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TECHNOLOGIES INC.

March 24, 2005

Our File: 108US-01321-021-001
108US-ACNU05-0017L
Your File: Project No. 722

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

Attention: Ms. A. Cubbage
Project Manager, ACR

Re: **Submittal of Report 108US-30000-LS-001 on Reactor Coolant Pressure Boundary**

Enclosed for NRC staff review as part of the ACR-700 pre-application scope of work for Design Certification is the following report:

"Codes, Standards and Acceptance Criteria For ACR-700 Reactor Coolant Pressure Boundary and On-Line Fueling Components and Systems", 108US-30000-LS-001, Rev. 0

The ACR-700 reactor and on-line fuel handling systems have unique characteristics relative to the power reactor designs licensed in the US. The purpose of the Codes, Standards and Acceptance Criteria document enclosed with this letter is to present to the NRC a requirements framework for the following systems and components comparable to the NRC's Standard Review Plan (NUREG-0800) and demonstrating compliance with the US Regulatory requirements established in 10 CFR 50:

- fuel channel (including pressure tube (PT));
- fuel handling system;
- calandria and end shield assembly; and
- reactivity control units (control rods).

Our expectations for the NRC's review of the Codes, Standards and Acceptance Criteria Report are as follow:

1. AECLT is requesting a Safety Evaluation Report (SER) for the specific issues identified in Item 2 below. In addition, for the components discussed in Item 3 below, AECLT is requesting an assessment analogous to the recently issued Pre-Application Safety Assessment Report (PASAR). Our expectation is that NRC staff will issue RAIs, and following receipt and review of our responses, NRC staff will issue an SER for the issues identified in item 2 below and an assessment report for the issues identified in item 3 below. We request that the SER and assessment report be issued by September 2005.

2. AECLT requests that an SER be prepared, which will contain the specific issues and conclusions identified as follows:

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a. The design of the ACR-700 Pressure Tubes and End Fittings meets the primary pressure boundary design requirements of 10CFR50.55a through the use of the ASME Code and the additional design requirements contained in this report and its references.

b. The design of the ACR-700 Reactor Coolant Pressure Boundary components, specifically the Pressure Tube, End Fitting, Closure Plug and Fueling Machine, satisfies the acceptance criteria of the Standard Review Plan sections 5.2.1.1, 5.2.1.2, 5.2.2, 5.2.3, 5.2.4, 5.2.5, 5.3.1 and 5.3.2.

c. The Code Classification of the Reactor Coolant Pressure Boundary components, specifically the Pressure Tube, End Fitting, Closure Plug, Fueling Machine and interfacing water systems are consistent with the guidance contained in Regulatory Guide 1.26.

3. AECLT is seeking agreement on the proposed application of the NRC's Standard Review Plan to the Calandria, End Shield and Reactivity Control Units. AECLT is not requesting an SER addressing these components.

In addition to the enclosed report, three attachments to this letter address issues raised by the NRC in the PASAR. Only those issues relevant to the enclosed report are included.

I propose that we arrange a technical meeting to walk NRC staff through the enclosed document, provide clarification on areas needed to facilitate NRC staff review, describe how the contents of the report address items identified in the PASAR, and provide any needed clarification of our expectations of NRC staff review of the document.

If you have any questions regarding this letter please contact me at (301) 332-9152.

Yours sincerely,

Glenn H. Archinoff
Manager ACR Licensing

/Enclosures:

1. "Codes, Standards and Acceptance Criteria For ACR-700 Reactor Coolant Pressure Boundary and On-Line Fueling Components and Systems", 108US-30000-LS-001, Rev. 0

/Attachments:

1. AECL's Response to Section 1 of the NRC's Pre-Application Safety Assessment Report (PASAR) for the ACR-700 – Class 1 Pressure Boundary Design
2. AECL's Response to Section 5 of the NRC's Pre-Application Safety Assessment Report (PASAR) for the ACR-700 - Canadian Design Codes and Quality Assurance Standards
3. AECL's Response to Section 7 of the NRC's Pre-Application Safety Assessment Report (PASAR) for the ACR-700 - On-Power Fueling

ATTACHMENT 1
(Letter G. Archinoff to A. Cubbage, "Submittal of Report 108US-30000-LS-001 on Reactor Coolant Pressure Boundary", March 24, 2005)

AECL's Response to Section 1 of the NRC's Pre-Application Safety Assessment Report (PASAR) for the ACR-700 – Class 1 Pressure Boundary Design

This attachment includes AECL's responses to specific issues identified in Section 1 (Focus Topic 1 – Class1 Pressure Boundary Design) of the NRC's Pre-Application Safety Assessment Report (PASAR) for the ACR-700. Items addressed herein are those PASAR items and issues that are related to reports "Codes, Standards and Acceptance Criteria for ACR-700 Reactor Coolant Pressure Boundary and On-Line Fueling", 108US-30000-LS-001 and "ACR-700 Pressure Tubes Integrity", 108US-31110-LS-001[1].

