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OAK RIDGE NATIONAL LABORATORY

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NUCLEAR DIVISION

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August 29, 1980

Mr. Stefan Pawlicki, Chief
Materials Engineering Branch
Division of Engineering
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Steve:

I have enclosed three copies of the revised draft SER and attachments for Task 1 under our contract. The list of materials for reactor coolant pressure boundary and safety systems, as well as a list of deviations from the SRP, will follow next week.

Sincerely,

Randy K. Nanstad

Randy K. Nanstad, Program Manager
NRC Licensing Assistance, Engineering

RKN/bdk

Enclosures

cc: R. J. Beaver
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4.5.1 Control Rod Drive Structural Materials

General Design Criterion 26, Appendix A, 10 CFR Part 50, requires, in part, that one of the reactivity control systems shall use control rods and shall be capable of reliably controlling reactivity changes to assure that fuel design limits are not exceeded under conditions of normal operation, including anticipated operational occurrences. Materials considerations in the design of the control rod drive mechanism are reviewed to assure compliance with General Design Criteria 26.

The mechanical properties of structural materials selected for the control rod system components exposed to the reactor coolant satisfy Appendix I of Section III of the ASME Code, or Part A of Section II of the Code.

The controls imposed upon the austenitic stainless steel of the system satisfy the intent of the recommendations of our position on Regulatory Guide 1.31, "Control of Stainless Steel Welding," and Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel," except for the following:

1. FSAR Section 4.5.1.1 contains austenitic stainless steel items with carbon contents in excess of 0.03 wt %. The applicant must identify those materials containing greater than 0.03 wt % C which are welded and submit justification, such as test results of base metal and heat-affected zones, to assure that susceptibility to intergranular corrosion is not significant as described in Regulatory Guide 1.44;

2. Additional information must be submitted relative to in-service inspection procedures of nitrided hard-surfaced components to assure fabrication and heat-treatment practices performed in accordance with these recommendations provide added assurance that stress corrosion cracking will not occur during the design life of the component; and

3. FSAR Section 4.5.1.1 identifies ASTM Specifications A269 and A249, and ASME Specification SA 312 for the austenitic stainless steel tubular items. It is not clear that the items indicated below represent welded tubular items:

a. Cylinder of the cylinder, tube and flange assembly;

b. Outer tube of the cylinder, tube and flange assembly;

c. Piston tube of the piston tube assembly;

d. Index tube of the drive assembly;

e. Collet piston of the collet assembly; and

f. Guide cap of the collet assembly.

The applicant must identify whether or not the above items are welded tubular parts. If any are welded tubular parts, the applicant must provide information that they were nondestructively examined in accordance with the nondestructive test requirements for welded tubular products as required by NC 2500 of the ASME Code, Section III. If they represent wrought seamless tubular products, the applicant must provide information to assure that the material was examined nondestructively similar to the requirements of NB 2500 of the ASME Code, Section III to identify adverse internal defects.

The compatibility of all materials used in the control rod system in contact with the reactor coolant satisfies the criteria of Articles NB-2160 and NB-3120 of Section III of the ASME Code. Both martensitic and precipitation-hardening stainless steels have been given tempering or aging treatments in accordance with staff positions.

Conformance with the codes, standards, and Regulatory Guides indicated above, and conformance with the staff positions on the allowable maximum yield strength of cold-worked austenitic stainless steel, and the tempering or aging temperatures of martensitic and precipitation-hardened stainless steels constitute an acceptable basis for meeting in part the requirements of General Design Criterion 26.

4.5.2 Reactor Internals Materials

General Design Criteria 1 and 14, Appendix A, 10 CFR Part 50, require that structures, systems, and components important to safety shall be designed, fabricated, and tested to quality standards commensurate with the importance of the safety functions to be performed.

Materials considerations in the design of major components of the reactor internals, which consist of the core support structures and the internal structures, are reviewed to assure compliance with the General Design Criteria.

The materials for construction of components of the core support structure have been identified by specification and found to be in conformance with the requirements of Section III of the ASME Code.

The materials for construction of components of the reactor internals other than the core support structure have been identified by specification to be adequate for the service intended with the following exceptions:

1. FSAR Section 4.5.2.1 identifies ASTM specification ASTM A370, Grades E38 and E55 for the pin and insert of the inlet-mixer in the jet pump assemblies. Specification A370 relates to testing of materials, not to material specifications. Clarification is required from the applicant;

2. The applicant must identify by specification and provide further justification for the use of nitrided type 304 stainless steel in jet pump beam bolts; and

3. FSAR Section 4.5.2.3 states, "For core support structures, wrought seamless tubular products were supplied in accordance with applicable ASME material specifications. These specifications require examination of the tubular product by radiographic and/or ultrasonic methods according to paragraph NG-2550 of the ASME Code Section III." The ASME material specifications do not automatically call out examination to the requirements of NG-2550. The applicant must provide specific information as to exactly how the nondestructive testing requirements of NG-2550 were specified in the procurement documents.

Regulatory Guide 1.71, "Welder Qualification for Areas of Limited Accessibility," requires that both radiography and sectioning be performed for mockup welding. FSAR Section 4.5.2.4 indicates that radiography or sectioning was used and the applicant must clarify its position or provide further justification for the assurance of the acceptability of welds fabricated in restricted areas.

With resolution of the uncertainties described above, the following conclusions can be drawn:

1. The materials for reactor internals exposed to the reactor coolant have been identified and all materials are compatible with the expected environment, as proven by extensive testing and satisfactory

performance. General corrosion on all materials is expected to be negligible.

2. The controls imposed on reactor coolant chemistry provide reasonable assurance that the reactor internals will be adequately protected during operation from conditions which could lead to stress corrosion of the materials and loss of component structural integrity.

3. The controls imposed upon components constructed of austenitic stainless steels, as used in the reactor internals, satisfy the recommendations of our position on Regulatory Guide 1.31, "Control of Stainless Steel Welding," Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel," and Regulatory Guide 1.71, "Welder Qualification for Areas of Limited Accessibility."

Material selection, fabrication practices, examination procedures, and protection procedures performed in accordance with these recommendations provide reasonable assurance that the austenitic stainless steel used for reactor internals will be in a metallurgical condition which precludes susceptibility to stress-corrosion cracking during service. The use of materials proven to be satisfactory by actual service experience and conformance with the recommendations of these regulatory guides constitutes an acceptable basis for meeting the applicable requirements of General Design Criteria 1 and 14.

5.2.3 Reactor Coolant Pressure Boundary Materials

General Design Criteria 1, 13 and 14, Appendix A, 10 CFR Part 50, require, in part, that the reactor coolant pressure boundary be designed, fabricated, erected, and tested to quality standards commensurate with the safety function to be performed and that instrumentation be provided to monitor the variables that can affect the integrity of the reactor coolant pressure boundary. General Design Criterion 31 requires, in part, that the reactor coolant pressure boundary be designed with sufficient margin to ensure that, when stressed under operating, maintenance, testing, and postulated accident conditions, the boundary behaves in a nonbrittle manner and the probability of rapidly propagating failure is minimized. Materials considerations in the design of the reactor coolant pressure boundary, other than the reactor pressure vessel reviewed in Section 5.3.1, are reviewed to assure compliance with the General Design Criteria. The fracture toughness properties of ferritic materials used for pressure-retaining components of the reactor coolant pressure boundary are reviewed in Section 5.3.1, Reactor Vessel Materials, for conformance with General Design Criterion 31.

