COMBUSTION ENGINEERING

October 4, 1989
LD-89-110

Project No. 675

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D. C. 20555

Subject: Response to NRC Request for Additional Information Concerning Chapters 4 and 5 Materials Engineering Branch.

Reference: Letter, G. S. Vissing (NRC) to A. E. Scherer (C-E), dated October 26, 1988.

Dear Sirs:

The Reference requested that Combustion Engineering provide additional information concerning the Combustion Engineering Standard Safety Analysis Report - Design Certification (CESSAR-DC), Chapters 4 and 5. Enclosure I to this letter provides our responses and Enclosure II provides the proposed corresponding revisions to CESSAR-DC.

Should you have any questions, please feel free to contact me or Mr. S. E. Ritterbusch of my staff at (203) 285-5206.

Very truly yours,

COMBUSTION ENGINEERING, INC.

A. E. Scherer
Director
Nuclear Licensing

AES:jeb
Enclosures: As Stated
cc: F. Ross (DOE-Germantown)
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RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION

CONCERNING CHAPTERS 4 AND 5,

MATERIALS ENGINEERING BRANCH

Question 251.11 (Re. Section 4.5.1)

A comparison with the Standard Review Plan indicates that the topics described below are not included in the identified CESSAR-DC sections.

4.5.1 Control Rod Drive Structural Materials

Section 4.5.1 should satisfy Regulatory Guide 1.85, "Code Case Acceptability ASME Section III Materials".

Section 4.5.2.1. The applicant must show that the material specification in this section satisfies the requirements of ASME Code, Section III, NG-2000.

Section 4.5.2.2. The welds of components for core support structures and reactor internals must be fabricated in accordance with the ASME Code, Section III, NG-4000.

Response 251.11 (Re. Section 4.5.1)

Section 4.5.1.1 will be revised to show conformance with Regulatory Guide 1.85.

Section 4.5.2.1 will be revised to show conformance with the requirements of the ASME Code, Section III, NG-2000.

Section 4.5.2.2 will be revised to indicate conformance with the ASME Code, Section III, NG-4000.

Section 4.5.1.1.A is also corrected to delete material ASTM A276, Type 440C, as a material in the upper pressure housing of the control element drive mechanism. This material is actually part of a seal, internal to the pressure housing vent line. Accordingly, this material will be added to the list of control element drive mechanism materials in contact with the primary coolant (Section 4.5.1.1.B).

Question 251.11 (Re. Section 5.2.3)

A comparison with the Standard Review Plan indicates that the topics described below are not included in the identified CESSAR-DC sections.

5.2.3 Reactor Coolant Pressure Boundary Materials

Section 5.2.3.1 should satisfy Regulatory Guide 1.85; ASME Code Section II, Parts A, B and C; ASME Code Section III, Appendix I.

Section 5.2.3.2.2 should satisfy Regulatory Guide 1.44 and the ASME Code, Section III, NB-3120.

Section 5.2.3.3.1.1.3. This section describes an exception to Appendix G to 10 CFR 50. The reactor vessel beltline weldments in the test specimen will not necessarily come from the actual production plate. The applicant must submit a detailed explanation for the exception.

Section 5.2.3.4 should satisfy Regulatory Guide 1.36.

Response 251.11 (Re. Section 5.2.3)

(Re. Section 5.2.3.1) It is acknowledged in Section 5.2.1.2 that C-E will comply with the intent of Regulatory Guide 1.85 in determining suitable ASME Code cases.

From Table 5.2-2 it can be seen that all Reactor Coolant Pressure Boundary materials are ASME Code Section II approved materials, that is, SA-, SB- or SFA- materials listed in Parts A, B, or C respectively.

Also, it is stated in Section 5.2.1.2 that all Reactor Coolant Pressure Boundary components are fabricated in accordance with the ASME Code, Section III. Appendix I, "Design Stress Intensity Values, Allowable Stresses, Material Properties and Design Fatigue Curves" is a part of Section III. The Design Stress Values listed in Appendix I are the basis of C-E engineering calculations.

