5.3 REACTOR VESSEL

5.3.1 Reactor Vessel Materials

5.3.1.1 Materials Specifications

The materials used in the reactor pressure vessel and appurtenances are shown in Table 5.2-4 together with the applicable specifications.

5.3.1.2 Special Processes Used for Manufacturing and Fabrication

The reactor pressure vessel is primarily constructed from low alloy, high strength steel plates and forgings. Plates are ordered to ASME SA 533 Grade B, Class 1, and forgings to ASME SA 508, Class 2. These materials are melted to fine grain practice and are supplied in the quenched and tempered condition. Further restrictions include a requirement for vacuum degassing to lower the hydrogen level and improve the cleanliness of the low alloy steels.

Studs, nuts, and washers for the main closure flange are ordered to the requirements on Table 5.2-4. Welding electrodes are low hydrogen type ordered to ASME SA 316.

All plates, forgings, and boltings are 100% ultrasonically tested and surface examined by magnetic particle methods or liquid penetrant methods in accordance with ASME Section III, Subsection NB standards. Fracture toughness properties are also measured and controlled in accordance with subsection NB requirements.

All fabrication of the reactor pressure vessel is performed in accordance with GE approved drawings, fabrication procedures, and test procedures. The shells and vessel heads are made from formed plates, and the flanges and nozzles from forgings. Welding performed to join these vessel components is in accordance with procedures qualified per ASME Section III and IX requirements. Weld test samples are required for each procedure for major vessel full penetration welds. Tensile and impact tests are performed to determine the properties of the base metal, heat affected zone and weld metal.

Submerged arc and manual stick electrode welding processes are employed. Electroslag welding is not permitted. Preheat and interpass temperatures employed for welding of low alloy steel meet or exceed the requirements of ASME Section III, Subsection NA. Post weld heat treatment at 1100°F minimum is applied to all low alloy steel welds.

Radiographic examination is performed on all pressure containing welds in accordance with Paragraph N624 of the 1968 ASME Section III Code including 1970 addenda. In addition, all welds are given a supplemental ultrasonic examination.

The materials, fabrication procedures, and testing methods used in the construction of BWR reactor pressure vessels meet the requirements of ASME Section III Class I vessels, 1968 Edition with Summer 1970 Addenda. Paragraph NB-338.2(d)(4) of the Winter 1971 Addenda shall supercede Paragraph I-613(d) of the 1968 Edition, and Paragraph NB-2400 of the 1971 Edition shall apply for all fabrication performed at the Susquehanna site.

5.3.1.3 Special Methods for Nondestructive Examination

The materials and welds on the reactor pressure vessel were examined in accordance with methods prescribed and met the acceptance requirements specified by ASME Boiler and Pressure Vessel Code, Section III. In addition, the pressure retaining welds were ultrasonically examined using manual techniques. The ultrasonic examination method, including calibration, instrumentation, scanning sensitivity, and coverage was based on the requirements imposed by ASME Code, Section XI in Appendix I. Acceptance standards were equivalent or more restrictive than required by ASME Code, Section XI.

5.3.1.4 Special Controls For Ferritic and Austenitic Stainless Steels

5.3.1.4.1 Compliance With Regulatory Guides

5.3.1.4.1.1 Regulatory Guide 1.31, (Rev 1) Control of Stainless Steel Welding

Controls on stainless steel welding are discussed in Subsection 5.2.3.4.2.1

5.3.1.4.1.2 Regulatory Guide 1.34, (12/72) Control of Electroslag Weld Properties

Electroslag welding was not employed for the reactor pressure vessel fabrication.

5.3.1.4.1.3 Regulatory Guide 1.43, (5/73) Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components

Reactor pressure vessel specifications require that all low alloy steel be produced to fine grain practice. Regulatory Guide 1.43 applies to RPV steels that have been manufactured to coarse grain steel making practice. The SSES vessels were manufactured to fine grain steel making practice; therefore, this regulatory guide does not apply to the SSES vessels.

5.3.1.4.1.4 Regulatory Guide 1.44, (5/73) Control of the Use of Sensitized Stainless Steel

Controls to avoid severe sensitization are discussed in Subsection 5.2.3.4.1.1.

5.3.1.4.1.5 Regulatory Guide 1.50 (5/73), Control of Preheat Temperature for Welding Low-Alloy Steel

Preheat controls are discussed in Subsection 5.2.3.3.2.1.

5.3.1.4.1.6 Regulatory Guide 1.71, (12/73) Welder Qualification for Areas of Limited Accessibility

Qualification for areas of limited accessibility is discussed in Subsection 5.2.3.3.2.3.

5.3.1.4.1.7 Regulatory Guide 1.99, (Rev. 2) Radiation Embrittlement of Reactor Vessel Materials

Predictions for changes in transition temperature and upper shelf energy were made in accordance with the requirements of Regulatory Guide 1.99, Revision 2.

5.3.1.5 Fracture Toughness

5.3.1.5.1 Compliance with 10CFR50 Appendix G

A major condition necessary for full compliance to Appendix G is satisfaction of the requirements of the Summer 1972 Addenda to Section III. This is not possible with components which were purchased to earlier Code requirements. For the extent of compliance, see Tables 5.3-1a and 5.3-2a.

Ferritic material complying with 10 CFR 50, Appendix G, must have both drop weight tests and Charpy V-notch (CVN) tests with the CVN specimens oriented transverse to the principal material working direction to establish the RT_{NDT}. The CVN tests must be evaluated against both an absorbed energy and lateral expansion criteria. The maximum acceptable RT must be determined in accordance with the analytical procedures of ASME Code Section III, Appendix G. Appendix G of 10 CFR 50 requires a minimum of 75 ft-lbs upper shelf CVN energy for beltline material. It also requires at least 45 ft-lbs CVN energy and 25 mils lateral expansion for bolting material at the lower of the preload or lowest service temperature.

By comparison, material for the Susquehanna SES reactor vessels was qualified by either drop weight tests and/or longitudinally oriented CVN tests (both not required), confirming that the material nil-ductility transition temperature (NDT) is at least 60°F below the lowest service temperature. When the CVN test was applied, a 30 ft-lbs energy level was used in defining the NDT. There was no upper shelf CVN energy requirement on the beltline material. The bolting material was qualified to a 30 ft-lbs energy requirement at 60°F below the minimum preload temperature.

From the previous comparison it can be seen that the fracture toughness testing performed on the SSES reactor vessel material in some cases cannot be shown to comply with 10 CFR 50, Appendix G. However, to determine operating limits in accordance with 10 CFR 50, Appendix G, estimates of the beltline material RT_{NDT} and the highest RT_{NDT} of all other material were made, as explained in Subsection 5.3.1.5.1.2. The method for developing these operating limits is also described therein.

On the basis of the last paragraph on page 19013 of the July 17, 1973, Federal Register, the following is considered an appropriate method of compliance.

5.3.1.5.1.1 Intent of Proposed Approach

The intent of the proposed special method of compliance with Appendix G for this vessel is to provide operating limitations on pressure and temperature based on fracture toughness. These operating limits assure that a margin of safety against a nonductile failure of this vessel is nearly the same as that for a vessel built to the Summer 1972 Addenda.

The specific temperature limits for operation when the core is critical are based on 10 CFR 50, Appendix G, January 1998.

5.3.1.5.1.2 Operating Limits Based on Fracture Toughness

Operating limits which define minimum reactor vessel metal temperatures vs reactor pressure during normal heatup and cooldown and, during in-service hydrostatic testing, were established using the methods of Appendix G of Section XI of the ASME Boiler and Pressure Vessel Code, 1992 Edition. The results are shown in Technical Specification Figures 3.4.10-1, 3.4.10-2, and 3.4.10-3 for each Unit. These figures have been modified and updated based on surveillance capsule tests that were performed after the first specimens were removed from the reactors and tested after 6 EFPY and calculations based on ASME Code Cases N-588 and N-640.

Estimated RT_{NDT} values and temperature limits are given in this section for the limiting locations in the reactor vessel.