1. **Section 1.2, Page 1-4, Regulatory Review**

   AECL provided a description of the extent of the RCPB in a March 31, 2004, submission to the NRC ... Figure 91-1 of the submission shows the extent of the Class 1 RCPB when the fueling machine is not attached. The RCPB includes reactor components, such as the PTs, end fittings, and closure plugs. The RCPB also includes those portions of the major components in contact with the reactor coolant, such as the steam generators, pumps, pressurizer, and connecting piping. Some of the piping attached to the main coolant loop show the Class 1 designation only out to the first isolation valve. According to AECL, some of the valve classifications currently shown for the ACR-700 may change to meet U.S. regulations, as stated above.

   AECL Response

   Double isolation valves have been included in the ACR design where necessary for compliance with US regulations for the RCPB. This feature is now shown on Figure 2-2 of 108US-30000-LS-001. Also, double isolation valves have been included in small piping connected to the fueling machine (Figure 2-3) that is considered part of the RCPB.

2. **Section 1.2, Page 1-4, Regulatory Review**

   The ASME Code Class 1 piping rules do not allow the use of threaded joints without an additional pressure boundary sealing mechanism.

   AECL Response

   Threaded joints in the ACR design have been equipped with related seals that allow the design to meet ASME Code Class 1 piping rules. This is documented in AECL Report, "Codes, Standards and Acceptance Criteria for ACR-700 Reactor Coolant Pressure Boundary and On-Line Fueling", 108US-30000-LS-001 Section 5.1.2.2.

1 AECL Report "ACR-700 Pressure Tubes Integrity", 108US-31110-LS-001 is designated PROTECTED-Proprietary and is submitted to the NRC under separate cover.

3. **Section 1.2, Page 1-4, Regulatory Review**

AECL further indicated that parts of the fueling machine pressure boundary, not designated as Class 1, are acceptable, according to the definition of the RCPB provided in 10 CFR 50.55a(c). In order to satisfy this requirement, AECL should either extend the Class 1 boundary to the second isolation valve for each line attached to the RCPB, or demonstrate that coolant loss from failure of the line will not exceed the normal makeup capacity. Application of 10 CFR 50.55a(c)(2) for small-diameter penetrations will require further review during design certification.

AECL Response

Double isolation valves are included in small piping connected to the fueling machine that is considered part of the RCPB (see Figure 2-3 of 108US-30000-LS-001). All materials used in the fueling machine Class 1 pressure boundary are ASME –accepted Class 1 materials.

4. **Section 1.2, Page 1-4, Regulatory Review**

As indicated above, safety classification of the ACR-700 RCPB components is an issue that will require further review during the design certification. AECL must demonstrate that the RCPB classification satisfies the requirements of 10 CFR 50.55a(c)(2).

AECL Response

This requirement is addressed in AECL Report, "Codes, Standards and Acceptance Criteria for ACR-700 Reactor Coolant Pressure Boundary and On-Line Fueling", 108US-30000-LS-001 Section 5.1.2.1.

5. **Section 1.2, Page 1-5, Regulatory Review**

The applicant's March 31, 2004, submission indicates that it will use the 2001 edition of the ASME Code to design the RCPB piping. The staff will determine applicable ASME Code editions and addenda at the time of design certification, in accordance with the requirements of 10 CFR 50.55a(c)(3).

AECL Response

RCPB piping design will comply with the ASME Code Edition and addenda as specifically incorporated by reference by the NRC in 10 CFR 50.55a(b)(1).

6. **Section 1.4, Page 1-6, Technical Issues**

In order to satisfy GDC 14, AECL must demonstrate that the rolled joint remains leaktight throughout its service life.

AECL Response

This is addressed in AECL Reports, "Codes, Standards and Acceptance Criteria for ACR-700 Reactor Coolant Pressure Boundary and On-Line Fueling", 108US-30000-LS-001 Sections 2.1.3 and 5 and "ACR-700 Pressure Tubes Integrity", 108US-31110-

LS-001, Sections 1.2.2 and 7.6.

7. **Section 1.4, Page 1-6, Technical Issues**

The performance testing of the prototype joints must be performed in conditions that simulate actual service conditions in order to demonstrate that the joint can remain leaktight throughout its design life.