The materials of construction of the reactor coolant pressure boundary (RCPB) have been identified by specification and found to be in conformance with the requirements of Section III of the 1971 ASME Code, Summer 1971 Addendum with the following exceptions for Table 5.2.4:

1. Submerged-arc welding materials for major reactor vessel welds are not specified in the table, while this process is referenced in other parts of the FSAR;
2. SA 420 Grade WPL1 is not an allowable grade per Section II, Summer 1971 Addenda, or any Addenda since then;

The following specified grades are not identifiable and are believed to be typographical errors:

3. SFA 5.1 Grade E-705;
4. SA 240 Grade F316; and
5. SA 182, Grade F216.

The following are general comments about materials of construction for the RCPB:

6. SFA 5.5 E7010A1 electrodes for shielded metal-arc welding are not low hydrogen and should not be used for critical applications; and
7. SA 540, Grade B24 is identified in FSAR Section 5.3.1.5.1.2 for reactor vessel closure bolting. That specification is not given in Table 5.2.4.

The applicant must make necessary changes to the FSAR or provide justifications as required before we can complete our review of materials selection.

The materials of construction of the RCPB exposed to the reactor coolant have been identified and all of the materials are compatible with the expected environment, as proven by extensive testing and satisfactory performance. General corrosion of all materials, except unclad carbon and low alloy steel, is negligible. For these materials, conservative corrosion allowances have been provided for all exposed surfaces in accordance with the requirements of the ASME Code, Section III.

The materials of construction for the RCPB are compatible with the thermal insulation used in these areas and are in conformance with the recommendations of Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steels."

The controls imposed on austenitic stainless steels are not in conformance with all of the requirements of Regulatory Guide 1.44, "Control of Sensitized Stainless Steel." These controls are discussed in FSAR Sections 5.2.3.4.1 and 3.13 (Regulatory Guide 1.44), and in PP&L Letter PLA-291 dated September 25, 1978 to Olan D. Parr, "A Reaction to Cracking of Austenitic Stainless Steel Piping in Boiling Water Reactors (including Susquehanna SES Design Modifications)." All austenitic stainless steels were purchased in the solution heat treated condition, with yield strengths less than 90 ksi, and low carbon grades were used when possible. Welding heat inputs (preheat, interpass and energy input) were limited to minimize stresses and sensitization. For critical areas special fabrication techniques were used, including weld overlaying areas adjacent to a weld seam with corrosion resistant materials and solution heat treating before making the final pressure boundary weld. However, there is no indication that nonsensitization of any of these welded materials has been verified using an approved procedure, as outlined in Regulatory Guide 1.44.

The controls imposed on reactor coolant chemistry are not in complete conformance with the recommendations of Regulatory Guide 1.56, "Maintenance of Water Purity in BWR's." Appropriate limits for conductance, chloride content and pH of reactor water have been established. An appropriate condensate treatment system and reactor water cleanup system has been added to control oxygen levels during other than normal operation periods (PP&L Letter, PLA-291, dated September 25, 1978, to Olan D. Parr). The instrumentation and sampling provisions for monitoring reactor coolant water chemistry provide adequate measurement capability for detecting significant changes on a timely basis. However, it is stated in FSAR Section 5.2.3.2.2(2)3 that 50% of the theoretical ion exchange capacity will be maintained in the demineralizers, while Regulatory Guide 1.56 requires that 60% of the initial capacity should be maintained. Also, the GE Report, NEDO-10899, which is referenced in the FSAR and which establishes General Electric's position on water purity, has not been officially submitted to, or accepted by, the NRC.

The issues discussed in the above two paragraphs must be satisfactorily addressed by the applicant before we can complete our review in this area. Conformance with the recommendations of Regulatory Guides 1.44 and 1.56 will provide reasonable assurance that the reactor coolant pressure boundary components will be adequately protected during operation from conditions that could initially lead to stress-corrosion cracking of the materials and loss of structural integrity of a component.

The requirements of Regulatory Guide 1.34 are not applicable here, since no electroslag welding was used for the RCPB materials.

The controls imposed on welding preheat are in conformance with the recommendations of Regulatory Guide 1.50, "Control of Preheat Temperature for Welding Low Alloy Steels," with two exceptions:

1. The guide calls for maintaining preheat until postweld heat-treatment, while the FSAR Section 5.2.3.3.2.1 indicates preheat was sometimes "held for an extended period of time at preheat temperature to assure removal of hydrogen." This statement does not provide justification for the extended preheat, nor does it provide reasonable assurance that cracking will not occur during fabrication; and

2. There is no assurance in the FSAR that welding procedures were qualified at the minimum preheat temperature.

The applicant must provide additional information concerning time at preheat and qualification of welding procedures at minimum preheat temperatures. Satisfactory resolution of the above issues will provide reasonable assurance that cracking of components made from low alloy steels will not occur during fabrication and minimize the possibility of subsequent cracking due to residual stresses being retained in the weldment.

The controls imposed upon components constructed of austenitic stainless steel used in the RCPB conform to the recommendation of Regulatory Guide 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal." Appropriate limits have been established for ferrite content to provide reasonable assurance that hot-cracking will not occur during welding and that degradation of properties will not occur as a result of subsequent high temperature exposures.

Regulatory Guide 1.71, "Welder Qualification for Areas of Limited Accessibility," requires that both radiography and sectioning be performed for mock-up welding. FSAR Section 5.2.3.3.2.3 indicates that radiography or sectioning was used. The applicant must clarify its position or provide further justification for the assurance of the acceptability of welds fabricated in restricted areas.

5.2.4 Reactor Coolant Pressure Boundary Inservice Inspection and Testing

General Design Criterion 32, Appendix A, 10 CFR Part 50, requires, in part, that components which are part of the reactor coolant pressure boundary be designed to permit periodic inspection and testing of important areas and features to assess their structural and leak tight integrity.

Design considerations relating to the Inservice Inspection (ISI) Program for ASME Code Class 1 components are reviewed to assure compliance with General Design Criterion 32.

To ensure that no deleterious defects develop during service, selected welds and weld heat-affected zones will be inspected periodically at the Susquehanna SES. The design of the ASME Code Class 1 and 2 components of the reactor coolant pressure boundary in the Susquehanna SES incorporates provisions for access for inservice inspection in accordance with Section XI of the ASME code. Methods have been developed to facilitate the remote inspection of those areas of the reactor vessel not readily accessible to inspection personnel.

Section 50.55a(g) of 10 CFR Part 50 defines the detailed requirements for the preservice and inservice inspection programs for light water cooled nuclear power facility components. Based upon a construction permit date of November 1973, this section of the Code of Federal Regulations requires that a preservice inspection program be developed and implemented using at least the Edition and Addenda of

Section XI of the ASME Code in effect six months prior to the date of issuance of the construction permit. Also, the initial inservice inspection program must comply with the requirements of the latest Edition and Addenda of Section XI of the ASME code in effect twelve months prior to the data of issuance of the operating license, subject to the limitations and modifications listed in section 50.55a(b) of 10 CFR Part 50.

The applicant is currently developing the preservice inspection program for Susquehanna Steam Electric Station - Unit 1. Deviations are required to be evaluated and acceptable according to the requirements of Section XI of the ASME code and in accordance with Section 50.55a of 10 CFR Part 50 before issuance of an operating license. Evaluation of the deviations from the ASME Section XI requirements will be presented in a supplement to the Safety Evaluation Report. The inservice inspection program will be evaluated after the applicable ASME Code Edition and Addenda have been determined and before the initial inservice inspections are performed.

The conduct of periodic inspections and hydrostatic testing of pressure retaining components of the reactor coolant pressure boundary in accordance with the requirements of Section XI of the ASME Boiler and Pressure Vessel Code and 10 CFR Part 50 will provide reasonable assurance that evidence of structural degradation or loss of leak tight integrity occurring during service will be detected in time to permit corrective action before the safety functions of a component are compromised. Compliance with the inservice inspections required by the Code and 10 CFR Part 50 constitutes an acceptable basis for satisfying the inspection requirements of General Design Criterion 32.