All the vessel shell and head areas remote from discontinuities, all other shell and head areas, flanges, and the feedwater nozzles were evaluated: the operating limit curves are based on the limiting locations. The boltup limits for the flange and adjacent shell region are based on a minimum metal temperature of RT_{NDT} + 60°. The maximum through-wall temperature gradient from continuous heating or cooling at 100°F per hour was considered. The safety factors applied were as specified in ASME Code, Section XI, Appendix G.

For the purpose of setting these operating limits, the reference temperature, RT_{NDT}, is determined from the toughness test data taken in accordance with requirements of the Code to which this vessel is designed and manufactured. This toughness test data, Charpy V-notch (CVN) and/or drop-weight nil-ductility transition temperature (NDT) is analyzed to permit compliance with the intent of 10 CFR 50, Appendix G. Because all toughness testing needed for strict compliance with Appendix G was not required at the time of vessel procurement, some toughness results are not available. For example, longitudinal CVN's, instead of transverse, were tested, usually at a single test temperature of +10°F or +40°F, for absorbed energy. Also, at the time either CVN or NDT testing was permitted; therefore, in some cases both tests were not performed as is currently required. To substitute for this absence of certain data, toughness property correlations were derived for the vessel materials in order to operate upon the available data to give a conservative estimate of RT_{NDT}, compliant with the intent of Appendix G criteria.

These toughness correlations vary, depending upon the specific material analyzed, and were derived from the results of WRC Bulletin 217, "Properties of Heavy Section Nuclear Reactor Steels," and from toughness data from the Susquehanna SES vessels and other reactors. In the case of vessel plate material (SA-533 Grade B, Class 1), the predicted limiting toughness property is either NDT or transverse CVN 50 ft-lbs temperature minus 60°F. CVN and NDT data are available for all of the beltline plates. Where NDT results are missing, NDT is estimated as the longitudinal CVN 35 ft-lbs transition temperature. The transverse CVN 50 ft-lbs transition temperature is estimated from longitudinal CVN data in the following manner. The lowest longitudinal CVN ft-lb value is adjusted to derive a longitudinal CVN 50 ft-lbs transition temperature by adding 2°F per ft-lb to the test temperature. If the actual data equal or exceed 50 ft-lbs, the test temperature is derived by interpolation or conservatively taken as the transition temperature. Once the longitudinal 50 ft-lbs temperature is derived, an additional

30°F is added to account for orientation effects and to estimate the transverse CVN 50 ft-lbs temperature minus 60°F, as described above.

For forgings (SA-508 Class 2), the predicted limiting property is the same as for vessel plates. CVN and NDT values are available for the vessel flange, closure head flange, and feedwater nozzle materials for Susquehanna SES. RT_{NDT} is estimated in the same way as for vessel plates.

For the vessel weld metal, the predicted limiting property is the CVN 50 ft-lbs transition temperature minus 60°F, as the NDT values are -50°F or lower for these materials. This temperature is derived in the same way as for the vessel plate material, except the 30°F addition for orientation effects is omitted since there is no principal working direction. When NDT values are available, they are also considered and the RT_{NDT} is taken as the higher of NDT or the 50 ft-lbs temperature minus 60°F. When NDT is not available, the RT_{NDT} shall not be less than -50°F, since lower values are not supported by the correlation data.

For vessel weld heat affected zone (HAZ) material, the RT_{NDT} is assumed the same as for the base material as ASME Code weld procedure qualification test requirements, and post weld heat treatment indicates this assumption is valid.

Closure bolting material (SA-540 Grade B24) toughness test requirements for Units 1 and 2 were for 30 ft-lbs at 60°F below the bolt-up temperature. Current Code requirements are for 45 ft-lbs and 25 mils lateral expansion at the preload or lowest service temperature, including bolt-up. The reactor vessel closure studs for Unit 1 have a minimum Charpy impact energy of 40 ft-lbs and 25 mils lateral expansion at 10°F. The lowest service temperature for the closure studs is 70°F for Unit 1. For Unit 2, the closure studs have a minimum Charpy impact energy of 48 ft-lbs and 27 mils lateral expansion at 10°F; therefore, the lowest service temperature for the Unit 2 closure studs is +10°F.

Using the above general approach, an initial RT_{NDT} of +18°F was established for the core beltline region for Unit 1 and +10°F for Unit 2.

The effect of the main closure discontinuity was considered by adding 60°F to the RT_{NDT} to establish the minimum temperature for boltup and pressurization. The minimum bolt-up temperature of +70°F for Units 1 and 2, which is required in Technical Bases 3.4.10 and is shown on Figures 3.4.10-1 through 3.4.10-3 in the Technical Specification for each Unit, is based on an initial RT_{NDT} of +10°F for the closure flange forgings.

The effect of the vessel nozzle and bottom head discontinuities is considered by developing separate curves for the bottom head and non-beltline regions. These separate curves utilized the appropriate Susquehanna SES nozzle forging and bottom head RT_{NDT}'s. For Unit 1, the controlling discontinuity limits are based on the recirculation inlet nozzles (RT_{NDT}=40°F) for feedwater nozzle limits and the bottom head penetrations (RT_{NDT}=34°F) for the CRD penetration limits. For Unit 2, the controlling discontinuity limits are based on the steam outlet nozzles (RT_{NDT}=30°F) for the feedwater nozzle limits and the recirculation outlet nozzle (RT_{NDT}=24°F) for the CRD penetration limits.

5.3.1.5.1.3 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 3.4.10-1 in Technical Specifications for each Unit (Reference 5.3-1, Task A-15). Curves labeled C on Technical Specification Figures 3.4.10-3 apply whenever the core is critical.

5.3.1.5.1.4 Temperature Limits For ISI Hydrostatic or Leak Pressure Tests

The fracture toughness analysis for in-service inspection or leak pressure tests resulted in the curves in Technical Specification Figure 3.4.10-1 for each Unit.

5.3.1.5.1.5 Adjusted Reference Temperature for Limiting Core Beltline Material

Adjusted reference temperature (ART) is a prediction of the effect of fast neutron fluence on RT_{NDT} at 1/4 of the vessel wall thickness. The applicable shift in RT_{NDT} including applicable effects of power uprate, is added to the pressure-temperature curve of each of the limiting beltline materials to produce the Unit 1 and 2 "Core Beltline" curves in Technical Specification Figures 3.4.10-1, 3.4.10-2, and 3.4.10-3.

The limiting beltline material for Unit 1 is plate heat C2433-1, which has an initial RT_{NDT} of +18°F and a 35.7-EFPY ART of 61.4°F (References 5.3-2 and 5.3-6).

The limiting beltline material for Unit 2 is plate heat C2421-3, which has an initial RT_{NDT} of -10°F and a 30.2-EFPY ART of 46.7°F (References 5.3-3 and 5.3-6).

5.3.1.5.1.6 Temperature Limits for Boltup

A minimum temperature of 70° for Unit 1 and of 10° for Unit 2 is required for closure studs. A sufficient number of studs may be tensioned at 70°F to seal the closure flange O-rings for the purpose of raising reactor water level above the closure flanges in order to assist in warming them. The flanges and adjacent shell are required to be warmed to a minimum temperature of 70°F before they are stressed by the full intended bolt preload. The fully preloaded bolt-up limits are shown on Technical Specification Figures 3.4.10-1, 3.4.10-2, and 3.4.10-3 for Unit 1 and Unit 2.

5.3.1.5.1.7 Reactor Vessel Annealing

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

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.

Materials for the program are selected to represent materials used in the reactor beltline region. The 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 metal, and the transition zone between base metal and weld. 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.

The surveillance program includes three capsule holders per reactor vessel. Charpy impact specimens for the reactor specimens for the reactor vessel surveillance programs are of the longitudinal orientation consistent with the ASME requirements prior to the issuance of the Summer 1972 Addenda and ASTM-E-185-82. Based on GE experience, the amount of shift measured by these irradiated logitudinal test specimens will be essentially the same as the shift in an equivalent transverse specimen.

