~~00046~~

0130

592

8004090 ~~589~~

CONTENTS

APPENDIX 4

~~00047~~

0131

Amendment 3

assumed. The crack arrest temperature through the thickness of the wall was developed on a stress-temperature coordinate system. The actual quench-induced, stress-temperature condition through the thickness of the wall at several times during the quench was developed and plotted (Figures 4A.1-4 and 4A.1-5). The maximum depth at which the material in the vessel wall would be in tension or at which the stress in the material would be in excess of the threshold stress for crack initiation (5-8 ksi) was determined by comparison of the plots. Comparison shows that a crack could propagate only through the inner 35 percent of the wall thickness if a crack initiation threshold of 5-8 ksi is applicable, and further that a crack could propagate only through the inner 43 percent of the wall thickness if a crack initiation threshold of zero were assumed.

The foregoing method of analysis is essentially a stress analysis approach which assumes the worst conceivable material properties and a flaw size large enough to initiate a crack. Actually, the outer 83 percent of the vessel wall is at a temperature above the RTT (NDTT + 60 F) when credit is taken for the neutron shielding, and for the original RTT profile through the wall thickness. The analysis is conservative in that it does not deny that cracks can be initiated, and in that it assumed a crack from 1 to 2-ft long to exist in the vessel wall at the time of the accident. Therefore, it can be concluded that, if a crack were present in the worst location and orientation (such as a circumferentially oriented crack on the inside of the vessel wall), it could not propagate through the vessel wall.

A fracture mechanics analysis was conducted which assumed a continuous surface flaw to exist on the inside surface of the vessel wall. The criterion used for the analysis is that a crack cannot propagate when the stress intensity at the tip of the crack is below the critical crack stress intensity factor (K_{IC}). Using conservative values of K_{IC} (for fully irradiated cold 302-Grade B steel K_{IC} equals 30,000 psi) [3] and the method of Emery [4] to calculate stress intensity factors, K_I, in the variable thermal transient stress field, it was found that the crack propagating energy is below that required for crack propagation when the crack reaches a depth of less than 3 in. or 35 percent of the wall thickness.

4A.1.1 The geometry of the plate and the cooling method assumed in the analysis,

ANSWER The analysis assumed a long cylindrical section which was insulated on the outside and subjected to a uniform flow of constant temperature (90 F) cold water flowing past the inner wall of the reactor vessel and outer wall of the thermal shield. For general dimensions of the thermal model and flow patch description, refer to Figure 4A.1-1.

0132 ~~00048~~

QUESTION THERMAL SHOCK ON REACTOR VESSEL

4A.1 With regard to thermal shock on the primary system components, induced by operation of the emergency core cooling system (ECCS), provide details of an analysis which indicates that the reactor vessel can accommodate without failure the rapid temperature change at the end of its design life. The analysis should consider both the ductile yielding and the brittle fracture modes of failure, and should include the following specific information: (See these further specific items following the answer to Question 4A.1.)

ANSWER
Refer to
4.3.1.1

The state of stress in the reactor vessel during the loss-of-coolant accident has been evaluated for an initial vessel temperature of 603 F. The inside of the vessel wall is rapidly subjected to 90 F injection water at the maximum flow rate obtainable. The results of this analysis show that the integrity of the vessel is not violated.

The assumed modes of failure are ductile yielding and brittle fracture. The modes of failure are considered separately as follows:

a. Ductile Yielding

The criterion for this mode of failure is that there shall be no gross yielding across the vessel wall using the minimum specified yield strength in the ASME Code, Section III. The analysis considered the maximum combined thermal and pressure stresses through the vessel wall thickness as a function of time during the safety injection. Comparison of calculated stresses to the material yield stress indicated that local yielding may occur in the inner 14.7 percent of the vessel wall thickness.

b. Brittle Fracture

Since the reactor vessel wall in the core region is subjected to neutron flux resulting in embrittlement of the steel, this area was analyzed from both a transition temperature and a fracture mechanics approach. The results of the two methods of analysis compare favorably and show that pressure vessel integrity is not lost.

The criterion used in the transition temperature analysis is that a crack cannot propagate beyond any point where the applied stress is below the threshold stress for crack initiation (5-8 ksi) or when the stress is compressive. [1, 2] This approach involves making the very conservative assumption that all of the vessel material could propagate a crack by a low energy absorption or cleavage mode. End-of-life vessel conditions were

0133 ~~00049~~

4A.1.3 The initial temperature of the vessel as a function of time delay in injecting the cold water,

ANSWER The reactor vessel wall is protected against radiation heating from the hot reactor core by three solid barriers: (a) the core shroud, (b) the core support barrel, and (c) the thermal shield. Each of these barriers is separated by a steam gap so that the reactor vessel is in a sense insulated from the hot core. In addition, the core barrel assembly and the thermal shield have considerable mass, i.e., 63,750 lbs and 47,500 lbs respectively, that must be heated before the reactor vessel wall is affected. The arrangement of these barriers is shown on Figures 3.2-59 and 3.2-60 of the PSAR.

Calculations show that the reactor vessel wall temperature will not increase as a function of time during the first several hundred seconds of an LOCA. The various component temperatures at 500 sec and at 1,400 sec are:

Component	Temperature (at 500 sec), F	Temperature (at 1400 sec), F
Core shroud	731	1,770
Core support barrel	579	770
Thermal shield	576	582
Reactor vessel	576	576

4A.1.4 The effect of axial temperature gradient in the vessel, during filling with cold water, on the total stress intensity,

ANSWER Figure 4A.1-3 shows the temperature profile through the vessel wall when the core flooding water impinges on a section of the vessel wall considering an abrupt line of demarcation between fluid and steam. The use of such an abrupt line of demarcation between fluid and steam is conservative. The conduction of heat through the vessel produces the gradual temperature change as shown on the isotherm plot on Figure 4A.1-3. This temperature distribution has been analyzed using the Seal Shell Computer Program,[9] and the results of this analysis are shown as a stress profile on Figure 4A.1-3. This stress profile shows that the worst stress condition is remote from the line of demarcation between fluid and steam, and that the axial conduction has more than offset any adverse influence of the uncooled portion of the vessel wall. Therefore, the original analysis, assuming a long cylinder subjected to a uniform quench, has presented the worst condition because the effect of the axial gradient will locally decrease the stress produced by ECCS operation in the LOCA.

4A.1.5 The effect of a circumferentially nonuniform cooling of the vessel shell, by the cold water entering the vessel through the injection nozzles, on the stresses and distortion in the vessel,

~~00050~~

The cooling method assumed in the analysis is as follows:

a. The metal in the vessel wall and thermal shield is cooled by conduction.

b. The heat transferred to the fluid is by forced convection.

4A.1.2 The heat transfer coefficient used, its experimental basis, and the degree of conservatism involved,

ANSWER The analysis used a water film heat transfer coefficient of 3,000 Btu/hr-ft^2-F. Using the classical (text book) approach,[5] the water film heat transfer coefficient was calculated to be about 900 Btu/hr-ft^2-F. However, when the water film heat transfer coefficient reaches a value of 2,000 to 3,000 Btu/hr-ft^2-F or more, the heat transfer properties of the metal, i.e., the metal conductivity, will govern the heat transfer rate, and consequently the shape and variation of the temperature profile through the thickness of the vessel wall with time (reference Figure 4A.1-2).

The experimental basis and degree of conservatism for the use of a water film heat transfer coefficient of 3,000 Btu/hr-ft^2-F is as follows:

a. The most severe condition that could possibly be postulated would be to quench the cylindrical portion of the vessel in a quench tank. Much experimental work has been done to determine the water film heat transfer coefficient for this condition.[6,7,8]

   Using Reference 6, the water film heat transfer coefficient (f) is calculated as follows:

$$f = 2H_G k$$

$$f = 2 \times 4 \times 277$$

$$f = 2,216 \text{ Btu/hr-ft}^2\text{-F}$$

   where:

   f = water film heat transfer coefficient, Btu/hr-ft^2-F

   H_G = Grossman's Severity of Quench (= 4 in violently agitated water)

   k = thermal conductivity of the material, Btu/hr-ft^2-F/in. (= 277 for SA302GB)

b. The comparison of our assumed water film heat transfer coefficient to the coefficient as calculated by Reference 5 yields a conservative ratio of 3.32, and a comparison to the water film heat transfer coefficient, calculated by Reference 6, yields a conservative ratio of 1.35.

0135

~~00051~~

4A.1.11 The value of the yield stress used as the failure criterion in the ductile yielding analysis.

ANSWER The analysis used the minimum yield strength values as a function of temperature, as listed in Table N-424 of Section III of the ASME Code. The values of yield strength for SA 302, Grade B, are as follows:

Temperature, F	Stress, psi
100	50,000
200	47,150
300	45,250
400	44,500
500	43,200
600	42,000

REFERENCES

1 Pellini, W. S. and Puzak, P. P., Practical Considerations in Applying Laboratory Fracture Test Criteria to Fracture Safe Design of Pressure Vessels, NRL 6030.