5.3.1 Reactor Vessel Materials

General Design Criterion 31, "Fracture Prevention of Reactor Coolant Pressure Boundary," Appendix A, 10 CFR Part 50, requires, in part, that the reactor coolant pressure boundary be designed with sufficient margin to ensure that, when stressed under operating, maintenance, testing, and postulated accident conditions, the boundary behaves in a nonbrittle manner and the probability of rapidly propagating fracture is minimized. General Design Criterion 32, "Inspection of Reactor Coolant Pressure Boundary," Appendix A, 10 CFR Part 50, requires, in part, that the reactor coolant pressure boundary be designed to permit an appropriate material surveillance program for the reactor pressure boundary.

We have reviewed the materials selection, toughness requirements, and extent of materials testing conducted by the applicant to provide assurance that the ferritic materials used for pressure retaining components of the reactor coolant pressure boundary possess adequate toughness under operating, maintenance, testing, and anticipated transient conditions. The ferritic materials for the reactor vessels were specified to meet the toughness requirements of the 1968 Edition including Addenda through Summer 1970 of the ASME Boiler and Pressure Vessel Code, Section III.

In the area of materials selection, we have reviewed Table 5.2-4 of the Susquehanna FSAR in which the reactor coolant pressure boundary materials are listed. Most of the materials identified for construction of the reactor vessel and its appurtenances are in conformance with Section III of the ASME Code, but there are a few exceptions (most notable is the lack of a submerged arc welding specification for the major vessel welds), and we have requested the additional information from the applicant necessary to complete review in this area.

The guidelines specified for the fracture toughness requirements for the ferritic materials of the reactor coolant pressure boundary are defined in Appendix G, "Fracture Toughness Requirements," of 10 CFR Part 50. The construction permit for both Units 1 and 2 of the Susquehanna Power Station was issued on November 2, 1973. Pursuant to paragraph 50.55a(c)(2) of 10 CFR Part 50, the applicable ASME Boiler and Pressure Vessel Code for the reactor vessels for b[illegible] Susquehanna Power Station Unit 1 and Unit 2 is the 1971 Edition, inc[illegible]ding the Summer 1971 Addendum. Based on information supplied by the applicant concerning anticipated deviations from the codes and standards rules of the above mentioned paragraph and on certain additional commitments relative to the reactor pressure vessels, the AEC granted on June 20, 1974 in accordance with paragraph 50.55a(a)(2)(ii), approval for relief from the rule for the pressure vessels and certain other components and acceptance of the ASME Section III Code for 1968 including Addenda through Summer 1970 for the reactor pressure vessels. In addition, the requirements of Sections NB-2152 and NB-2400 of the 1971 ASME Code Section III including the Summer 1971 Addendum must be met for work at the Susquehanna Power Station site as well as performing an ASME site audit as required by the 1971 Edition of the ASME Code Section III.

Since the applicable ASME Code Edition and Addendum described above, as defined in 10 CFR 50.55a, preceeded the publication of Appendices G and H, 10 CFR Part 50, some of the fracture toughness tests for the ferritic materials in the primary coolant pressure boundary were not conducted to demonstrate explicit compliance with the current requirements of Appendices G and H. Alternate methods for compliance have been proposed by the Pennsylvania Power and Light Company (PP&L).

We have reviewed the information submitted to date in the applicant's FSAR in conjunction with their proposed alternate methods and have evaluated their degree of compliance with the fracture toughness requirements set forth in Appendix G, 10 CFR Part 50. The results of our evaluation indicate that the applicant can meet the requirements of the appendix if sufficient additional justification of their methods and materials properties data are submitted to support the levels of fracture toughness described in the Susquehanna FSAR. We also found that, while not explicitly meeting the current requirements regarding testing equipment, personnel, calibration, and record keeping, that the requirements which were met were adequate. All of these areas are discussed below.

Section III of Appendix G requires that ferritic materials in the reactor coolant pressure boundary be shown to comply with fracture toughness requirements by testing Charpy V-notch specimens and, when required, by testing drop-weight specimens. The Code in effect at the time of the construction of the vessel, however, required only Charpy V-notch or drop-weight testing. Hence, the applicant has stated that all of the above information is not available and to date has submitted only limited test results on weld and base metal and none on heat-affected zones (HAZ). Before review of this area can be completed, we require that the results of those fracture toughness tests (Charpy V-notch, drop-weight, and tensile) performed on base metal, heat-affected-zone material, weld metal, and bolting material required by Appendix G be submitted so that an evaluation of the degree of compliance with this regulation can be made. These results shall include details of location, orientation, and actual parent material for all specimens.

Compliance with Section IV of Appendix G requires that for the vessel, exclusive of bolting, a reference temperature, RT_{NDT}, be defined and used as a basis for providing adequate margins of safety for reactor operations. The value of RT_{NDT}, as defined by Appendix G with references to the ASME Code, is the higher of either the nil-ductility temperature (NDT) as determined by the drop-weight test, or a temperature 60°F less than the temperature at which 50 ft-lb energy and 35 mils lateral expansion is achieved by Charpy impact tests. The Charpy impact tests are to be conducted using specimens oriented in the transverse direction.

Material tests for Susquehanna Power Station Units 1 and 2 were performed to meet the requirements of Section III of the ASME Code, 1968 Edition, including Addenda through Summer 1970. In accordance with these code requirements, the NDT for the vessel materials was determined by drop-weight test. However, there are not sufficient data to define RT_{NDT} exactly because these earlier codes did not require enough Charpy impact tests to define the temperature at which 50 ft-lbs energy would be achieved. Instead, three impact tests were required to be conducted at a single temperature equal to 60°F below the lowest service temperature. The test temperature chosen in this manner typically was 10°F. Further, the tests were conducted using samples whose orientations were longitudinal rather than transverse.

To define RT_{NDT} and demonstrate compliance with the requirements of Appendix G to 10 CFR Part 50, PP&L has used the existing impact test data in conjunction with various data correlations to define the temperature at which 50 ft-lb is achieved for specimens of transverse orientation. The temperature at which 50 ft-lb would be achieved was first estimated from the available longitudinal data by using a temperature-impact energy correlation of 2°F per ft-lb to extrapolate the energy level obtained at 10°F to the temperature corresponding to the 50 ft-lb energy level. This temperature was then increased by 30°F to account for the transverse specimen orientation now required by the ASME Code. The 2°F per ft-lb correlation was devised recently by the applicant using data contained in WRC Bulletin 217, "Properties of Heavy

Section Nuclear Reactor Steels" and additional data for similar materials from other reactor facilities. The 30°F temperature increase used to adjust from longitudinal to transverse specimen orientation is a commonly used empirical correlation that previously has been obtained from comparisons of longitudinal and transverse Charpy data using similar reactor vessel steels.

We have reviewed Units 1 and 2 data, WRC Bulletin 217 test data, and similar test data reported for several heats of steel in Electric Power Research Institute Report, EPRI-NP-121, Volume II, Part One, April 1976. Our review of these data indicate that the longitudinal Charpy V-notch (CVN) 50 ft-lbs transition temperature, which is adjusted by adding 2°F per ft-lb to the test temperature, is conservative only if it is used to extrapolate from low energy levels to higher energy levels. Extrapolation from high to lower energy levels may lead to nonconservative estimates of the temperature at which 50 ft-lb impact energy would be achieved. Consequently, the 2°F per ft-lb correlation cannot be used for all data to define RT_{NDT} as required by Appendix G.