(Re. Section 5.2.3.2.2) Consistency with the recommendations of Regulatory Guide 1.44 is discussed extensively in Section 5.2.3.4.1.1. Primary concerns, such as solution heat treatment, test for susceptibility to intergrannular corrosion, control of carbon content and welding procedures, were covered in order to avoid sensitization.

All Reactor Coolant Pressure Boundary components in contact with the reactor coolant are either made of corrosion resistant alloys or clad with stainless steel, in accordance with the special consideration on corrosion mentioned in Subarticle NB-3120 of the ASME Code.

(Re. Section 5.2.3.3.1.1.3) There was an exception to 10 CFR 50, Appendix G, paragraph III.C.2 (Federal Register, V.38, No. 138, July 17, 1973) cited previously in CESSAR-DC, Section 5.2.3.3.1.1-3; however, this exception has been deleted. The current version of 10 CFR 50, Appendix G (Federal Register, V.48, No. 104, May 27, 1983) stipulates testing in accordance with the ASME Code and 10 CFR 50, Appendix H, which involves ASTM Standard Practice E185-82. ASTM E185-82, paragraph 5.4, requires the weld metal test material to be fabricated "...with the same heat of weld wire and lot of flux and by the same welding practice as that used for the selected beltline weld". There is no stipulation in ASTM E185-82 that the base metal adjacent to the weld come from the actual production (beltline) plate. However, C-E's practice continues to be the use of base metal with the same P-number classification in conformance with the ASME Code. Therefore, there is no need to repeat the exception and Section 5.2.3.3.1.1-3 was deleted from CESSAR-DC in Amendment D (September 30, 1988).

(Re. Section 5.2.3.4) As mentioned in Section 5.2.3.2.3, only small sections of non-metallic insulation are used around stainless steel nozzles in the reactor vessel head, and the quantity of leachable halogens is limited in accordance with the intent of Regulatory Guide 1.36.

Question 251.11 (Re. Section 5.3.1)

A comparison with the Standard Review Plan indicates that the topics described below are not included in the identified CESSAR-DC sections.

5.3.1 Reactor Vessel Materials

The applicant should demonstrate that Section 5.3.1 satisfies GDC 1, 4, 14, 30 and 31; Appendix G and H to 10 CFR 50; and Regulatory Guides 1.37 and 1.65.

Section 5.3.1.3 must satisfy the requirements in Section V of the ASME Code.

Section 5.3.1.4 should satisfy Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associate Components of Water-Cooled Nuclear Power Plants".

Section 5.3.1.5 is not detailed enough for the staff review. The applicant should demonstrate that Section 5.3.1.5 conforms to requirements of SRP 5.3.1.II.5.

5.3.1.6 Reactor Vessel Material Surveillance Program

The applicant should discuss the chemistry or material content of base metal, weld metal, and heat-affected zone material. The copper, nickel, and phosphorus contents (in percentage) of each metal should be included.

The staff could not find Table 5.3-8 in Section 5.3.1.6.

Response 251.11 (Re. Section 5.3.1)

Conformance with all the GDC's is discussed in Section 3.1. These discussions summarize the manner in which the design features meet the individual criteria.

C-E complies with 10 CFR 50, Appendix G, as discussed in Section 5.3.1.5, in that the reactor vessel beltline materials have a minimum upper-shelf energy of 75 ft-lbs from transversely oriented specimens.

C-E complies with 10 CFR 50, Appendix H, as discussed in Section 5.3.1.6, in that the System 80+ reactor vessel surveillance program adheres to all of the requirements of ASTM E185-82.

The cleaning and contamination control for NSSS components, as recommended by Regulatory Guide 1.37, is stipulated by the procedures presented in Section 5.2.3.4.1.2.1.

The reactor vessel fastener material, SA-540 B23 or B24, Class 3, as mentioned in Section 5.3.1.7, conforms to the intent of Regulatory Guide 1.65.

(Re. Section 5.3.1.3) Section 5.3.1.3 stipulates nondestructive examinations based upon Section III of the ASME Code and Subarticle NB-5111 of Section III requires that nondestructive examination be conducted in accordance with the examination methods of Section V. Therefore, the requirements in Section V of the ASME Code are strictly followed in nondestructive examination of the reactor vessel.

