RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

APR1400 Design Certification

Korea Electric Power Corporation / Korea Hydro & Nuclear Power Co., LTD

Docket No. 52-046

RAI No.:	**509-8591**
SRP Section:	**16 - Technical Specification**
Application Section:	**16 - Technical Specification**
Date of RAI Issue:	**08/01/2016**

Question No. 16-218

Paragraph (a)(11) of 10 CFR 52.47 states that a design certification (DC) applicant is to propose Technical Specifications (TS) prepared in accordance with 10 CFR 50.36 and 50.36a. NUREG-1432, “Standard Technical Specifications (STS)-Combustion Engineering Plants,” Rev. 4, provides NRC guidance on format and content of technical specifications as one acceptable means to meet 10 CFR 50.36 requirements. Staff needs to evaluate all technical differences from standard TS (STS) NUREG-1432, STS Combustion Engineering Plants, Rev. 4, which is referenced by the DC applicant in DCD Tier 2 Section 16.1, and the docketed rationale for each difference because conformance to STS provisions is used in the safety review as the initial point of guidance for evaluating the adequacy of the generic TS to ensure adequate protection of public health and safety, and the completeness and accuracy of the generic TS Bases.

The Writer’s Guide for Plant-Specific Improved Technical Specifications (TSTF-GG-05-01) also provides guidance for the format and content of the TS. There are format and content differences between the DCD and the Writer’s Guide. These following corrections are necessary to ensure the completeness and accuracy of the TS and Bases.

Justify the omission of Reviewer’s Notes in various sections of Technical Specification (TS) 5.0 Administrative Controls.

The Reviewer’s Notes could prove beneficial to subsequent COL applicants who choose to reference the APR1400 design and to the NRC staff performing said reviews.

The omitted Reviewer’s Notes are as follows:
- 5.1 (2 Reviewer’s Notes) – refer to the unit staff.
- 5.2.2.a – refers to non-licensed operator staffing requirements.
- 5.3 – refers to minimum qualifications for the unit staff.
- 5.5.3 – refers to Post-Accident Sampling.

- 5.5.9 (2 Reviewer's Notes) within the Steam Generator (S/G) Program
- 5.5.11 within the Ventilation Filter Testing Program (VFTP)
- 5.5.17 – refers to other LCOs as they relate to the Battery Monitoring and Maintenance Program
- 5.5.19.b – discusses NRC safety evaluation reports that should be listed
- 5.5.19.d – discusses the instrument functions that should be listed
- 5.6.4.c – discusses the NRC approval of the methodology for the calculation of the P-T Limits

This justification is required to ensure the completeness of the TS, and to facilitate potential COL applications referencing the APR1400 design.

Response

The identified Reviewer's Notes above will be added as shown in attached markup. Also, the APR1400 Technical Specification format and contents will be revised to align both STS Combustion Engineering Plants (NUREG-1432, Rev. 4) and the Writer's Guide for Plant-Specific Improved Technical Specifications (TSTF-GG-05-01).

Impact on DCD

Same as changes described in Impact on Technical Specifications section.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

Technical Specification 5.0 Administrative Controls will be revised as shown in attached markup.

Impact on Technical/Topical/Environmental Report

There is no impact on any Technical, Topical, or Environment Reports.

5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

5.1.1 The plant manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence. The plant manager or his designee shall approve, prior to implementation, each proposed test, experiment, or modification to systems or equipment that affect nuclear safety.

5.1.2 The [Shift Supervisor (SS)] shall be responsible for the control room command function. During any absence of the [SS] from the control room while the unit is in MODE 1, 2, 3, or 4, an individual with an active Senior Reactor Operator (SRO) license shall be designated to assume the control room command function. During any absence of the [SS] from the control room while the unit is in MODE 5 or 6, an individual with an active SRO license or Reactor Operator license shall be designated to assume the control room command function.

---REVIEWER'S NOTES---

1. Titles for members of the unit staff shall be specified by use of an overall statement referencing an ANSI Standard acceptable to the NRC staff from which the titles were obtained, or an alternative title may be designated for this position. Generally, the first method is preferable; however, the second method is adaptable to those unit staffs requiring special titles because of unique organizational structures.