Based on our evaluation of the test data, we conclude that PP&L has not provided a conservative estimate of all the existing data to determine the temperature at which 50 ft-lb energy is achieved. Consequently, before we can complete our evaluation, PP&L, in addition to supplying the necessary fracture toughness test results specified above, must (1) identify those materials where the 2°F per ft-lb correlation was used to extrapolate from high to lower energy levels, and (2) provide conservative estimates of the temperature at which 50 ft-lb will be achieved for the materials identified in (1).

Section IV.B of Appendix G to 10 CFR Part 50 requires a minimum of 75 ft-lbs upper shelf Charpy V-notch impact energy for the beltline material. The applicant reported in the Susquehanna FSAR, Revision No. 13, that neither base metal nor the material in the weld seams in the reactor vessel beltline were tested over a temperature range that would allow the upper shelf to be adequately defined.

In accordance with 10 CFR 50.55a, the tests for the beltline material were conducted to Code editions that preceeded Appendix G to 10 CFR Part 50. These earlier Code editions did not require that the upper shelf energy be established but that the tests be run at a single temperature equal to 60°F below the lowest service temperature. The applicant has recommended acceptance of the material based upon the lowest longitudinal Charpy V-notch test energies and corresponding percent shear obtained in that testing. No justification for this recommendation has yet been submitted and is required before review in this area can be completed.

To provide an acceptable basis for meeting the 75 ft-lbs upper shelf energy requirement of Section VI.B, the applicant can provide an analysis showing that the proposed use of Charpy energy and percent shear on the specimen fracture face to extrapolate existing test data to the 75 ft-lbs level is conservative. If the analysis is based on data from the literature, it must be shown that the data obtained for the Susquehanna pressure vessels are consistent with other similar reactor vessel steels and weldments, and the analysis should demonstrate the capability to discriminate between materials with high and low upper shelf energy. As one alternative, the applicant can attempt to locate archive material for all reactor vessel steels and weldments in the beltline of the vessel and perform additional Charpy tests to directly measure the upper shelf energy.

Information describing the magnetic particle, dye penetrant, and visual inspections required by the ASME Code, paragraph NB 2583, and specified in paragraph C.2 of Regulatory Guide 1.65, "Materials and Inspections for Reactor Vessel Closure Studs," but not yet included in the Applicants FSAR, shall be submitted for evaluation.

On January 13, 1980, PP&L notified the NRC of the unauthorized attachment of a pipe hanger by welding to the SSES Unit 1 reactor pressure vessel. Since this represents significant deviation from approved practices for a safety related component, we require the applicant to correct this deficiency. Technical details of the unauthorized attachment as well as resolution of this deficiency, including justification of the corrective action, shall be submitted for our review.

Paragraphs III.B.3 through III.B.5 of Appendix G, 10 CFR Part 50 specify requirements related to calibration of test instruments, qualification of test personnel, and records and certifications. The purpose of these requirements is to ensure that data of sufficient quantity and quality are developed to fulfill the goals of Appendix G. Given that the actual ordering of the materials, fabrication of the Susquehanna reactor vessels and development of the testing program for the materials samples occurred well before Appendix G became effective, there is only limited value in a step by step comparison of the procedures of Appendix G cited above and the actual procedures employed. We have instead reviewed the procedures actually employed on their merits. These procedures are contained in the 1968 edition of the Code, paragraphs N-331 and N-332, and by reference in ASTM E 208 and ASTM A 370. These are long standing procedures that have been utilized successfully over many years for nuclear and non-nuclear components.

Based on our review we have concluded that the practices followed by the applicant in relation to each of the matters cited above would provide data of a quantity and quality sufficient to accurately characterize the fracture toughness properties of the materials being tested. The goals of Appendix G to 10 CFR Part 50 would, therefore, be fulfilled with respect to these matters.

The toughness properties of the reactor vessel beltline materials shall be monitored throughout the service life of the Susquehanna Nuclear Station by a materials surveillance program that will satisfy the requirements of ASTM Standard E185-73, "Standard Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels," and

Appendix H, 10 CFR Part 50. We have evaluated the applicant's degree of compliance with the requirements of Appendix H, 10 CFR Part 50. The results of our evaluation indicate that the applicant meets the requirements of this appendix, except that the materials used in the reactor vessel surveillance program have not been sufficiently identified, and the orientation and number of specimens in the surveillance program do not satisfy the specific requirements of Appendix H. These points are addressed individually below.

The applicant has indicated in the Susquehanna FSAR that the material from which surveillance specimens will be fabricated is not in complete agreement with the requirements of Appendix H, 10 CFR Part 50. Therefore, in order to demonstrate compliance with Appendix H, we require that the applicant submit information that will identify the origin and use of the materials in the surveillance program for both baseline and irradiated fracture toughness measurements. If these materials are not those which will be limiting, an analysis including effects of initial toughness levels and impurity levels is also required to show that the results from the materials used can be employed to conservatively predict the maximum expected temperature shift due to irradiation.

The Charpy V-notch specimens to be used in the Susquehanna Power Station surveillance program are of a longitudinal orientation, whereas transversely oriented specimens are required by Appendix H by reference to the ASTM Standard Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels, E 185-73. The longitudinal orientation was that specified in the corresponding ASTM Standard, E 185-70, in use at the time the vessel was under construction. The test data that will be obtained from the specimens with longitudinal orientation will provide sufficient data to predict the relative shift in RT_{NDT} due to neutron irradiation for the specific materials tested since the experience indicates that relative shift is not greatly sensitive to specimen orientation. Hence, adequate information concerning radiation damage can be obtained from specimens of a longitudinal orientation.

The number of Charpy V-notch impact specimens required by Appendix H is 12 each for weld metal, HAZ material, and base metal for each surveillance capsule and 15 for each material for out-of-reactor baseline testing. Since the applicant has stated that it will include fewer specimens in some of the capsules, we require them to submit for review their testing and evaluation plans for the specimens in these capsules. These plans shall demonstrate that a similar level of assurance in measuring the shift of RT_{NDT} and drop in upper shelf can be obtained as that provided by the 12 specimen testing sets. The number of specimens for their out-of-reactor Charpy-V-notch baseline testing shall be included along with, if necessary, a justification for any discrepancies from the requirements of Appendix H.

Appendix G, "Protection Against Non-Ductile Failure," Section III of the ASME Boiler and Pressure Vessel Code, will be used, together with the fracture toughness test results required by Appendices G and H, 10 CFR Part 50, to calculate the reactor coolant pressure boundary pressure-temperature limitations for the Susquehanna Nuclear Station.

The fracture toughness tests required by the ASME Code and by Appendix G of 10 CFR Part 50 provide reasonable assurance that adequate safety margins against the possibility of non-ductile behavior or rapidly propagating fracture can be established for all pressure retaining components of the reactor coolant boundary. The use of Appendix G, Section III of the ASME Code, as a guide in establishing safe operating procedures, and use of the results of the fracture toughness tests performed in accordance with the ASME Code and NRC regulations, will provide adequate safety margins during operating, testing, maintenance, and anticipated transient conditions. Compliance with these Code provisions and NRC regulations constitutes an acceptable basis for satisfying the requirements of General Design Criterion 31.

The materials surveillance program required by Appendix H, 10 CFR Part 50, will provide information on material properties and the effects of irradiation on material properties so that changes in the fracture toughness of the material in the Susquehanna reactor vessel beltline caused by exposure to neutron radiation can be properly assessed, and adequate safety margins against the possibility of vessel failure can be provided.

Compliance with ASTM E 185-73 and Appendix H, 10 CFR Part 50 assures that the surveillance program constitutes an acceptable basis for monitoring radiation-induced changes in the fracture toughness of the reactor vessel material and satisfies the requirements of General Design Criteria 31 and 32.