(Re. Section 5.3.1.4) Section 5.3.1.4 will be revised to indicate conformance with the intent of Regulatory Guide 1.37.

(Re. Section 5.3.1.5) Conformance with the guidance of SRP 5.3.1.II.5 is presented in CESSAR-DC, Section 3.1.27, "Criterion 31- Fracture Prevention of Reactor Coolant Pressure Boundary." CESSAR-DC, Section 5.3.1.5, requires that reactor vessel beltline material must comply with 10 CFR 50, Appendix G (i.e., fracture toughness requirements). Further, this section states that tests will be performed on beltline forgings to demonstrate such compliance. These tests are performed on a plant-specific basis and, hence, compliance with Appendix G must be confirmed for each plant. This is the same approach used for the System 80 design described in CESSAR-F and approved in Section 5.3.1 of Safety Evaluation Report, NUREG-0852.

(Re. Section 5.3.1.6) The effect of residual elements on fracture toughness of beltline materials is discussed in Section 5.2.3.1. The limits of residual elements are specified in Section 5.2.3.1.

(Re. Section 5.3.1.6) Table 5.3-8 in CESSAR-F was originally provided to support the design curve for transition temperature shift prediction. Since CESSAR-DC now refers to Regulatory Guide 1.99, Rev. 02, Table 5.3-8 is no longer necessary and has been deleted from the text of CESSAR-DC.

Question 251.11 (Re. Section 5.3.1.7)

A comparison with the Standard Review Plan indicates that the topics described below are not included in the identified CESSAR-DC sections.

5.3.1.7 Reactor Vessel Fasteners

The applicant must specify the ultimate yield strength of the fasteners. The applicant must perform nondestructive examination(s) according to Section III of the ASME Code, Subarticle NB-2580.

Response 251.11 (Re. Section 5.3.1.7)

Based upon a review of SRP 5.3.1.II.7.b and Regulatory Guide 1.65, it is Combustion Engineering's understanding that the concern is the ultimate tensile strength of reactor vessel fasteners. The minimum value of the ultimate tensile strength of SA-540 B23 or B24, Class 3, is specified as 145 ksi in Section II of the ASME Code. The minimum of the yield strength for these materials is specified as 130 ksi.

Section 5.3.1.7 will be revised to indicate conformance with Subarticle NB-2580 of Section III of the ASME Code.

Question 251.11 (Re. Section 5.3.2.1)

A comparison with the Standard Review Plan indicates that the topics described below are not included in the identified CESSAR-DC sections.

5.3.2.1 Limit Curves

The applicant must discuss and satisfy the requirements for the pressure-temperature curves as described in 10 CFR 50, Appendix G. For example, the applicant should discuss that the minimum permissible temperature is 60 °F above the reference temperature of the closure flange regions that are highly stressed by the bolt preload.

The applicant should discuss and satisfy the requirements for the construction of pressure-temperature curves per SRP 5.3.2. Therefore, Section 5.3.2.1 should include all operating conditions as described in SRP 5.3.2. This section should also include a discussion on the calculation of RT_{NDT} as described in Regulatory Guide 1.99, Revision 2. The discussion should include the derivation of initial RT_{NDT}, RT_{NDT} shift, and margin of safety.

The pressure-temperature curves should be included in this section and not in chapter 16.

Response 251.11 (Re. Section 5.3.2.1)

Section 5.3.2.1 has been revised (Amendment E) to provide the requested information.

Question 251.11 (Re. Section 5.4.1.1)

A comparison with the Standard Review Plan indicates that the topics described below are not included in the identified CESSAR-DC sections.

5.4.1.1 Pump Flywheel Integrity

Pump flywheel integrity should satisfy the requirements in Regulatory Guide 1.14.

