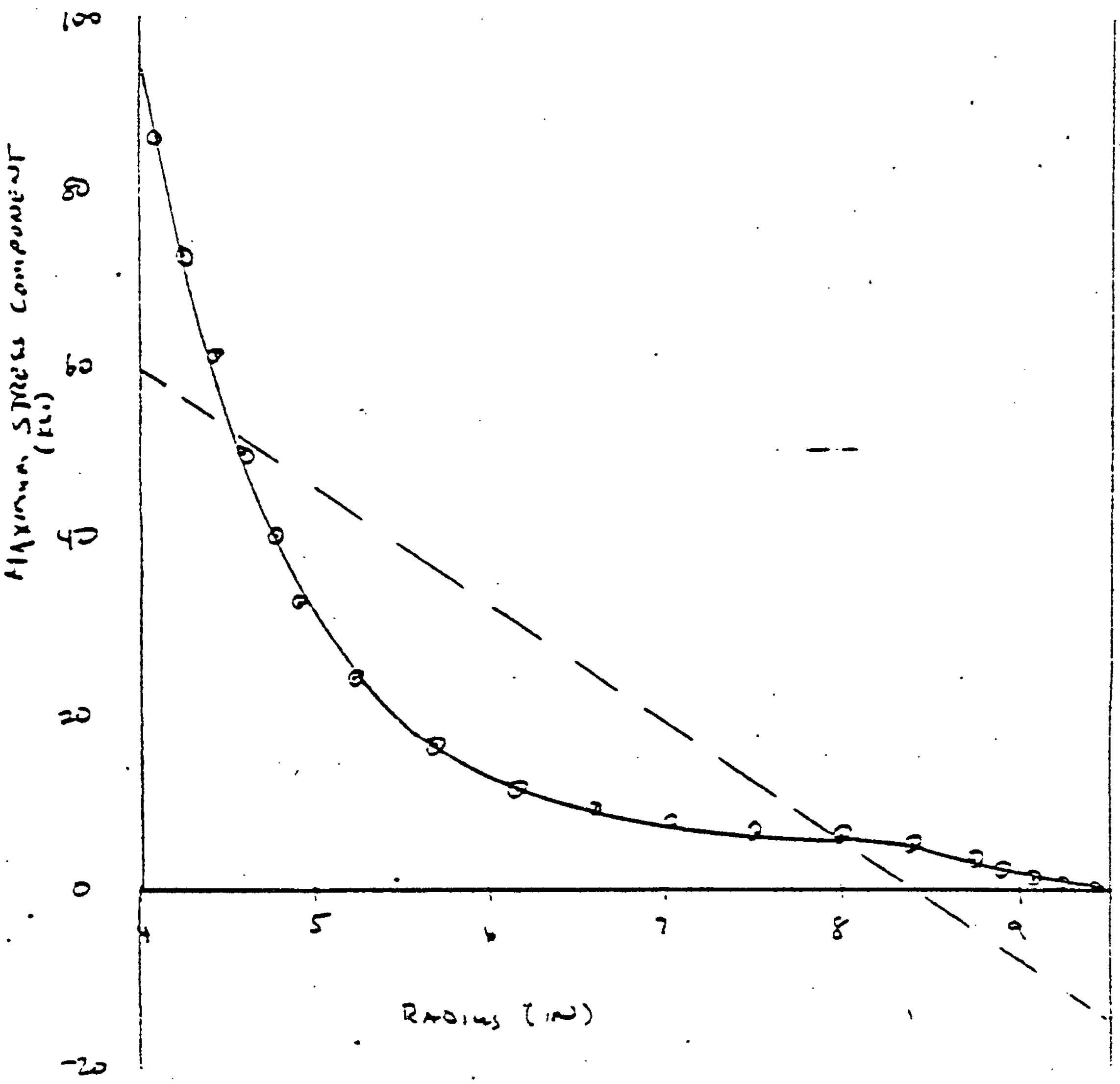

Figure 5. Stress distribution through vessel wall in the region of the penetration at 140.8 seconds of transient.

ENCLOSURE 3

Independent Nondestructuve Examination
of the
Reactor Vessel Lower Shell, Stub Tube, and Conduit Welds
from
Indian Point Unit 2
of
Consolidated Edison Company of New York

Report No. IE-122
February 3, 1981

Prepared for: United States Nuclear Regulatory Commission
Office of Inspection and Enforcement

NRC Contract 05-80-251
PAR: NRC-IE-80/81, Task 06

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Consulting Engineers
Elm Grove, Wisconsin

Reviewed: Kenneth A. Ristau
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Reviewed: Walter J. Foley
Walter J. Foley, P.E.