2 Pellini, W. S. and Puzak, P. P., Fracture Analysis Diagram Procedures for the Fracture Safe Engineering Design of Steel Structures, NRL 5920

3 Landerman, E., Yanichko, S. E., and Hazelton, W. S., An Evaluation of Radiation Damage to Reactor Vessel Steels Using Both the Transition Temperature and Fracture Mechanics Approaches, WAPD.

4 Emery, A. F., "Stress Intensity Factors for Thermal Stresses in Thick Hollow Cylinders," Journal of Basic Engineering, March 1966.

5 Hsu, S. T., Engineering Heat Transfer, Van Nostrand, 1963, pg. 301.

6 Grossman, M. A., Elements of Hardenability.

7 Austin, J. B., Heat Flow in Metals, ASM Publication.

8 Russell, T. F., Russell's Tables.

9 WAPD-TM-398.

~~00352~~

0136

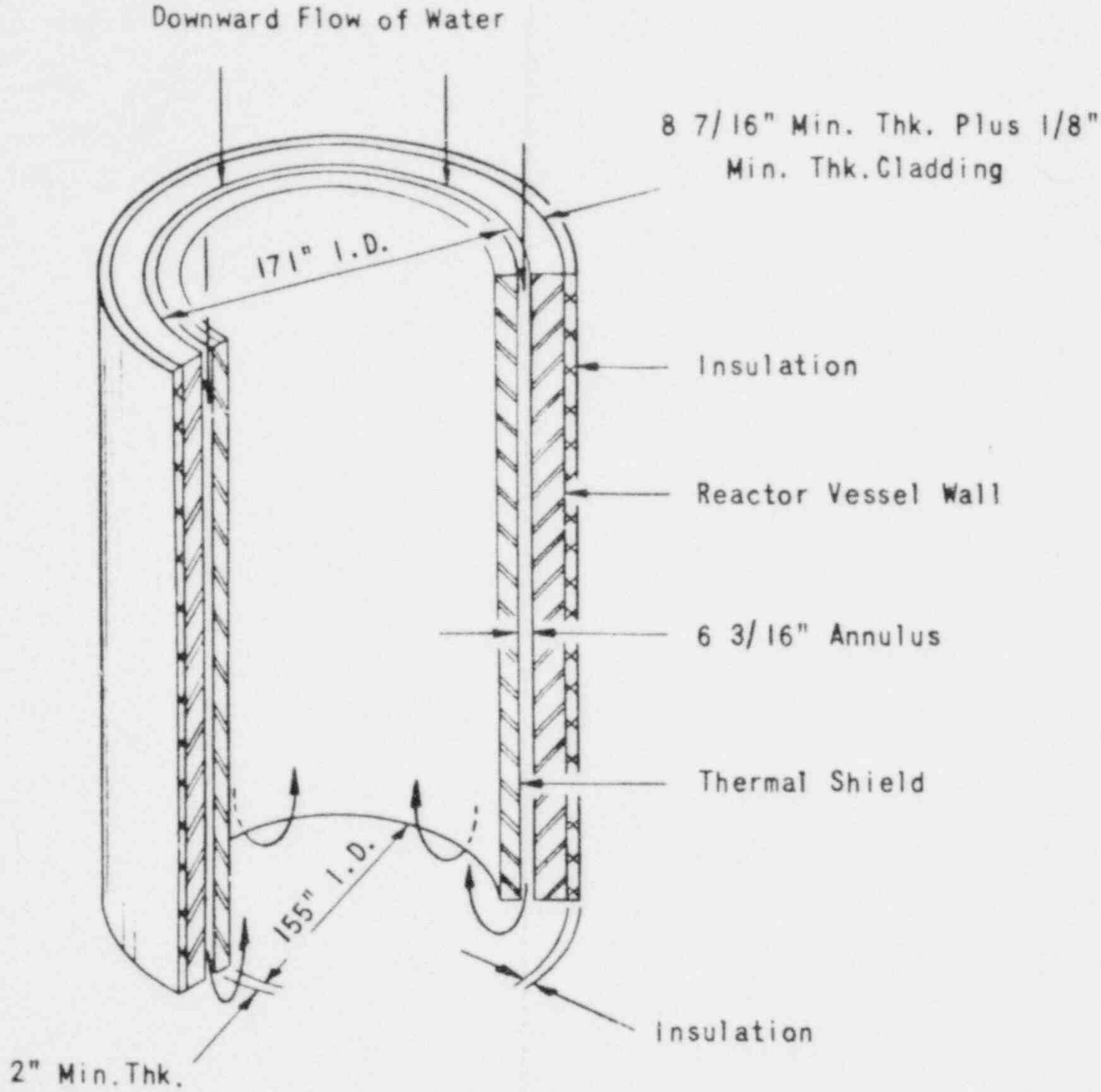

FIGURE 4A.1-1
SCHEMATIC OF THERMAL MODEL
FOR ECCS OPERATION

~~00353~~
0137

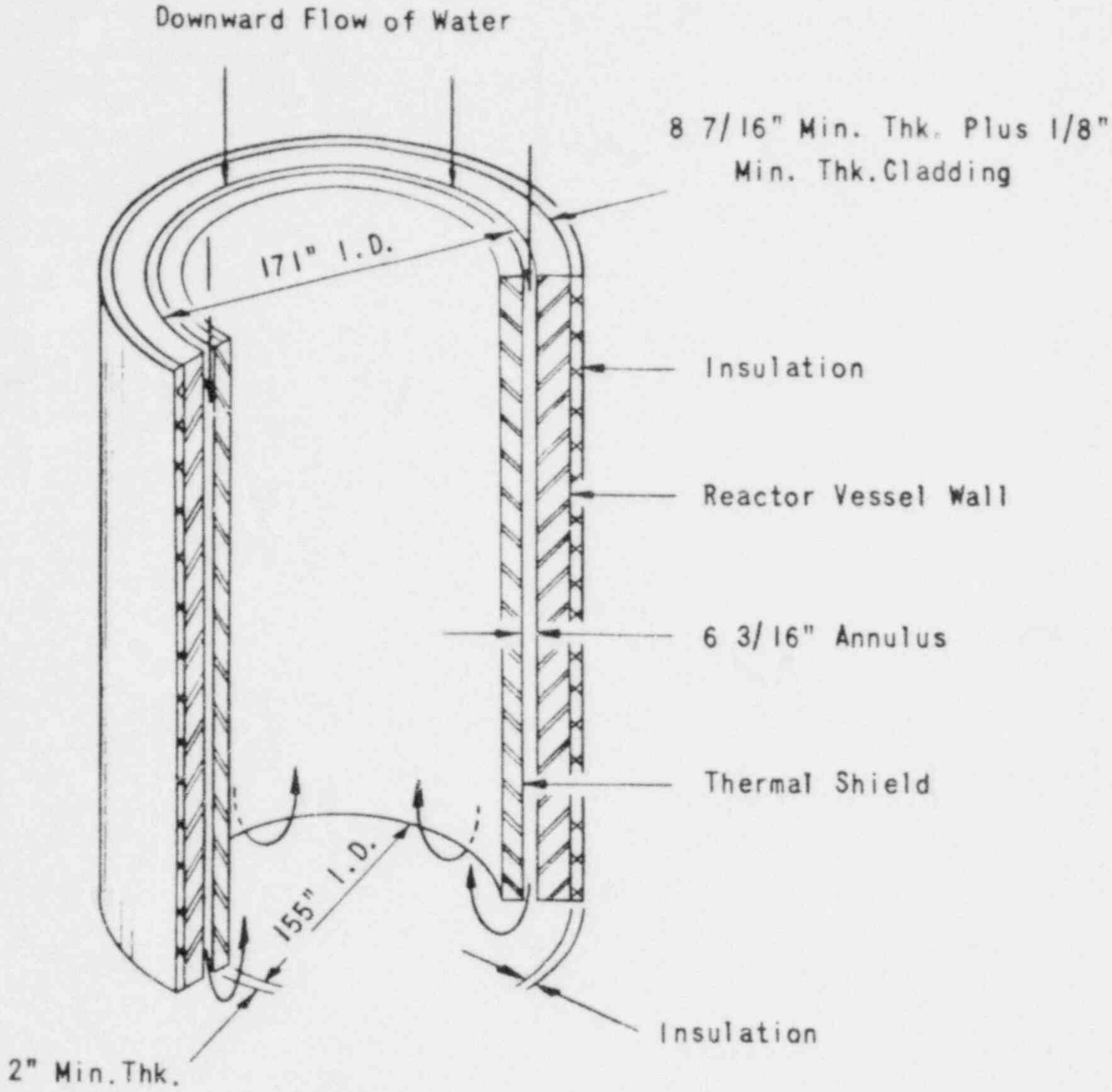

FIGURE 4A.1-1
SCHEMATIC OF THERMAL MODEL
FOR ECCS OPERATION

SMUD
SACRAMENTO MUNICIPAL UTILITY DISTRICT