Paragraph 5.4.1.1c states that minimum fracture toughness of the flywheel material will be equivalent to a dynamic stress intensity factor (K_{IC} dynamic) of at least 100 ksi/in (this unit should be written ksi$\sqrt{in}$). Also paragraph 5.4.1.1.C.2 uses a lower bound fracture toughness curve and dynamic K_{IC} of 45 ksi$\sqrt{in}$ [sic] to comply to the minimum fracture toughness. SRP 5.4.1.1.II.2. requires a minimum static stress intensity factor of at least 150 ksi$\sqrt{in}$ and that the normal operating temperature must be at least 100°F above the RT_{NDT}.

The applicant should either a) use the requirements in SRP 5.4.1.1.II.2 verbatim, or b) justify that the stress intensity factors in paragraph 5.4.1.1.C or satisfy the SRP requirements.

Paragraph 5.4.1.1.2.a should include the following statement "or 1/3 of the measured yield strength in the weak direction of the material if appropriate tensile tests have been performed on the actual material of the flywheel."

Paragraph 5.4.1.1.2.b should state that the design overspeed of flywheel should be at least 10% above the highest anticipated overspeed. The applicant should either satisfy SRP 5.1.1.4.b requirements or justify paragraph 5.4.1.1.2.b.

Paragraph 5.4.1.1.2.c should include the following statement "or 2/3 of the measured yield strength in the weak direction of appropriate tensile tests have been performed on the actual material of the flywheel."

Paragraph 5.4.1.1.1.d should include the following statement "The ultrasonic examination of the areas of high stress concentration at the bore and keyway at about 3 1/2 years interval, during the refueling or maintenance shutdown coinciding with the inservice inspection schedule as required by the ASME code, Section XI. Removal of the flywheel is not required."

Paragraph 5.4.1.1.1.f should include the following statement "The liquid penetrant examination or magnetic particle methods and 100% volumetric examination by ultrasonic methods should be conducted at about ten years intervals during the plant shutdown coinciding with the inservice inspection schedule as required by the ASME Code, Section XI."

The applicant should review Section 5.4.1.1 to ensure it satisfies all requirements in SRP 5.4.1.1.

Response 251.11 (Re. Section 5.4.1.1)

Paragraph 5.4.1.1 of CESSAR-DC meets the intent of Regulatory Guide 1.14 in the same way it was satisfied for the System 80 design (see Section 5.4.1.1 of the Safety Evaluation Report, NUREG-0852).

(Re. Paragraph 5.4.1.1c) 5.4.1.1(1.)c is the designation in CESSAR-F. The same material is now presented in Section 5.4.1.1.A.3 of CESSAR-DC, Amendment D.

For typical flywheel materials the static stress intensity factor values are higher than dynamic stress intensity factor values at any given flywheel operating temperature. In other words a material which demonstrates a minimum static stress intensity factor of 150 ksi$\sqrt{\text{in}}$ at flywheel operating temperature would not provide a minimum dynamic stress intensity of 100 ksi$\sqrt{\text{in}}$. Therefore, use of the 100 ksi$\sqrt{\text{in}}$ dynamic stress intensity factor in CESSAR-DC Section 5.4.1.1.A.3 is conservative and satisfies the SRP recommendation. This same information was approved for the System 80 design described in CESSAR-F. CESSAR-DC will be revised to correctly state the stress intensity units, as indicated in the question.

(Re. Paragraph 5.4.1.1.2.a) 5.4.1.1.2.a is the designation in CESSAR-F. The same material is now presented in Section 5.4.1.1.B.1 of CESSAR-DC, Amendment D. Paragraph 5.4.1.1.B.1 will be revised to include the NRC statement provided in the question.

(Re. Paragraph 5.4.1.1.2.b) 5.4.1.1.2.b is the designation in CESSAR-F. The same material is presented in Section 5.4.1.1.B.3 of CESSAR-DC, Amendment D.

Paragraph 5.4.1.1.B.2 of CESSAR-DC correctly states that the flywheel design speed will be 125% of normal operating speed. This design speed includes 10% margin above the highest anticipated overspeed of the pump which is 115% of normal speed due to a turbine-generator overspeed condition. Due to NRC acceptance of the RCS piping leak before break concept (revision of GDC4), a pump discharge line pipe break is no longer considered as a credible source for predicting pump overspeeds.