2. The ANSI Standard shall be the same ANSI Standard referenced in Section 5.3, Unit Staff Qualifications. If alternative titles are used, all requirements of these Technical Specifications apply to the position with the alternative title as apply with the specified title. Unit staff titles shall be specified in the Final Safety Analysis Report or Quality Assurance Plan. Unit staff titles shall be maintained and revised using those procedures approved for modifying/revising the Final Safety Analysis Report or Quality Assurance Plan.

5.2 Organization

5.2.2 Unit Staff

The unit staff organization shall include the following:

a. A non-licensed operator shall be assigned to each reactor containing fuel and an additional non-licensed operator shall be assigned for each control room from which a reactor is operating in MODE 1, 2, 3, or 4.

b. Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i) and 5.2.2.a and 5.2.2 e for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements.

c. A radiation protection technician shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

d. The operations manager or assistant operations manager shall hold an SRO license.

e. An individual shall provide advisory technical support to the unit operations shift crew in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. This individual shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift.

--------------------------------------REVIEWER'S NOTE---------------------------------------
Two unit sites with both units shutdown or defueled require a total of three nonlicensed operators for the two units.

5.0 ADMINISTRATIVE CONTROLS

5.3 Unit Staff Qualifications

5.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of [NRC RG 1.8, Revision 2, 1987, or more recent revisions, or ANSI Standard acceptable to the NRC staff]. [The staff not covered by NRC RG 1.8 shall meet or exceed the minimum qualifications of Regulations, NRC RGs, or ANSI Standards acceptable to NRC staff].

5.3.2 For the purpose of 10 CFR 55.4, a licensed Senior Reactor Operator (SRO) and a licensed Reactor Operator (RO) are those individuals who, in addition to meeting the requirements of Specification 5.3.1, perform the functions described in 10 CFR 50.54(m).

--REVIEWER'S NOTE---

Minimum qualifications for members of the unit staff shall be specified by use of an overall qualification statement referencing an ANSI Standard acceptable to the NRC staff or by specifying individual position qualifications. Generally, the first method is preferable; however, the second method is adaptable to those unit staffs requiring special qualification statements because of unique organizational structures.

5.5 Programs and Manuals

5.5.2 Primary Coolant Sources Outside Containment

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The systems include Containment Spray System, Safety Injection System, Chemical and Volume Control System, Gaseous Waste Management System and Containment Hydrogen Control System. The program shall include the following:

a. Preventive maintenance and periodic visual inspection requirements and

b. Integrated leak rate test requirements for each system at least once per 18 months.

The provisions of SR 3.0.2 are applicable.

5.5.3 Post-Accident Sampling

This program provides controls that ensure the capability to obtain and analyze reactor coolant, radioactive gases, and particulates in plant gaseous effluents and containment atmosphere samples under accident conditions. This program shall include the following:

a. Training of personnel,

b. Procedures for sampling and analysis, and

c. Provisions for maintenance of sampling and analysis equipment.

---REVIEWER'S NOTE--

This program may be eliminated based on the implementation of Topical Report CE NPSD-1157, Rev. 1, "Technical Justification for the Elimination of the Post-Accident Sampling System from the Plant Design and Licensing Basis for CEOG Utilities," and the associated NRC Safety Evaluation dated May 16, 2000, and implementation of the following commitments:

1. [Licensee] has developed contingency plans for obtaining and analyzing highly radioactive samples of reactor coolant, containment sump, and containment atmosphere. The contingency plans will be contained in emergency plan implementing procedures and implemented with the implementation of the License amendment. Establishment of contingency plans is considered a regulatory commitment.

2. The capability for classifying fuel damage events at the Alert level threshold has been established for [PLANT] at radioactivity levels of 300 mCi/cc dose equivalent iodine. This capability may utilize the normal sampling system and/or correlations of sampling or letdown line dose rates to coolant concentrations. This capability will be described in emergency plan implementing procedures and implemented with the implementation of the License amendment. The capability for classifying fuel damage events is considered a regulatory commitment.

3. [Licensee] has established the capability to monitor radioactive iodines that have been released to offsite environs. This capability is described in our emergency plan implementing procedures. The capability to monitor radioactive iodines is considered a regulatory commitment..

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Program (continued)

--------------------------------------REVIEWER'S NOTE---------------------------------------
Alternate tube repair criteria currently permitted by plant technical specifications are listed here. The description of these alternate tube repair criteria should be equivalent to the descriptions in current technical specifications and should also include any allowed accident induced leakage rates for specific types of degradation at specific locations associated with tube repair criteria.