Amendment 1

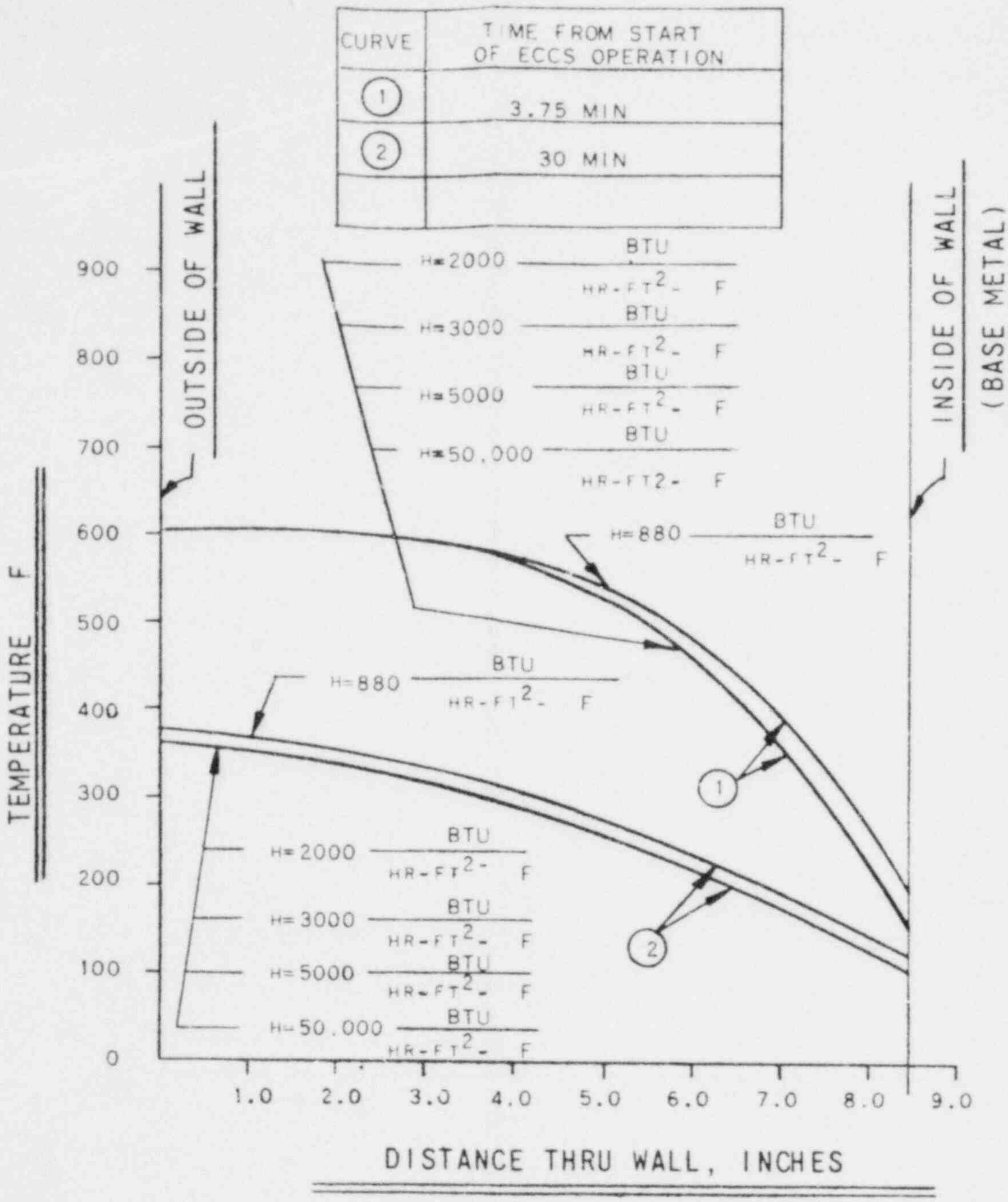

FIGURE 4A.1-2
EFFECT OF VARYING WATER FILM
HEAT TRANSFER COEFFICIENT

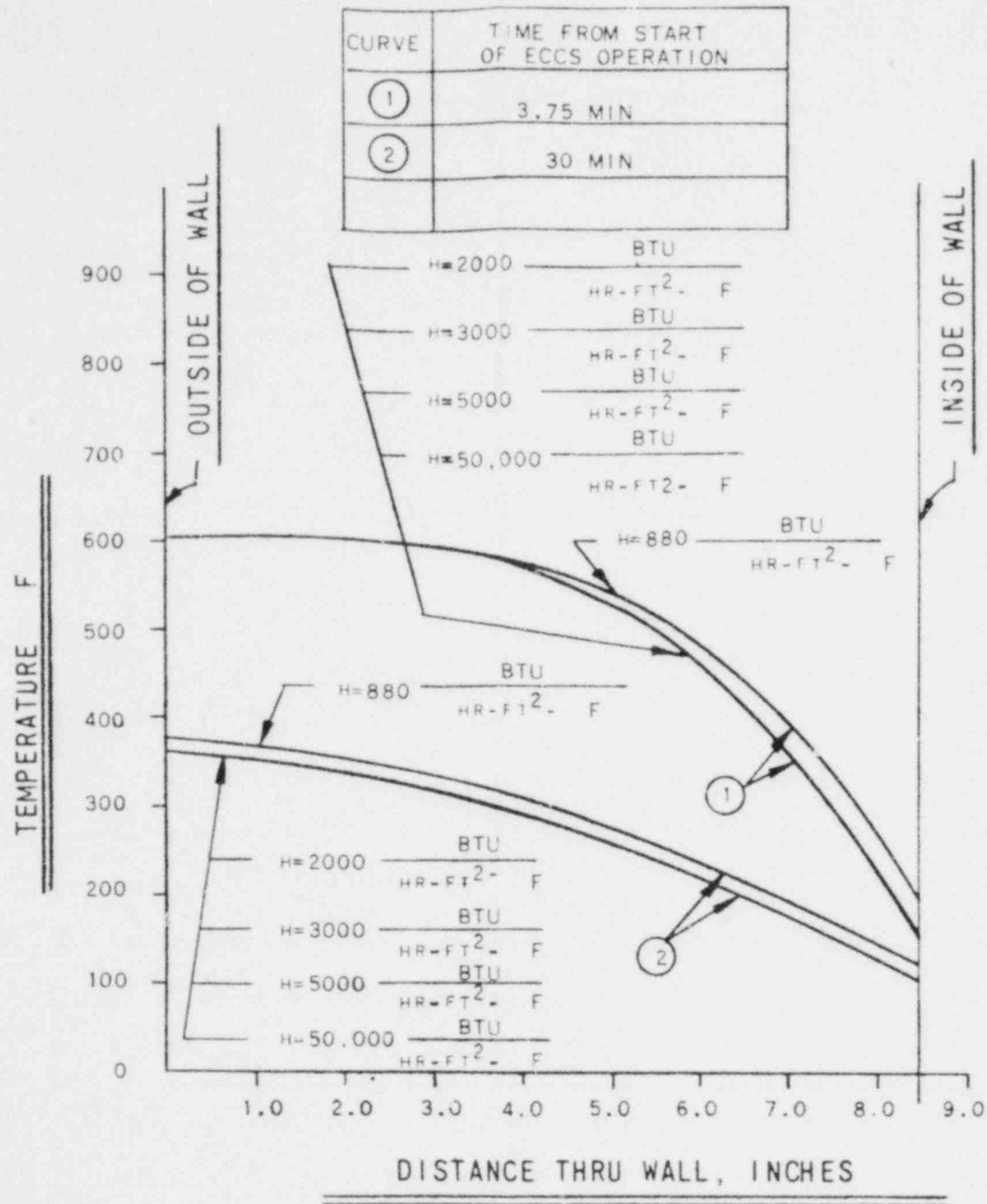

FIGURE 4A.1-2
EFFECT OF VARYING WATER FILM
HEAT TRANSFER COEFFICIENT

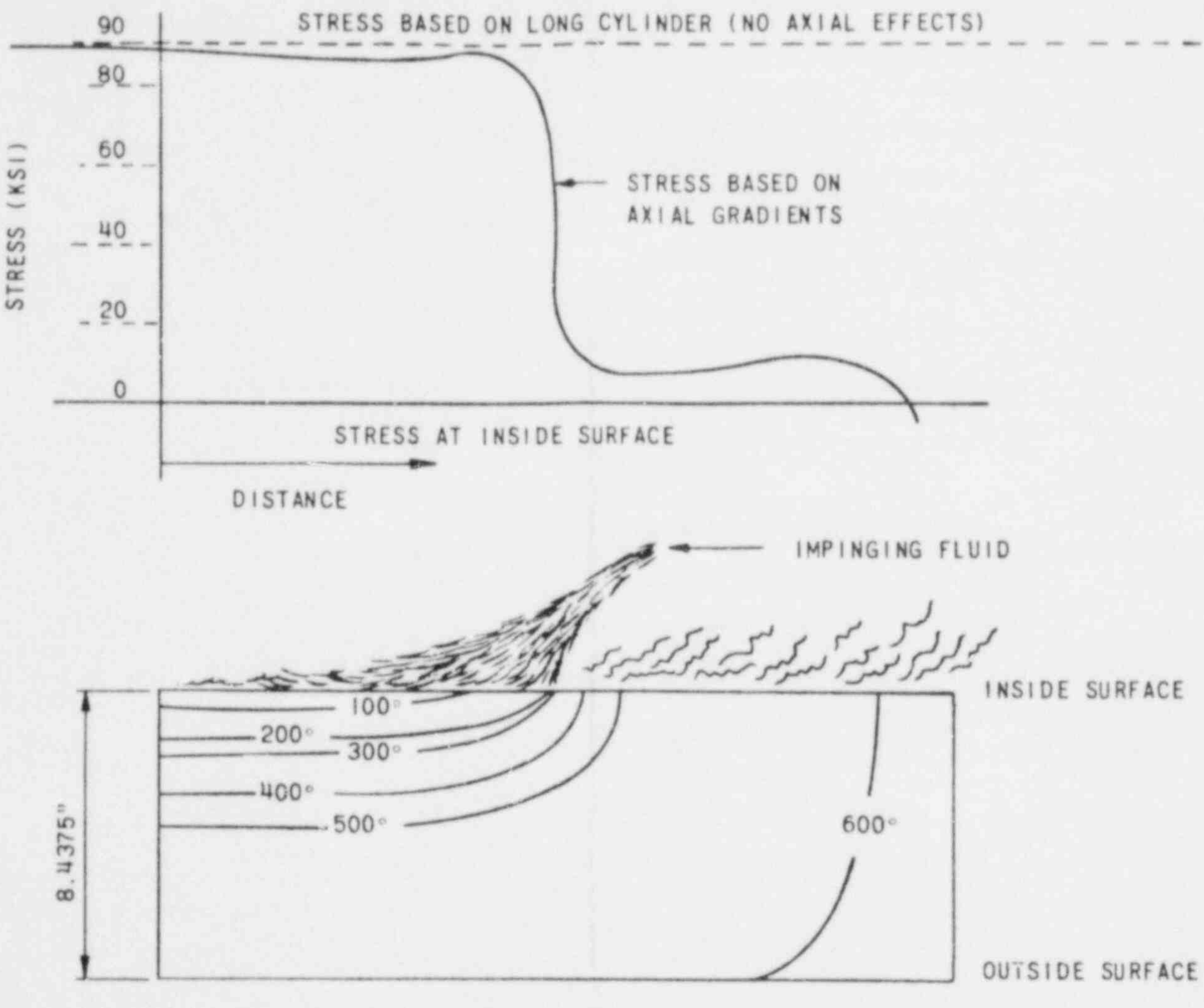

FIGURE 4A.1-3
TEMPERATURE AND MAXIMUM STRESS
FOLLOWING ECCS OPERATION (409 SEC)

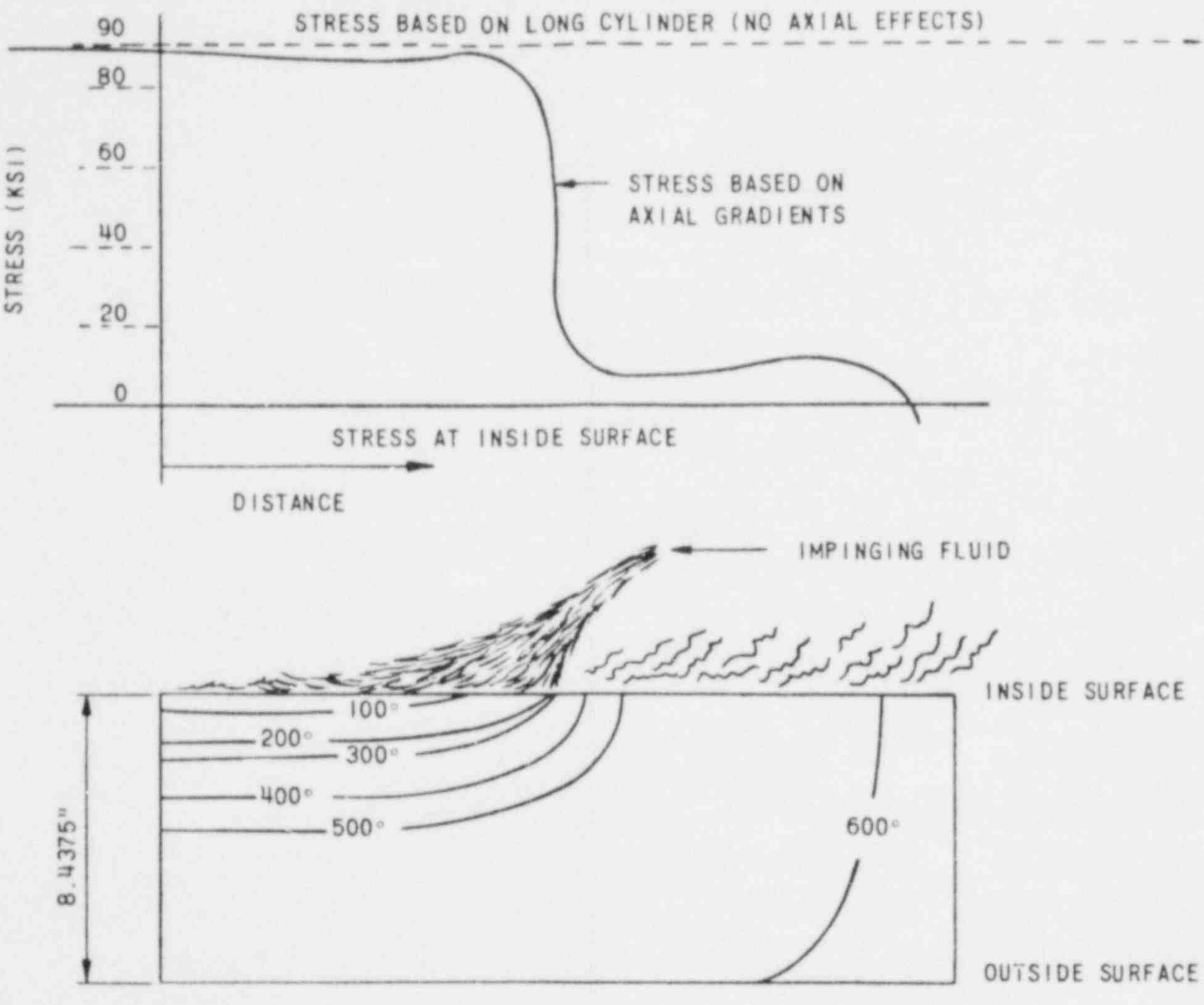

FIGURE 4A.1-3
TEMPERATURE AND MAXIMUM STRESS
FOLLOWING ECCS OPERATION (409 SEC)