NOTICE

Report No. IE-122

Section 1

PARAMETER, Inc.

Introduction and Summary of Results

Parameter, Inc.
CONSULTING ENGINEERS
ELM GROVE, WISCONSIN

Table of Contents

Note: For description of Appendices A through F, refer to cover sheet of Section 2.

Parameter, Inc.
CONSULTING ENGINEERS
ELM GROVE, WISCONSIN

Scope of Work

Independent nondestructive examination of the Indian Point Unit 2 reactor vessel lower shell welds, stub tube welds and conduit welds that were exposed to leaking service water is provided, reported and documented in accordance with the Specified Statement of Work included in Section 1 of this report.

The report of nondestructive examination (Section 2 of this report) is reviewed independently to make sure that documentation is adequate to show that all assigned tasks have been performed and that personnel, equipment, material and procedures have been qualified and certified in accordance with the Specified Statement of Work.

Background (Excerpted from NRC-IE Assignment of Task 06)

The Indian Point Unit 2 plant, owned by the Consolidated Edison Company of New York, was operating at full power early on October 17 when a nuclear instrument malfunctioned. The plant was shut down at 4:15 a.m., and workers entered the containment to investigate the problem at about 11 a.m. A large quantity of water was found on the containment building floor and subsequently also in the cavity under the reactor vessel. The containment sumps were filled with water. The total quantity has been estimated to be about 100,000 gallons. The principal source of leakage was from fan cooler units which are used to cool the air inside containment. Subsequent to the 4:15 a.m. shut down on October 17, the reactor was restarted twice on that day, and again on the morning of October 20, before the plant was placed in a cold shut down condition on October 22.

The cavity under the vessel accumulated service water which resulted in wetting the bottom of the vessel to a height of about nine feet.

Parameter, Inc.
CONSULTING ENGINEERS
ELM GROVE, WISCONSIN

Specified Statement of Work (Excerpted from NRC-IE Assignment of Task 06)

The required nondestructive examinations and evaluations required are as follows:

a. Provide a technical evaluation of the suitability of performing magnetic particle examination of the Indian Point Unit 2 (IP-2) reactor vessel lower head without removal of the protective coating.

b. Perform procedure qualification to demonstrate the magnetic particle examination methodology to be used is capable of detecting the flaws in the licensee's coated calibration standard.

c. Supplying the necessary qualified (SN-TC-1A) personnel and equipment, perform and evaluate the results of magnetic particle inspections (ASME Section V) using the AC yoke method of the following IP-2 reactor vessel welds:

   (1) The circumferential lower head to shell weld.

   (2) The meridional welds (orange peel) in the lower head.

   (3) The circumferential weld (dollar piece) in the lower head.