LCO 3.4.12, "RCS Operational LEAKAGE."

c. Provisions for SG tube repair criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40 % of the nominal tube wall thickness shall be plugged.

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

1. Inspect 100 % of the tubes in each SG during the first refueling outage following SG installation.

2. Inspect 100 % of the tubes at sequential periods of 144, 108, 72, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50 % of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50 % by the refueling outage nearest the end of the period. No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected.

--------------------------------------REVIEWER'S NOTE---------------------------------------
Plants are to include the appropriate Frequency (e.g., select the appropriate Item 2.) for their SG design. The first Item 2 is applicable to SGs with Alloy 600 mill annealed tubing. The second Item 2 is applicable to SGs with Alloy 600 thermally treated tubing. The third Item 2 is applicable to SGs with Alloy 690 thermally treated tubing.

5.5 Programs and Manuals

5.5.11 Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems in accordance with NRC RG 1.52, Revision 4, ASME N511-2007, and AG-1-2009 at the system flow

--------------------------------------REVIEWER'S NOTE--------------------------------------

The use of any standard other than ASTM D3803-1989 to test the charcoal sample may result in an overestimation of the capability of the charcoal to adsorb radioiodine. As a result, the ability of the charcoal filters to perform in a manner consistent with the licensing basis for the facility is indeterminate.

ASTM D 3803-1989 is a more stringent testing standard because it does not differentiate between used and new charcoal, it has a longer equilibration period performed at a temperature of 30°C (86°F) and a relative humidity (RH) of 95% (or 70% RH with humidity control), and it has more stringent tolerances that improve repeatability of the test.

Allowable Penetration = [(100% - Methyl Iodide Efficiency * for Charcoal Credited in Licensee's Accident Analysis) / Safety Factor]

When ASTM D3803-1989 is used with 30°C (86°F) and 95% RH (or 70% RH with humidity control) is used, the staff will accept the following:

Safety factor 2 for systems with or without humidity control.

Humidity control can be provided by heaters or an NRC-approved analysis that demonstrates that the air entering the charcoal will be maintained less than or equal to 70 percent RH under worst case design basis conditions.

If the system has a face velocity greater than 110 percent of 0.203 m/s (40 ft/min), the face velocity should be specified.

*This value should be the efficiency that was incorporated in the licensee's accident analysis which was reviewed and approved by the staff in a safety evaluation.

ABCAEES	5,097 cmh (3,000 cfm)

c. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in NRC RG 1.52, Revision 4, shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of 30 °C (86 °F) and the relative humidity specified below:

ESF Ventilation System	Penetration	RH
CREACS	0.5 %	70 %
FHAEES	0.5 %	70 %
ABCAEES	0.5 %	70 %

5.5 Programs and Manuals

5.5.16 Containment Leakage Rate Testing Program (continued)

d. Leakage rate acceptance criteria are:

1. Containment leakage rate acceptance criterion is ≤ 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are < 0.60 L_a for the Type B and C tests and ≤ 0.75 L_a for Type A tests.

2. Air lock testing acceptance criteria are:

i. Overall air lock leakage rate is ≤ 0.05 L_a when tested at ≥ P_a.

ii. For each door, leakage rate is ≤ 0.01 L_a when pressurized to ≥ 10 psig.

e. The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

f. Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J.

5.5.17 Battery Monitoring and Maintenance Program

This Program provides controls for battery restoration and maintenance. The program shall be in accordance with IEEE Standard (Std) 450-2002, “IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications,” as endorsed by NRC RG 1.129, Revision 2 (RG), with RG exceptions and program provisions as identified below:

a. The program allows the following RG 1.129, Revision 2 exceptions:

1. Battery temperature correction may be performed before or after conducting discharge tests.

2. RG 1.129, Regulatory Position 1, Subsection 2, “References,” is not applicable to this program.

3. In lieu of RG 1.129, Regulatory Position 2, Subsection 5.2, “Inspections,” the following shall be used: “Where reference is made to the pilot cell, pilot cell selection shall be based on the lowest voltage cell in the battery.”

-------------------------------------REVIEWER’S NOTE---
This program and the corresponding requirements in LCO 3.8.4, LCO 3.8.5, and LCO 3.8.6 require providing the information and verifications requested in the Notice of Availability for TSTF-500, Revision 2, "DC Electrical Rewrite - Update to TSTF-360," (76FR54510).