0142

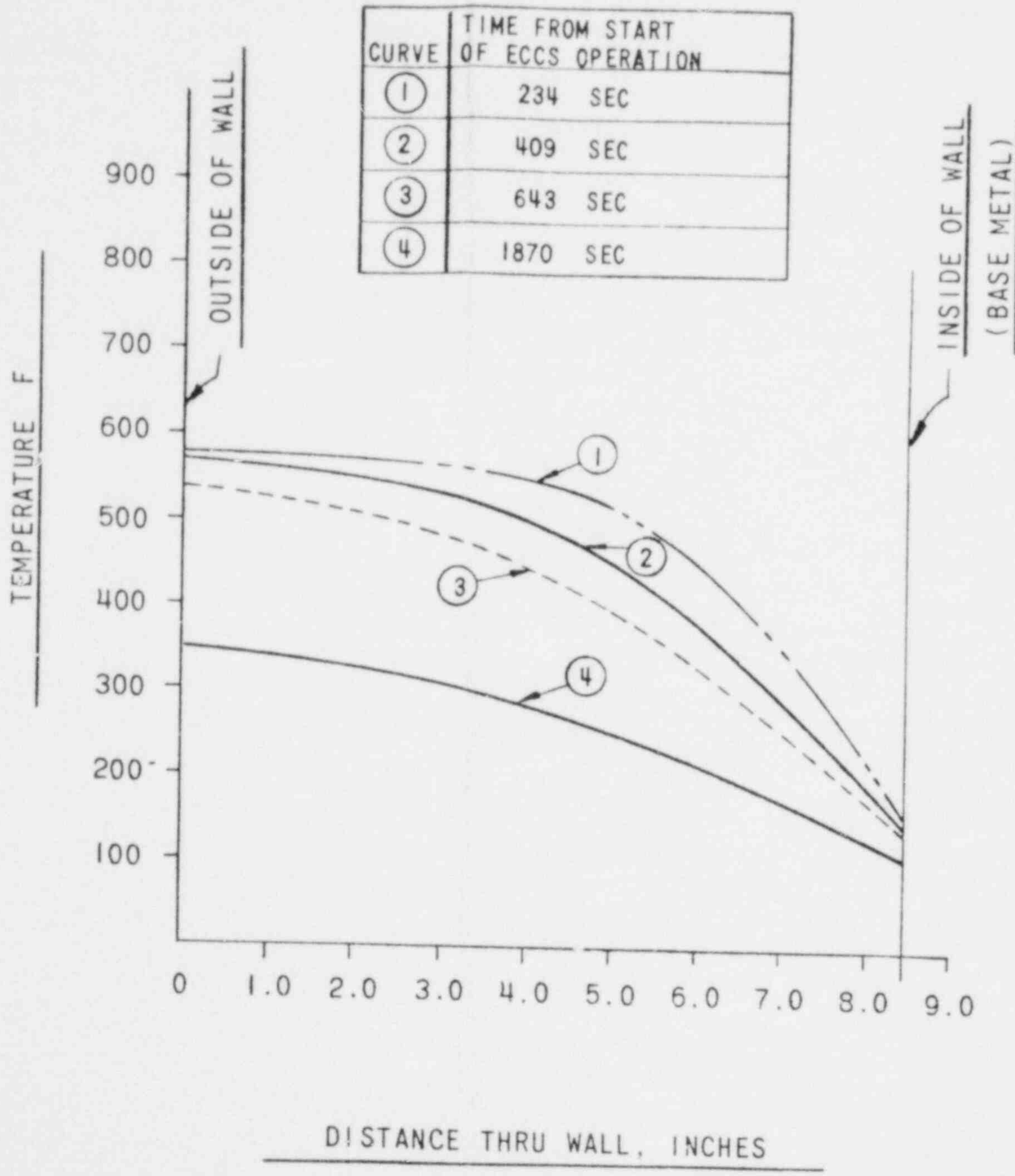

FIGURE 4A.1-4
TEMPERATURE PROFILES DUE
TO ECCS OPERATION

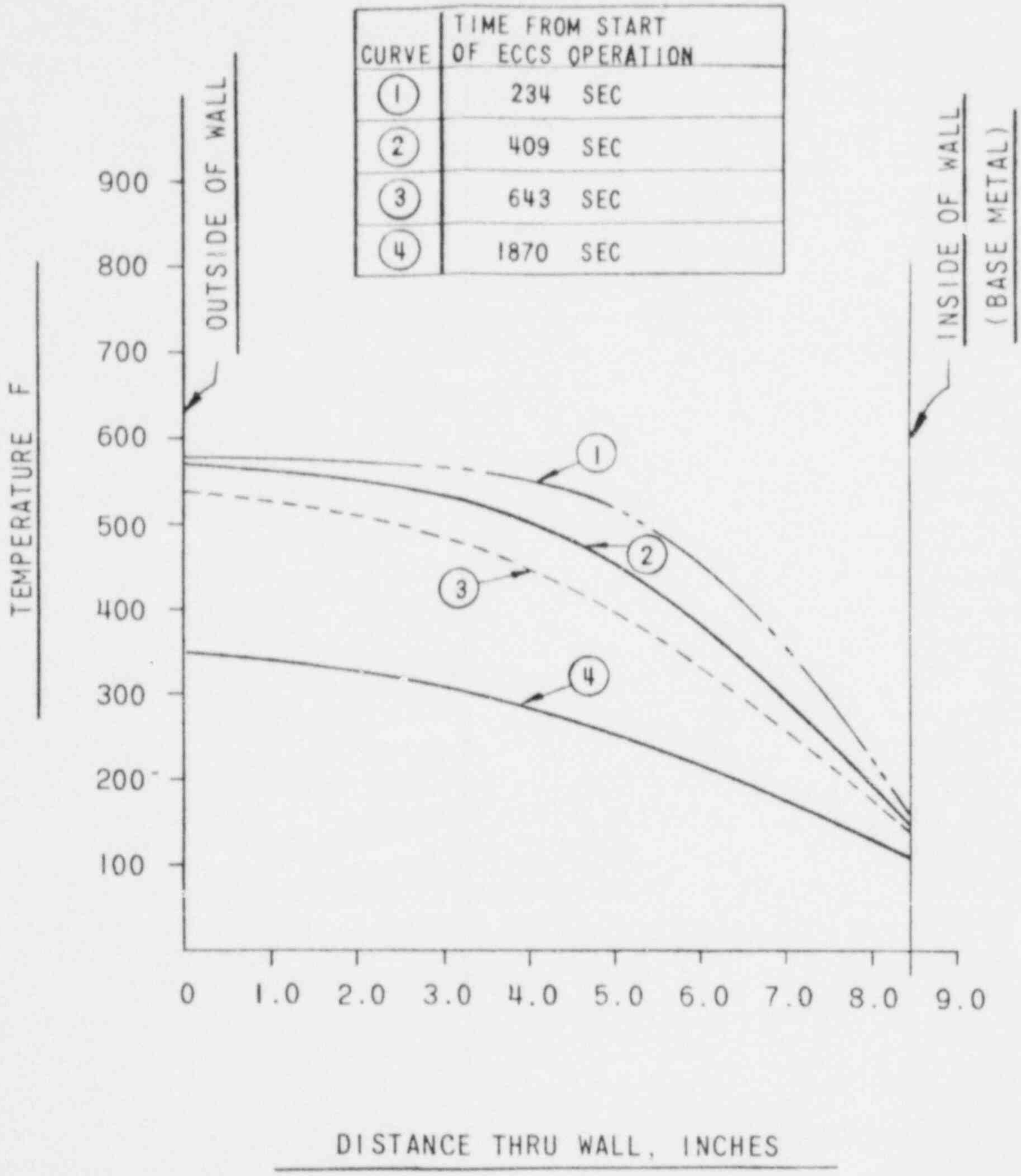

FIGURE 4A.1-4
TEMPERATURE PROFILES DUE
TO ECCS OPERATION

SMUD
SACRAMENTO MUNICIPAL UTILITY DISTRICT

Amendment 1

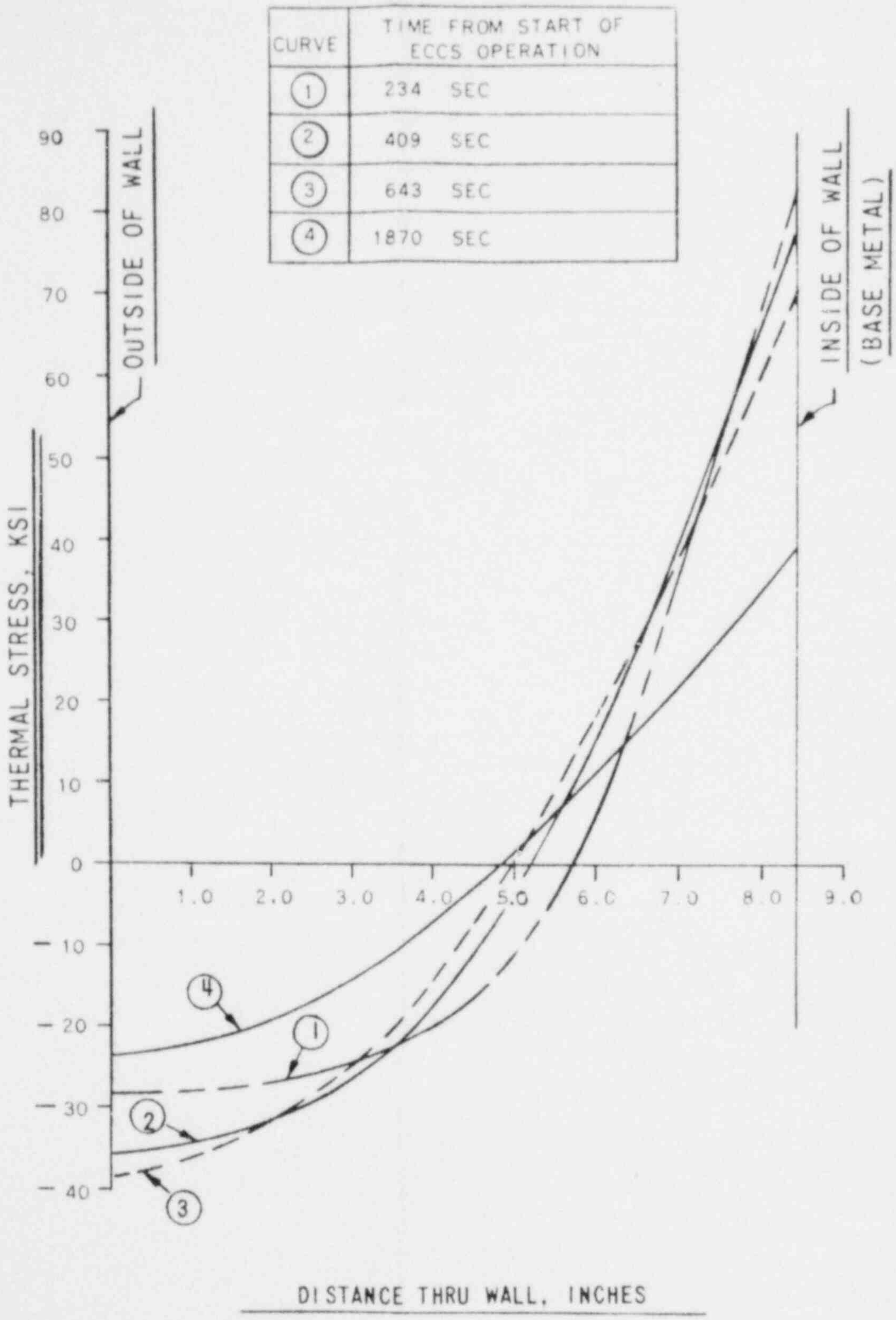

FIGURE 4A.1-5
THERMAL STRESS DUE TO ECCS OPERATION

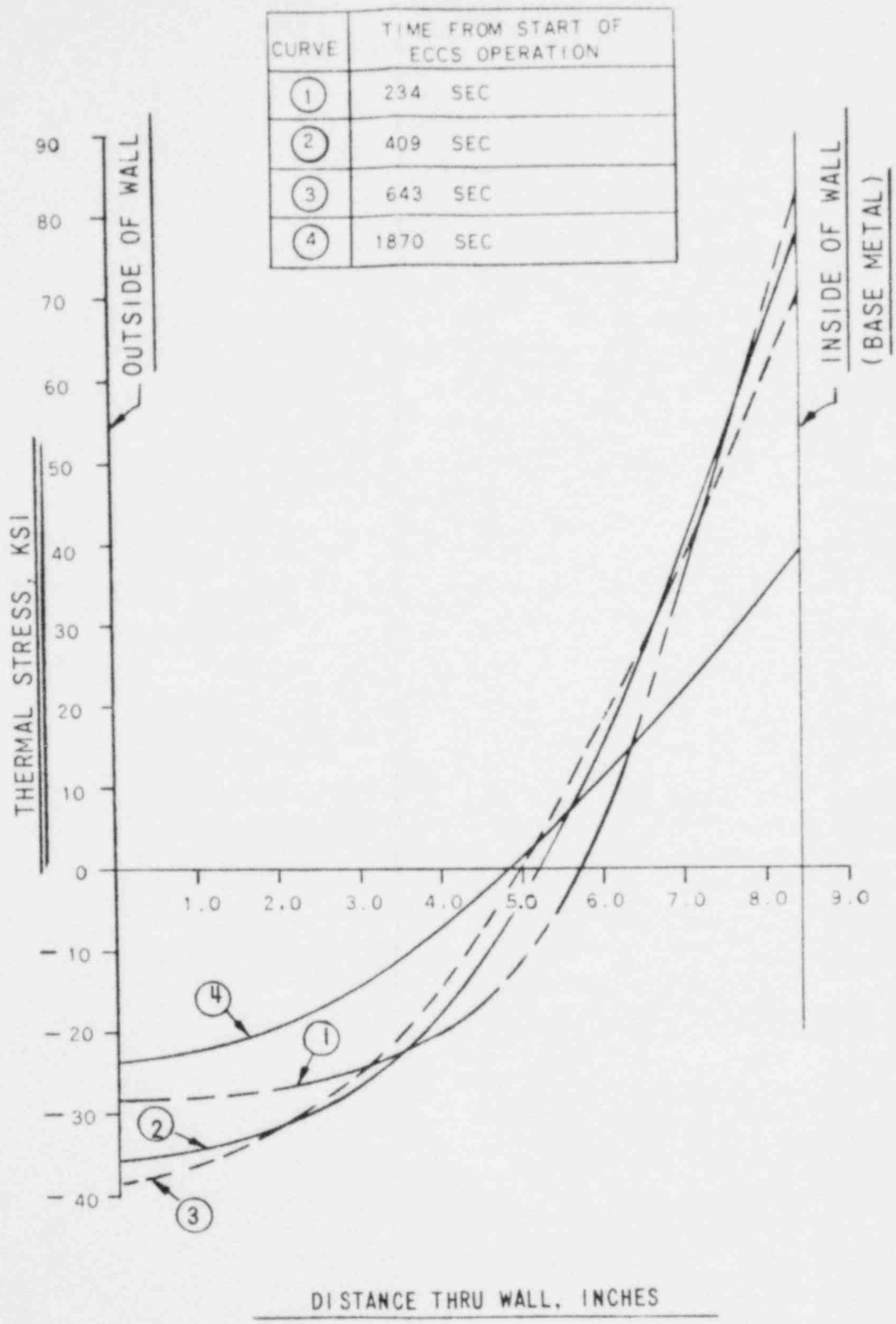

FIGURE 4A.1-5
0146 THERMAL STRESS DUE TO ECCS OPERATION

SMUD
SACRAMENTO MUNICIPAL UTILITY DISTRICT

~~00062~~

Amendment 1

QUESTION 4A.2 With respect to the brittle fracture mode of failure provide the following additional information:

4A.2.1 The assumed distribution of the initial NDT temperature through the plate thickness. State also the experimental basis for this assumption and the degree of conservatism involved.

ANSWER The distribution of NDTT through the plate thickness was assumed to be a constant value of +10 F. The +10 F was assumed because the B&W Material Specification requires that material in the core region will have, as a maximum, an initial NDTT of +10 F at a depth below the surface equal to 1/4 T. The use of a constant value through the thickness of the plate is conservative when consideration is given to the recent work from Lehigh University, [1] B&W, [2] and others. [3] From the references cited it is found that, for all practical purposes, the NDTT at 1/4 T is the same as the NDTT at 1/2 T for plates in the thickness range of 8 to 12 in. Our analysis is conservative in that it did not consider the benefit which could be gained by considering the enhanced properties which exist at the surface. From References 1 and 2 the NDTT at the surface would be expected to be -50 F.

The AEC, with the cooperation of Industry, is at present engaged in a program of material characterization which will further substantiate the data presented here.

4A.2.2 The assumed time-integrated neutron flux (nvt) at the reactor vessel inner diameter.

ANSWER The assumed time-integrated neutron flux (nvt) at the reactor vessel inner wall is 3×10^{19} n/cm^2 (E > 1 Mev). This value is stated in the PSAR Section 4.1.4.1, page 4.1-8.

4A.2.3 The profile of the NDT temperature shift through the thickness of the plate.

ANSWER The NDTT profile at the end of Station life was assumed to be a constant value of 250 F through the thickness of the reactor vessel wall. This value was stated in the PSAR Section 4.1.4.1, page 4.1-8. The use of a constant value for NDTT shift is very conservative because the analysis did not consider the beneficial effect which can be realized by considering the self-shielding [4] of the material to radiation damage.

0147

REFERENCES

[1] Strunk, S. S., Pense, A. W., and Stout, R. D., The Properties and Microstructure of Spray-Quenched Thick-Section Steels, Welding Research Council Bulletin No. 120, February 1967. (Appendix A)

[2] B&W Data on SA-302GB Material. (Appendix B)

[3] Naval Reactors Program Data (Classified).

[4] NRL Memo No. 1731, p. 15 - 21.