The program for implementation of the scheduling and testing of the surveillance specimen is governed and controlled by the BWR Integrated Surveillance Program (ISP) [References 5.3-7 through 5.3-10]. The Unit 1 second holder (131C7717(G2) will be pulled in accordance with the schedule in the ISP. For Unit 2, all the information will come from the other plants in the Integrated Surveillance Program. No capsules are scheduled to be withdrawn from Unit 2. Other plants will remove and test specimens in accordance with the ISP. The results from these test will provide the necessary data to monitor embrittlement for Unit 2. Since the predicted adjusted reference temperature of the reactor vessel beltline steel is less than 100°F at end of life, the use of the capsules per the ISP meets the requirements of 10 CFR 50, Appendix H, and ASTM-E-185-82. The withdrawl schedule and other requirements are provided in the ISP.

For the extent of compliance to 10 CFR 50, Appendix H, see Tables 5.3-1 b and 5.3-2b.

Each holder is loaded with capsules which contain the following surveillance specimens and dosimeter wires:

First holder (131C7717G3):

36 Charpy impact specimens including 12 base metal, 12 weld metal, and 12 heat affected zone metal specimens; 10 tensile specimens including 3 base metal, 4 weld metal, and 3 weld heat affected zone metal specimens; 9 metal wire dosimeters including 3 iron, 3 nickel, and 3 copper.

After the first capsule holders (for both Units 1 and 2) were withdrawn and the specimens tested (see references 5.3-4 and 5.3-5), the broken specimens were remachined as miniature specimens and reloaded in the vessels during the next refueling outages. The contents of the new “reconstituted” capsules (for both Units 1 and 2) are as follows:

2 Charpy specimen packets each containing 12 Charpy specimens – 1 packet for base metal specimens and 1 for weld metal specimens. (EXCEPTION: The Unit 1 weld metal capsule only has 11 specimens).

Copper, Iron and Niobium flux wires are included in the capsules with the Charpy specimens.

2 tensile specimen tubes – 1 containing one tensile capsule with four 0.113 inch diameter miniature tensile specimens, the other containing 1 capsule with one 0.113 inch diameter miniature tensile specimen and one 0.250 inch diameter original weld metal tensile specimen.

The new holders have the same geometry as the original capsule holders.

Second holder (131C7717G2):

24 Charpy impact specimens including 8 base metal, 8 weld metal, and 8 weld heat affected zone metal specimens; 8 tensile specimens including 3 base metal, 3 weld metal, and 2 weld heat affected zone metal specimens; 6 metal wire dosimeters including 2 iron, 2 nickel, and 2 copper.

Third holder (131C7717G1):

24 Charpy impact specimens including 8 base metal, 8 weld metal and 8 weld heat affected zone metal specimens; 6 tensile specimens including 2 base metal, 2 weld metal, and 2 weld heat affected zone metal specimens; 6 metal wire dosimeters including 2 iron, 2 nickel, and 2 copper.

A set of out-of-reactor baseline Charpy V-notch specimens is provided with the surveillance test specimens.

5.3.1.6.2 Neutron Flux and Fluence Calculations

A description of the methods of analysis is contained in Subsections 4.1.4.5 and 4.3.2.8.

5.3.1.6.3 Positioning of Surveillance Capsules and Method of Attachment

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 5.3-3. The capsule holder brackets allow the capsule holder to be removed at any desired time in the life of the plant for specimen testing. These brackets are designed, fabricated and analyzed to the requirements of Section III of the 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.

5.3.1.6.4 Time and Number of Dosimetry Measurements

GE has provided neutron dosimetry wires in each of the specimen holders. In addition, one holder in each vessel is designed with a separately removable dosimeter, to be removed after one fuel cycle. The first cycle dosimeter was removed from Unit 1 in 1986 and analyzed. A first cycle dosimeter was not available for removal from Unit 2. However, the first cycle dosimetry for Unit 1 provides a good estimate of flux for Unit 2, because vessel geometries and core power shapes are very similar.

The first cycle dosimetry provides a means of calibrating the flux distribution calculations to actual vessel conditions. Dosimetry will be updated as holders are removed and tested. The holder withdrawal schedule is listed in Table 5.3-3.

5.3.1.7 Reactor Vessel Fasteners

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 its 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. The design and analysis of this area of the vessel is in full compliance with all Section III Class I Code requirements. The material for studs, nuts and washers is SA-540 Grade B23 or B24. The maximum reported ultimate tensile stress for the bolting material was 163,500 psi, which is less than the 170,000 psi limitation in Regulatory Guide 1.65 (10/73). Also the Charpy impact test results for the closure studs are given in Subsection 5.3.1.5.1.2. Hardness tests are performed on all main closure bolting to demonstrate that heat treatment has been properly performed. Studs, nuts, and washers are ultrasonically examined in accordance with Section III, N-322 and the following additional requirements:

(1) Examination is performed after heat treatment and prior to machining threads.

(2) Straight beam examination is performed on 100 percent of each stud. Reference standard for the radial scan is a 1/2-inch diameter flat bottom hole having a depth equal to 10 percent of the material thickness. For the end scan, the reference is a 1/4-inch flat bottom hole having a standard depth of 1/2-inch.

(3) Nuts and washers are examined by angle beam from the outside circumference in both the axial and circumferential directions.

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 on these surfaces.

5.3.2 PRESSURE-TEMPERATURE LIMITS

5.3.2.1 Limit Curves

Limits on pressure and temperature for in-service leak and hydrostatic tests, normal operation (including heatings and cooldown), and reactor core operation are shown in Technical Specification Figures 3.4.10-1, 3.4.10-2, and 3.4.10-3. The basis used to determine these limits is described in Subsection 5.3.1.5.1.2.

5.3.2.2 Operating Procedures

By comparison of the pressure vs. temperature limits in Subsection 5.3.2.1 with intended normal operating procedures for the most severe upset transient, it is shown that the 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 (loss of AC power) yields a minimum fluid temperature of 250°F and a maximum pressure peak of 1218 psig. Scram automatically occurs with initiation of this upset condition, so the applicable operating limits are given by the curves labeled A in Figure 3.4.10-1 in the Technical Specification for each Unit. For a temperature of 250°F, the maximum allowable pressure exceeds 1218 psig for the intended margin against nonductile failure. The maximum transient pressure of 1218 psig is, therefore, within the specified allowable limits.

5.3.3 REACTOR VESSEL INTEGRITY

The reactor vessels were fabricated for General Electric's Nuclear Energy Division by Chicago Bridge and Iron (CB&I), and were subject to the requirements of General Electric's Quality Assurance program.

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's office.

5.3.3.1 Design

5.3.3.1.1 Description

5.3.3.1.1.1 Reactor Vessel

The reactor vessel shown in Figure 5.3-1 is a vertical, cylindrical pressure vessel of welded construction. The vessels for Units 1 and 2 are designed, fabricated, tested, inspected, and stamped in accordance with the ASME Code Section III, Class A including the Summer Addenda 1970 except that: Paragraph NB-338.2(d)(4) of the Winter 1971 Addenda shall supercede Paragraph I-613(d) of the 1968 Edition, and paragraph NB-2400 of the 1971 Edition shall apply for all fabrication performed at the Susquehanna site. Design of the reactor vessel and its support system meets Seismic Category I equipment requirements. Quality control methods used during the fabrication and assembly of the reactor vessel and appurtenances assure that design specifications are met.

The materials used in the reactor pressure vessel are shown in Table 5.2-4. The cylindrical and bottom head sections of the reactor vessel are fabricated of low alloy steel, the interior of which is clad with stainless steel weld overlay. Nozzle and nozzle weld zones are unclad except for those mating to stainless steel piping systems.

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. Radiation embrittlement is not a problem outside of the vessel beltline region because the neutron fluence in those areas is less than 1 x 10^{17} n/cm^2 with neutron energies in excess of 1 MeV.

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. This support is designed to carry the weight of the shroud, shroud head, 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

Under normal operating conditions, the Boiling Water Reactor does not use borated water for reactivity control.

