THE CLEVELAND ELECTRIC ILLUMINATING COMPANY
P.O. BOX 5000 - CLEVELAND, OHIO 44101 - TELEPHONE (216) 622-9800 - ILLUMINATING BLDG. — 55 PUBLIC SQUARE
Serving The Best Location in the Nation

Dalwyn R. Davidson
VICE PRESIDENT
SYSTEM ENGINEERING AND CONSTRUCTION

September 22, 1982

Mr. A. Schwencer, Chief
Licensing Branch No. 2
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Perry Nuclear Power Plant
Docket Nos. 50-440; 50-441
Confirmatory Issue - No. 16
Limiting Weld Material

Dear Mr. Schwencer:

This letter and its attachments are provided in response to the Perry SER confirmatory issue numbered 16. This response addresses limiting beltline weld materials as well as revises the responses to the NRC questions 123.1, 123.2, and 123.3.

We believe that this letter and its attachments should resolve the confirmatory issue of material surveillance capsules for limiting reactor beltline materials for the next Perry Supplementary Safety Evaluation Report (SSER No. 2).

Very truly yours,

Dalwyn R. Davidson

Dalwyn R. Davidson
Vice President
System Engineering and Construction

DRD:mb

Attachment

cc: Jay Silberg, Esq.
John Stefano
Max Gildner
Barry Elliott

B001

123.1 To demonstrate compliance with Paragraph III.B.3, certify that the calibration schedule for temperature instruments, drop weight, and Charpy V-notch machines comply with the requirements of Paragraph NB-2360 of the ASME Code. If they are not in compliance, indicate the schedule used and provide a basis for granting an exemption to the exact requirements of NB-2360 of the ASME Code.

RESPONSE

Calibration of temperature instruments, and Charpy V-notch impact test machines used in impact testing was required in accordance with the requirements of paragraph NB-2360 of the ASME Code.

123.2 To demonstrate compliance with the requirements of Paragraph IV.A.3 of Appendix G, 10 CFR Part 50, certify that all ferritic materials used for reactor coolant pressure boundary piping and valves which are in balance of plant and the nuclear steam supply system meet the requirements of NB-2300 of the ASME Code.

RESPONSE

All ferritic materials in the reactor coolant pressure boundary piping and valves in balance of plant and nuclear steam supply system are Safety Class 1 and are procurred in accordance with the material requirements of ASME Section III, Subsection NB-2300.

For the balance of plant, fracture test data, as applicable, is available and auditable if required.

The specific components, applicable code requirements, and impact test temperatures of the nuclear steam supply system are listed below.

1. Main Steam Pipe - ASME Section III, 1974 Edition, +60°F.

2. Flued Head Fittings - ASME Section III, 1974 Edition, +60°F.

3. HPCS Gate Valve - ASME Section III, 1971 Edition Through 1973 Addenda, +40°F.

4. Main Steam Isolation Valve - ASME Section III, 1974 Edition, +60°F maximum.

5. Safety Relief Valve (8" x 10") - ASME Section III, 1974 Edition Through Summer 1976 Addenda, +60°F maximum.

123.3 To demonstrate the surveillance capsule program complies with Paragraphs II.B and III.C of Appendix H.

(a) Provide a sketch showing the azimuthal location of each material surveillance capsule.

(b) Identify each plate specimen in each capsule by heat number and chemical composition, especially copper and phosphorus.

(c) Identify each weld specimen in each capsule by weld wire type and heat, flux type, lot identification, and chemical composition, especially copper and phosphorus.

(d) Identify the lead factor for each surveillance capsule.

RESPONSE

(b) See New FSAR Table 5.3-3.

(c) See New FSAR Table 5.3-3.

(d) The calculated lead factors for Perry 1 and 2 are the quotient of neutron fluence greater than 1 MeV at the surveillance capsule, divided by the neutron fluence greater than 1 MeV at the peak point at one quarter the depth diameter, or the peak point at one quarter the depth into the vessel. By symmetry, all of the surveillance samples have the same lead factor. The following lead factors are provided:

$$\text{Lead factor} \quad \frac{\text{capsule}}{\text{vessel ID}} = 0.40$$

$$\text{Lead factor} \quad \frac{\text{capsule}}{\frac{1}{4}\text{t vessel}} = 0.58$$

Surveillance capsule contents and locations are:

		CHARPY SPECIMENS			FLUX WIRES	
Capsule No.	Azimuth	Transverse Base	HAZ	Weld	Fe	Cu
1	3°	12	12	12	2	2
2	177°	12	12	12	2	2
3	185°	12	12	12	2	2

Three Fe flux wires are also contained in a separate neutron dosimeter at the 3° location.

The most limiting plate and weld materials is identified in revised FSAR Section 5.3.1.6.3. Review of the documentation has confirmed the surveillance specimens were taken from the most limiting plate material. A similar documentation review for the weld materials has confirmed the limiting beltline weld materials is included in the RPV surveillance capsule.

Two of the three test plate heats that were used in construction beltline weld seams are considered the limiting beltline weld materials in terms of estimated E-O-L RT_{NDT}. There are heat 627260 E-O-L RT_{NDT} of +31 and heat 62667 E-O-L RT_{NDT} of +30. The third construction weld material was heats not used in construction welds are equivalent (as defined by equivalency criteria in Annex A1 of ASTM E 185-73) to the two production weld heats not employed in test plate fabrication. However, CBI has informed GE's Quality Control that stick electrode used to seal the back-up bars in the core region weld seams as well as the test plate weld, were removed by back gouging of the seams. Thus, the construction beltline welds and the related surveillance program test plate weld contain primarily submerged arc welding material from 5SP6214B. Certified test reports for the plate and weld materials are available for review.

f. Regulatory Guide 1.71, Welder Qualification for Areas of Limited Accessibility

Qualification for areas of limited accessibility is discussed in Section 5.2.3.4.2.

g. Regulatory Guide 1.99, Effects of Residual Elements on Predicted Radiation Damage to Reactor Pressure Vessel Materials

Predictions for changes in transition temperature and upper shelf energy were made in accordance with the requirements of Regulatory Guide 1.99.

5.3.1.5 Fracture Toughness

5.3.1.5.1 Compliance with 10 CFR 50 Appendix G

Appendix G of 10 CFR 50 is interpreted for Class I RCPB components of the BWR 6 reactor design and complied with as discussed in Section 5.3.2 and below with the following exceptions:

1. The specific temperature limits for operation when the core is critical are based on a proposed modification to 10 CFR 50, Appendix G, Paragraph IV, A.2.C. The proposed modification and the justification for it, together with the results of an NRC review, are given in GE Licensing Topical Report NEDO-21778-A.

2. A minimum boltup and pressurization temperature of 70°F is called for, which is at least 60°F above the flange region RT_{NDT}. This exceeds the minimum RT_{NDT} temperature required by ASME Code Section III, Paragraph 2222(c), Summer 1976 and later editions. A flange region flaw size less than 10% of the wall thickness can be detected at the outside surface of the flange to shell and head junctions where stresses due to boltup are most limiting.

a. Method of Compliance

The following items 1 through 7 are the interpretations and methods used to comply with Appendix G of 10 CFR 50. Item 8 reports the fracture toughness test results and the background information used as the basis to show compliance with 10 CFR 50, Appendix G.

1. Records and Procedures for Impact Testing (Refer to 10 CFR 50, Appendix G-III B.4 and G-III B.5)

   Personnel conducting fracture toughness testing were qualified by experience and training that demonstrated competency to perform tests in accordance with required procedures. No record of qualification of individuals performing these tests were required at that time as the order of the Perry components predates the requirements of Appendix G.

2. Specimen Orientation for Original Qualification Versus Surveillance (Refer to 10 CFR 50 Appendix G-III A)

   The second sentence of G-III A is understood to be independent of the first sentence; that is, Charpy-V-Notch tests as defined in NB-2321.2 are to be conducted on both unirradiated and irradiated ferritic materials; however, the special beltline longitudinally oriented Charpy specimens required by the general reference NB-2300 and, specifically, NB-2322.2(a)(6) are not included in the surveillance program base metal because the transverse specimens are limiting with regard to toughness.

3. Charpy-V Curves for the RPV Beltline (Refer to 10 CFR 50 Appendix G-III C & H-III B)

   It is understood that the orientation of impact test specimens for the G-III C1 requirements shall comply with the requirements of NB-2322(a)4 (transverse specimen) for plate material as opposed to NB-2322(a)(6) (longitudinal specimen). This understanding of the

general reference to NB-2322 in G-III C1 results in meaningful and conservative beltline curves of unirradiated materials for comparison with the results of surveillance program testing of irradiated transverse base metal specimens and also allow this curve to comply with ASTM E-185-73.

It is understood that the number, type, and locations of specimens necessary for the full curves of G-III C(1) are those required to comply with Paragraphs 4.3 and 4.4 of ASTM E-185-73. This interpretation is considered necessary to assure that the adjusted reference temperature of irradiated base metal, heat-affected zone and weld metal called for in H-III B can be based on directly comparable data for the unirradiated reference temperature.

In regard to C-III C(2), the procedures of ASTM E-185-73 were used for selection of surveillance specimen base material to provide a conservative adjusted reference temperature for the beltline base material. The test plate weld materials are equivalent to beltline construction weld materials. The weld test plate for the surveillance program specimens had the principal working direction normal to the weld seam to assure that heat affected zone specimens are oriented such that they parallel actual production weld conditions.