AECL Response

Qualification testing of prototype joints is performed using temperature cycling and helium leak checks, which are more severe than water leak checks. AECL's successful history of continuous active monitoring has demonstrated the adequacy of the design and the qualification methods used. AECL documents 108US-30000-LS-001, Sections 1.2.2 and 5.2, and 108US-31110-LS-001 provide justification for these qualification methods.

8. **Section 1.4, Page 1-6, Technical Issues**

As stated previously, CAN/CSA-N285.2 requires that the PT end fitting rolled joints be designed in accordance with the requirements of Section III, paragraph NB-3200, of the ASME Code. Several technical issues related to the design of the PTs and end fittings must be resolved during design certification. These technical issues include the effect of the reactor coolant environment on the fatigue life of the PTs and end fittings, the creep behavior of the PTs, and the potential for delayed hydride cracking (DHC) of the PTs. Section 1.9.3 of this PASAR provides an additional discussion of issues related to creep and DHC of the PTs.

AECL Response

The effects of reactor coolant system environment on the PT, End Fitting and Rolled Joint are addressed in 108US-30000-LS-001 Section 5.1.1.4 and 108US-31110-LS-001 Section 6.

9. **Section 1.4, Page 1-6, Technical Issues**

Paragraph NB-3200 of Section III of the ASME Code requires an evaluation of the PTs and end fittings for cyclic operation. Although the prototype testing of the rolled joint discussed above can provide additional assurance that the joint will remain leaktight during service conditions, it will not demonstrate that ASME Code design criteria have been satisfied for the PT or end fitting. Therefore, an ASME Code evaluation for cyclic operation is necessary. CAN/CSA-N285.2 requires development of irradiated fatigue curves for the evaluation of the PTs, since the existing ASME Code design fatigue curves do not apply to the irradiated PT material.

AECL Response

Past operating experience for CANDU reactors demonstrates that the pressure tube design complies with applicable requirements. This is discussed in 108US-30000-LS-

001 and 108US-31110-LS-001. Fatigue curves will be developed for the ACR and provided to the NRC staff for review when AECL applies for Design Certification of the ACR design.

10. **Section 1.4, Page 1-7, Technical Issues**

AECL needs to develop fatigue curves that account for the conditions of the ACR PT environment, including oxygen level, temperature, and radiation effects. Development of design fatigue curves that apply to the ACR environmental conditions is a topic that will require further review during design certification.

AECL Response

See AECL Response to Issue 9 in this document.

11. **Section 1.4, Page 1-7, Technical Issues**

The staff notes that AECL did not consider environmental effects in the fatigue design for the end fitting and closure plug material. The current ASME Code does not take into account the reduction in fatigue life caused by the environmental conditions of the reactor water. Austenitic SSs and some nickel-alloy components have been observed to experience a significant reduction in fatigue life at low levels of oxygen in pure water in experiments conducted under LWR conditions. The effect of the ACR-700 coolant environment on the fatigue life of the end fittings and closure plugs will require review during design certification.

AECL Response

Appropriate fatigue curves will be developed for the ACR and included in the DCD when AECL applies for Design Certification of the ACR design.

12. **Section 1.4, Page 1-7, Technical Issues**

The FCTR indicates that the tensile strength of the PT material also increases with irradiation. The report further indicates that credit is taken for part of the strength increase caused by irradiation in order to maintain the intended design margins at the end-of-design-life conditions. The FCTR does not specify the amount of strengthening that will be credited in the evaluation. No staff guidance is applicable to the review of radiation strengthening of pressure boundary material. This topic will require further review during design certification.

AECL Response

ACR design analysis does not credit the increase in pressure tube tensile strength due to irradiation. Refer to 108US-31110-LS-001 Section 6.8.1.

13. **Section 1.4, Page 1-8, Technical Issues**

Although the load combinations and stress limits proposed by AECL are consistent with the guidance provided in SRP Section 3.9.3, the stress limits for earthquake loads appear to conflict with the criteria referenced in the FCTR. The FCTR indicates that a more conservative stress limit is used for the seismic design of the fuel channel assembly. The stress limit used for load combinations that include earthquake loads will require clarification during design certification.

AECL Response

For the seismic design of the fuel channel assembly, the Level D stress limit will be used in the DCD for load combinations that include earthquake loads (see AECL report 108US-30000-LS-001).