5.5 Programs and Manuals

5.5.19 Setpoint Control Program

This program shall establish the requirements for ensuring that setpoints for automatic protective devices are initially within and remain within the assumptions of the applicable safety analyses, provides a means for processing changes to instrumentation setpoints, and identifies setpoint methodologies to ensure instrumentation will function as required. The program shall ensure that testing of automatic protective devices related to variables having significant safety functions as delineated by 10 CFR 50.36(c)(1)(ii)(A) verifies that instrumentation will function as required.

a. The program shall list the Functions in the following specifications to which it applies:

1. LCO 3.3.1, “Reactor Protection System (RPS) Instrumentation – Operating”

2. LCO 3.3.2, “Reactor Protection System (RPS) Instrumentation – Shutdown”

3. LCO [3.3.3, “Control Element Assembly Calculators (CEACs)”]

4. LCO 3.3.5, “Engineered Safety Features Actuation System (ESFAS) Instrumentation”

5. LCO 3.3.7, “Emergency Diesel Generator (EDG) – Loss of Voltage Start (LOVS)”

6. LCO 3.3.8, “Containment Purge Isolation Actuation Signal (CPIAS)”

7. LCO 3.3.9, “Control Room Emergency Ventilation Actuation Signal (CREVAS)”

8. LCO 3.3.10, “Fuel Handling Area Emergency Ventilation Actuation Signal (FHEVAS)”

9. LCO 3.3.13, “Logarithmic Power Monitoring Channels”

b. The program shall require the Nominal Trip Setpoint (NTSP), Allowable Value (AV), As-Found Tolerance (AFT), and As-Left Tolerance (ALT) (as applicable) of the Functions described in paragraph a. are calculated using the NRC approved setpoint methodology, as listed below. In addition, the program shall contain the value of the NTSP, AV, AFT, and ALT (as applicable) for each Function described in paragraph a. and shall identify the setpoint methodology used to calculate these values.

------------------------------------- Reviewer's Note ---------------------------------------
List the NRC safety evaluation report by letter, date, and ADAMS accession number (if available) that approved the setpoint methodologies.

5.5 Programs and Manuals

5.5.19 Setpoint Control Program (continued)

c. The program shall establish methods to ensure that Functions described in paragraph a. will function as required by verifying the as-left and as-found settings are consistent with those established by the setpoint methodology.

d. The program shall identify the Functions described in paragraph a. that are automatic protective devices related to variables having significant safety functions as delineated by 10 CFR 50.36(c)(1)(ii)(A). These Functions shall be demonstrated to be functioning as required by applying the following requirements during CHANNEL CALIBRATIONS and CHANNEL FUNCTIONAL TESTS that verify the NTSP.

1. The as-found value of the instrument channel trip setting shall be compared with the previous as-left value or the specified NTSP.

2. If the as-found value of the instrument channel trip setting differs from the previous as-left value or the specified NTSP by more than the pre-defined test acceptance criteria band (i.e., the specified AFT), then the instrument channel shall be evaluated before declaring the SR met and returning the instrument channel to service. This condition shall be entered in the plant corrective action program.

3. If the as-found value of the instrument channel trip setting is less conservative than the specified AV, then the SR is not met and the

4.

----------------------------------REVIEWER'S NOTE-------------------------------------

A license amendment request to implement a Setpoint Control Program must list the instrument functions to which the program requirements of paragraph d. will be applied. Paragraph d. shall apply to all Functions in the Reactor Protection System and Engineered Safety Feature Actuation System specifications unless one or more of the following exclusions apply:

1. Manual actuation circuits, automatic actuation logic circuits or to instrument functions that derive input from contacts which have no associated sensor or adjustable device, e.g., limit switches, breaker position switches, manual actuation switches, float switches, proximity detectors, etc. are excluded. In addition, those permissives and interlocks that derive input from a sensor or adjustable device that is tested as part of another TS function are excluded.

2. Settings associated with safety relief valves are excluded. The performance of these components is already controlled (i.e., trended with as-left and as-found limits) under the ASME Code for Operation and Maintenance of Nuclear Power Plants testing program.

3. Functions and Surveillance Requirements which test only digital components are normally excluded. There is no expected change in result between SR performances for these components. Where separate as-left and as-found tolerance is established for digital component SRs, the requirements would apply.