Paragraph 5.4.1.1.B.2 was revised to reflect the RCS piping leak before break concept in Amendment D.

(Re. Paragraph 5.4.1.1.2.c) 5.4.1.1.2.c is the designation in CESSAR-F. The same material is now presented in Section 5.4.1.1.B.3 of CESSAR-DC, Amendment D. Paragraph 5.4.1.1.B.3 will be revised to include the NRC statement provided in the question.

(Re. Paragraph 5.4.1.1.1.d) 5.4.1.1.1.d and 5.4.1.1.1.f are designations in CESSAR-F. The same material is now presented in Sections 5.4.1.1.A.4 and 5.4.1.1.A.6 of CESSAR-DC, Amendment D.

As stated in 5.4.1.1.B, the flywheel will be accessible for 100 percent in-place volumetric ultrasonic inspection. The flywheel-motor assembly is designed to allow such inspection with a minimum of motor disassembly. Section 5.4.1.1.B will be revised to provide the requested statements as conditions for the plant-specific in-service inspection program.

C-E has reviewed Section 5.4.1.1 of CESSAR-DC and concludes that, with the above revisions, it satisfies the intent of all guidance in SRP 5.4.1.1.

PROPOSED REVISIONS TO THE

COMBUSTION ENGINEERING SAFETY ANALYSIS REPORT -

DESIGN CERTIFICATION

4.5 REACTOR MATERIALS

4.5.1 CONTROL ELEMENT DRIVE STRUCTURAL MATERIALS

4.5.1.1 Material Specifications

A. The materials used in the control element drive mechanism (CEDM) reactor coolant pressure boundary components are as follows:

1. Motor housing assembly

SA 182, Type 347 (austenitic stainless steel)

ASME Code Case N-4-11 (modified Type 403 martensitic stainless steel), and additional requirements of ASME SA-182 | D

SB 166 (nickel-chromium alloy)

2. Upper pressure housing

SA 213, Type 316 (austenitic stainless steel)

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SA 479, Type 316 (austenitic stainless steel) | D

~~ASTM A276, Type 440C (martensitic stainless steel with yield strength greater than 90,000 psi)~~

The above listed materials ~~with the exception of the ASTM A276, Type 440C material~~ are also listed in Section III of the ASME Boiler and Pressure Vessel Code. In addition, the materials comply with Sections II and IX of the ASME Boiler and Pressure Vessel Code. Code Case N-4-11 is acceptable per Regulatory Guide 1.85. | D
The functions of the above listed components are described in Section 3.9.4.1.

B. The materials in contact with the reactor coolant used in the CEDM motor assembly components are as follows:

1. Latch guide tubes

ASTM A269, Type 316 (austenitic stainless steel)

Chrome Oxide (plasma spray treatment)

2. Magnet and spacer

ASTM A276, Type 410 (martensitic stainless steel)

3. Latch and magnet housing

   ASTM A276, Type 316 (austenitic stainless steel)

   QQ-C-320, Class 2B (chrome plating) D

   ASTM A276, Type 440C (martensitic stainless steel)

4. Spacer

   ASTM A240, Type 304 (austenitic stainless steel)

5. Alignment Tab

   ASTM A276, Type 410 (martensitic stainless steel)

6. Spring

   AMS 5698B, Inconel X-750 (nickel base alloy)

7. Pin

   Haynes Stellite No. 6B (cobalt base alloy)

8. Dowel pin

   ASTM A314, Type 410 (martensitic stainless steel)

9. Spacer and screw

   ASTM A276, Type 321 (austenitic stainless steel)

10. Stop

    ASTM A276, Type 304 (austenitic stainless steel)

11. Latch and pin

    Haynes Stellite No. 36 (cobalt base alloy)

12. Locking cup and screws

    Type 300 Series austenitic stainless steel

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13. Steel Ball
ASTM A 276, Type 440 C

The functions of the CEDM motor assembly components are described in Section 3.9.4.1.