We have reviewed the toughness requirements, and the extent of materials testing for ferritic components of the reactor coolant pressure boundary, other than the reactor pressure vessels described earlier, in accordance with General Design Criterion 31. The ferritic materials were specified to meet the toughness requirements of the 1971 Edition of the ASME Boiler and Pressure Vessel Code, Section III, "Rules for Construction of Nuclear Power Plant Components."

Appendix G, "Fracture Toughness Requirements," of 10 CFR Part 50 specifies the fracture toughness requirements for the ferritic materials of the reactor coolant pressure boundary. The construction permit for both Units 1 and 2 of the Susquehanna Power Station was issued on November 2, 1973. Pursuant to paragraph 50.55a of 10 CFR Part 50, the applicable ASME Boiler and Pressure Vessel Code for the reactor recirculation piping is the 1971 Edition, including Addenda through Summer 1972; while, the applicable Code for the reactor recirculation system pumps, main steam line isolation valves and main steam safety/relief valves is the 1971 Edition including Addenda through

Winter 1971. Based on information supplied by the applicant concerning anticipated deviations from the codes and standards rules of the above mentioned paragraph, the AEC granted on June 20, 1974, in accordance with paragraph 50.55a(a)(2)(ii) approval for relief from the rule and acceptance of the ASME Section III Code for 1971 including the Summer 1971 Addendum for the above mentioned components.

Since the applicable ASME Code Edition and Addendum described above preceded the publication of Appendix G, 10 CFR Part 50, some of the fracture toughness tests for the ferritic materials in the reactor coolant pressure boundary were not conducted to demonstrate explicit compliance with the current requirements of Appendix G.

We have reviewed the information submitted to date in the applicant's FSAR to determine the degree of compliance with the fracture toughness requirements of Apendix G, 10 CFR Part 50. Our evaluation indicates that the applicant has met all requirements of Appendix G, 10 CFR Part 50, except for the following:

1. Paragraphs III.B.1-B.2 specify that fracture toughness tests be conducted with (a) impact test specimens which comply with the location and orientation requirements of paragraph NB-2322, Section III, of the ASME Code and (b) materials which are representative of the actual materials used in the reactor coolant pressure boundary. The applicant must supply information to indicate that these requirements have been met.

2. Paragraph IV.A.3 specifies that materials for piping, pumps, and valves meet the requirements of paragraph NB-2332, Section III, of the ASME Code, and materials for bolting and other fasteners meet the requirements of paragraph NB-2333, Section III, of the ASME Code. The applicant has not supplied the fracture toughness data required by NB-2332 or NB-2333 and, therefore, must submit this data in order that we may determine compliance with Appendix G.

3. The applicant stated in Section 5.2.3.3.1 of the FSAR that the main steam isolation valves were exempted from impact testing because the ASME Code Section III, Summer 1971 Addendum did not require brittle fracture testing on ferritic pressure boundary components when the system temperature is in excess of 250°F at 20% of the design pressure. Our review does not indicate that the applicant's referenced exception was approved for inclusion in the ASME Code, Section III. The comments addressed in Paragraph 2 above are, therefore, applicable to the main steam isolation valves as well.

The fracture toughness tests required by the ASME Code and Appendix G, 10 CFR Part 50, provide reasonable assurance that adequate safety margins against the possibility of non-ductile behavior or rapidly propagating facture can be established for all pressure retaining components of the reactor coolant pressure boundary. The use of Appendix G, Section III of the ASME Code, as a guide in establishing safe operating procedures, and the use of the fracture toughness test results performed in accordance with the ASME Code and NRC regulations, will provide adequate safety margins during operating, testing, maintenance, and anticipated transient conditions. Compliance with these Code provisions and NRC regulations will constitute an acceptable basis for satisfying the requirements of General Design Criterion 31.

5.3.2 Pressure Temperature Limits

Appendix G, "Fracture Toughness Requirements," and Appendix H, "Reactor Vessel Material Surveillance Program Requirements," 10 CFR Part 50, describe the conditions that require pressure-temperature limits and provide the general bases for these limits. These appendices specifically require that pressure-temperature limits must provide safety margins at least as great as those recommended in the ASME Boiler and Pressure Vessel Code, Section III, Appendix G, "Protection Against Non-Ductile Failure." Appendix G, 10 CFR Part 50 requires additional safety margins whenever the reactor core is critical, except for low-level physics tests.

The following pressure-temperature limits imposed on the reactor coolant pressure boundary during operation and tests are reviewed to ensure that they provide adequate safety margins against non-ductile behavior or rapidly propagating failure of ferritic components, as required by General Design Criterion 31:

A. Preservice hydrostatic tests,

B. Inservice leak and hydrostatic tests,

C. Heatup and cooldown operation, and

D. Core operation.

The applicant has proposed the use of an alternative method of calculating the shift in the reference temperature, as required by Appendices G and H, 10 CFR Part 50. This method estimates the shift in the reference temperature by extrapolating the methods in Regulatory

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The following pressure-temperature limits imposed on the reactor coolant pressure boundary during operation and tests are reviewed to ensure that they provide adequate safety margins against non-ductile behavior or rapidly propagating failure of ferritic components, as required by General Design Criterion 31:

1. Preservice hydrostatic tests,
2. Inservice leak and hydrostatic tests,
3. Heatup and cooldown operation, and
4. Core operation.

The applicant has proposed the use of an alternative method of calculating the shift in the reference temperature, as required by Appendices G and H, 10 CFR Part 50. This method estimates the shift in the reference temperature by extrapolating the methods in Regulatory

Guide 1.99 (Revision 1), "Effect of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials," to shifts of less than 50°F. This method estimates the shift in reference temperature conservatively for the first 10 effective full power years. Subsequent to operation, predictions of radiation damage can be based on the actual measured shifts in reference temperature that are from the results of the surveillance program at Susquehanna Power Station.

The additional safety margins which Appendix G, 10 CFR Part 50 require beyond those specified in the ASME Boiler and Pressure Vessel Code whenever the core is critical, other than for low level physics tests, are that the temperatures of the reactor vessel be at least 40°F higher than in similar non-nuclear heating conditions as well as above that required for the inservice system hydrostatic pressure test. The applicant has proposed a criticality temperature limit which is not restricted by the inservice system hydrostatic pressure test but, rather, by consideration of fracture prevention in the flange regions which are highly stressed by the bolt preload. This approach has been previously described in the General Electric Topical Report NEDO-21778-A, *Transient Pressure Rises Affecting Fracture Toughness Requirements for Boiling Water Reactors*, accepted November 13, 1978.

We have evaluated this alternative approach and have concluded that the method proposed by the applicant for setting the pressure-temperature limits based on fracture prevention in the highly pre-stressed flange regions of the reactor vessel at low temperatures provides a sufficient margin of safety and that the goals of Appendix G 10 CFR Part 50 would, therefore, be fulfilled in regard to these matters.

The applicant has specified initial limiting reference temperatures as a basis for the pressure-temperature limits it is proposing. Insufficient materials property data have been submitted to the staff to support the conclusion that the temperatures specified are indeed limiting. We require the applicant to supply the necessary supporting materials data described in detil in SER Section 5.3.1 which will be necessary to complete review in this area, and show, based on these data, that the limiting temperatures and materials are justified.

The pressure-temperature limits to be imposed on the reactor coolant system for all operating and testing conditions to assure adequate safety margins against non-ductile or rapidly propagating failure appear to be in conformance with established criteria, codes, and standards acceptable to the staff except for the area discussed above. However, before we can complete our review in this area, we require the applicant to submit the materials property data required by 10 CFR Part 50 and the ASME Code as detailed in SER Section 5.3.1.