This subsection is therefore not applicable.

5.3.3.1.2 Safety Design Bases

Design of the reactor vessel and appurtenances meet the following safety design bases:

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

(2) To minimize the possibility of brittle fracture of the nuclear system process barrier, the following are required:

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

b. 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.

c. 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 basis:

(1) The reactor vessel has been designed for a useful life of 40 years. Operation of the vessel for the period of extended operation was reviewed for license renewal and found to be acceptable.

(2) 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.

(3) Design of the reactor vessel and appurtenances allow for a suitable program of inspection and surveillance.

5.3.3.1.4 Reactor Vessel Design Data

Reactor vessel design data are contained in Tables 5.2-3 and 5.2-4.

5.3.3.1.4.1 Vessel Support

The reactor vessel support assembly consists of a ring girder and the various bolts and shims necessary to position and secure the assembly between the reactor vessel support skirt and the support pedestal. The concrete and steel support pedestal is constructed as an integral part of the building foundation. Steel anchor bolts are set in the concrete with their threads extending above the surface. The anchor bolts extend through the ring girder bottom flange. High strength bolts are used to secure the flange of the reactor vessel support skirt to the top flange of the ring girder. The ring girder is fabricated of ASTM A-36 Structural Steel according to AISC specifications.

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.

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 top head insulation is designed to permit complete submersion in water during shutdown without loss of insulating material, contamination of the water, or adverse effect on the insulation efficiency after draining. All reactor vessel insulation is the stainless steel, reflective type.

The top head insulation framework is designed to seismic category I requirements and is used as a structural support point for reactor vessel head spray and vent piping.
The insulation above the reactor vessel stabilizer brackets is close fitting, free-standing insulation designed to be 100% removable for inservice inspection of the reactor vessel.

The insulation below the stabilizer brackets is suspended from the brackets to allow a minimum of 8 inches annular clearance between the reactor vessel and the insulation for remote inservice inspection of the reactor vessel. The suspended insulation is also equipped with removable access ports.

Reactor vessel bottom head insulation includes horizontal flat panels connected to a cylindrical shell covering the inside of the reactor support skirt. The top row of the cylindrical shell panels are removable to expose the bottom head for inservice inspection.

Quick removable insulation is provided around all reactor vessel nozzles to allow manual or remote automatic examination of nozzle-to-vessel and nozzle-to-piping welds.

5.3.3.1.4.5 Reactor Vessel Nozzles

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

The vessel top head nozzle is provided with a flange with small groove facing. The drain nozzle is of the full penetration weld design. The recirculation inlet nozzles (located as shown in Figure 5.3-1), feedwater inlet nozzles, and core spray inlet nozzles all have thermal sleeves.

Nozzles connecting to stainless steel piping have safe ends, made of stainless steel. These safe ends are welded to the nozzles after the pressure vessel has been heat treated to avoid furnace sensitization of the stainless steel safe ends. 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.

5.3.3.1.4.6 Materials and Inspections

The reactor vessels were designed and fabricated in accordance with the appropriate ASME Boiler and Pressure Vessel Code as defined in Subsection 5.2.1. Table 5.2-4 defines the materials and specifications. Subsection 5.2.4 defines the compliance with reactor vessel Inservice Inspection program requirements.

5.3.3.1.4.7 Reactor Vessel Schematic

The reactor vessel schematic is illustrated in Figure 5.3-1.

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 I and SA508 Class 2. Special requirements for the low alloy steel plate and forgings are discussed in Subsection 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 shells and vessel heads 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 per ASME Section III and IX requirements. Weld test samples were required for each procedure for major vessel full penetration welds.

Submerged arc, gas tungsten arc and gas metal arc welding processes were employed. Electroslag welding was not permitted. Preheat and interpass temperatures employed for welding of low alloy steel met or exceeded the requirements of ASME Section III, subsection NA. Post weld heat treatment of 1100°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 periods up to 30 years and their service history is excellent. The vessel fabricator, CBI Nuclear Co., has had extensive experience with GE Co. reactor vessels and has been the primary supplier for GE 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 GE Co. reactor vessels dates back to 1966.

5.3.3.4 Inspection Requirements

All plates, forgings, and boltings were 100% 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 met the acceptance requirements specified by ASME Section III. In addition, the pressure retaining welds were ultrasonically examined using acceptance standards which were equivalent or more restrictive than required by ASME Section XI.

5.3.3.5 Shipment and Installation

The pressure vessel is shop fabricated in shell ring sections which are shipped to the site in a suitably protected condition. The shell rings are then field erected and finally assembled in place at the site. The completed reactor vessel is given a thorough cleaning and examination following the hydro test. The vessel is tightly sealed for storage to prevent entry of dirt or moisture. Preparations for shipment and storage were in accordance with detailed written procedures. Suitable measures are taken to assure that vessel integrity is maintained; for example, access controls are applied to personnel entering the vessel, weather protection is provided, and periodic inspections 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:

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

(2) If the coolant temperature difference between the dome (inferred from Psat) and the bottom head drain exceeds 145°F, neither reactor power level nor recirculation pump flow shall be increased.

(3) 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 1) assures that the vessel closure, closure studs, vessel support skirt, and control rod drive housing and stub tube stresses and usage remain within acceptable limits. The limit regarding a vessel temperature limit on recirculation pump operation and power level increase restriction (Item 2) augments the Item 1 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 recirculation pump operation or natural circulation (cold coolant can accumulate as a result of control drive in leakage and/or low recirculation flow rate during startup or hot standby). The Item 3 limit further restricts operation of the recirculation 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, ensure 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 discussed in Subsection 5.2.2.

5.3.3.7 Inservice Surveillance

Inservice inspection of the reactor pressure vessel will be in accordance with the requirements discussed in Subsection 5.2.4. The materials surveillance program will monitor 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. Specimens of actual reactor beltline material will be exposed in the reactor vessel and periodically withdrawn for impact testing. Operating procedures will be modified in accordance with test results to assure adequate brittle fracture control.

Material surveillance programs and inservice inspection programs are in accordance with applicable ASME Code requirements, and provide assurance that brittle fracture control and pressure vessel integrity will be maintained throughout the service lifetime of the reactor pressure vessel.

5.3.4 References

5.3-1 Faynshtein, K., and D. R. Pankratz, "Power Uprate Engineering Report for Susquehanna Steam Electric Station, Units 1 and 2," General Electric Report NEDC-32161P, as revised by PP&L Calculation EC-PUPC-1001, Revision 0, March, 1994.

5.3-2 Carey, R. G., "Susquehanna Steam Electric Station Unit 1 Vessel Surveillance Materials Testing and Fracture Toughness Analysis," General Electric Report GE-NE-523-169-1292, Revision 1, October, 1993. Attached to PP&L letter PLA-3953, R. G. Byram to C. L. Miller, NRC, "Susquehanna Steam Electric Station, Submittal of Reactor Vessel Material Surveillance Test Report per 10CFR50 Appendix H for Unit 1," April 8, 1993.

5.3-3 Contreras, G. W., "Susquehanna Steam Electric Station Unit 2 Vessel Surveillance Materials Testing and Fracture Toughness Analysis," General Electric Report GE-NE-523-107-0893, Revision 1, October, 1993. Attached to PP&L Letter PLA-4126, R. G. Byram to C. L. Miller, NRC, "Susquehanna Steam Electric Station, Submittal of Revision to Reactor Vessel Material Surveillance Test Report per 10CFR50 Appendix H for Unit 2," May 19, 1994.

5.3-4 DuBord, R.M., “Susquehanna Steam Electric Station Unit 1 Fabrication of New Surveillance Capsule with Reconstituted Charpy Specimens” General Electric Nuclear Energy report GE-NE-523-A054-0595, May 1995.

5.3-5 DuBord, R.M., “Susquehanna Steam Electric Station Unit 2 Fabrication of New Surveillance Capsule with Reconstituted Charpy Specimens” General Electric Nuclear Energy report GE-NE-523-A055-0595, May 1995.