4. Upper Shelf Energy for Beltline (Refer to 10 CFR 50 Appendix G-IV B)

All beltline material meets the Charpy-V-Notch test minimum upper shelf energy of 75 ft-lbs for Perry reactor pressure vessels.

5. Bolting Materials (Refer to 10 CFR 50 Appendix G-IV A.4)

See section 5.3.1.7.

6. Alternative Procedures for the Calculation of Stress Intensity Factor (Refer to 10 CFR 50 Appendix G-IV A.2(a) and G-IV A.2(b))

Stress Intensity Factors were calculated by the methods of Appendix G to Section III of the ASME Code. Discontinuity regions

were evaluated, as well as shell and head areas, as part of the detailed thermal and stress analysis in the vessel stress report. Equivalent margins of safety to those required for shells and heads were demonstrated using a 1/4 T defect at all locations, with the exception of the main closure flange to head and shell discontinuity locations. Here it was found that additional restriction on operating limits would be required for outside surface flaw size greater than 0.24 inches at the outside surface of the flange to shell joint (based on additional analyses made for BWR 6 reactor vessels). It has been demonstrated using a test mockup of these areas that smaller defects can be detected by the ultrasonic inservice examinations procedures required at the adjacent weld joint. Since the stress intensity factor is greatest at the outside surface of the flange to shell and head joints a flaw can also be detected by outside surface examination techniques.

7. Fracture Toughness Margins in the Control of Reactivity (Refer to 10 CFR 50 Appendix G-IV.A.2.c)

   Appendix G of the ASME Code, Section III (1971 Edition with Addenda to and including Winter 1972 or later), "Protection Against Non-ductile Failure," was used in determining pressure/temperature limitations for all phases of plant operation. Additionally, when the core is critical a 40°F temperature allowance is included in the reactor vessel operating pressure vs temperature limits to account for operational occurances in the control of reactivity as described in GE BWR Licensing Topical Report NEDO-21778-A and the U.S. Nuclear Regulatory Commission's acceptance basis which is included therein.

8. Results of fracture toughness tests are reported in Tables 5.3-1 and Table 5.3-2.

5.3.1.6 Material Surveillance

5.3.1.6.1 Compliance with "Reactor Vessel Material Surveillance Program Requirements"

The materials surveillance program monitors changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region resulting from their exposure to neutron irradiation and thermal environment.

Reactor vessel materials surveillance specimens are provided in accordance with requirements of ASTM E-185-73 and 10 CFR 50, Appendix H. Materials for the program are selected to represent materials used in the reactor beltline region. Specimens are manufactured from a plate actually used in the beltline region and a weld typical of those in the beltline region and thus represent base metal, weld material, and the weld heat affected zone material. The plate and weld are heat treated in a manner which simulates the actual heat treatment performed on the core region shell plates of the completed vessel.

Each in-reactor surveillance capsule contains 36 Charpy-V-Notch specimens. The capsule loading consists of 12 specimens each of base metal, weld metal, and heat affected zone material. A set of out-of-reactor baseline Charpy-V-Notch specimens and archive material are provided with the surveillance test specimens.

Three capsules are provided in accordance with Case A requirements of 10 CFR 50 Appendix H since the predicted end of life adjusted temperature of the reactor vessel steel is less than 100°F. The proposed withdrawal schedule is in accordance with 10 CFR 50, Appendix H.

First capsule - one-fourth service life
Second capsule - three-fourths service life
Third capsule - Standby

Fracture toughness testing of irradiated capsule specimens will be in accordance with requirements of 10 CFR 50, Appendix H.

5.3.1.6.2 Neutron Flux and Fluence Calculations

A description of the methods of analysis is contained in Sections 4.1.4.5 and 4.3.2.8. The peak fluence at 1/4 t depth of the vessel beltline material is 4.5×10^{18} n/cm^2 after 40 years of service. All predictions of radiation damage to the reactor vessel beltline material were made using peak fluence values.

5.3.1.6.3 Predicted Irradiation Effects on Vessel Beltline Materials

Estimated maximum changes in RT_{NDT} and upper shelf fracture energy as a function of the end of life (EOL) fluence at the 1/4 t depth of the vessel beltline materials are listed in Table 5.3-3. The predicted peak EOL fluence at the 1/4 t depth of the vessel beltline is 4.5×10^{18} n/cm^2 after 40 years of service. Transition temperature changes and changes in upper shelf energy were calculated in accordance with the guidelines of Regulatory 1.99. Reference temperatures were established in accordance with 10 CFR 50, Appendix G and NB-2330 of the ASME Code.

5.3.1.6.4 **Positioning of Surveillance Capsules and Methods of Attachment (Refer to 10 CFR 50, Appendix H.11 C(2))**

Surveillance specimen capsules are located at three azimuths at a common elevation in the core beltline region. The sealed capsules are not attached to the vessel but are in welded capsule holders. The capsule holders are mechanically retained by capsule holder brackets welded to the vessel cladding as shown in Figure 5.3-1. The capsule holder brackets allow the removal and reinsertion of capsule holders. These brackets are designed, fabricated and analyzed to the requirements of Section III ASME Code. A positive spring loaded locking device is provided to retain the capsules in position throughout any anticipated event during the lifetime of the vessel.

In areas where brackets, such as the surveillance specimen holder brackets, are located, additional non-destructive examinations are performed on the vessel base metal and stainless steel weld deposited cladding or weld buildup pads during vessel manufacture. The base metal is ultrasonically examined by straight beam techniques to a depth at least equal to the thickness of the bracket being joined. The area examined is the area of the subsequent attachment weld plus a band around this area of width equal to at least half the thickness of the part joined. The required stainless steel weld deposited

cladding is similarly examined. The full penetration welds are liquid penetrant examined to ASME Section III Standards. Cladding thickness is required to be at least 1/8 inch.

The above requirements have been successfully applied to a variety of bracket designs which are attached to weld deposited stainless steel cladding or weld buildups in many operating BWR reactor pressure vessels.

Inservice inspection examinations of core beltline pressure retaining welds are performed from the outside surface of the reactor pressure vessel. If a bracket for mechanically retaining surveillance specimen capsule holders were located at or adjacent to a vessel shell weld, it would not interfere with the straight beam or half node angle beam inservice inspection ultrasonic examinations performed from the outside surface of the vessel.

5.3.1.6.5 Time and Number of Dosimetry Measurements

GE provides a separate neutron dosimeter so that fluence measurements may be made at the vessel ID during the first fuel cycle to verify the predicted fluence at an early date in plant operation. This measurement is made over this short period to avoid saturation of the dosimeters now available. Once the fluence-to-thermal power output is verified, no further dosimetry is considered necessary because of the linear relationship between fluence and power output. It will be possible however to install a new dosimeter, if required, during succeeding fuel cycles.

5.3.1.7 <u>Reactor Vessel Fasteners</u>

The reactor vessel closure head (flange) is fastened to the reactor vessel shell flange by multiple sets of threaded studs and nuts. The lower end of each stud is installed in a threaded hole in the vessel shell flange. A nut and washer are installed on the upper end of each stud. The proper amount of preload can be applied to the studs by a sequential tensioning using hydraulic tensioners. Hardness tests are performed on all main closure bolting to demonstrate that heat treatment has been properly performed.

Regulatory Guide 1.65 defines acceptable materials and testing procedures with regard to reactor vessel closure stud bolting for light-water-cooled reactors. The PNPP Units 1 & 2 vessel order date preceded implementation of Regulatory Guide 1.65. The design and analysis of reactor vessel bolting materials is in full compliance with ASME Code Section III, Class I requirements. The reactor pressure vessel closure studs are SA-540, Grade B 23 or 24 (AISI4340). The maximum reported ultimate tensile strength is 174,000 psi. Also, the Charpy impact test requirements of 10CFR50 Appendix G, paragraph IV-A.4 were satisfied, as the lowest reported CV energy was 44 ft-lbs at +10°F, compared to the requirement of 45 ft-lbs at the lowest service temperature, and the lowest reported CV expansion was 25 mils at +10°F compared to the 25 mils required.

There are no metal platings applied to closure studs, nuts, or washers. A phosphate coating is applied to threaded areas of studs and nuts and bearing areas of nuts and washers to act as a rust inhibitor and to assist in retaining lubricant (either graphite/alcohol or nickel powder base lubricant) on these surfaces.

In relationship to regulatory position C.2.b., the bolting materials were ultrasonically examined in accordance with ASME Section III, NB-2585 after final heat treatment and prior to threading. The specified requirement for examination according to SA-388 was complied with. The procedures approved for use in practice were judged to insure comparable material quality and, moreover, were considered adequate on the basis of compliance with the applicable requirements of ASME Code Paragraph NB-2583. Straight beam examination was performed on 100 percent of cylindrical surfaces, and from both ends of each stud using a 3/4 maximum diameter transducer. In addition to the Code required notch, the reference standard for the radial scan contains a 1/2 inch diameter flat bottom hole with a depth of 10 percent of thickness, and the end scan standard contains a 1/4 diameter flat bottom hole 1/2 inch deep. Also, angle beam examination was performed on the outer cylindrical surface in both axial and circumferential directions. Any indication greater than the indication from the applicable calibration feature is unacceptable. A distance-amplitude correction curve per NB-2585 is used for the longitudinal wave examination. Surface examinations were performed on

the studs and nuts after final heat treatment and threading, as specified in the Guide, in accordance with NB-2583 of the applicable ASME Code.