~~00064~~

0148

QUESTION 4A.3 — An estimate of the effect of an initial vessel temperature higher than that assumed in the analysis on the extent of yielding and deformation of the vessel.

ANSWER — The analysis used an initial vessel wall temperature of 603 F. A sensitivity analysis considering various initial wall temperatures, up to 1,500 F, has also been completed. The results indicate that 31 percent of the material in the inner portion of the vessel wall thickness has yielded at 1,500 F. The analysis with an initial wall temperature of 603 F indicated that 14.7 percent of material in the inner portion of the vessel wall thickness had yielded. Thus, the sensitivity analysis indicated an increase in the ductile yielding of 16.3 percent when the initial wall temperature was assumed to be 1,500 F.

QUESTION 4A.4 — An estimate of the maximum allowable pressure stress, when combined with other stresses present in the vessel, which could be tolerated without failure.

ANSWER — The maximum pressure that B&W considered was 600 psi. This is based on the fact that the core flooding tanks will not operate until the reactor vessel pressure is at or below 600 psi. This internal pressure would only increase the depth of ductile yielding from 14.7 to 17.5 percent of the wall thickness.

QUESTION 4A.5 — An estimate of the maximum neutron flux exposure (nvt) of the vessel that could be tolerated without vessel failure.

ANSWER — The analysis considering the brittle fracture mode assumed the conservative approach in that the material would behave in a completely brittle manner, and thus the lower threshold stress was used for comparison with the imposed stresses. Therefore, the analysis as performed by B&W is insensitive to increased flux levels.

0149 ~~00065~~

QUESTION 4A.6 — The effect of potential local penetrations present in the vessel cladding, exposing the base metal to the coolant, on the results of the analysis.

ANSWER — Our analysis did not consider the beneficial effect of cladding. In regions where local penetrations in the clad surface are postulated to be potential occurrences, the actual temperature profile across the thickness will be virtually unchanged (because of the small difference in conductivity and the small thickness of clad), and the stresses at these points will be as they were originally calculated.

QUESTION 4A.7 — The number of thermal shock cycles, induced by ECCS operation, that the vessel could withstand at the end of its fatigue life.

ANSWER — B&W does not consider the ECCS operation as a cyclic occurrence. However, plastic deformation (ductile yielding) might safely be repeated without the integrity of the vessel being violated. If ECCS operation should occur when the vessel is in the brittle region, then further operation of the unit would be prohibited until an exhaustive examination of the vessel has been completed.

QUESTION 4A.8 — Experimental data on the thermal shock effects in thick plates under stress, tested below the NDT temperature.