5.3.3 Reactor Vessel Integrity

General Design Criteria 14, 30, 31 and 32, Appendix A, 10 CFR Part 50 require, in part, that the reactor coolant pressure boundary be designed, fabricated, erected and tested to the highest quality standards practical so as to minimize coolant leakage, minimize the probability of rapidly propagating fracture and maximize inspectability and surveillance capability to permit a continuing evaluation of structural integrity. Although most areas relating to reactor vessel integrity are evaluated separately in other sections of the Safety Evaluation Report, the reactor vessel is of such importance that a special summary review of all factors is warranted.

Information in the following areas has been reviewed to ensure completeness and that no inconsistencies exist which would compromise the integrity of Susquehanna Units 1 and 2 reactor pressure vessels.

1. Design (SER 5.2.3 and 5.3.1),
2. Materials of construction (SER 5.2.3 and 5.3.1),
3. Fabrication methods (SER 5.2.3 and 5.3.1),
4. Inspection requirements (SER 5.2.4), and
5. Operation conditions (SER 5.3.2).

We have reviewed the above factors contributing to the structural integrity of the reactor vessel and conclude that the applicant has complied with Appendices G and H, 10 CFR Part 50, except for the following:

1. The applicant has not reported all fracture toughness data as required by Appendix G;

2. The applicant has proposed various material property correlations to allow for conservative estimates of the fracture toughness requirements of Appendix G. Additional information and justification for these procedures must be supplied;

3. Information concerning corrective action for an unauthorized attachment to the Susquehanna Unit 1 reactor pressure vessel has not been submitted;

4. The materials used in the reactor vessel surveillance program have not been sufficiently identified, and the orientation and number of specimens do not satisfy the specific requirements of Appendix H. The applicant must submit additional information concerning the surveillance program for our review;

5. The applicant has not submitted a preservice inspection program plan for review; and

6. The applicant has not submitted sufficient materials property data to support their pressure-temperature limits.

Until the applicant has supplied the information necessary to complete our evaluation of Susquehanna Units 1 and 2 compliance with Appendices G and H, we cannot form a conclusion regarding the structural integrity of the reactor vessels.

6.1.1 Engineered Safety Feature Materials

Engineered Safety Features (ESF) are provided in nuclear plants to mitigate the consequences of design basis or loss-of-coolant accidents, even though the occurrence of these accidents is very unlikely. The General Design Criteria 16, 34, 35, 38, 41 and 44 of Appendix A, 10 CFR Part 50 require that certain systems be provided to serve as Engineered Safety Features. The materials and fabrication procedures used in the Engineered Safety Features are reviewed to assure compliance with the General Design Criteria.

The materials selected for the Engineered Safety Features satisfy Appendix I of Section III of the ASME Code, and Parts A, B and C of Section II of the Code, and the staff position that the yield strength of cold-worked stainless steels shall not exceed 90,000 psi.

Ferritic materials must satisfy the impact requirements of NB-2300, NC-2300, or ND-2300 of Section III of the Code. So that we may complete our review in this area, the applicant should confirm that ferritic materials were tested and found in conformance with the Code requirements. These fracture toughness tests and mechanical properties required by the Code provide reasonable assurance that ferritic materials will have adequate safety margins against the possibility of nonductile behavior or rapidly propagating fracture.

The controls on pH and chemistry of the reactor containment sprays and the Emergency Core Cooling water following a postulated loss-of-coolant or design basis accident are adequate to reduce the probability of stress-corrosion cracking of the austenitic stainless steel components and welds of the Engineered Safety Features systems during the accident to completion of cleanup. The controls on the use and fabrication of the austenitic stainless steels adequately satisfy the requirements of Regulatory Guide 1.31, "Control of Ferrite Content of Stainless Steel Weld Metal," but do not satisfy the requirements, regarding sensitization, of Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel," as indicated below.

The specific item not dealt with is tests to confirm the lack of sensitization of as-received materials and processed materials by ASTM A 262-70, "Recommended Practices for Detecting Susceptibility to Intergranular Attack in Stainless Steel." The applicant must confirm that the fabrication and heat treatment practices followed in fabricating these components provide adequate assurance that stress-corrosion cracking will not occur.

Cleaning procedures are in accordance with Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants," and the controls placed on concentrations of leachable impurities in nonmetallic thermal insulation used on components of the Engineered Safety Features are in accordance with Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel."

The control of the pH of the sprays and cooling water, in conjunction with controls of containment materials, is in accordance with Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident," and provides assurance that the sprays and cooling water will not give rise to excessive hydrogen gas evolution resulting from corrosion of containment metal or cause serious deterioration of the materials in containment.

Zinc compounds are applied in conformance with Regulatory Guide 1.54, "Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants," but are noted in FSAR Section 3.13 to not meet all the requirements of Regulatory Guide 1.54. The details and consequences of the total lack of conformance with Regulatory Guide 1.54 must be clarified by the applicant before we can complete our review in this area.

Conformance with the Codes and Regulatory Guides and with the staff positions mentioned above constitute an acceptable basis for meeting the requirements of General Design Criteria 16, 34, 35, 38, 41 and 44.

6.6 Inservice Inspection of Class 2 and 3 Components

General Design Criteria 36, 39, 42, and 45, Appendix A, 10 CFR Part 50, require, in part, that the emergency core cooling containment heat removal, containment atmosphere cleanups and cooling water systems be designed to permit appropriate periodic inspection of important component parts to assure system integrity and capability. Design considerations relating to the Inservice Inspection (ISI) Program for ASME Code Class 2 and Class 3 components are reviewed to assure compliance with the General Design Criteria.

Section 50.55a(g), 10 CFR Part 50, defines the detailed requirements for the preservice and inservice inspection programs for light water cooled nuclear power facility components. Based upon a construction permit date of November 1973, this section of the Code of Federal Regulations requires that a preservice inspection program be developed for Class 2 components and be implemented using at least the Edition and Addenda of Section XI of the ASME code in effect six months prior to the date of issuance of the construction permit. Also, the initial inservice inspection program must comply with the requirements of the latest Edition and Addenda of Section XI of the ASME Code in effect twelve months prior to the date of issuance of the operating license, subject to the limitations and modifications listed in Section 50.55a(b) of 10 CFR Part 50.

The applicant is currently developing the preservice inspection program for Susquehanna Steam Electric Station - Unit 1. Deviations are required to be evaluated and acceptable from the requirements of Section XI of the ASME Code and in accordance with 10 CFR Part 50.55a, before issuance of an operating license. Evaluation of the deviations from Section XI requirements will be presented in a supplement to the SER. The inservice inspection program for Class 2 and Class 3 components will be evaluated after the applicable ASME Code Edition and Addenda have been determined and before the initial inservice inspections are performed.

Compliance with the inservice inspections required by ASME Code and 10 CFR Part 50 constitutes an acceptable basis for satisfying applicable requirements of General Design Criteria 36, 39, 42 and 45.

10.3.6 Steam and Feedwater System Materials

General Design Criterion 1 requires, in part, that systems important to safety shall be designed to quality standards commensurate with the importance to safety of the functions to be performed. The materials utilized in the Steam and Feedwater System are reviewed for compliance with General Design Criterion 1.

The mechanical properties of materials selected for Class 1, 2, and 3 components of the steam and feedwater systems provided in FSAR, Section 10.3.6.2 satisfy Appendix I of Section III of the ASME Boiler and Pressure Vessel Code, and Parts A, B, or C of Section II of the Code. However, the list of materials is inadequate in that it does not identify materials used for specific portions of the system, such as valves, fittings, welds, etc., and the additional information must be provided before we can complete our review in this area.