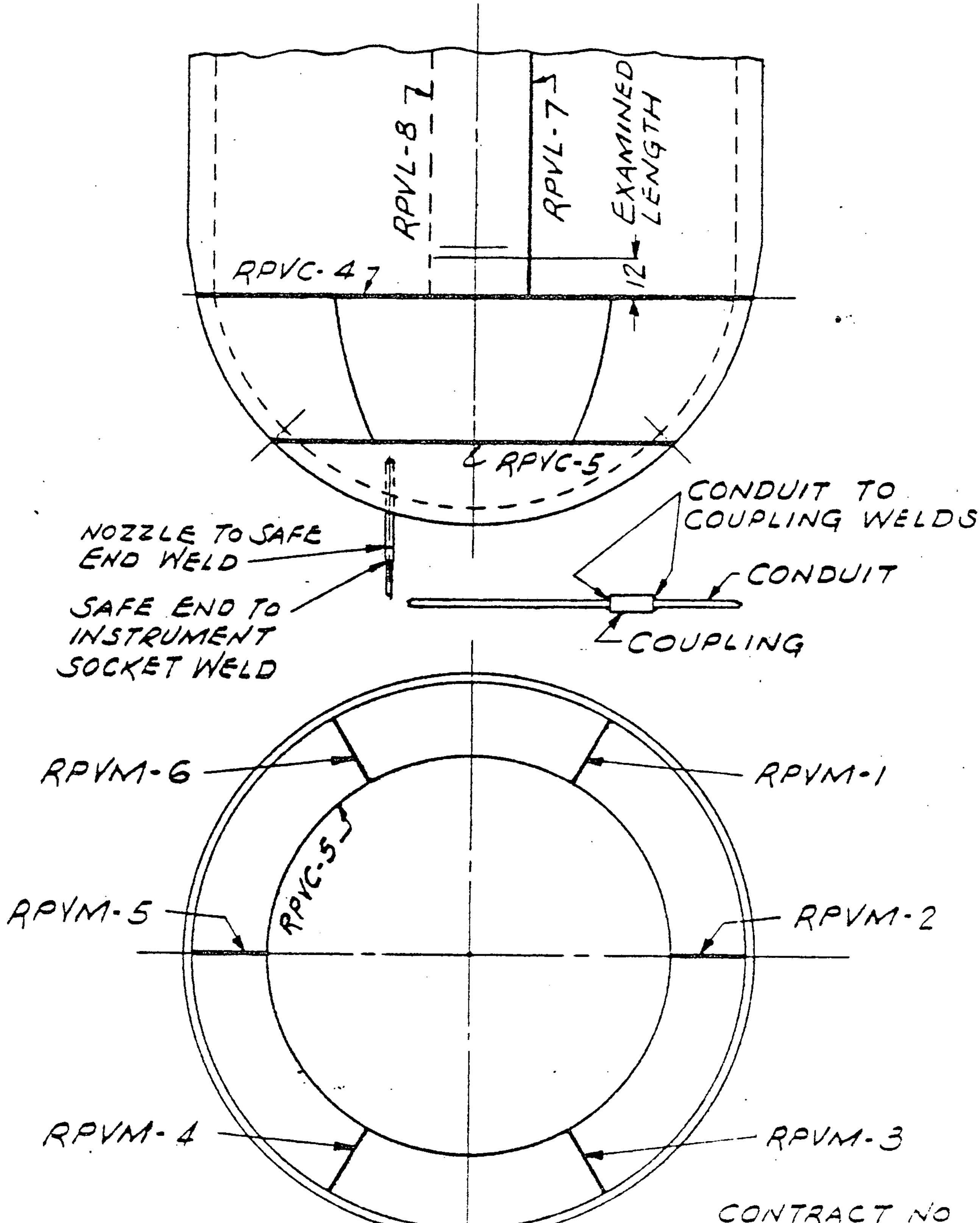
   (4) One foot of the longitudinal shell welds intersecting the circumferential shell to lower head weld.

d. Supplying the necessary qualified (SN-TC-1A) personnel and equipment, perform liquid penetrant inspections (ASME Section V) of the following:

   (1) 25% of the instrument nozzle to safe-end and safe-end to instrument socket welds. Those nozzle which are observed to have longitudinal marks should be included in the sample.

   (2) 10% of the conduit welds which could have been exposed to the leaking service water.

e. Other consultation and support as may be required for the evaluation of the instrument conduit piping or the IP-2 reactor vessel.