ANSWER — The demonstration of the adequacy of the reactor vessel to accommodate the thermal gradients, developed upon injection of emergency coolant following a loss-of-coolant accident, is a unique application of fracture mechanics and analysis involving stressed plates, thermal gradients, crack triggering by quenching, transition temperature gradients, and notch geometries.

Data relative to the individual parts of this problem are available. This data exists in the form of the Robertson Gradient Tests, routine practice in quenching heavy section shell forgings, and the transition temperature correlation work carried out by Pellini and Puzak at NRL. Also there is extensive work which is being conducted in the fracture mechanics field by such research establishments as ORNL, Westinghouse Research, and Universities. All of this data was valuable in developing the conservative methods which were used in the analysis as presented.

~~00066~~

0150

QUESTION 4A.9 An evaluation of the capability of the safety injection nozzles and accumulator piping to withstand the transient.

ANSWER B&W is considering the effect of ECCS operation in the analysis of the safety injection nozzle and accumulator piping. As soon as this analysis is completed, the results of the analysis will be presented.

QUESTION 4A.10 An evaluation of the effects of this transient on the core barrel and other internals with regard to assuring that distortion would not restrict the flow path of the emergency core coolant.

ANSWER A detailed analysis of the effects of emergency core coolant flow on the reactor internals has not been performed. However, preliminary analysis and previous similar experience indicate the following:

> The reactor internals are constructed of Type 304 stainless steel, and therefore are not subject to brittle fracture at temperatures of interest (some loss of impact strength has been observed at about -320 F). Further, the material is sufficiently ductile that many quenches of the expected magnitude can be withstood without initiation of a crack, or propagation of an assumed existing crack. Consequently, thermal shock fracture of the internals is not considered credible.

The reactor internals are being designed to conservative stress and deflection limits, so that failure or large deformations of the internals due to blowdown loadings will not occur.

A further degree of conservatism is provided by coolant inlet flow deflector vanes in the region of the emergency coolant inlets. These vanes are attached to the core support shield, and will prevent that shield from approaching within about 5 in. of the vessel ID in the region of the emergency coolant inlets.

~~00367~~

0151

QUESTION 4A.11 Provide a detailed outline of the research program required to verify the analysis methods on thermal shock effects in thick plates under stress below the NDT temperature. Identify any other area related to the pressure vessel and piping thermal shock problem that requires a research and development program for proof-of-principle, and outline the required program.

ANSWER For safety analysis purposes, B&W does not believe a research program can significantly affect the conclusions obtained by the methods used in the analysis of the thermal shock effects on the reactor vessel caused by the actuation of the ECCS due to an LOCA. However, as part of industry's continuing effort to improve the detailed knowledge of material behavior under all conceivable conditions, B&W has included this subject on the agenda for the PVRC meeting held January 16, 1968.

B&W does not consider that any area related to the pressure vessel and piping thermal shock problem requires a research and development program for proof-of-principle.

~~00068~~

0152

QUESTION 4A.12 (DRL 4.1) <u>Thermal Shock</u>

With regard to thermal shock on reactor components, induced by operation of the emergency core cooling system (ECCS), provide details of an analysis which indicates that the reactor vessel and reactor internals can withstand the rapid temperature change at the end of their design life. The analysis should include both the ductile yielding and the brittle fracture modes of failure.

4A.12.1 (DRL 4.1.1) The brittle fracture analysis for the vessel should assume an initial crack size just below the critical crack size corresponding to the stresses present during normal operation and transients. Since the initial crack is most likely to exist in a weld or a heat affected zone, the analysis should consider two cases: a circumferential crack, and a crack parallel to the axis of the reactor vessel. The details of the analysis should be provided including specific information on:

(a) The critical stress intensity factor (K_{IC}) assumed, and the basis for its selection,
(b) The assumed time-integrated neutron flux (nvt) at the reactor vessel inner diameter,
(c) The value of residual stresses assumed in the base metal and the weld areas,
(d) The initial crack geometry and size assumed in the analysis,
(e) Equations used to correlate crack size with the calculated stress intensity factor (K_I).

ANSWER

(a) This question was answered in the reply to Question 8.11.1 of the Florida Power Corporation PSAR (Docket No: 50-302 and -303.)

(b) This question has already been answered in Question 4A.2.2 in Amendment 1 to the Rancho Seco PSAR.

(c) This question has already been answered in Question 4A.1.8 in Amendment 1 to the Rancho Seco PSAR.

(d) This question was answered in the reply to Question 8.11.2 of the Florida Power Corporation PSAR (Docket Nos. 50-302 and -303.)

(e) This question was answered in the reply to Question 8.11.3 of the Florida Power Corporation PSAR (Docket Nos. 50-302 and -303.)

4A.12.2 (DRL 4.1.2) The details of the ductile yielding mode of analysis for the vessel should include the following information:

(a) The geometry of the plate and the cooling method assumed in the analysis,
(b) The heat transfer coefficient used, its experimental basis, and the degree of conservatism involved,
(c) The initial temperature of the vessel as a function of time delay in injecting the cold water,
(d) The effect of axial temperature gradient in the vessel, during filling with cold water, on the total stress intensity and the distortion of the vessel,
(e) The temperature profiles and the calculated thermal stress profiles through the thickness of the plate for several times during the cold water injection transient,
(f) The magnitude of the axial dead load stresses in the vessel,
(g) The magnitude of the stresses in the vessel shell due to potential simultaneous seismic loading,
(h) The value of the yield stress used as the failure criterion in the ductile yielding analysis.

ANSWER

(a) This question has already been answered in Question 4A.1.1 in Amendment 1 to the Rancho Seco PSAR.

(b) This question has already been answered in Question 4A.1.2 in Amendment 1 to the Rancho Seco PSAR.

(c) This question has already been answered in Question 4A.1.3 in Amendment 1 to the Rancho Seco PSAR.

(d) This question has already been answered in Question 4A.1.4 in Amendment 1 to the Rancho Seco PSAR.

(e) This question has already been answered in Question 4A.1.6 in Amendment 1 to the Rancho Seco PSAR.

(f) This question has already been answered in Question 4A.1.9 in Amendment 1 to the Rancho Seco PSAR.

~~00370~~

0154

(g) This question has already been answered in Question 4A.1.10 in Amendment 1 to the Rancho Seco PSAR.

(h) This question has already been answered in Question 4A.1.11 in Amendment 1 to the Rancho Seco PSAR.

4A.12.3 (DRL 4.1.3)

Based on the analyses for the vessel provide:

(a) An estimate of the maximum acceptable initial temperature of the vessel that could be tolerated without failure of the vessel,
(b) An estimate of the maximum neutron flux exposure (nvt) of the vessel that could be tolerated without vessel failure,
(c) An estimate of the maximum allowable pressure stress, when combined with other stresses present in the vessel, which could be tolerated without failure.

ANSWER

(a) This question has already been answered in Question 4A.3 in Amendment 1 to the Rancho Seco PSAR.

(b) This question has already been answered in Question 4A.5 in Amendment 1 to the Rancho Seco PSAR.

(c) This question has already been answered in Question 4A.4 in Amendment 1 to the Rancho Seco PSAR.

4A.12.4 (DRL 4.1.4)

Evaluate the capability of the piping, safety injection nozzles, and vessel nozzles to withstand the transient.

ANSWER

This question has already been answered in Question 4A.9 in Amendment 1 to the Rancho Seco PSAR.

4A.12.5 (DRL 4.1.5)

Evaluate the effects of this transient on the core barrel and other internals with regard to assuring that distortion would not restrict the flow path of the emergency core coolant.

ANSWER

This question was answered in Question 4A.10 in Amendment 1 of the Rancho Seco PSAR.

0155

~~00071~~

4A.12.6 Current status of the fracture mechanics analysis of the thermal shock problem.

ANSWER This problem was evaluated using two different analytical techniques and presented in the answers to Question 4A.1 in Appendix 4A. One of these techniques was based on ductile yielding data relative to the propagation of flaws in reactor vessel steels. The other was based on a fracture mechanics analysis of the problem. Both of these methods predicted consistent results which indicate that the reactor vessel would not crack through its thickness as a result of this thermal shock.

During the ACRS review of other reactor applications in January 1968, a third method of analysis was proposed. This proposed method can be found in ASTM STP-381. While it is
3 felt that the evaluation presented in the PSAR adequately demonstrates that a crack will not propagate through the vessel wall as a result of the thermal shock, this third evaluation was undertaken using the method suggested. The preliminary results of this third method of analysis confirm the results of the evaluation presented in the PSAR by demonstrating that the crack will not propagate through the wall of the vessel.

The assumptions in the original fracture mechanics analysis (Question 4.A.1) and in the third method of analysis differed primarily in that, in the original fracture mechanics analysis, the critical stress intensity factor was considered to be a variable and residual stresses were considered to remain constant. In the third method the critical stress intensity factor was considered to be a constant, and the residual stresses were considered to vary.

~~00372~~

0156

QUESTION 4A.13 (DRL 4.3)

Discuss the full power radiation environment with respect to corresponding damage thresholds for the control rod actuators and the primary loop pumps and pump motors. Consider the N-16 activity, the fission product activity in coolant, and the radiation streaming contributions.