Ferritic materials must satisfy the impact requirements of NB-2300, NC-2300, or ND-2300 of Section III of the Code. So that we may complete our review in this area, the applicant should confirm that ferritic materials were tested and found in conformance with the Code requirements. These fracture toughness tests and mechanical properties required by the Code provide reasonable assurance that ferritic materials will have adequate safety margins against the possibility of nonductile behavior or rapidly propagating fracture.

Since austenitic stainless steels are not used in the steam and feedwater systems, the numerous requirements specified for these materials do not apply to the Susquehanna Power Station. The applicant takes exception to the requirements of Regulatory Guide 1.71, "Welder Qualification for Areas of Limited Accessibility," and simply complied with Sections III and IX of the Code. The applicant must show that their procedures resulted in high quality welds.

The onsite cleaning and cleanliness controls used during fabrication are in conformance with the positions given in Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants," and the requirements of ANSI Standard N 45.2.1-1973, "Cleaning of Fluid Systems and Associated Components During Construction Phase of Nuclear Power Plants."

The applicant has not noted specifically controlling and monitoring the preheat and interpass temperatures during welding of carbon and low alloy steel components to conform to Regulatory Guide 1.50, "Control of Preheat Temperature for Welding Low-Alloy Steel," and must provide assurance of adherence to the requirements of the Regulatory Guide.

The ASME Code Section III, Paragraphs NB/NC/ND 2550-2570 provide requirements for nondestructive examination of tubular products. The applicant must identify all tubular products and provide information showing that they were nondestructively examined in accordance with the applicable requirements of the ASME Code.

Conformance with the codes, standards, and Regulatory Guides mentioned constitutes an accepable basis for assuring the integrity of steam and feedwater systems and for meeting in part the requirements of General Design Criterion 1.

4.5.1 Control Drive Structural Materials

1. FSAR Section 4.5.1.3 indicates that nitrided austenitic stainless steel parts of the control rod drive system (piston tube, index tube, collet piston and guide cap) are accessible for visual examination. Provide additional details of the inservice inspection procedures for all nitrided components including inspection methods, schedules and justification for same.

2. FSAR Section 4.5.1.1 identifies ASTM Specifications A269 and A249 as well as ASME Specification SA312 for various tubular components of the control rod drive systems constructed of austenitic stainless steel. The specifications are applicable for welded and seamless products. The nondestructive examination procedures of the 1971 ASME Code, Section III, paragraph NB-2550, are required for wrought seamless tubular products. For tubular products welded without filler metal, the requirements for Class 2 components are given by paragraph NC-2550, which references the requirements of NB-2550. For Class 2 tubular products welded with filler metal, paragraph NC-2560 of the 1971 ASME Code is applicable.

The FSAR does not identify which tubular components of the control rod drive system are welded and which are wrought seamless. Provide information delineating those welded and seamless tubular products as well as the nondestructive examination methods used to verify the components are free of internal defects.

4.5.2 Reactor Internals Materials

1. FSAR Section 4.5.2.1 identifies ASTM Specification A370, Grades E38 and E55 for the pin and insert of the inlet-mixer in the jet pump assemblies. Specification A370 relates to testing of materials, not to material specifications. Provide the correct material specification used for those components.

2. FSAR Section 4.5.2.1 identifies nitrided type 304 stainless steel as the material for jet pump beam bolts. Identify the material by specification and provide assurance that the material can be expected to perform satisfactorily under the expected service conditions in this application.

3. In FSAR Section 4.5.2.3, it is stated that the ASME material specifications used for the Susquehanna core support structures require nondestructive examination of wrought seamless tubular products according to paragraph NG-2550, Section III of the ASME Code. We do not necessarily agree that the ASME material specifications invoke the requirements of NG-2550. Provide assurance that the nondestructive examination for all wrought seamless tubular products used in the core support structures of Susquehanna Units 1 and 2 were performed in accordance with the requirements of NG-2550.

5.2.3 Reactor Coolant Pressure Boundary Materials

1. We have noted seven deficiencies in Table 5.2.4, "Reactor Coolant Pressure Boundary Materials (RCPB)," which require clarification or justification as follows:

a. Submerged-arc-welding (SAW) materials for major reactor vessel welds are not specified in the table, while the SAW process is referenced in other sections of the FSAR. Identify all SAW material specifications and specific applications in the RCPB.

b. SA 420, Grade WPL1 is identified in the table as a material specification for fittings. This is not an allowable grade in Section III, Summer 1971 Addendum or any subsequent addenda. Provide justification for the use of this material in the RCPB and assurance that it will perform satisfactorily under expected service conditions.

c. The table indicates that SFA 5.5 E7010 A1 shielded metal-arc electrodes are used for the RCPB. Since these materials are not low hydrogen electrodes, their use could lead to cold cracking problems and, thus, should not be used for fabrication of critical components. Identify those welds in the RCPB fabricated with E7010 A1 electrodes and provide assurance that the integrity of the RCPB will not be compromised.

d. SA 540, Grade B24 is identified in FSAR Section 5.3.1.5.1.2 for reactor vessel closure bolting. That specification is not reflected in Table 5.2-4 for any component. Verify that the above specification is used as indicated and, if so, change the table accordingly.

e. The following specified grades are not identifiable and are believed to be typographical errors:

(1) SFA 5.1 Grade E-705

(2) SA 240, Grade F316

(3) SA 182, Grade F216

Clarify the intended grades and make appropriate changes to Table 5.2-4.

2. Regulatory Guide 1.44, "Control of Sensitized Stainless Steel," recommends that nonsensitization of austenitic stainless steel weldments be verified using an approved procedure as outlined in Regulatory Guide 1.44. There is no indication in the FSAR that such weldments in the control rod drive system, reactor internals, reactor coolant pressure boundary or engineered safety features have been so verified. Provide assurance that these welds are not severely sensitized and that they will not experience intergranular stress-corrosion cracking in service.

3. Regulatory Guide 1.56, "Maintenance of Water Purity in BWR's," recommends that 60% of the theoretical ion exchange capacity be maintained in the demineralizers. FSAR Section 5.2.3.2.2(2)3 indicates that 50% of the initial capacity will be maintained. Provide justification for this reduction in ion exchange capacity.

Additionally, General Electric Report NEDO-10899, "Chloride Control in BWR Coolants," is referenced in various section of the FSAR as the document which establishes the applicant's position on water purity.

The report is used as justification for not performing verification tests for sensitization of austenitic stainless steel welds under certain conditions. That report has not been approved by the NRC as a referenceable topical report on the subject of water purity. Provide justification for acceptance of this report in providing assurance that intergranular stress-corrosion cracking will not occur in the expected operating environment.

4. Regulatory Guide 1.50, "Control of Preheat Temperature for Welding Low Alloy Steels," calls for maintaining preheat until postweld heat-treatment. FSAR Section 5.2.3.3.2.1 indicates preheat was sometimes "held for an extended period of time at preheat temperature to assure removal of hydrogen." This statement does not provide justification for the extended preheat, nor does it provide assurance that cracking will not occur during fabrication. Provide additional information concerning extended preheat treatments and justification for their use.

Additionally, there is no assurance given in the FSAR that welding procedures were qualified at the minimum preheat temperature as recommended by Regulatory Guide 1.50. Verify that welding procedures were properly qualified as discussed above or, if not, provide assurance that qualification procedures used are satisfactory for fabrication of components in the reactor coolant pressure boundary and the steam and feedwater system in general.