5.3-6 Structural Integrity Associates Report No. SIR-00-167, Revision 0, “Revised Pressure-Temperature Curves for Susquehanna Units 1 and 2”, January 2001.

5.3-7 BWRVIP-86: BWR Vessel Internals Project, BWR Integrated Surveillance Program Implementation Plan, February 2002 including future revisions.

5.3-8 BWRVIP-78: BWR Vessel Internals Project, BWR Integrated Surveillance Program Plan, February 2002.

5.3-9 BWRVIP Responses to NRC Staff RAIs dated December 22, 2001 and May 30, 2001.

5.3-10 NRC Safety Evaluation regarding the Integrated Surveillance Program, February 1, 2002.

TABLE 5.3-1a

APPENDIX G MATRIX FOR SUSQUEHANNA SES UNIT 1

Appendix G Para. No.	Topic	Comply Yes/No or N. A.	Alternate Actions or Comments
I, II	Introduction; Definitions	--	
III.A	Compliance With ASME Code, Section NB-2300	Yes	See Section 5.3.1.5.1.2 for discussion.
III.B.1	Location & Orientation of Impact Test Spec	Yes	See III.A, above.
III.B.2	Materials Used to Prepare Test Specimens	No	Compliance except for CVN orientation and CVN upper shelf.
III.B.3	Calibration of Temp. Inst. and Charpy Test Machines	No	Paragraph NB-2360 of the ASME B&PV code, Section IV, was not in existence at the time of purchase of the Susquehanna SES Unit 1 reactor pressure vessel. However, the requirements of the 1971 edition of the ASME B&PV Section III Code, Summer 1971 addenda, were met. For the discussions of the GE interpretations of compliance and NRC acceptance see References 1 and 2. The temperature instruments and Charpy Test Machines calibration data are retained until the next recalibration. This is in accordance with Reg. Guide 1.88 Rev. 2, GE Alternative Position 1.88, and ANSI N45.2.9, 1974. Therefore, the instrument calibration data for Susquehanna SES Unit 1 would not be currently available.
III.B.4	Qualification of Testing Personnel	No	No written procedures were in existence as required by the Regulation; however, the individuals were qualified by on-the-job training and past experience. For the discussion of the GE interpretation of compliance and NRC acceptance see References 1 and 2.
III.B.5	Test Results Recording and Verification	Yes	See References 1 and 2.
III.C.1	Test Conditions	No	See III.A, III.B.2, above.
III.C.2	Materials Used to Prepare Test Specimens for Reactor Vessel Beltline	Yes	Compliance on base metal and weld metal tests. Test weld not made on same heat of base plate, necessarily.
IV.A.1	Acceptance Standard of Materials	--	
IV.A.2.a	Calculated Stress Intensity Factor	Yes	
IV.A.2.b	Requirements for Nozzles, Flanges and Shell Region Near Geometric Discontinuities	No	Plus 60°F was added to the RT_{NDT} for the reactor vessel flanges. For feedwater nozzles, the results of the BWR/6 analysis were adjusted to Susquehanna SES Unit 1 RT_{NDT} conditions.
IV.A.2.c	RPV Metal Temperature Requirement When Core is Critical	Yes	

TABLE 5.3-1a

APPENDIX G MATRIX FOR SUSQUEHANNA SES UNIT 1

Appendix G Para. No.	Topic	Comply Yes/No or N. A.	Alternate Actions or Comments
IV.A.2.d	Minimum Permissible Temp. During Hydro Test	Yes	
IV.A.3	Materials for Piping, Pumps and Valves	No	Main steamline piping is in compliance. See Subsection 5.2.3.3.1 for discussions on pumps and valves.
IV.A.4	Materials for Bolting and Other Fasteners	Yes	Closure studs tested at preload or lowest service temperature minus 60°F to give lowest values of 40 ft-lb and 25 mils at +10°F. This is equivalent to meeting current requirements of 45 ft-lb and 25 mils at +70°F.
IV.B	Minimum Upper Shelf Energy for RPV Beltline	No	No upper shelf tests run. However, recommend acceptance based upon lowest longitudinal CVN's for plates at +10°F of 37 ft-lb (40% shear) for heat C0776-1 (0.12% Cu), 31 ft-lb (30% shear) for heat C2433-1 (0.10% Cu), and 44 dt-lb (50% shear) for heat B5083-1 (0.14% Cu). Lowest CVN's for welds are 51 ft-lb (no % shear records) at +10°F for 0.04% Cu. End-of-life upper shelf values (100% shear) are predicted to be in excess of 50 ft-lb, based upon preceding data and Regulatory Guide 1.99.
IV.C	Requirement for Annealing when RT_{NDT} 200°F	N/A	
V.A	Requirements for Material Surveillance Program	See App. H	
V.B	Conditions for Continued Operation	Yes	See section 5.3.1.5.1.1, 5.3.1.5.1.2., 5.3.1.5.1.3., 5.3.1.5.1.4, 5.3.1.5.1.6, 5.3.1.6 and Table 5.3-lb
V.C	Alternative if V.B Cannot be Satisfied	---	---
V.D	Requirement for RPV Thermal Annealing if V.C Cannot be Met	N/A	
V.E	Reporting Requirement for V.C and V.D	N/A	

References:

1. Letter MFN-414-77, G. G. Sherwood (GE) to Edson G. Case (NRC) dated October 17, 1977.

2. Letter, Robert B. Minogue (NRC) to G. G. Sherwood (GE) dated February 14, 1978.

TABLE 5.3-1b

APPENDIX H MATRIX FOR SUSQUEHANNA SES UNIT 1

APPENDIX H PARA. NO.	TOPIC	COMPLY YES/NO OR N.A.	ALTERNATE ACTIONS OR COMMENTS
I	Introduction	N/A	
II.A	Fluence $<10^{17}$ n/cm² – Surveillance Program Not Required	N/A	
II.B	Standards Requirements (ASTM) for Surveillance	No	Noncompliance with ASTM E185-73 in that the surveillance specimens are not necessarily from the limiting beltline material. Specimens are from representative beltline material, however, and can be used to predict behavior of the limiting material. Heat and heat/lot numbers for surveillance specimens are to be supplied.
II.C.1	Surveillance Specimen Shall be Taken from Locations Alongside the Fracture Test Specimens (Section III.B of Appendix G)	No	Noncompliance in that specimens may not have necessarily been taken from along-side specimens required by Section III of Appendix G and transverse CVNs may not be employed. However, representative materials have been used, and RT_{NDT} shift appears to be independent of specimen orientation.
II.C.2	Locations of Surveillance Capsules in RPV	Yes	Code basis is used for attachment of brackets to vessel cladding. See Section 5.3.1.6.4.
II.C.3.a	Withdrawal Schedule of Capsules, $RT_{NDT}<100°F$	Yes	Starting RT_{NDT} of limiting material is based on alternative action (see Paragraph III.A of Appendix G). One capsule complete. Other capsules are scheduled and tested in accordance with Reference 5.3-7 through 5.3-10..
II.C.3.b	Withdrawal Schedule of Capsules, $RT_{NDT}<200°F$	N/A	
II.C.3.c.	Withdrawal Schedule of Capsules, $RT_{NDT}<200°F$	N/A	
III.A	Fracture Toughness Testing Requirements of Specimens	Yes	See Section 5.3.1.6
III.B	Method of Determining Adjusted Reference Temp. for Base Metal, HAZ and Weld Metal	Yes	See Section 5.3.1.5
IV.A	Reporting Requirements of Test Results	Yes	See Section 5.3.1.6
IV.B	Requirement for Dosimetry Measurement	Yes	See Section 5.3.1.6.2, 5.3.1.6.4
IV.C	Reporting Requirements of Press/Temp. Limits	Yes	See Section 5.3.2