In relationship to regulatory position C.3, GE practice allows exposure to stud bolting surfaces to high purity fill water; nuts and washers are dry stored during refueling.

5.3.2 PRESSURE-TEMPERATURE LIMITS

5.3.2.1 Limit Curves

The limit curves presented in Figures 5.3-2 and 5.3-3 are based on the requirements of 10 CFR 50, Appendix G with the modification to Paragraph IV A.2C per GE BWR Licensing Topical Report NEDO-21778-A. All the vessel shell and head areas remote from discontinuities plus the feedwater nozzles were evaluated, and the operating limit curves are based on the limiting location. The boltup limits for the flange and adjacent shell region are based on a minimum metal temperature of RT_{NDT} and 60°F. The maximum through-wall temperature gradient from continuous heating or cooling at $100^{o}F$ per hour was considered. The safety factors applied were as specified in ASME Code Appendix G and GE Licensing Topical Report NEDO-21778-A.

5.3.2.1.1 Temperature Limits for Boltup

A minimum temperature of $70^{o}F$ is required for the closure studs. A sufficient number of studs may be tensioned at $70^{o}F$ to seal the closure flange O-rings for the purpose of raising the reactor water level above the closure flagnes in order to assist in warming them. The flanges and adjacent shell are required to be warmed to minimum temperature of $70^{o}F$ before they are stressed by the full intended bolt preload. The fully preloaded boltup limits are shown on Figures 5.3-2 and 5.3-3.

5.3.2.1.2 Temperature Limits for Preoperational System Hydrostatic Tests and ISI Hydrostatic or Leak Pressure Tests

Based on 10 CFR 50 Appendix G IV.A.2.d, which allows a reduced safety factor for tests prior to fuel loading, the preoperational system hydrostatic test at 1563 psig may be performed at a minimum temperature of 126°F for Perry Units 1 and 2.

The fracture toughness analysis for system pressure tests resulted in the curves labeled A shown in Figures 5.3-2 (Perry 1) and 5.3-3 (Perry 2). The curves labeled "core beltline" are based on an initial RT_{NDT} of 0°F for the plate material for Perry 1, and an initial RT_{NDT} of -20°F for the plate material for Perry 2.

The predicted shift in the RT_{NDT} from Figure 5.3-5 (based on neutron fluence at 1/4 T of the vessel wall) must be added to the beltline curve to account for the effect of fast neutrons.

5.3.2.1.3 Operating Limits During Heatup, Cooldown and Core Operation

The fracture toughness analysis was done for the normal heatup or cooldown rate of 100°F/hour. The temperature gradients and thermal stress effects corresponding to this rate were included. The results of the analyses are a set of operating limits for non-nuclear heatup or cooldown shown as curves labeled B on Figures 5.3-2 and 3. Curves labeled C on Figures 5.3-2; and 5.3-3 apply whenever the core is critical. The basis for curve C is described in GE BWR Licensing Topical Report NEDO-21778-A.

5.3.2.1.4 Reactor Vessel Annealing

In place annealing of the reactor vessel because of radiation embrittlement is unnecessary because the predicted value in transition of adjusted reference temperature does not exceed 200°F (10 CFR 50, Appendix G, Paragraph IV.C).

5.3.2.1.5 Predicted Shift in RT_{NDT}

(Refer to 10 CFR 50, Appendix G II.G and V.B)

The adjusted reference temperature for BWR vessels is predicted using Figure 5.3-5. This figure is the same as Regulatory Guide 1.99, Rev. 1, Figure 1 with the exception that the curves have been extrapolated to 20°F. The extrapolation is based on data contained in GE Licensing Topical Report NEDO 21708, "Radiation Effects in Boiling Water Reactor Pressure Vessel Steels," October 1977.

5.3.2.2 Operating Procedures

By comparison of the pressure vs. temperature limit in Section 5.3.2.1 with intended normal operating procedures for the most severe upset transient, it is shown that these limits will not be exceeded during any foreseeable upset condition. Reactor operating procedures have been established such that actual transients will not be more severe than those for which the vessel design adequacy has been demonstrated. Of the design transients, the upset condition producing the most adverse temperature and pressure condition anywhere in the vessel head and/or shell areas yields a minimum fluid temperature of 250°F and a maximum pressure peak of 1,180 psig. Scram automatically occurs as a result of this event, prior to the reduction in fluid temperature, so the applicable operating limits are given in curves B and B' in Figures 5.3-2 and 5.303. For a temperature of 250°F, the maximum allowable pressure exceeds 1,600 psig for the intended margin against noductile failure. The maximum transient pressure of 1,180 psig is therefore within the specified allowable limits.

5.3.3 REACTOR VESSEL INTEGRITY

Each reactor vessel was fabricated for General Electric's Nuclear Energy Division by CBI Nucl ar Co., and was subject to the requirements of General Electric's quality assurance program.

The CBI Nuclear Co. has had extensive experience with GE reactor vessels and has been the primary supplier of GE domestic reactor vessels and some foreign vessels since the company was formed in 1972 from a merger. Prior experience by the Chicago Bridge and Iron Company with an agreement between Chicago Bridge and Iron Co. and General Electric GE reactor vessels dates back to 1966.

Assurance was made that measures were established requiring that purchased material, equipment, and services associated with the reactor vessels and appurtenances conform to the requirements of the subject purchase documents. These measures included provisions, as appropriate, for source evaluation and selection, objective evidence of quality furnished, inspection at the vendor source and examination of the completed reactor vessels.

General Electric provided inspection surveillance of the reactor vessel fabricator's in-process manufacturing, fabrication, and testing operations in accordance with GE's Quality Assurance Program and approved inspection procedures. The reactor vessel fabricator was responsible for the first level inspection of his manufacturing, fabrication, and testing activities and General Electric is responsible for the first level of audit and surveillance inspection.

Adequate documentary evidence that the reactor vessel material, manufacture, testing, and inspection conforms to the specified quality assurance requirements contained in the procurement specification is available at the fabricator plant site.

Regulatory Guide 1.2 states that a suitable program be followed to assure the reactor pressure vessel will behave in a non-brittle manner under loss of coolant accident (LOCA) conditions. Should it be considered that the margin of safety against reactor pressure vessel brittle fracture due to emergency core cooling system operation at any time during vessel life is unacceptable, the Regulatory Guide states that an engineering solution, such as annealing, could be applied to assure adequate recovery of the fracture toughness properties of the vessel material.

An analysis of the structural integrity of boiling water reactor pressure vessels during a design basis accident (DBA) has been performed.

The analysis included:

(1) Description of the LOCA event.

(2) Thermal analysis of the vessel wall to determine the temperature distribution at different times during the LOCA.

(3) Determination of the stresses in the vessel wall including thermal, pressure, and residual stresses.

(4) Consideration of radiation effect on material toughness (NDTT Shift and changes in toughness).

(5) Fracture mechanics evaluation of vessel wall for different postulated flaw sizes.

This analysis incorporated conservative assumptions in all areas (particularly in the areas of heat transfer, stress analysis, effects of radiation on material toughness, and crack tip stress intensity factor evaluation.) The analysis concluded that even in the presence of large flaws, the vessel will have considerable margin against brittle fracture following a loss of coolant accident.

5.3.3.1 Design

5.3.3.1.1 Description

5.3.3.1.1.1 Reactor Vessel

Each reactor vessel shown in Figure 5.3-6 is a vertical, cylindrical pressure vessel of welded construction. The vessel is designed, fabricated, tested, inspected, and stamped in accordance with the ASME Code Section III, Class I requirements including the addenda in effect at the date of order placement, Unit I: Winter 1972 and Unit 2: Winter 1972. Design of the reactor vessel and its support system satisfies Seismic Category I equipment requirements. The materials used in the reactor pressure vessel are shown in Table 5.2-4.

The cylindrical shell and top and bottom heads of the reactor vessel are fabricated of low alloy steel, the interior of which is clad with stainless steel weld overlay, except for the top head and nozzle and nozzle weld zones.

In-place annealing of the reactor vessel is unnecessary because shifts in transition temperature caused by irradiation during the 40-year life can be accommodated by raising the minimum pressurization temperature and the

predicted value of adjusted reference temperature does not exceed 200°F. Radiation embrittlement is not a problem outside of the vessel beltline region because of the low fluence in those areas.

Quality control methods used during the fabrication and assembly of the reactor vessel and appurtenances assure that design specifications are met.

The vessel top head is secured to the reactor vessel by studs and nuts. These nuts are tightened with a stud tensioner. The vessel flanges are sealed with two concentric metal seal-rings designed to permit no detectable leakage through the inner or outer seal at any operating condition, including heating to operating pressure and temperature at a maximum rate of 100°F/hr in any one hour period. To detect seal failure, a vent tap is located between the two seal-rings. A monitor line is attached to the tap to provide an indication of leakage from the inner seal-ring seal.

5.3.3.1.1.2 Shroud Support

The shroud support is a circular plate welded to the vessel wall and to a cylinder supported by vertical stilt legs from the bottom head. This support is designed to carry the weight of peripheral fuel elements, neutron sources, core plate, top guide, the steam separators, the jet pump diffusers, and to laterally support the fuel assemblies. Design of the shroud support also accounts for pressure differentials across the shroud support plate, for the restraining effect of components attached to the support, and for earthquake loadings. The shroud support design is specified to meet appropriate ASME code stress limits.

5.3.3.1.1.3 Protection of Closure Studs

The boiling water reactor does not use borated water for reactivity control during normal operation.