5. Regulatory Guide 1.71, "Welder Qualification for Areas of Limited Accessibility," recommends that both radiography and sectioning be performed for mock-up welding. FSAR Sections 4.5.2.4 and 5.2.3.3.2.3 indicate that only one method, radiography or sectioning, was performed. Verify the accuracy of the referenced statements in the FSAR and, if accurate, provide justification for deviation from the recommendations of Regulatory Guide 1.71 to assure that welder qualification methods used resulted in sound welds in areas of limited accessibility existing in reactor internals, the reactor coolant pressure boundary and the steam and feedwater system in general.

5.3.1 Reactor Vessel Materials — 5.3.2 Pressure Temperature Limits

1. The answer to Question 121.1 is not adequate as no information on any of the material outside of the beltline region of either reactor is included. Supply the information required to fully answer Question 121.1 for the remainder of the Susquehanna reactor vessels. Also, the statement that no records were kept on the use of specific welding electrodes in particular welds is totally unacceptable as this represents a major departure from QA requirements and standard practice. Provide a thorough description of the exact method of electrode accountability used and a technical justification including analytical and/or experimental results as necessary to show that acceptable material properties were obtained in every weld in the Susquehanna reactor vessels.

2. The information submitted in the FSAR through Revision 13 and including answers to Questions 121.2 and 121.3 is still not sufficient to satisfy that the requirements for fracture toughness of the reactor coolant pressure boundary of the Susquehanna plant as defined in Appendix G to 10 CFR Part 50 have been met. The applicant shall submit additional material test results as required to show that points identified below are in strict compliance to the requirements of Appendix G, 10 CFR Part 50. In the event that the requirements cannot be specifically met, it shall be demonstrated, using additional materials test results and/or technical justification, that the proposed alternatives provide acceptable margins of safety relative to Appendix G requirements.

(1) The fracture toughness data on the ferritic material in the reactor coolant pressure boundary outside of the beltline required in paragraphs IIIA and IVA is completely lacking and must be supplied. Include results of testing for piping, pumps, valves and bolting used in the boundary (including main steam isolation valves).

(2) Fracture toughness data on the reactor vessel heat-affected zone (HAZ) material is lacking and must be supplied. The statement in FSAR Section 5.3.1.5.1.2 that the RT_{NDT} for the HAZ is the same as for the base material is unacceptable without supporting technical justification and materials test data.

(3) It has not been demonstrated that the location and identification of the materials used to determine fracture toughness results for the beltline materials complies with the requirements of paragraph III.C.2. Provide an exact description of the materials used for this purpose including the locations from which they were obtained.

(4) The 2°F per ft-lb correlation used in FSAR Section 5.3.1.5.1.2 to estimate the longitudinal CVN 50 ft-lb level as an intermediate step in establishing transverse CVN 50 ft-lb temperature for RCPB materials has not been shown to be conservative in extrapolating from higher to lower energy levels. Identify materials where this was done and provide conservative estimates of the temperatures at which 50 ft-lb will be achieved for these materials.

(5) The recommendation made in FSAR Table 5.3.1.A for meeting the 75 ft-lb upper shelf requirement for beltline material specified in paragraph IV.B on the basis of lowest longitudinal CVN energies and percent shear is insufficient. Justify this basis for acceptance or provide an alternative basis.

3. Insufficient information concerning magnetic particle, dye penetrant, and visual examinations of the reactor vessel closure bolting material was included in the Susquehanna FSAR Section 5.3.1.7. Supply information demonstrating that the bolting material for the Susquehanna vessels have met the requirements of the ASME Code Section III, paragraph N325 or NB2583 for such examinations as well as those specified in Regulatory Guide 1.65, "Materials and Inspections for Reactor Vessel Closure Studs," paragraph C.2.d.

4. To demonstrate compliance with Appendix H to 10 CFR Part 50, include in the Susquehanna FSAR and Technical Specifications a table that provides the following information for each surveillance specimen capsule:

(1) The actual surveillance materials in each capsule,

(2) The beltline material from which each surveillance material was obtained,

(3) The test specimen type(s) and their orientation for each surveillance material,

(4) The actual location of each capsule in the reactor vessel,

(5) The lead factor for each capsule calculated with respect to the 1/4 wall thickness location,

(6) The proposed loading schedule of the capsules into the Susquehanna reactor vessel, and

(7) The proposed time of capsule withdrawal (calendar years and effective full power years).

State whether or not the materials identified in parts (1) and (2) were chosen according to the ASTM Standard Recommended Practice E185-73, "Surveillance Tests for Nuclear Reactors." If they were not, supply the full details of the selection method along with technical justification for its use.

Further state whether or not the materials identified in parts (1) and (2) are the materials expected to be limiting. If they are not, include an analysis showing how the test results with these materials will be used to conservatively predict the maximum expected shift of RT_{NDT}.

5. Since the Susquehanna FSAR states in Section 5.3.1.6.1 that only eight Charpy V-notch specimens per material are to be included in the second and third surveillance capsules, provide justification that these eight specimens will provide the similar margin of safety as that required by Appendix H, 10 CFR Part 50, by reference to ASTM Standard Recommended Practice E185-73, "Surveillance Tests for Nuclear Reactors," which specifies that 12 specimens shall be included.

6. In the Susquehanna FSAR Section 5.3.1.6.4, which describes the positioning and method of attachment of the surveillance capsules within the reactor vessels, no mention is made of the in-service inspection plans required by paragraph II.C.2 Appendix H, 10 CFR Part 50, by reference to Section XI of the ASME Code for permanent structural attachments.

Supply the inspection plans required by the above mentioned paragraph for our review. Also, the answer to question 121.4 (SSES FSAR Revision 11) is incorrect with respect to the code applicability of the surveillance capsule attachment brackets as well as inconsistent with FSAR Section 5.3.1.6.4 (Revision 13). Revise the answer accordingly.

7. With reference to the unauthorized attachment of a pipe hanger by welding to the SSES Unit 1 reactor pressure vessel, it is necessary that the deficiency be corrected. Technical details of the attachment (i.e., description of hanger, welding procedure, depth of penetration, location on vessel, etc.), as well as plans for corrective action and justification for same, must be submitted for our review.

6.1.1 Engineered Safety Features Materials

1. Ferritic materials used in Engineered Safety Features are required to meet the fracture toughness requirements of paragraphs NB-2300, NC-2300 or ND-2300, Section III of the ASME Code for Class 1, Class 2 and Class 3 components, respectively. Section 6.1 of the FSAR does not discuss the toughness of those materials. Clarify whether impact testing was performed on those materials and provide assurance that the ferritic materials used for the Engineered Safety Features meet the minimum requirements specified in the Code.

2. In Section 3.13 of the FSAR it is noted that the zinc compounds do not meet all the recommendations of Regulatory Guide 1.54, "Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants." List the specific areas in which the coatings do not meet the requirements of Regulatory Guide 1.54 and note the anticipated operational consequences of this lack of conformance.

10.3.6 Steam and Feedwater System Materials

1. FSAR Section 10.3.6 does not discuss the nondestructive examination of tubular products used in the steam and feedwater system. Describe the examination procedures used for tubular products of Class 1, Class 2 and Class 3 components. Provide justification for any deviation from the requirements of Section III, ASME Code.

2. Section 10.3.6 of the FSAR indicates that the ferritic materials specified for main steam and feedwater piping were not impact tested. Section III, NB-2300, NC-2300 and ND-2300 require impact testing be performed for Class 1, Class 2 and Class 3 components, respectively. Clarify the statement concerning impact testing in FSAR Section 10.3.6 and provide assurance that ferritic materials used for steam and feedwater system components meet the minimum requirements specified in the Code.

3. The list of materials given in FSAR Section 10.3.6.2 is incomplete. Provide a comprehensive list of materials used for Class 1, Class 2, and Class 3 components of the Steam and Feedwater System to include valves, fittings, bolting, pumps, pipes, welding, etc.