BOTTOM VIEW

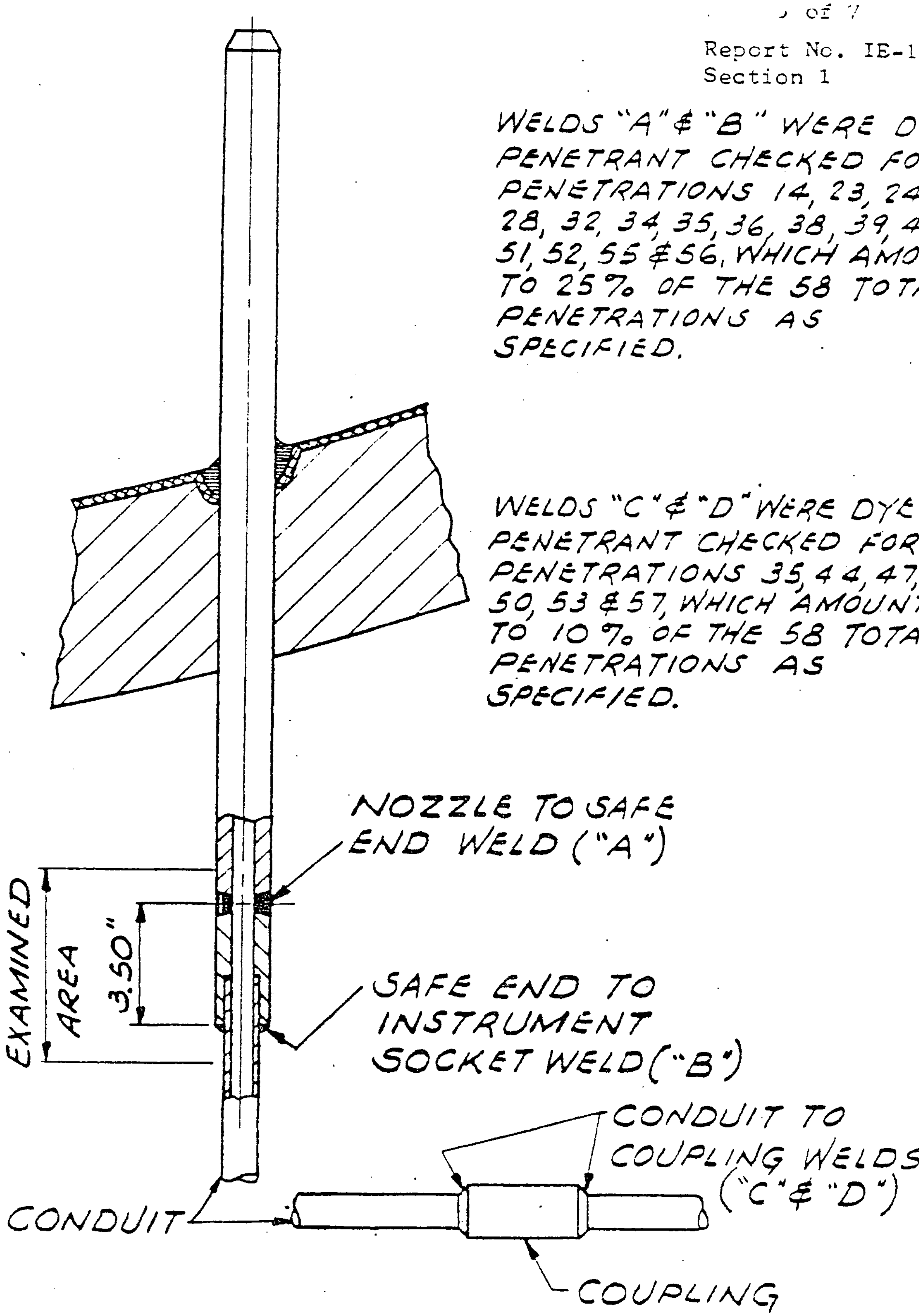
BOTTOM HEAD WELDS

CONTRACT NO
NRC-05-80-251
PARAMETER, INC
NRC-IE-80/81
TASK ORDER NO. 6

Report No. IE-122
Section 1

WELDS "A" & "B" WERE DYE PENETRANT CHECKED FOR PENETRATIONS 14, 23, 24, 28, 32, 34, 35, 36, 38, 39, 42, 51, 52, 55 & 56, WHICH AMOUNT TO 25% OF THE 58 TOTAL PENETRATIONS AS SPECIFIED.

WELDS "C" & "D" WERE DYE PENETRANT CHECKED FOR PENETRATIONS 35, 44, 47, 50, 53 & 57, WHICH AMOUNT TO 10% OF THE 58 TOTAL PENETRATIONS AS SPECIFIED.



INSTRUMENT NOZZLE & CONDUIT WELDS

CONTRACT NO. NRC-05-80-251
PAR: NRC-IE-80/81
TASK ORDER NO.

Parámeter, Inc.
CONSULTING ENGINEERS
ELM GROVE, WISCONSIN

Review of Documentation & Procedures

1. Documentation provided by Peabody Testing Services in Section 2 of this report shows that the tasks required by the Specified Statement of Work in Section 1 have been performed as required.

2. Documentation provided by Peabody Testing Services in Section 2 of this report shows that personnel, equipment, procedures and materials have been qualified and certified as required by the Specified Statement of Work in Section 2.

3. Peabody Testing Services nondestructive examination procedures 21.A.3-4 Rev. 1 (Appendix "E") and 23.A. 1-4 Rev. 1 (Appendix "F") are adequate for the tasks required by the Specified Statement of Work.

Reviewer: Kenneth A. Ristau
Kenneth A. Ristau
NDE Level III Examiner
- Magnetic Particle
- Dye Penetrant
- Radiography
- Ultrasonic

Parámeter, Inc.
CONSULTING ENGINEERS
ELM GROVE, WISCONSIN

Summary of Results

1. The procedure for performing magnetic particle examination of the lower head without removing the coating of paint has been qualified, certified and documented in accordance with the Specified Statement of Work.

2. No indications of discontinuities were identified by magnetic particle examination of lower head welds.

3. No rejectable indications were identified by dye penetrant examination of instrument nozzle and conduit welds.

4. Documentation is adequate to show that all assigned tasks have been performed and that personnel, equipment, material and procedures have been qualified and certified in accordance with the Specified Statement of Work.

Report No. IE-122

Section 2

Peabody Testing Services

Summary Report of Nondestructive Examination

Report No. IE-122

Section 2

Peabody Testing Services

Nondestructive Examination Report

Contents

Note: Original documentation of qualification and certification of personnel is on file in offices of Peabody Testing Services. Copies of these documents are included in Appendix "B".