5.3.3.1.2 Safety Design Basis

The design of the reactor vessel and appurtenances meets the following safety design bases:

a. The reactor vessel and appurtenances will withstand adverse combinations of loading and forces resulting from operation under abnormal and accident conditions.

b. To minimize the possibility of brittle fracture of the nuclear system process barrier, the following are required:

   1. Impact properties at temperatures related to vessel operation have been specified for materials used in the reactor vessel.

   2. Expected shifts in transition temperature during design life as a result of environmental conditions, such as neutron flux, are considered in the design. Operational limitations assure that NDT temperature shifts are accounted for in reactor operation.

   3. Operational margins to be observed with regard to the transition temperature are specified for each mode of operation.

5.3.3.1.3 Power Generation Design Basis

The design of the reactor vessel and appurtenances meets the following power generation design bases:

a. The reactor vessel has been designed for a useful life of 40 years.

b. External and internal supports that are integral parts of the reactor vessel are located and designed so that stresses in the vessel and supports that result from reactions at these supports are within ASME Code limits.

c. Design of the reactor vessel and appurtenances allows for a suitable program of inspection and surveillance.

5.3.3.1.4 Reactor Vessel Design Data

The reactor vessel design pressure is 1,250 psig and the design temperature is 575°F. The maximum installed test pressure is 1,563 psig.

5.3.3.1.4.1 Vessel Support

5.3.3.1.4.2 Control Rod Drive Housings

The control rod drive housings are inserted through the control rod drive penetrations in the reactor vessel bottom head and are welded to the reactor vessel. Each housing transmits loads to the bottom head of the reactor. These loads include the weights of a control rod, a control rod drive, a control rod guide tube, a four-lobed fuel support piece, and the four fuel assemblies that rest on the fuel support piece. The housings are fabricated of Type 304 austenitic stainless steel, and Inconel 600.

5.3.3.1.4.3 In-Core Neutron Flux Monitor Housings

Each in-core neutron flux monitor housing is inserted through the in-core penetrations in the bottom head and is welded to the inner surface of the bottom head.

An in-core flux monitor guide tube is welded to the top of each housing and either a source range monitor/intermediate range monitor (SRM/IRM) drive unit or a local power range monitor (LPRM) is bolted to the seal/ring flange at the bottom of the housing (Section 7.6).

5.3.3.1.4.4 Reactor Vessel Insulation

The reactor vessel insulation is of the reflective type and is constructed completely of metal. The outer surface temperature of the insulation is expected to be at 160° F and the heat transfer rate through the insulation is

approximately 65 $Btu/hr\text{-}ft^2$ under normal operating conditions. The insulation consists of several self-contained assemblies latched together, each of which can be easily removed and replaced. The insulation assemblies are designed to remain in place and resist permanent damage during a safe shutdown earthquake.

The reactor top head insulation is supported from a structure secured on the bulkhead by means of temporary fasteners. During refueling, the support structure along with the top head insulation is removed. The insulation for the reactor vessel cylindrical surface is supported by brackets welded on the shield wall liner plate.

5.3.3.1.4.5 Reactor Vessel Nozzles

All piping connected to the reactor vessel nozzles has been designed so as not to exceed the allowable loads on any nozzle.

The vessel top head nozzles are provided with flanges having small groove facing. The drain nozzle is of the full penetration weld design. The recirculation inlet nozzles (located as shown in Figure 5.3-6), feedwater inlet nozzles, core spray inlet nozzles and LPCI nozzles, all have thermal sleeves. Nozzles connecting to stainless steel piping have safe ends, or extensions made of stainless steel. These safe ends or extensions were welded to the nozzles after the pressure vessel was heat treated to avoid furnace sensitization of the stainless steel. The material used is compatible with the material of the mating pipe.

The nozzle for the standby liquid control pipe is designed to minimize thermal shock effects on the reactor vessel, in the event that use of the standby liquid control system is required.

The solution of the feedwater nozzle cracking problems involve several elements including nozzle clad removal and thermal sleeve redesign. A description of these changes and appropriate analysis is available in Reference 2.

5.3.3.1.4.6 Materials and Inspections

The reactor vessel was designed and fabricated in accordance with the appropriate ASME Boiler and Pressure Vessel Code as defined in Section 5.2.1. Table 5.2-4 defines the materials and specifications. Section 5.3.1.6 defines the compliance with reactor vessel material surveillance program requirements.

5.3.3.1.4.7 Reactor Vessel Schematic (BWR)

The reactor vessel schematic is contained in Figure 5.3-6. Trip system water levels are indicated as shown in Figure 5.3-7.

5.3.3.2 Materials of Construction

All materials used in the construction of the reactor pressure vessel conform to the requirements of ASME Code Section II materials. The vessel heads, shells, flanges, and nozzles are fabricated from low alloy steel plate and forgings purchased in accordance with ASME specifications SA533 Grade B Class 1 and SA508 Class 2. Special requirements for the low alloy steel plate and forgings are discussed in Section 5.3.1.2. Cladding employed on the interior surfaces of the vessel consists of austenitic stainless steel weld overlay.

These materials of construction were selected because they provide adequate strength, fracture toughness, fabricability, and compatibility with the BWR environment. Their suitability has been demonstrated by long term successful operating experience in reactor service.

5.3.3.3 Fabrication Methods

The reactor pressure vessel is a vertical cylindrical pressure vessel of welded construction fabricated in accordance with ASME Code, Section III Class I requirements. All fabrication of the reactor pressure vessel was performed in accordance with GE approved drawings, fabrication procedures, and test procedures. The shell and vessel head were made from formed low alloy

steel plates, and the flanges and nozzles from low alloy steel forgings. Welding performed to join these vessel components was in accordance with procedures qualified in accordance with ASME Section III and IX requirements. Weld test samples were required for each procedure for major vessel full penetration welds.

Submerged arc and manual stick electrode welding processes were employed. Electroslag welding was not permitted. Preheat and interpass temperatures employed for welding of low alloy steel satisfied or exceeded the requirements of ASME Section III, subsection NA. Post weld heat treatment of 1,100°F minimum was applied to all low alloy steel welds.

All previous BWR pressure vessels have employed similar fabrication methods. These vessels have operated for an extensive number of years and their service history is excellent.

The vessel fabricator, CBI Nuclear Co., has had extensive experience with General Electric Co. reactor vessels and has been the primary supplier for General Electric domestic reactor vessels and some foreign vessels since the company was formed in 1972 from a merger agreement between Chicago Bridge and Iron Co. and General Electric Co. Prior experience by the Chicago Bridge and Iron Co. with General Electric Co. reactor vessels dates back to 1966.

5.3.3.4 Inspection Requirements

All plate, forgings, and bolting were 100 percent ultrasonically tested and surface examined by magnetic particle methods or liquid penetrant methods in accordance with ASME Section III requirements. Welds on the reactor pressure vessel were examined in accordance with methods prescribed and satisfy the acceptance requirements specified by ASME Section III. In addition, the pressure retaining welds were ultrasonically examined using acceptance standards which are required by ASME Section XI.

5.3.3.5 Shipment and Installation

The completed reactor vessel was given a thorough cleaning and examination prior to shipment. The vessel was tightly sealed for shipment to prevent entry of dirt or moisture. Preparations for shipment are in accordance with detailed written procedures. On arrival at the reactor site the reactor vessel was carefully examined for evidence of any contamination as a result of damage to shipping covers. Suitable measures were taken during installation to assure that vessel integrity was maintained; for example, access controls were applied to personnel entering the vessel, weather protection is provided and periodic cleanings are performed.

5.3.3.6 Operating Conditions

Procedural controls on plant operation are implemented to hold thermal stresses within acceptable ranges. These restrictions on coolant temperature are:

a. The average rate of change of reactor coolant temperature during normal heatup and cooldown shall not exceed 100°F during any one-hour period.

b. If the coolant temperature difference between the dome (inferred from $P_{(sat)}$ and the bottom head drain exceeds 100°F, neither reactor power level nor recirculation pump flow shall be increased.

c. The pump in an idle reactor recirculation loop shall not be started unless the coolant temperature in that loop is within 50°F of average reactor coolant temperature.

The limit regarding the normal rate of heatup and cooldown (Item a) assures that the vessel rod drive housing and stub tube stresses and usage remain within acceptable limits. The limit regarding a vessel temperature limit on recirculating pump operation and power level increase restriction (Item b) augments the Item a limit in further detail by assuring that the vessel bottom head region will not be warmed at an excessive rate caused by rapid sweep out of cold coolant in the vessel lower head region by recirculating pump

operation or natural circulation (cold coolant can accumulate as a result of control drive inleakage and/or low recirculation flow rate during startup of hot standby). The Item c limit further restricts operation of the recirculating pumps to avoid high thermal stress effects in the pumps and piping, while also minimizing thermal stresses on the vessel nozzles.

The above operational limits when maintained insure that the stress limits within the reactor vessel and its components are within the thermal limits to which the vessel was designed for normal operating conditions. To maintain the integrity of the vessel in the event that these operational limits are exceeded the reactor vessel has also been designed to withstand a limited number of transients caused by operator error. Also, for abnormal operating conditions where safety systems or controls provide an automatic temperature and pressure response in the reactor vessel, the reactor vessel integrity is maintained since the severest anticipated transients have been included in the design conditions. Therefore, it is concluded that the vessel integrity will be maintained during the most severe postulated transients, since all such transients are evaluated in the design of the reactor vessel. The postulated transient for which the vessel has been designed is shown on Figure 5.2-5 and discussed in Section 5.2.2.