TABLE 5.3-2a

APPENDIX G MATRIX FOR SUSQUEHANNA UNIT 2

Appendix G Para. No.	Topic	Comply Yes/No or N. A.	Alternate Actions or Comments
I, II	Introduction; Definitions	--	
III.A	Compliance With ASME Code, Section NB-2300	Yes	See Subsection 5.3.1.5.1.2 for discussion.
III.B.1	Location & Orientation of Impact Test Spec	Yes	See III.A, above.
III.B.2	Materials Used to Prepare Test Specimens	No	Compliance except for CVN orientation and CVN upper shelf.
III.B.3	Calibration of Temp. Inst. and Charpy Test Machines	No	Paragraph NB-2360 of the ASME B&PV code, Section IV, was not in existence at the time of purchase of the Susquehanna SES Unit 2 reactor pressure vessel. However, the requirements of the 1971 edition of the ASME B&PV Section III Code, Summer 1971 addenda, were met. For the discussions of the GE interpretations of compliance and NRC acceptance see References 1 and 2. The temperature instruments and Charpy Test Machines calibration data are retained until the next recalibration. This is in accordance with Reg. Guide 1.88 Rev. 2, GE Alternative Position 1.88, and ANSI N45.2.9, 1974. Therefore, the instrument calibration data for Susquehanna SES Unit 2 would not be currently available.
III.B.4	Qualification of Testing Personnel	No	No written procedures were in existence as required by the Regulation; however, the individuals were qualified by on-the-job training and past experience. For the discussion of the GE interpretation of compliance and NRC acceptance see References 1 and 2.
III.B.5	Test Results Recording and Certification	Yes	See References 1 and 2.
III.C.1	Test Conditions	No	See III.A, III.B.2, above.
III.C.2	Materials Used to Prepare Test Specimens for Reactor Vessel Beltline	Yes	Compliance on base metal and weld metal tests. Test weld not made on same heat of base plate, necessarily.
IV.A.1	Acceptance Standard of Materials	--	
IV.A.2.a	Calculated Stress Intensity Factor	Yes	
IV.A.2.b	Requirements for Nozzles, Flanges and Shell Region Near Geometric Discontinuities	Yes	Plus 60°F was added to the RT_{NDT} for the reactor vessel flanges. For other nozzles and discontinuities, the results of the BWR/6 analysis were adjusted to Susquehanna SES Unit 2 RT_{NDT} conditions.
IV.A.2.c	RPV Metal Temperature Requirement When Core is Critical	Yes	

TABLE 5.3-2a

APPENDIX G MATRIX FOR SUSQUEHANNA UNIT 2

Appendix G Para. No.	Topic	Comply Yes/No or N. A.	Alternate Actions or Comments
IV.A.2.d	Minimum Permissible Temp. During Hydro Test	Yes	
IV.A.3	Materials for Piping, Pumps and Valves	No	Main steamline piping is in compliance. See Subsection 5.2.3.3.1 for discussions on pumps and valves.
IV.A.4	Materials for Bolting and Other Fasteners	Yes	Current toughness requirements for closure head studs are met at 10°F.
IV.B	Minimum Upper Shelf Energy for RPV Beltline	No	No upper shelf tests run. However, recommend acceptance based upon lowest longitudinal CVN's for plates at +10°F of 45 ft-lb (50% shear) for heat C2421-3 (0.10% Cu), 50 ft-lb (50% shear) for heat C2929-1 (0.13% Cu), and 39 ft-lb (40% shear) for heat C2433-2 (0.10% Cu). Lowest CVN's for welds are 22, 30, 31, 43, 55 ft-lb (no % shear records) at -20°F with 0.06% Cu. The scatter in energy data at -20°F indicates transition behavior and the probability that upper shelf is in excess of 50 ft-lb (for 100% shear). End-of-life upper shelf values (100% shear) are predicted to be in excess of 50 ft-lb, based upon preceding data and Regulatory Guide 1.99.
IV.C	Requirement for Annealing when RT_{NDT} 200°F	N/A	
V.A	Requirements for Material Surveillance Program	See App. H	
V.B	Conditions for Continued Operation	Yes	See section 5.3.1.5.1.1, 5.3.1.5.1.2., 5.3.1.5.1.3., 5.3.1.5.1.4, 5.3.1.5.1.6, 5.3.1.6 and Table 5.3-2b
V.C	Alternative if V.B Cannot be Satisfied	---	---
V.D	Requirement for RPV Thermal Annealing if V.C Cannot be Met	N/A	
V.E	Reporting Requirement for V.C and V.D	N/A	

References:

1. Letter MFN-414-77, G. G. Sherwood (GE) to Edson G. Case (NRC) dated October 17, 1977.
2. Letter, Robert B. Minogue (NRC) to G. G. Sherwood (GE) dated February 14, 1978.

*This table references the 1980 Appendices G and H requirements. The engineering rationale used to meet the 10CFR50 Appendices G and H requirements of 1980 are equally applicable to meeting the 10CFR50 Appendices G and H Requirements of 1983.

TABLE 5.3-2b

APPENDIX H MATRIX FOR SUSQUEHANNA SES UNIT 2

APPENDIX H PARA. NO.	TOPIC	COMPLY YES/NO OR N.A.	ALTERNATE ACTIONS OR COMMENTS
I	Introduction	N/A	
II.A	Fluence <10^{17} n/cm^2 – Surveillance Program Not Required	N/A	
II.B	Standards Requirements (ASTM) For Surveillance	No	Noncompliance with ASTM E185-73 in that the surveillance specimens are not necessarily from the limiting beltline material. Specimens are from representative beltline material, however, and can be used to predict behavior of the limiting material. Heat and heat/lot numbers for surveillance specimens are to be supplied.
II.C.1	Surveillance Specimen Shall be Taken from Locations Alongside the Fracture Test Specimens (Section III.B of Appendix G)	No	Noncompliance in that specimens may not have necessarily been taken from alongside specimens required by Section III of Appendix G and transverse CVN's may not be employed. However, representative materials have been used, and RT_{NDT} shift appears to be independent of specimen orientation.
II.C.2	Locations of Surveillance Capsules in RPV	Yes	Code basis is used for attachment of brackets to vessel cladding. See Subsections 5.3.1.6.4.
II.C.3.a	Withdrawal Schedule of Capsules, RT_{NDT}<100°F	Yes	Starting RT_{NDT} of limiting material is based on alternative action (see Paragraph III.A of Appendix G). One capsule completed. Other capsules are scheduled and tested in accordance with References 5.3-7 through 5.3-10.
II.C.3.b	Withdrawal Schedule of Capsules, RT_{NDT}<200°	N/A	
II.C.3.c	Withdrawal Schedule of Capsules, RT_{NDT}<200°	N/A	
III.A	Fracture Toughness Testing Requirements of Specimens	Yes	See Section 5.3.1.6
III.B	Method of Determining Adjusted Reference Temp. for Base Metal, HAZ and Weld Metal	Yes	See Section 5.3.1.5
IV.A	Reporting Requirements of Test Results	Yes	See Section 5.3.1.6
IV.B	Requirement for Dosimetry Measurement	Yes	See Section 5.3.1.6.2, 5.3.1.6.4
IV.C	Reporting Requirements of Press/Temp. Limits	Yes	See Section 5.3.2