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rubber-bonded aluminum oxide or silicon carbide wheels that have not previously been used on materials other than Type 300 series stainless steel alloys.

Internal surfaces of completed components are cleaned to the extent that grit, scale, corrosion products, grease, oil, wax gum, adhered or embedded dirt, or extraneous material are not visible to the unaided eye.

Cleaning is effected by either solvents (acetone or isopropyl alcohol) or inhibited water (300-200 ppm hydrazine). Water will conform to the following requirements:

Halides	
Chloride, ppm	< 0.60
Fluoride, ppm	< 0.40
Conductivity, μmhos/cm	< 5.0
pH	6.0 - 8.0
Visual clarity	No turbidity, oil or sediment

To prevent halide-induced integranular corrosion that could occur in an aqueous environment with significant quantities of dissolved oxygen, flushing water is inhibited via additions of hydrazine. Experiments have proven these inhibitors to be effective. Operational chemistry specifications preclude halides and oxygen (both prerequisites for intergranular attacks) and are shown in Section 9.3.4.

4.5.2 REACTOR INTERNALS MATERIALS

4.5.2.1 Material Specifications

The materials used in fabrication of the reactor internal structures are primarily Type 304 stainless steel. The flow skirt is fabricated from Inconel. Welded connections are used where feasible; however, in locations where mechanical connections are required, structural fasteners are used which are designed to remain captured in the event of a single failure. Structural fastener material is typically a high strength austenitic stainless steel; however, in less critical applications Type 316 stainless steel is employed. Hardfacing of Stellite material is used at wear points. The effect of irradiation on the properties of the materials is considered in the design of the reactor internal structures. Work hardening properties of austenitic stainless steels are not used.

For reactor internals, the material specifications satisfy the requirements of Article NG-2000 in Section III of the ASME Code.

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E. Bolt and pin material

ASTM-A-453 and ASTM-A-638, Grade 660 material (trade name A-286) is used for bolting and pin applications. This alloy is heat treated in accordance with the ASTM specifications by precipitation hardening at 1300-1400°F for 16 hours to a minimum yield strength of 85,000 psi. Its corrosion properties are similar to those of the Type 300 series austenitic stainless steels. It is austenitic in all conditions of fabrication and heat treatment. This alloy was used for bolting in previous reactor systems and test facilities in contact with primary coolant and has proven completely satisfactory. D

F. Chrome plating and hardfacing

Chrome plating or hardfacing are employed on reactor internals components or portions thereof where required by function. Chrome plating complies with Federal Specification No. QQ-C-320. The hardfacing material employed is Stellite 25. D

All of the materials employed in the reactor internals and in-core instrument support system have performed satisfactorily in operating reactors such as Palisades (Docket-50-255), Fort Calhoun (Docket-50-285) and Maine Yankee (Docket-50-309).

4.5.2.2 Welding Acceptance Standards

Welds employed on reactor internals and core support structures meet the acceptance standards delineated in article NG-5000, Section III, Division I, and control of welding is performed in accordance with Section III, Division I, and Section IX of the ASME Code. In addition, consistency with the recommendations of Regulatory Guides 1.31 and 1.44 is described in Section 4.5.2.3.

4.5.2.3 Fabrication and Processing of Austenitic Stainless Steel

The following information applies to unstabilized austenitic stainless steel as used in the reactor internals.

4.5.2.3.1 Control of the Use of Sensitized Austenitic Stainless Steel

The recommendations of Regulatory Guide 1.44, as described in Sections 4.5.2.3.1.1 through 4.5.2.3.2.5, are followed except for the criterion used to demonstrate freedom from sensitization. The ASTM A708 Strauss Test is used in lieu of the ASTM A262 Method E, Modified Strauss Test, to demonstrate freedom from

are fabricated in accordance with Article NG-4000 in Section III, and

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Upon completion of all post-weld heat treatments, the reactor vessel is hydrostatically tested, and all accessible ferritic weld surfaces, including those used to repair material, are magnetic-particle inspected in accordance with Section III of the ASME Boiler and Pressure Vessel Code. | D

5.3.1.4 **Special Controls for Ferritic and Austenitic Stainless Steels**

Special controls for ferritic and austenitic stainless steels are as follows:

A. Regulatory Guide 1.31, Control of Ferrite Content in Stainless Steel Weld Metal, is addressed in Section 5.2.3.4. | D

B. Regulatory Guide 1.34, Control of Electroslag Weld Properties, is addressed in Section 5.2.3.3.

D ~~C~~. Regulatory Guide 1.43, Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components, is addressed in Section 5.2.3.3.