5.3.4 References for Section 5.3

1. Brandt, F. A., "Mechanical Property Surveillance of Reactor Pressure Vessels for General Electric BWR 6 Plants", NEDO-20651, March 1975.

2. Watanabe, H., "Boiling Water Reactor Feedwater Nozzle/Sparger, Final Report", August 1979 (NEDE-21821-2 and NEDO-21821-2).

3. Cooke, F., "Transient Pressure Rises Affecting Fracture Toughness Requirements for BWR", NEDO-21778-A, December 1978.

TABLE 5.3-1

CHARPY TEST RESULTS AND CHEMICAL COMPOSITION

A. PERRY UNIT 1

I. VESSEL BELTLINE MATERIAL IDENTIFICATION

A. Number 2 Shell Ring

Plates - Pc. 22-1-1, Heat C2557, Slab 1
Pc. 22-1-2, Heat B6270, Slab 1
Pc. 22-1-3, Heat A1155, Slab 1

B. Welds in No. 2 Shell Ring Vertical Seams

Seam BD - Type E8018NM, Heat 627260, Lot B322A27AE
Type E8018NM, Heat 626677, Lot C301A27AF
Type RACO-1NMM, Heat 5P6214B, Lot 0331

Seam BE - Type E8018NM, Heat 624063, Lot C228A27A
Type E8018NM, Heat 626677, Lot C301A27AF
Type E8018NM, Heat 627069, Lot C312A27A
Type RACO-1NMM, Heat 5P6214B, Lot 0331

Seam BF - Type E8018NM, Heat 627260, Lot B322A27AE
Type E8018NM, Heat 626677, Lot C301A27AF
Type RACO-1NMM, Heat 5P6214B, Lot 0331

II. CHEMICAL ANALYSES FOR BELTLINE MATERIAL

A. Plates	C	Mn	P	S	Cu	Si	Ni	Mo	V	Al
Pc. 22-1-1 Ht C2557	.23	1.32	.010	.025	.06	.27	.61	.54	.001	.039
Pc. 22-1-2, Ht B6270	.20	1.28	.012	.015	.06	.23	.63	.53	0	.039
Pc. 22-1-3, Ht A1155	.20	1.33	.010	.013	.06	.28	.63	.54	.002	.031

B. Welds	C	Mn	Ni	Si	Mo	Cu	P	S	V
Ht. 627260 Lot B322A27AE	.04	1.25	1.08	.56	.64	.06	.020	.022	.02
Ht. 626677 Lot C301A27AF	.048	1.10	.85	.56	.45	.010	.015	.022	.009
Ht. 5P6214B Lot 0331	.051	1.39	.82	.53	.52	.02	.013	.017	.004
Ht. 624063 Lot D228A27A	.041	1.12	1.00	.41	.54	.03	.009	.018	.01
Ht. 627069 Lot C312A27AG	.037	1.07	.94	.60	.52	.010	.013	.019	.012

TABLE 5.3-1 (Continued)

B. PERRY UNIT 2

I. VESSEL BELTLINE MATERIAL IDENTIFICATION

A. Number 2 Shell Ring

Plates - Pc. 22-1-1, Heat C3474, Slab 1
Pc. 22-1-2, Heat C3560, Slab 1
Pc. 22-1-3, Heat C3560, Slab 2

B. Welds in No. 2 Shell Ring Vertical Seams

Seam BD - Type E8018NM, Heat 422K8511, Lot G313A27AD
Type E8018NM, Heat 492L4871, Lot A421B27AE
Type E8018NM, Heat 04T981, Lot A423B27AG
Type RACO-1NMM, Heat 5P6756, Lot 0342

Seam BE - Type RACO-1NMM, Heat 5P6756, Lot 0342
Type E8018NM, Heat 492L4871, Lot A421B27AE
Type E8018NM, Heat 422K8511, Lot G313A27AD

Seam BF - Type E8018NM, Heat 422K8511, Lot G313A27AD
Type E8018NM, Heat 492L4871, Lot A421B27AE
Type E8018NM, Heat 04T931, Lot A423B27AG
Type RACO-1NMM, Heat 5P6756, Lot 0342

II. CHEMICAL ANALYSES FOR BELTLINE MATERIAL

A. Plates	C	Mn	P	S	Cu	Si	Ni	Mo	V	Al
Pc. 22-1-1, Ht C3474-1	.19	1.32	.011	.014	.09	.23	.63	.56	0	.022
Pc. 22-1-2, Ht C3560-1	.18	1.32	.010	.015	.07	.24	.61	.55	0	.036
Pc. 22-1-3, Ht C3560-2	.18	1.32	.010	.015	.06	.24	.61	.55	0	.036

B. Welds	C	Mn	Ni	Si	Mo	Cu	P	S	V
Ht. 422K8511, Lot G313A27AD	.06	1.21	1.00	.31	.54	.01	.016	.013	.02
Ht. 492L4871, Lot A421B27AE	.07	1.06	.95	.37	.50	.04	.018	.025	.02
Ht. 04T931, Lot A423B27AG	.05	1.03	1.00	.28	.53	.03	.020	.024	.01
Ht. 5P6756, Lot 0342(1)	.063	1.27	.93	.57	.45	.08	.010	.011	.006
Ht. 5P6756, Lot 0342(2)	.078	1.24	.92	.53	.46	.09	.010	.012	.006

(1) Single Wire (2) Tandem Wire

TABLE 5.3-2

UNIRRADIATED FRACTURE TOUGHNESS PROPERITES

A. PERRY UNIT 1

Plates Ht.No.	Drop Wt. NDT(°F)	Transverse CVN Ft-lbs	MLE	Temp.	Reference Temp.(°F)	Upper Shelf (Ft-lb)
C2557-1						
Top	-20	55,50,52	42,46,42	+70F	0	84
Bottom	-20	54,64,76	63,53,46	+60F		
B6270-1						
Top	-40	53,78,56	43,58,44	+20F		
Bottom	30	63,63,64	51,51,52	+30F	-30	94
A1155-1						
Top	-20	65,63,67	54,60,52	+50F	-20	114
Bottom	-20	54,66,85	68,55,44	+40F		

Weld Metal	Drop Wt. NDT(°F)	CVN Ft-lbs	MLE	Temp.	Reference Temp.(°F)	Upper Shelf (Ft-lb)
Ht. 627260 Lot B322A27AE	-40	52,56,51	36,37,35	+30F	-36	104
Ht. 626677 Lot C301A27AF	-40	53,51,54	36,37,35	+40	-20	90
Ht. 5P6214B	-50	56,50,54	45,41,46	+10	-50	88
Lot 0331	-40	50,61,65	46,50,52	+10	-40	96
Ht. 624063 Lot 228A27A	-60	57,59,68	37,38,46	+10	-50	105
Ht. 627069 Lot C312A27A	-60	72,64,78	52,48,56	0	-60	112

TABLE 5.3-2 (Continued)

B. PERRY UNIT 2

Plates Ht.No.	Drop Wt. NDT(°F)	Transverse CVN Ft-lbs	MLE	Temp.	Reference Temp.(°F)	Upper Shelf (Ft-lb)
C3474-1						
Top	-30	54,54,53	44,44,44	+40F	-20	95
Bottom	-40	51,63,69	41,45,51	+20F		
C3560-1						
Top	-30	92,86,68	71,56,62	+30F		
Bottom	-30	54,74,64	58,62,46	+40F	-20	125
C3560-1						
Top	-30	70,52,61	44,54,49	+30F		
Bottom	-30	60,66,76	62,52,48	+40F	-20	107

Weld Metal	Drop Wt. NDT(°F)	Transverse CVN Ft-lbs	MLE	Temp.	Reference Temp.(°F)	Upper Shelf (Ft-lb)
Ht. 422K8511 Lot G313A27AD	-80	65,74,127	44,48,76	-20F	-80	143
Ht. 492L4871 Lot A421B27AE	-90	50,51,57	36,38,40	0F	-60	151
Ht. 04T931 Lot A423B27AG	-90	65,69,72	52,48,50	0F	-60	148
Ht. 5P6756 Lot 0342 (Single Wire)	-60	55,66,63	54,51,57	0F	-60	89
Ht. 5P6756 Lot 0342 (Tandem Wire)	-50	64,79,77 80,72	35,72,55 69,60	+10F	-50	95

TABLE 5.3-3

PERRY 1

EOL BELTLINE PLATE RT_{ndt} & WELDS

A. Beltline Plates

Heat	WT %Cu	WT%P	ASME NB-2300 Start RT_{ndt}	Reg. Guide 1.99 Extrap. ΔRT_{ndt} (°F)	Estimated EOL RT_{ndt} (°F)
C2557-1**	.06	.010	0	34	+34 Limiting Plate
B6270-1	.06	.012	-30	40	+10
A1155-1	.06	.010	-20	34	+14