14. **Section 1.4, Page 1-8, Technical Issues**

AECL also indicated that the ASME special stress limits in ASME Code, Section III, paragraph NB-3227.3, applicable to progressive distortion of nonintegral connections, will be satisfied for the PT/end fitting rolled joints. The rolled joint relies on the residual compressive stresses between the PT and end fitting produced by the rolling process to ensure a leaktight joint. The staff believes that this joint constitutes a nonintegral connection. A nonintegral connection is subject to loosening of the mating parts if the material yields under the applied loads. The ASME Code provision limits the primary plus secondary stress intensity to the material yieldstress to prevent progressive loosening of the mating parts. The staff considers it important that these stress limits be satisfied for all load combinations to assure that the rolled joints remain leaktight under all postulated design conditions.

AECL Response

As confirmed in 108US-30000-LS-001 Section 5.1.1.1.3, the special stress limits in paragraph NB-3227.3 will be applied to the design of the PT/end fitting rolled joints.

15. **Section 1.7.2, Page 1-13, 10 CFR 50.55a, "Codes and Standards"**

Some of the materials specified in the ASME Code could potentially be suitable for use as PTs or end fittings. If the ASME Code does specify materials that are suitable, and AECL is proposing alternative materials, the staff will review the use of these non-ASME Code materials in the ACR-700 as alternatives to the ASME Code, Section III, requirements pursuant to 10 CFR 50.55a(a)(3)(i). If the ASME Code does not specify any materials suitable for use as PTs or end fittings, the staff will consider the ASME Code requirements for the materials selection for PTs and/or end fittings as not relevant to the ACR-700 design. As part of design certification, AECL should provide the basis for its determination of which of these avenues is correct for both the PTs and end fittings.

AECL Response

The end-fitting material is an approved ASME Code case, as discussed in 108US-30000-LS-001 Section 4.1.4. The pressure tube materials are as discussed in 108US-31110-LS-001.

16. **Section 1.7.2, Page 1-13, 10 CFR 50.55a, "Codes and Standards"**

As part of the design certification review, AECL will need to identify to the NRC whether there are any additional areas in which 10 CFR 50.55a contains rules that are not technically relevant to the ACR-700 design. Such areas may be documented as not technically relevant, in accordance with 10 CFR 52.48. However, if it is determined that these areas are not in compliance with the 10 CFR 50.55a requirements, the staff can authorize alternatives to these requirements. This regulatory issue will need to be addressed during the design certification review.

AECL Response

Report 108US-30000-LS-001 includes a review against the design and inspection requirements of 10 CFR 50.55a for primary pressure boundary components. Those cases where additional acceptance criteria are required to supplement the ASME Code are identified in 108US-30000-LS-001 Section 4 and evaluated in Section 5.

17. **Section 1.7.2, Page 1-14, 10 CFR 50.55a, "Codes and Standards"**

AECL relies upon CAN/CSA-N285.4, "Periodic Inspection of CANDU Nuclear Power Components," for minimum inspection requirements for pressure boundary components, such as fuel channel components, not addressed by ASME Code, Section XI. The staff requests that during the design certification review, AECL explain the Canadian standards and the basis for their acceptability. The staff's review of this information may lead to technical issues not yet identified. Resolution of these issues may require the NRC to develop limitations or additional requirements.

AECL Response

AECL documents 108US-30000-LS-001 and 108US-31110-LS-001 provide an explanation of the minimum inspection requirements for pressure boundary components that are not addressed by ASME Section XI, and the basis for their acceptability.

The COL applicant will develop detailed in-service testing and in-service inspection programs. These IST and ISI programs will comply with 10 CFR 50.55a(f) and (g) and will include the additional requirements from 108US-31110-LS-001. The COL applicant will follow one of the processes specified in Section 50.55a for departures from the ASME Code, or will request an exemption under 10 CFR 50.12.

18. **Section 1.7.2, Page 1-14, 10 CFR 50.55a, "Codes and Standards"**

Accordingly, Canadian standards adopted to address those technical areas for which the ASME Code rules are either not applicable or have to be supplemented will need an appropriate level of regulatory control. For example, the Canadian standards with limitations or additions, as appropriate, could be incorporated by reference as requirements in the design certification appendix to 10 CFR Part 52 applicable to the ACR-700. This approach would provide regulatory control by making the Canadian standards, with limitations or additions as appropriate, NRC requirements.