E ~~D~~. Regulatory Guide 1.44, Control of the Use of Sensitized Stainless Steel, is addressed in Section 5.2.3.4.

F ~~E~~. Regulatory Guide, 1.50, Control of Preheat Temperature for Welding of Low-Alloy Steel, is addressed in Section 5.2.3.3.

G ~~F~~. Regulatory Guide 1.71, Welder Qualification for Areas of Limited Accessibility, is addressed in Section 5.2.3.3.

H ~~G~~. Regulatory Guide 1.99, Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials, is addressed in Section 5.3.1.6.7. | B

5.3.1.5 **Fracture Toughness**

In accordance with 10 CFR 50 Appendix G, Paragraph IV A, the reactor vessel beltline materials have minimum upper-shelf energy, as determined from Charpy V-notch tests on unirradiated specimens in accordance with Paragraphs NB-2322.2(a) of the ASME Code, of 75 ft-lbs. Charpy impact tests will be performed on transversely (weak direction) oriented specimens from the beltline forgings to establish RT_{NDT} as required by 10 CFR 50, Appendix G. | D

C. Regulatory Guide 1.37, "Quality assurance Requirements for cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants", is addressed in Section 5.2.3.4.1.2.1.

the intent of

5.3.1.7 **Reactor Vessel Fasteners**

The bolting material for the reactor vessel closure head is fabricated from SA 540, B23 or B24, Class III material. This material conforms to the requirements of 10 CFR 50, Appendix G and Regulatory Guide 1.65, "Materials and Inspections for Reactor Vessel Closure Studs."

Nondestructive examination will be performed according to Subarticle NB-2580 of Section III of the ASME Code, during the manufacturing process.

C-E specifies the use of a manganese phosphate coating on threads of studs, nuts and washers to improve anti-galling properties and resistance to corrosion. In addition, Super Moly lubricant (containing molybdenum disulfide) is specified to be added to threads and bearing surfaces at installation to further enhance anti-galling properties. Laboratory testing and field experience to date have shown no evidence of deleterious breakdown of either phosphate coating or lubricant.

5.3.2 **PRESSURE-TEMPERATURE LIMITS**

All components in the Reactor Coolant System (RCS) are designed to withstand the effects of cyclic loads due to RCS temperature and pressure changes. These cyclic loads are introduced by normal unit load transients, reactor trips, and startup and shutdown operation.

During unit startup and shutdown, the rates of temperature and pressure changes are limited by the Technical Specifications. The design number of cycles for heatup and cooldown is based on a rate of 100°F/hr.

The maximum allowable RCS pressure at the corresponding minimum allowable temperature is based upon the stress limitations for brittle fracture. These limitations are derived using linear elastic fracture mechanics (LEFM) principles, the procedures prescribed by Appendix G to Section III of the ASME Code, "Protection Against Nonductile Fracture," Appendix G to 10 CFR 50, "Fracture Toughness Requirements," NRC Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials," and the procedures recommended by WRC Bulletin 175, "PVRC Recommendations on Toughness Requirements for Ferritic Materials." Compliance with Appendix H to 10 CFR 50, "Reactor Vessel Material Surveillance Program Requirements," is discussed in Section 5.3.1.6. E

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3. The minimum fracture toughness of the material at the normal operating temperature of the flywheel will be equivalent to a dynamic stress intensity factor (K_{IC} dynamic) of at least 100 ~~ksi/in~~ ksi√in. Compliance will be demonstrated by either of the following:

   a. Testing of the actual material of the flywheel to establish the K_{IC} (dynamic) value at the normal operating temperature, or

   b. Use of a lower bound fracture toughness curve obtained from tests on the same type of material. The curve will be translated along the temperature coordinate until the K_{IC} (dynamic) value of 45 ~~ksi/in~~ ksi√in. is indicated at the NDT of the material, as obtained from drop-weight tests.