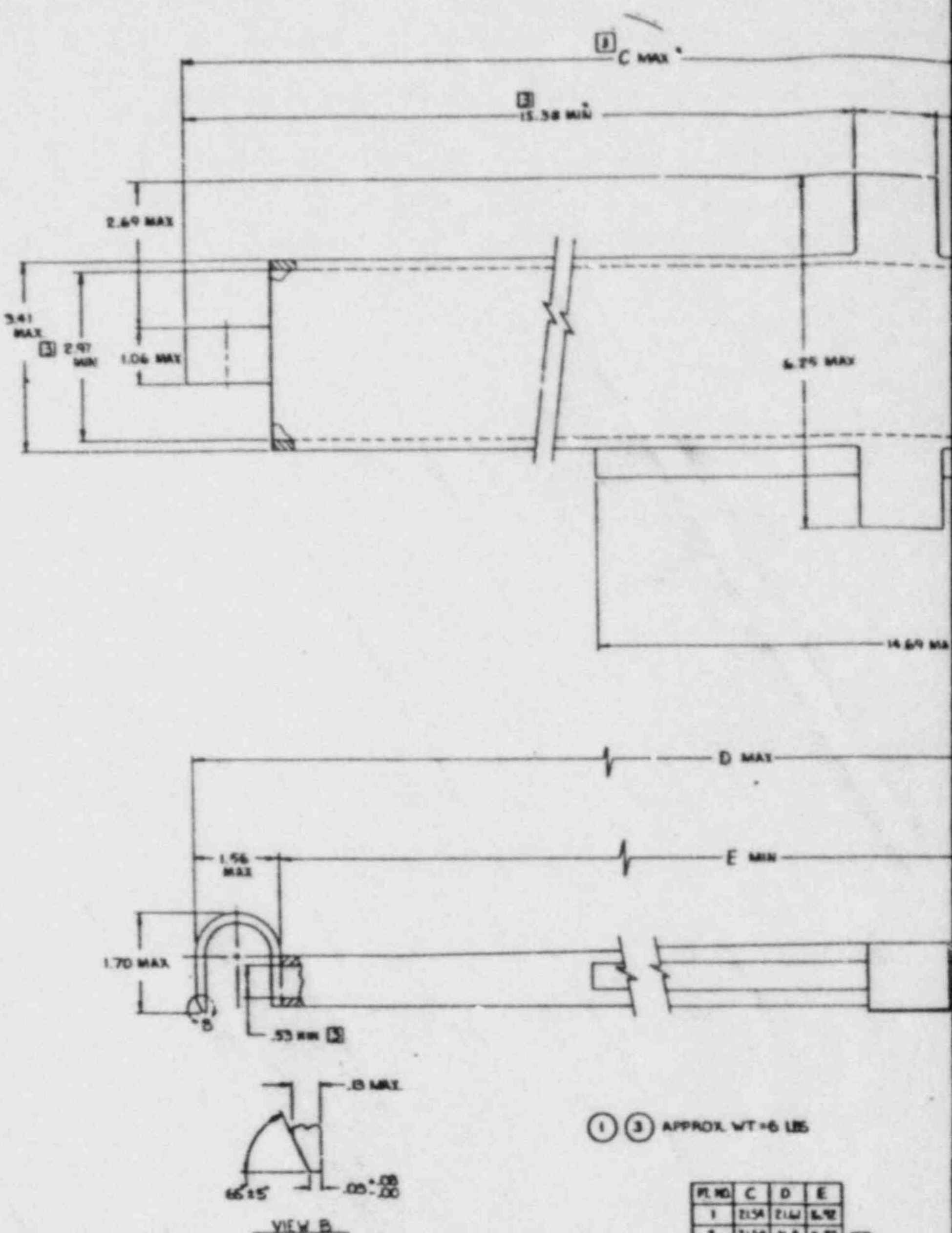
B. Beltline Welds

Weld Seam	Heat/Lot	WT.%Cu	WT.%P	ASME NB-2300 Start RT_{ndt}	Reg. Guide 1.99 Extrap ΔRT_{ndt} (°F)	Estimated EOL RT_{ndt} (°F)	Weld Wire Type	Flux Type
BD, BF	627260/B322A27AE***	.06	.02	-36	67	+31	E8018NM	--
BD, BE, BF	626677/C301A27AF***	.01	.015	-20	50	+30	E8018NM	--
BE	624063/D22BA27A	.03	.009	-50	30	-20	E8018NM	--
BE	627069/C312A27A	.01	.013	-60	43	-17	E8018NM	--
BD, BE, BF	5P6214B/0331* ***	.02	.013	-40	43	+3	RACO 1 NMM (Single Wire)	Linde 124

C. Test Plate Weld Mat'l

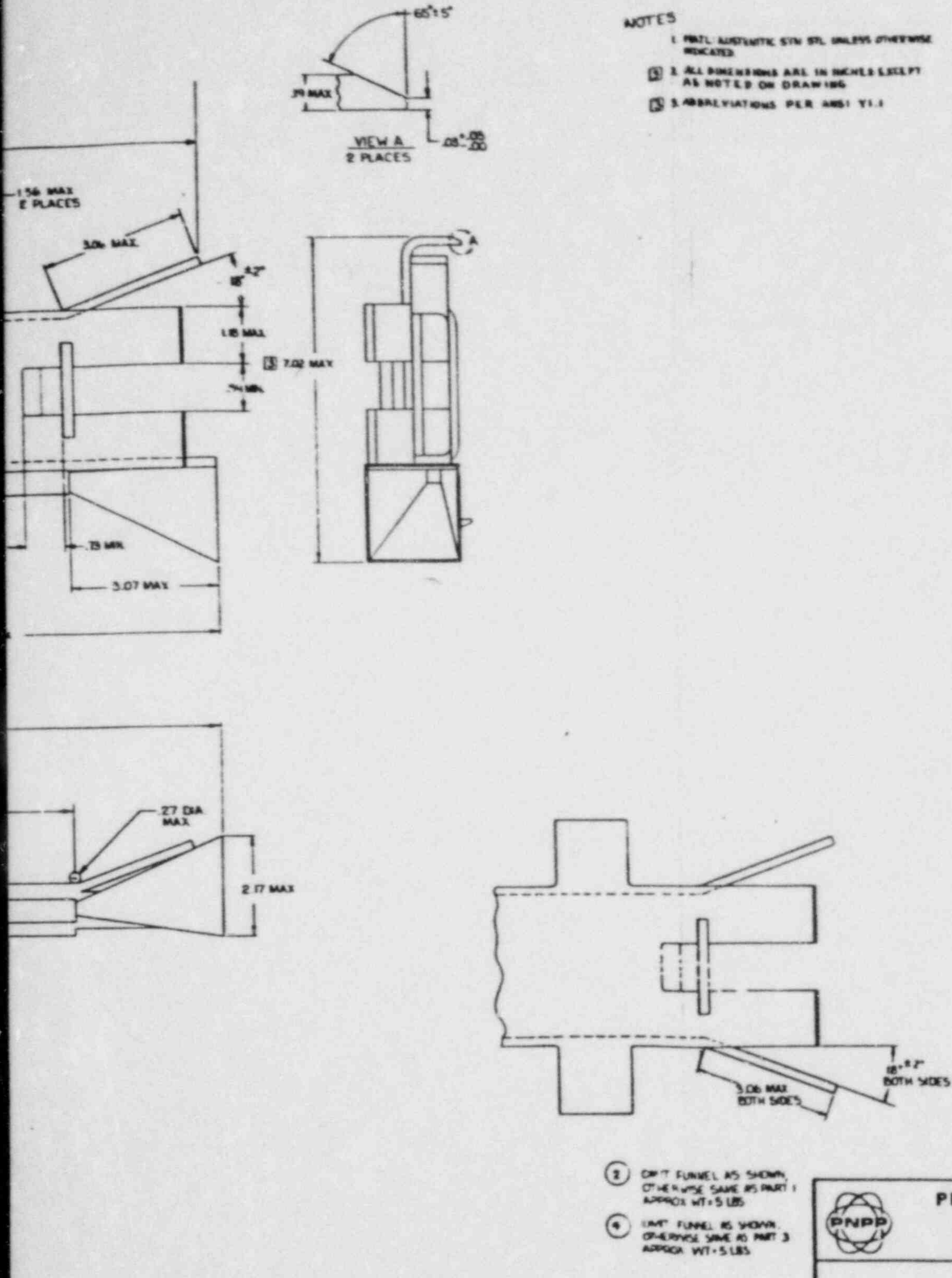
Heat/Lot	WT.%Cu	WT.%P	ASME NB-2300 Start RT_{ndt}	Reg. Guide 1.99 Extrap. ΔRT_{ndt} (°F)	Estimated EOL RT_{ndt} (°F)
624039/D205A27A***	.028	.015	-90	50	-40
07R458/S403B27AG***	.04	.02	-60	67	+7
401P2871/H430B27AF***	.03	.013	-50	43	-7