NONDESTRUCTIVE EXAMINATION REPORT

PARAMETER, INC. CONTRACT NO. NRC-05-80-251

TASK ORDER NO. 6

INTRODUCTION

Peabody Testing Services Division of Magnaflux Corporation were engaged by Parameter, Inc., Consulting Engineers, Elm Grove, Wisconsin to perform Nondestructive Examination in accordance with Contract No. NRC-05-80-251 Task Order No. 6.

The testing crew consisted of the following Peabody Nondestructive test technicians:

Level III-Michael Sherwin
Level III-Joseph Gagnon
Level II -Henry Sibits
Level II -Ron Belline
Level II -Dennis Saskowski
Level II -John Lyons

The purpose of inspection was to nondestructively examine the Indian Point Unit #2 Reactor vessel lower shell welds, stub tube welds and conduit welds that were exposed to leaking service water for approximately nine feet up from the bottom of the vessel.

RESULTS OF NONDESTRUCTIVE

EXAMINATION CONTRACT #NRC-05-80-251

TASK ORDER #6

The following results of nondestructive examination and evaluation are described as they would follow in the order required by the statement of work of this contract.

A) Suitability for performing Magnetic Particle examination of Indian Point Unit #2 (IP-2) reactor vessel lower head without removal of the protective coating was established by Michael Sherwin, Peabody Testing Level III.

B) Magnetic Particle examination procedure qualification was performed and documented on Appendix "A" by Mike Sherwin, Peabody Testing Level III, and witnessed by R. McBrearty US-NRC. The purpose for this procedure qualification was to demonstrate that the Magnetic Particle Yoke examination procedure used at Indian Point Unit #2 (IP-2) was definitely capable of detecting flaws in the licensee's coated calibration standard. The Procedure Qualification was performed in the down position, however, overhead and various out of position testing was qualified by the ability to detect defects in the standard thru paint and the addition of 40 mil of tape on top of the paint.

C) Necessary nondestructive examination personnel certified and qualified in accordance with SNT-TC-1A 1975 edition and ASME Section V, Boiler and Pressure Vessel Code Requirements (Sec. III) were provided to perform the testing in accordance with the statement of work for contract NRC-05-80-251 Task Order #6 para. C. Copies of personnel and equipment and material certifications are located in Appendix "B" of this report. Actual Magnetic Particle results are as follows:

Magnetic Particle AC Yoke examination was performed in accordance with ASME Boiler and Pressure Vessel Code requirements Section V and Section III and in accordance with Peabody Testing Services written A.C. Yoke procedure 21.A.304 Rev. 1. The following reactor pressure vessel welds were examined:

VERTICAL WELDS		CIRCUMFERENTIAL WELDS
RPVM-1		RPVC-5 (Dollar Weld)
RPVM-2		RPVC-4 (Head to shell weld)
RPVM-3		
RPVM-4	12 in. of RPVL-7	
RPVM-5	12 in. of RPVL-8	
RPVM-6		

C) Continued

No indications of discontinuities were identified at this time. Copies of actual Magnetic Particle data sheets for the above inspection are located in Appendix "C" of this report.

D) Necessary nondestructive test personnel certified and qualified in accordance with SNT-TC-1A 1975 edition and ASME Boiler and Pressure Vessel Code requirements (Section III) and (Section V) were provided to perform the testing in accordance with the statement of work for contract NRC-05-80-251 penetrant examination.

Copies of personnel, equipment, and materials certifications utilized during penetrant examination are located in Appendix "B" of this report.

Actual penetrant test results are as follows:

Penetrant examination was performed on the following welds in accordance with Peabody Testing Services Penetrant Procedure 23.A.1-4 Rev. 1.

Instrument Nozzle to Safe-End and Safe-End to Instrument Socket Welds

Weld #51	Weld #25	Weld #52
Weld #34	Weld #28	Weld #32
Weld #38	Weld #39	Weld #55
Weld #14	Weld #23	Weld #56
Weld #42	Weld #24	Weld #36

The above 15 pairs of welds represent more than 25% of the total instrument nozzle to safe-end and safe-end to instrument socket welds as required by the scope of work in this contract. No rejectable indications were identified on any of the above 15 pairs of welds.

Conduit Welds

Bay-4 #35	Bay-4 #50
Bay-4 #44	Bay-4 #53
Bay-4 #47	Bay-4 #57

The above 6 pairs of welds represent more than 10% of the total conduit-to-coupling welds as required by the scope of work in this contract. No rejectable indications were identified on any of the above 6 pairs of welds.

Copies of penetrant data sheets of the above inspections are located in Appendix "D" of this report.