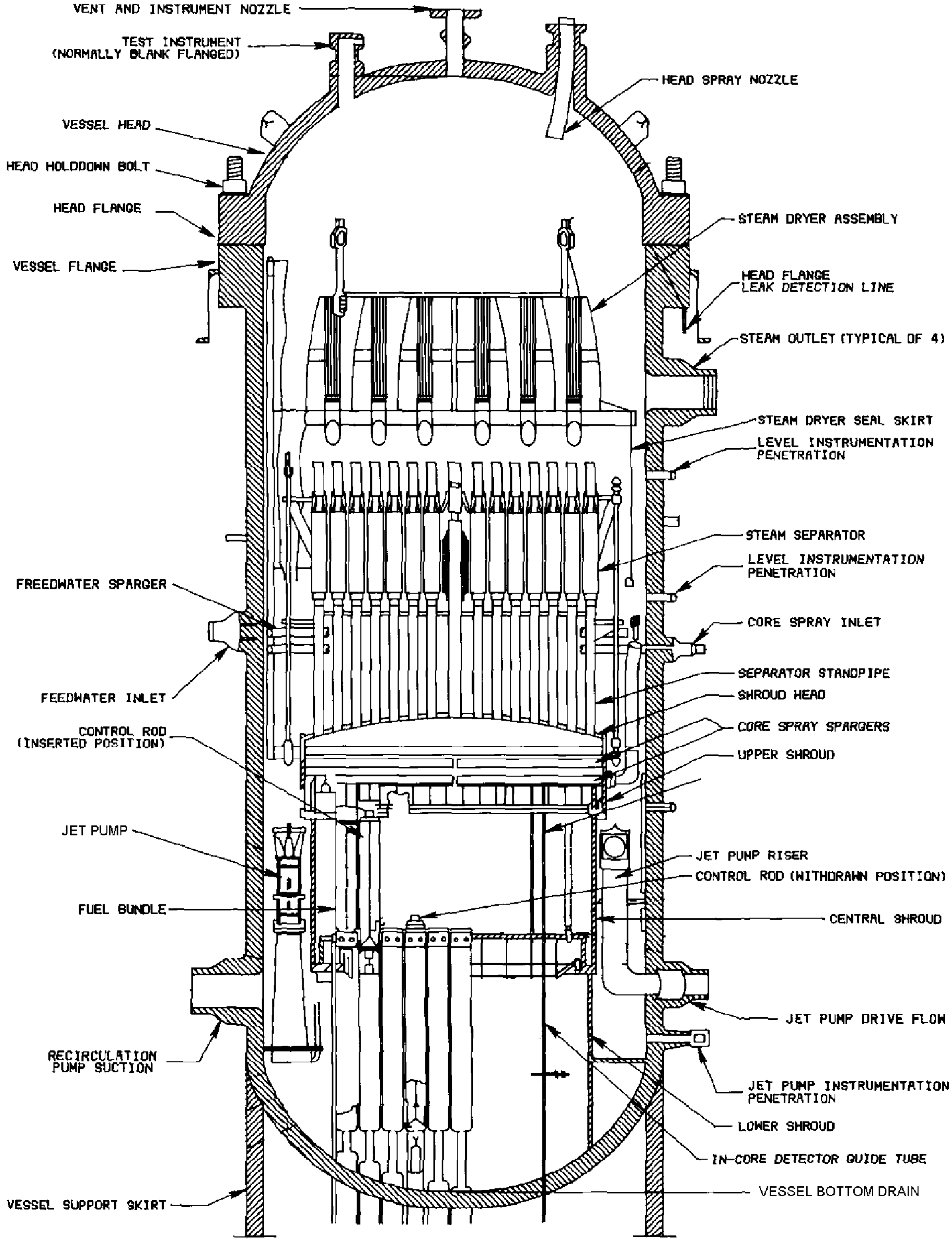
TABLE 5.3-3

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM-WITHDRAWAL SCHEDULE

Specimen Holder	Vessel Location	Lead Factor *	Withdrawal Time (EFPY)
UNIT 1			
131C7717G1	300°	1.20	Spare
131C7717G2	120°	1.20	22
131C7717G3	30°	1.20	6 (Actual Date - Fall 1992)
G3 Reconstituted Specimens	30°	1.20	Spare
UNIT 2			
131C7717G1	300°	1.20	Spare
131C7717G2	120°	1.20	Spare
131C7717G3	30°	1.20	6 (Actual Date - Fall 1992)
G3 Reconstituted Specimens	30°	1.20	Spare

* At 1/4 T.

\# Withdrawal Time is in accordance with Reference 5.3-7 through 5.3-10..



REACTOR VESSEL
COMPOSITE DRAWING

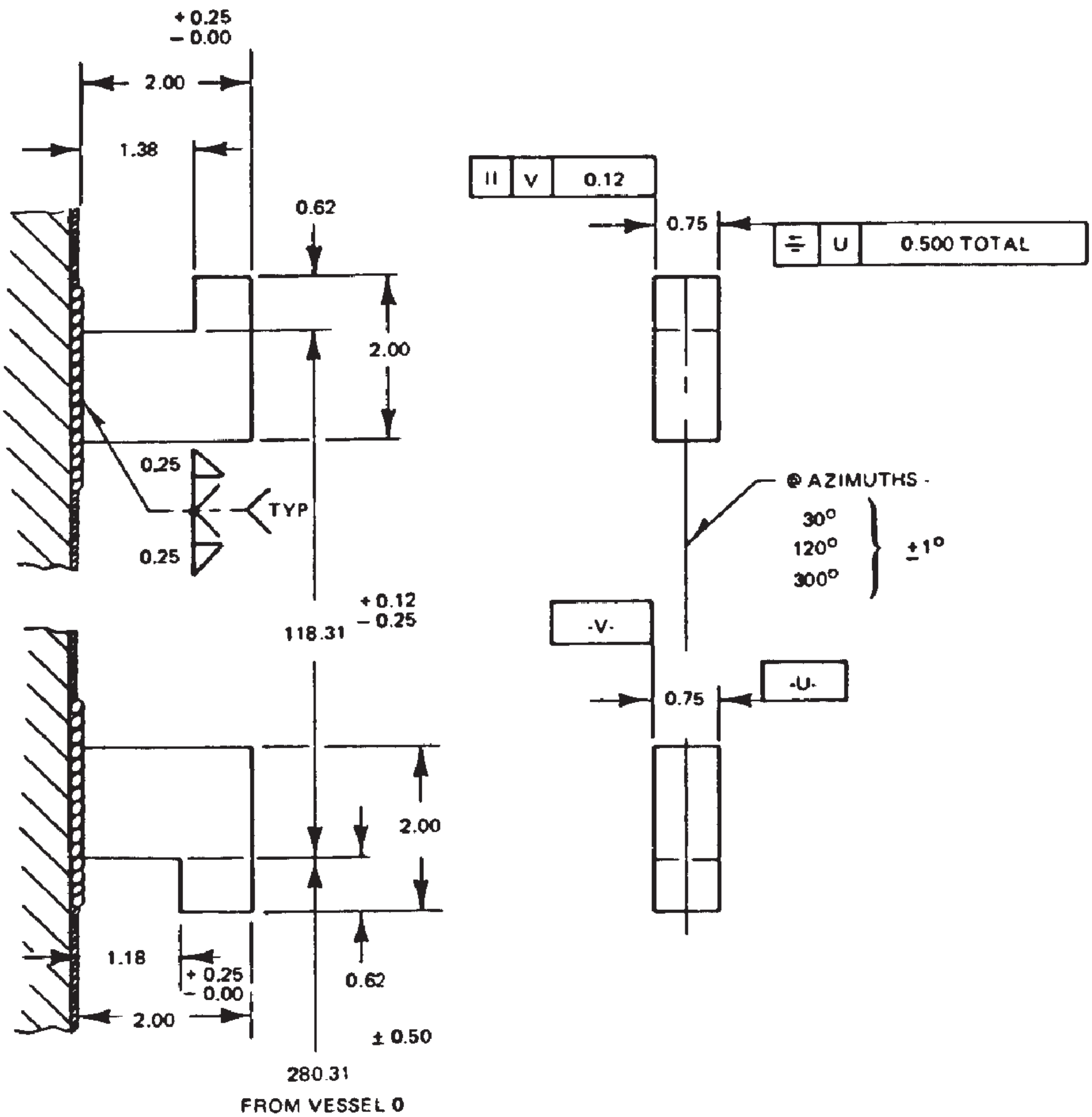
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