*Submerged ARC welding/**Surveillance Test Plate/***Surveillance Weld Material



PT. NO.	C	D	E
1	21.54	21.61	16.92
2	21.54	N A	16.92
3	23.79	22.50	19.17
4	23.79	N A	19.17

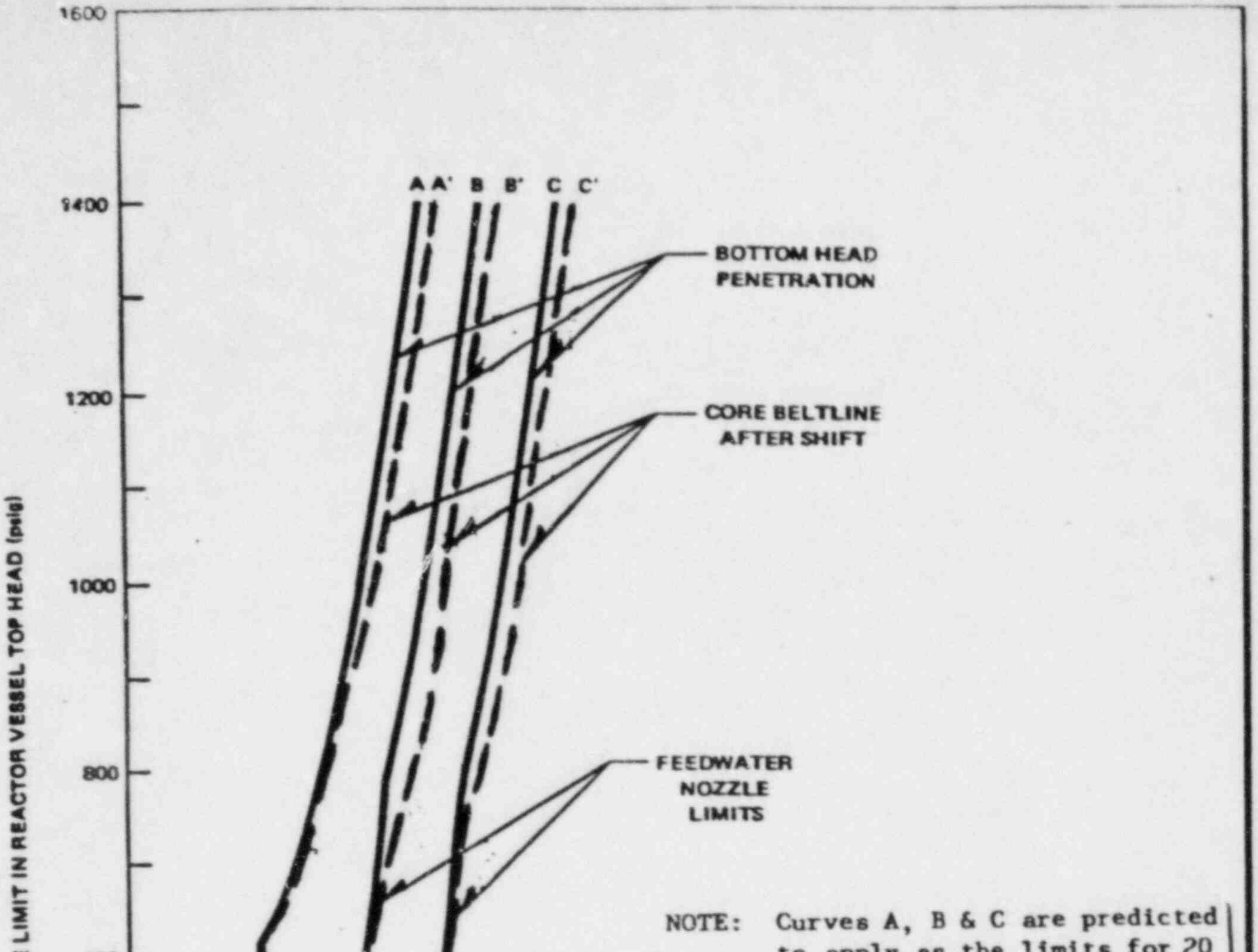
3



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Surveillance Bracket

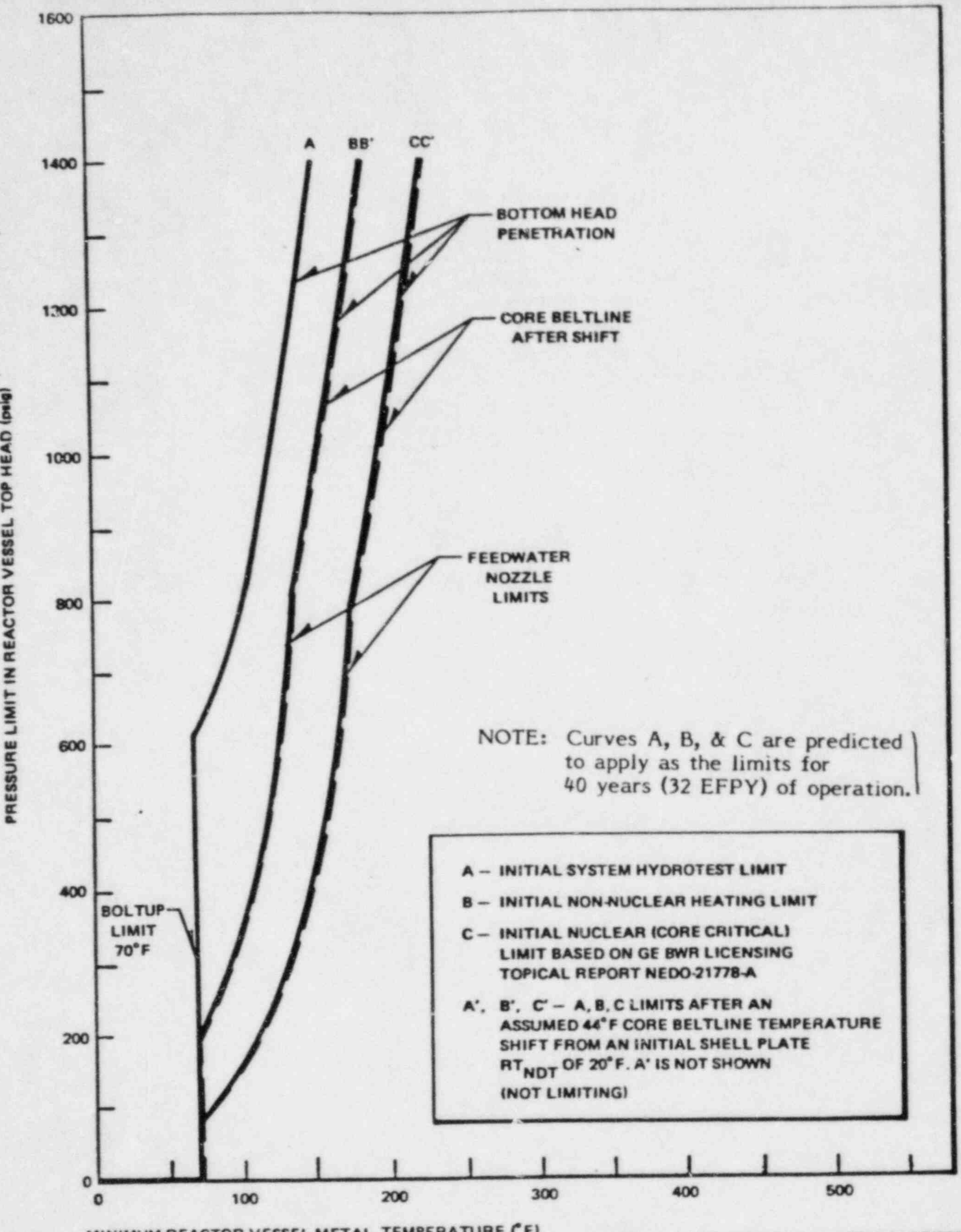
Figure 5.3-1



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

Minimum Temperatures Required vs Reactor Pressure, Unit 1

Figure 5.3-2



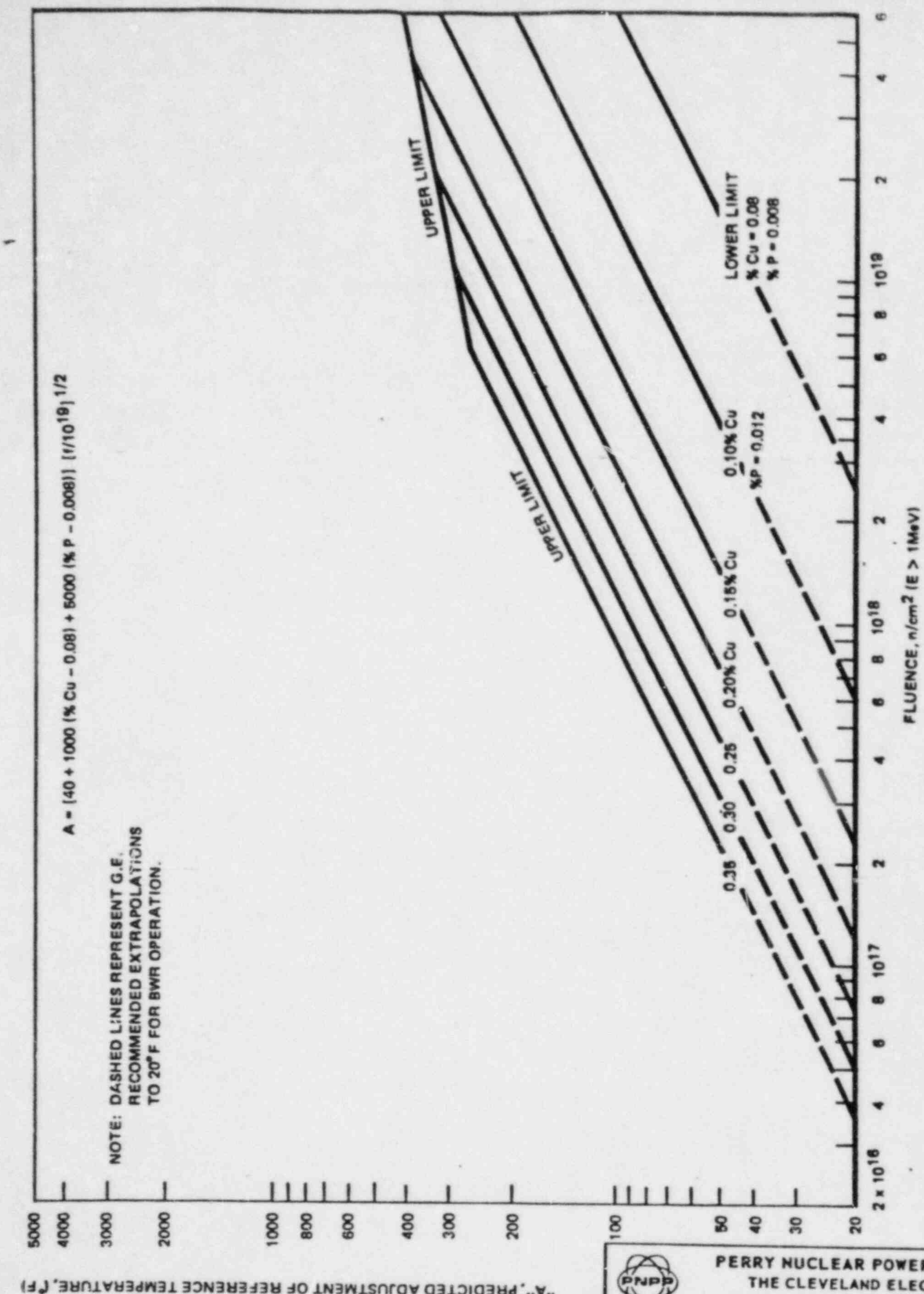
PNPP

PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