5.6 Reporting Requirements

5.6.4 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heatup, cooldown, low temperature operation, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

3.4.3, "RCS Pressure and Temperature (P/T) Limits"

3.4.6, "RCS Loops – MODE 4"

3.4.7, "RCS Loops – MODE5 (Loops Filled)"

3.4.10, "Pressurizer Pilot Operated Safety Relief Valves (POSRVs)"

3.4.11, "Low Temperature Overpressure Protection (LTOP) System"

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

APR1400-Z-M-NR-14008-P, "Pressure-Temperature Limits Methodology for RCS Heatup and Cooldown."

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

---REVIEWER'S NOTE---

The methodology for the calculation of the P-T limits for NRC approval should include the following provisions:

1. The methodology shall describe how the neutron fluence is calculated (reference new Regulatory Guide when issued).
2. The Reactor Vessel Material Surveillance Program shall comply with Appendix H to 10 CFR 50. The reactor vessel material irradiation surveillance specimen removal schedule shall be provided, along with how the specimen examinations shall be used to update the PTLR curves.
3. Low Temperature Overpressure Protection (LTOP) System lift setting limits for the Power Operated Relief Valves (PORVs), developed using NRC approved methodologies may be included in the PTLR.
4. The adjusted reference temperature (ART) for each reactor beltline material shall be calculated, accounting for radiation embrittlement, in accordance with Regulatory Guide 1.99, Revision 2.
5. The limiting ART shall be incorporated into the calculation of the pressure and temperature limit curves in accordance with NUREG-0800 Standard Review Plan 5.3.2, Pressure-Temperature Limits.
6. The minimum temperature requirements of Appendix G to 10 CFR Part 50 shall be incorporated into the pressure and temperature limit curves.
7. Licensees who have removed two or more capsules should compare for each surveillance material the measured increase in reference temperature (RT_{NDT}) to the predicted increase in RT_{NDT}; where the predicted increase in RT_{NDT} is based on the mean shift in RT_{NDT} plus the two standard deviation value (2∂) specified in Regulatory Guide 1.99, Revision 2. If the measured value exceeds the predicted value (increase in $RT_{NDT} + 2\partial$), the licensee should provide a supplement to the PTLR to demonstrate how the results affect the approved methodology.

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Question No. 16-220

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The Writer's Guide for Plant-Specific Improved Technical Specifications (TSTF-GG-05-01) also provides guidance for the format and content of the TS. There are format and content differences between the DCD and the Writer's Guide. These following corrections are necessary to ensure the completeness and accuracy of the TS and Bases.

Clarify the text in Technical Specification (TS) 5.5.5, which deviates from the STS.

The STS reads "…to track the FSAR, Section [ ], cyclic…" The APR1400 text reads "…to track the Chapter 3, cyclic…" The intended reference should include the document and the specific section/chapter, not just the chapter.

This clarification is required to ensure that all references to documents are done accurately and to align the text with the STS.

Response

Section 5.5.5 will include reference as specific section/chapter, not just the chapter as shown in attached markup.
Also, APR1400 DCD16 format and contents will be revised to align the Writer's Guide for Plant-Specific Improved Technical Specifications (TSTF-GG-05-01).

Impact on DCD

Same as changes described in Impact on Technical Specifications section.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

DCD 16 Section 5.5.5 will be revised as shown in attached markup.

Impact on Technical/Topical/Environmental Report

There is no impact on any Technical, Topical, or Environment Reports.

FSAR Section 3.9 and Table 3.9-1

5.5 Programs and Manuals

5.5.5 Component Cyclic or Transient Limit

This program provides controls to track the ~~Chapter 3~~, cyclic and transient occurrences to ensure that components are maintained within the design limits.

5.5.6 Pre-Stressed Concrete Containment Tendon Surveillance Program

This program provides controls for monitoring any tendon degradation in prestressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial operations. The Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with Section XI, Subsection IWL of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50.55a, except where an alternative, exemption, or relief has been authorized by the NRC.

The provisions of SR 3.0.3 are applicable to the Tendon Surveillance Program inspection frequencies.

5.5.7 Reactor Coolant Pump Flywheel Inspection Program

This program shall provide for the inspection of each reactor coolant pump flywheel per the recommendation of regulatory position c.4.b of NRC RG 1.14, Revision 1, August 1975.