REACTOR VESSEL

FIGURE 5.3-1, Rev.54

AutoCAD: Figure Fsar 5_3_1.dwg



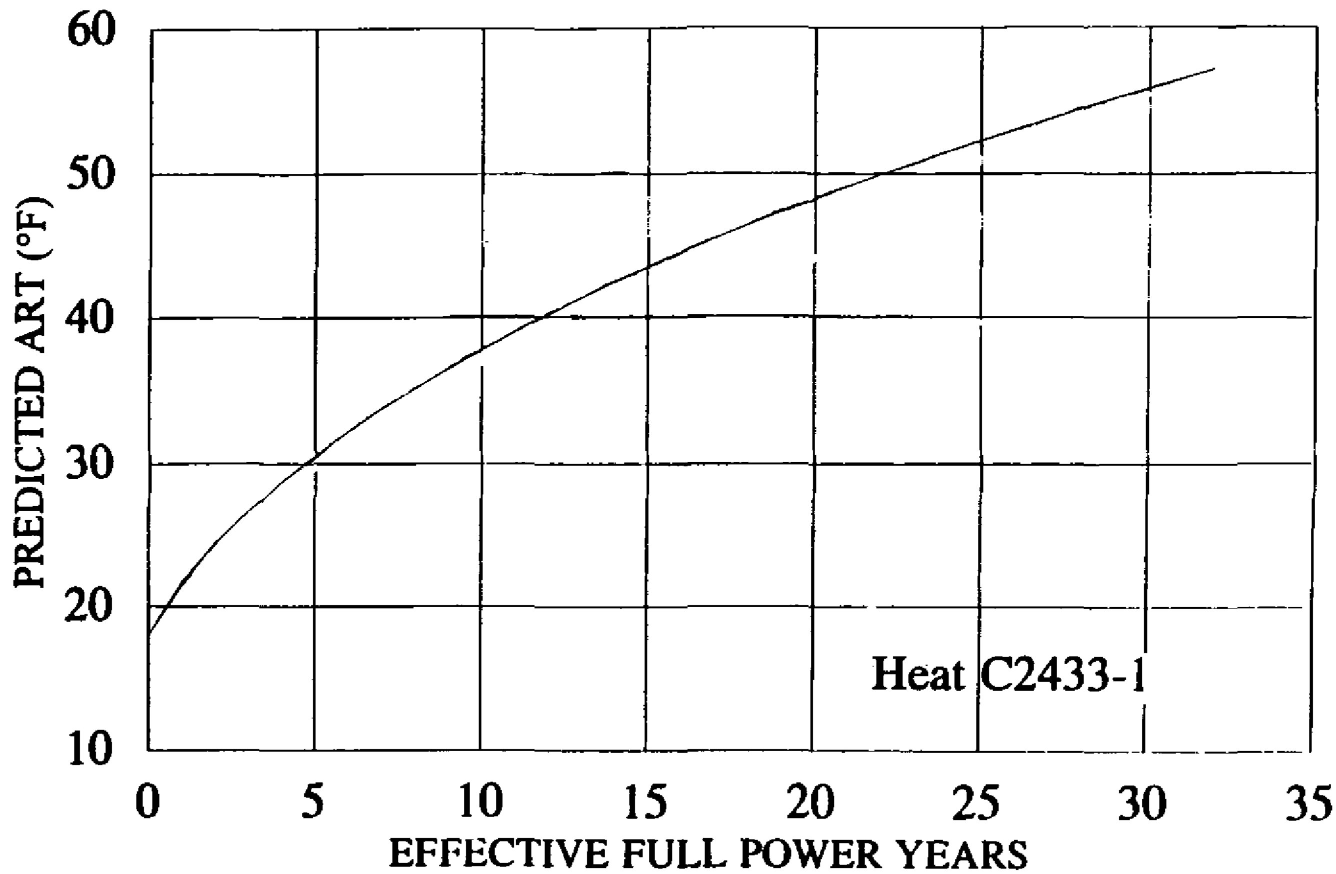
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

BRACKET FOR HOLDING
SURVEILLANCE CAPSULE

FIGURE 5.3-3, Rev.47

AutoCAD: Figure Fsar 5_3_3.dwg



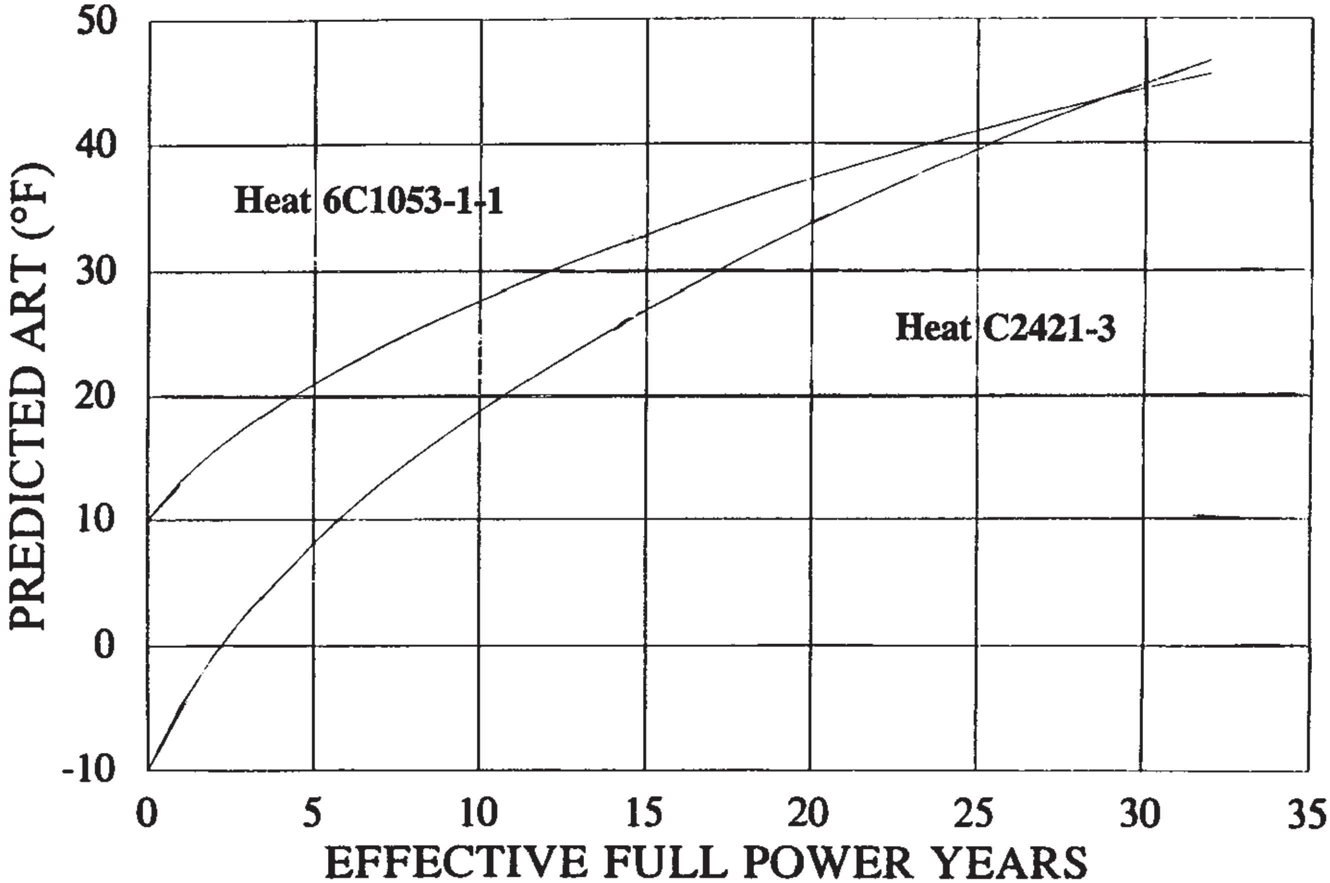
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

UNIT 1
PREDICTED ADJUSTED REFERENCE
TEMPERATURE VS. EFFECTIVE FULL
POWER YEARS OF OPERATION

FIGURE 5.3-4C, Rev.50

AutoCAD: Figure Fsar 5_3_4C.dwg



FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

UNIT 2
PREDICTED ADJUSTED REFERENCE
TEMPERATURE VS. EFFECTIVE FULL
POWER YEARS OF OPERATION

FIGURE 5.3-4D, Rev.50

AutoCAD: Figure Fsar 5_3_4D.dwg