Minimum Temperatures Required vs Reactor Pressure, Unit 2

Figure 5.3-3

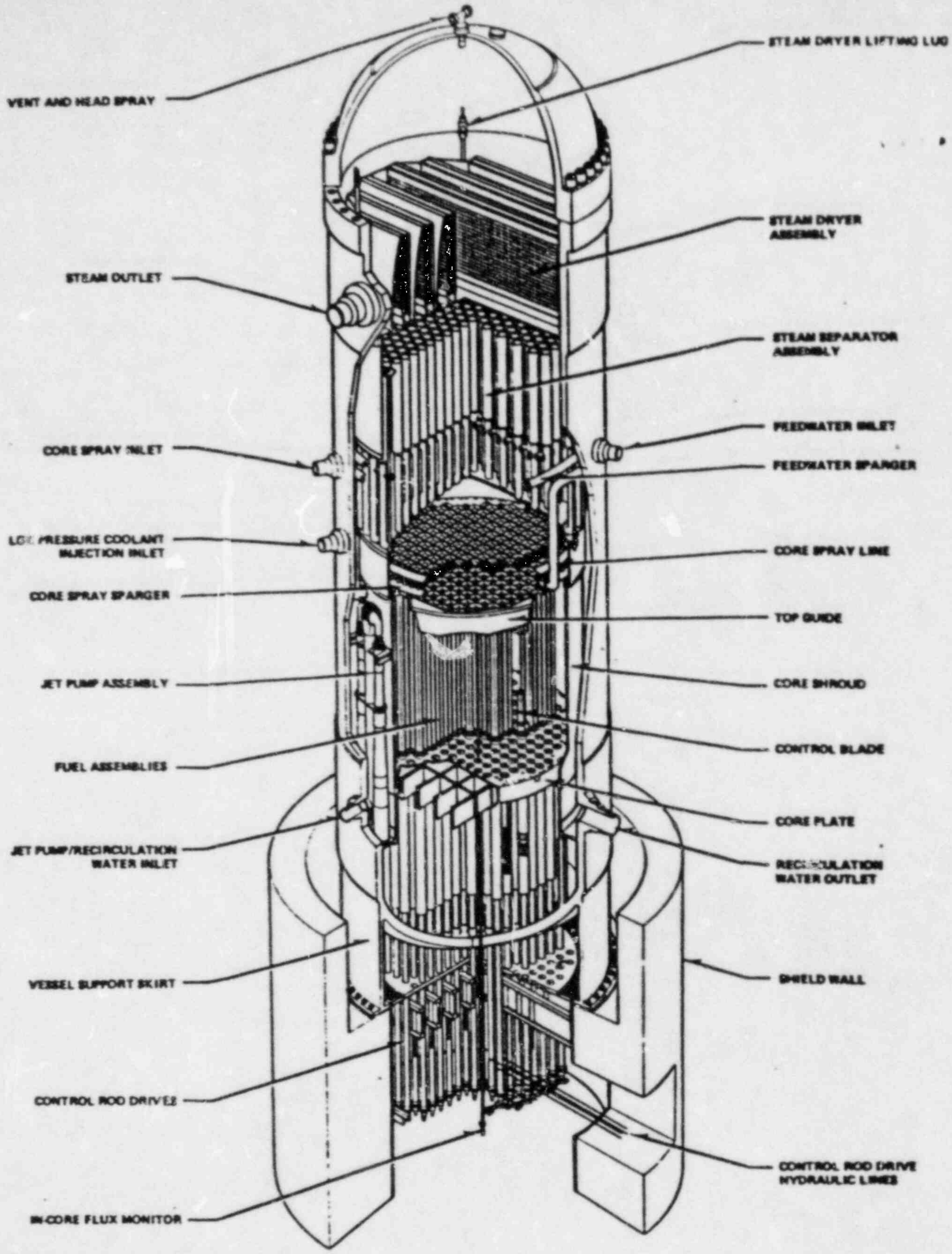
FIGURE 5.3-4 HAS BEEN DELETED



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

Predicted Adjustment of Reference Temperature

Figure 5.3-5

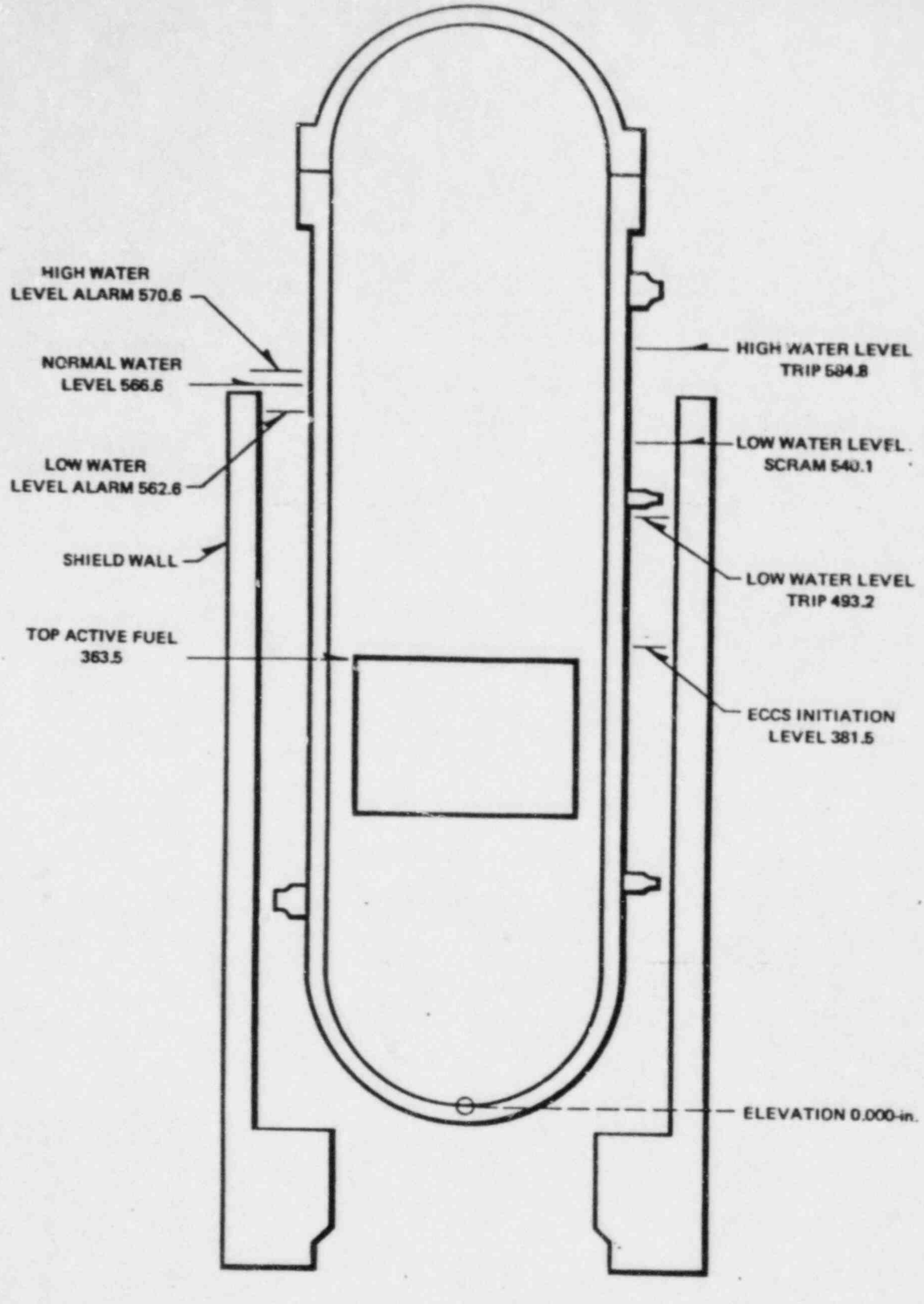


PNPP

PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Reactor Vessel Cutaway Diagram

Figure 5.3-6



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Reactor Vessel Nominal Water
Level Trip and Alarm Elevations

Figure 5.3-7