4. Each finished flywheel will be subjected to a 100 percent volumetric ultrasonic inspection from the flat surface per ASME BPVC Section III.

   This inspection will be performed on the flywheel after final machining and the overspeed test.

5. If the flywheel is flame cut, at least 1/2 inch of stock will be left on the outer and bore radii, for machining to final dimensions.

6. The flywheel will be subjected to a magnetic particle or liquid-penetrant examination per "Section III" before final assembly. The inspection will be performed on finished machined bores, keyways, and on both flat surfaces to a radial distance of 8 inches minimum beyond the final largest machined bore diameter but not including small drilled holes. There will be no stress concentrations such as stamp marks, center punch marks, or drilled or tapped holes within 8 inches of the edge of the largest flywheel bore.

B. The flywheels will be designed to withstand normal operating conditions, anticipated transients, and the largest mechanistic pipe break size remaining after application of leak before break as described in Section 3.6, combined with the Safe Shutdown Earthquake. D

The following criteria will be satisfied:

1. The combined stress, both centrifugal and interference, at normal operating speed will not exceed one-third of the minimum specified yield strength ~~for the material selected in the direction of maximum stress~~.

or 1/3 of the measured yield strength in the weak direction of the material if appropriate tensile tests have been performed on the actual material of the flywheel

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a 2/3 of the measured yield strength in the weak direction if appropriate tensile tests have been performed on the actual material of the flywheel.

2. The design speed of the flywheel will be 125 percent of normal operating speed.

   The lowest of the critical speeds of the flywheel will be at least 10% above the highest anticipated overspeed of the pump. The highest anticipated overspeed is predicted for the largest break size remaining after applicationn of leak before break as described in Section 3.6. D

3. The combined centrifugal and interference stresses at the design speed will be limited to two-thirds of the minimum specified yield strength.[a] Design speed is defined as 125 percent of normal operating speed.

4. The motor and pump shaft or bearings and coupling will withstand any combination of normal operating loads or anticipated transients, and the largest remaining pipe break after application of leak before break as described in Section 3.6, combined with the Safe Earthquake Shutdown. D

Each flywheel will be tested at design speed, 125 percent of normal operating speed, as defined in B.2 above.

The flywheel will be accessible for 100 percent in-place volumetric ultrasonic inspection. The flywheel-motor assembly is designed to allow such inspection with a minimum of motor disassembly. Insert A

5.4.1.2 **Description**

Table 5.4.1-1 lists the principal parameters of the reactor coolant pumps and Figure 5.4.1-1 depicts the arrangement of the pump and motor. Reactor coolant pump supports are discussed in Section 5.4.14. The pump piping and instrument diagram is given in Figure 5.1.2-2.

The four reactor coolant pumps are vertical, single stage bottom suction, horizontal discharge, motor-driven centrifugal pumps. The pump impeller is keyed and locked to its shaft. Pump shaft alignment is maintained by a water lubricated radial bearing within the pump and by radial and thrust bearings located in the motor stand. The pump and motor shafts are directly connected by a coupling.

The shaft seal assembly consists of two face-type, mechanical seals in series, with controlled leakage bypass to provide the same pressure differential across each seal. The seal assembly is designed for 2500 psi differential and to reduce the leakage pressure from Reactor Coolant System pressure to the volume

INSERT A

The in-service inspection program will include ultrasonic examinations of the areas of high stress concentration at the bore and keyway at about 3 1/3 year intervals, during the refueling or maintenance shutdown coinciding with the in-service inspection schedule as required by the ASME code, Section XI. Removal of the flywheel is not required.

Liquid penetrant examination or magnetic particle methods and 100% volumetric examination by ultrasonic methods will be conducted at about ten-year intervals during the plant shutdown coinciding with the in-service inspection schedule as required by the ASME Code, Section XI.