ANSWER

The dose to primary loop pumps and motors from all gamma and neutron sources is computed to be about 2×10^7 rads at the end of 32 effective full power years. Of this total dose approximately 60% is from N-16, 20% from fission and corrosion products, and 20% from streaming through primary shield penetrations. 3

With the exception of the lubricating oil all materials in the pumps and motors are rated as being capable of withstanding at least 10^8 rads exposure before exhibiting any signs of radiation damage. At 10^8 rads the lubricating oil shows about a 10% increase in viscosity. However, this oil will be replaced every 5 to 10 years resulting in a maximum exposure to the oil of about 5×10^6 rads.

The radiation damage thresholds for all materials in the control rod actuators have not been identified at present. The correlation of damage thresholds with radiation levels will be determined, and the design will take cognizance.

0157 ~~00073~~

QUESTION 4A.14 (DRL 4.4)

Provide a tabulation of all the nuclear pressure vessels in the Class I (seismic design) systems in the facility. The tabulation should include a notation of whether the vessel design is complete, the stage of fabrication of the vessel, and the extent to which each of the vessels will comply with each of the 34 supplementary criteria in "Tentative Regulatory Supplementary Criteria for ASME Code-Constructed Nuclear Pressure Vessels", issued by AEC Press Release No. IN-817, dated August 25, 1967.

For each vessel, provide a discussion that represents the reason why total compliance is not feasible for each criterion not met in its entirety.

ANSWER

The following will provide information relative to the status of design and fabrication of the nuclear steam supply
3 system components fabricated by The Babcock & Wilcox Company and their compliance with the AEC Supplementary Criteria.

a. Nuclear Pressure Vessels - Class I (Service Design)

Class I equipment in the Rancho Seco Nuclear Generating Station is defined in Appendix 5A of the PSAR. Vessels in the reactor coolant system are designed and classified in accordance with appropriate and existing codes as listed in Section 4.1 and Table 4.1-9 of the PSAR. Vessels in the auxiliary systems are designed and classified in accordance with appropriate and existing codes as listed in Section 9 of the PSAR.

The nuclear pressure vessels in the Class I (seismic design) systems in the facility are tabulated below:

	Vessel	Design Complete	Status of Fabrication
	Reactor Vessel	Yes	Material ordered
	Steam generators	Yes	Material ordered
	Pressurizer	Yes	Material ordered
	Core Flooding Tanks	Yes	Material ordered
3	Control Rod Drive Pressure Housing	No	Purchase Orders Not Yet Placed
	Decay Heat Coolers	No	"
	Letdown Coolers	No	"
	Makeup Tank	No	"
	Purification Demineralizers	No	"
	Purification Filters	No	"
	Seal Return Coolers	No	"

~~00074~~

0158

b. Supplementary Criteria

The AEC's "Tentative Regulatory Supplementary Criteria for ASME Code-Constructed Nuclear Pressure Vessels", which were issued by AEC Press Release No. IN-817 (August 25, 1967), have been reviewed by Atomic Industrial Forum ad hoc group and the ASME, as well as several industrial concerns. The outcome of comments by these organizations is awaited. Further reference is made to the B&W Company comments forwarded directly to the AEC on this subject. 3

Specific comments applicable to the B&W furnished Rancho Seco vessel are contained in B&W letter to Dr. Harold Price dated April 2, 1968.

Excluding the equipment which is being fabricated by The Babcock & Wilcox Company (reactor vessel, pressurizer, steam generators and core flooding tanks), no purchase orders have been placed for equipment.

For other equipment, a detailed answer to this question prior to selecting vendors and placing purchase orders is considered premature.

Following is a tabulation of the Criteria in compliance and notations on those Criteria not in compliance.

Criterion		B&W Compliance	Comments
1.10	Classification of Vessels	No	Letdown Cooler - Class C
1.11	Conditions for Design	Yes	
1.12	Certification of Stress Reports	Yes	
1.13	Conditions with Unspecified Design Rules	Yes	
1.14	Vessel Owner's Responsibility for Inspection	No	Not practical
1.15	Manufacturer's Responsibility for Quality Control	Yes	
1.16	Vessel Fabrication Report	No	Do not literally comply with weld repair records.

3

0159

~~00075~~

Criterion		B&W Compliance	Comments
1.17	Boundary between Vessel and Piping	Yes	
1.20	Vessel Material Property Improvement	No	Use design consideration and cleanliness and property requirements in lieu of Criterion
1.21	Material Test Coupons	Yes	
1.22	Nondestructive Examination of Reactor Vessel Plates	No	Do not agree technically.
1.23	Nondestructive Examination and Repairs of Material	No	Impractical technically - follow Code requirements.
1.24	Examination of Reactor Vessel Bolts	No	Criterion impractical - use better method.
1.25	Ductile Brittle Transition Properties	No	No excess material, not practical.
1.26	Exclusion of Repairs in Bolting Material	Yes	
1.30	Fracture Mechanics	No	Not on all materials.
1.31	Design for Cyclic Loading	Yes	
1.32	Bolting Design Requirements	No	Not applicable on small connections.
1.33	Earthquake Load	No	Use loads when specified in 1.34.
1.34	Design Conditions - Combination Loading	No	No fatigue analysis for earthquake - impractical

3

0160 ~~00076~~

Criterion		B&W Compliance	Comments
1.35	Computer Programs	Yes	
1.36	Environmental Effects	No	Not practical.
1.37	Design for inspecta-bility	No	As welded clad-ding surface.
1.38	Attachments to Reactor Vessel	Yes	
1.39	Reactor Vessel Core Support	No	Do not agree technically.
1.40	Chemical Analysis of Weld Wire	No	Code require-ment considered adequate.
1.41	Cutting Plates	No	Not practical.
1.42	Welding Qualification Procedure Requirements	No	Code require-ment considered adequate.
1.43	Precautions for Welding	No	Do not agree technically.
1.44	Welding Requirements	No	Do not agree technically.
1.50	Final Inspection and Examination	No	As welded clad-ding.
1.51	Nondestructive Examina-tion and Responsibilities	No	No approval of procedures.
1.60	Hydrostatic Testing Requirements	No	Not practical.
1.70	New Materials	Yes	

3

0161 ~~00077~~

QUESTION 4A.15 (DRL 4.5) Submit Certified Code Design Specifications for component parts of the Class I systems as required by the ASME Code Section III, paragraph N-141 (passed 6-23-67).

3 ANSWER Refer to B&W proprietary topical report CS(F)-3-22-T submitted separately by SMUD.

(Deleted)

~~00078~~

0162

APPENDIX B

B&W DATA

TYPICAL NDTT DATA FOR

SA-302 B PLATE MODIFIED TO

CODE CASE 1339 PARAGRAPH 1

1. Material from Shell plate 6-1/4" thick

Aust. 1675-1725F, 6-1/4 hours, B.Q.
Aust. 1600-1650F, 6-1/4 hours, B.Q.
Temper 1175-1225F, 6-1/4 hours, B.Q.
Lab S.R. 1100-1150F, 18 hours, F.C.

All specimens longitudinal to final rolling direction.

Chemistry	
C	.18
Mn	1.08
P	.005
S	.012
Si	.24
Cr	.16
Ni	.44
Mo	.45
Cu	.17

A. As Tempered Properties - Surface

Charpy Tests

Test Temp.	Ft-Lbs	Lat. Exp., Mils	Est. % Shear
0F	61, 65, 70	48, 50, 51	30, 30, 30
-30F	52,100,100	39, 62, 68	25,100,100
-30F	69, 85, 94	50, 60, 61	25, 80, 90
-60F	53, 67	38, 47	13, 30
-90F	19, 33, 48	15, 27, 33	3, 5, 10

B. As Tempered - Just Below 1/4T Properties

Charpy Tests

Test Temp.	Ft-Lbs	Lat. Exp., Mils	Est. % Shear
+40F	48, 63,69	40,51,54	30, 40, 50
+10F	55, 56, 60	43,43,47	25, 25, 30
-20F	30, 33, 34	25,25,26	5, - 10
-40F	10, 20, 30	7,17,21	0, - 5

0163

2. Material from Shell plate 9-3/4" Thick Mn-Mo-Ni Plate (A533B)

Air Cool from 1675-1725F
Quenched from 1675-1725F
Tempered from 1200-1225F, Air Cool
Stress Relieved 60 hours 1100-1150F, Furnace Cooled

A. As Tempered Properties - Surface

Charpy Tests

Test Temp.	Ft-Lbs
-80F	7, 9,
-50F	10, 22, 37
-20F	28, 45, 45
+10F	35, 60, 62
+40F	70, 83
+300F	134,134

B. As Tempered - Just Below 1/4T Properties

Charpy Tests

Test Temp.	Ft-Lbs.
-40F	16
-20F	11, 14, 29
0F	18, 20, 28
+10F	25, 32, 37, 38, 42, 43
+40F	47, 50, 55
+300F	130,131

C. As Tempered - Just Below 1/2T Properties

Charpy Tests

Test Temp.	Ft-Lbs.
-40F	11
-20F	13, 16, 18
+10F	21, 33, 35, 35, 40, 42
+40F	38, 46, 48
+300 F	127,120

0164 ~~00380~~