AECL Response

In cases where the use of Canadian Standards is required, the following action plan is proposed:

1. *The DCD will not directly refer to a Canadian standard to satisfy 10 CFR 50.55a.*
2. *The detailed requirements from Canadian standards will be established in separate licensing submittals that can be referenced in the DCD (Referenced Reports).*
3. *Referenced Reports will be identified in the DCD as Tier 2 material.*

19. **Section 1.7.2, Page 1-14, 10 CFR 50.55a, "Codes and Standards"**

The areas of authorization of alternatives, requests for relief, and ISI program updates for the ACR-700 present unique challenges that the staff may need to address during the design certification review. Specifically, the NRC has not addressed the issue of using and updating Canadian standards to regulate ISI of the reactor assembly and on-power fueling system. It may be appropriate to discuss how AECL proposes that a COL applicant would obtain regulatory approval of alternatives to and relief from Canadian standards, and address revised ISI requirements in updated Canadian standards applied to periodic inspection of the ACR-700 reactor assembly and on-power fueling system. If the design certification appendix applicable to the ACR-700 codifies the Canadian standards, the staff may need to consider incorporating provisions in the design certification rule for authorizing alternatives to, and granting relief from, Canadian standards that are adopted as NRC requirements. The Canadian standards are updated and reissued periodically, although not as frequently as the ASME Code. Further, if the design certification appendix applicable to the ACR-700 codifies the Canadian standards, the staff may need to consider incorporating a mechanism for periodically updating those requirements of the Canadian standards that are incorporated by reference into the NRC regulations.

AECL Response

AECL documents 108US-30000-LS-001 and 108US-31110-LS-001 provide an explanation of the minimum inspection requirements for pressure boundary components that are not addressed by the ASME Code, Section XI, and the basis for their acceptability.

The COL applicant will develop detailed in-service testing and in-service inspection programs. These IST and ISI programs will comply with 10 CFR 50.55a(f) and (g) as well as report 108US-31110-LS-001 Section 7. Where the IST and ISI programs rely on CSA standards, the COL applicant will follow one of the processes specified in Section 50.55a for departures from the ASME Code, or will request an exemption under 10 CFR 50.12.

20. **Section 1.7.2, Page 1-15, 10 CFR 50.55a, "Codes and Standards"**

As an alternative to option 1, specific Canadian standards with limitations and additions as agreed upon by the NRC could be identified as Tier 2* material. This Tier 2* approach would provide regulatory control by requiring the licensee to obtain NRC approval to change/modify these standards. The Tier 2* approach would also permit the 10 CFR Part 52 licensee amendment process to be used for the approval of alternatives, granting of relief, or owner updating of the applicable edition of the standard.

AECL Response

See AECL's response to issue 18 in this document.

21. **Section 1.7.2, Page 1-15, 10 CFR 50.55a, "Codes and Standards"**

As another alternative, Tier 2* material could include specific provisions from the CAN/CSA-N Series standards that are related to the design and inspection requirements for those areas in which ASME Code Sections III and XI rules are not applicable or need to be supplemented. The staff refers to this alternative as option 3. This option differs from option 2 in that the Tier 2* material would include the specifics of design and inspection based on the CAN/CSA-N Series standards, with limitations and additions agreed to by the NRC; in option 2, the Tier 2* material would include limitations and additions to referenced CAN/CSA-N Series standards.

AECL Response

See AECL's response to issue 18 in this document.

22. **Section 1.7.3, Page 1-16, 10 CFR 50.60, 10 CFR 50.61, and Appendices G and H to 10 CFR Part 50**

However, under the assumption that 10 CFR 50.60, 10 CFR 50.61, and Appendices G and H to 10 CFR Part 50 would legally apply to the ACR-700 design, the NRC staff proceeded with a review of the technical relevance of these regulations to the ACR-700 design.

AECL Response

Given the scope of the above regulations, AECL proposes the following rationale be used for establishing their applicability:

- *10CFR50.61 is clearly specified for 'reactor vessels' only and is therefore not applicable to fuel channel reactors.*
- *10CFR50.60 simply invokes Appendix G and H and is specified for all 'light water reactors' and as such, it is applicable. The ACR-700 shall satisfy this regulation to the extent that Appendix G and H are applicable to fuel channel reactors.*
- *Appendix G is applicable to all Ferritic components of the ACR-700 RCPB. It is concluded that meeting the ASME Section III fracture toughness rules*

satisfies the requirements of Appendix G. Such rules are contained in part NB-2000 of the Code. The additional requirements stated in Appendix G relating specifically to 'reactor vessels' are considered not applicable to the ACR-700.

- *Appendix H is only for 'reactor vessels' and is therefore not applicable to the ACR-700.*

In addition, although 10CFR50.61 is not directly applicable, a material surveillance and testing program is proposed as an alternative to 10CFR50.61 as described in report 108US-31110-LS-001 Section 7.

23. **Section 1.7.3, Page 1-18, 10 CFR 50.60, 10 CFR 50.61, and Appendices G and H to 10 CFR Part 50**

To the extent that Appendix G to 10 CFR Part 50 requires that all ferritic materials (per the definition of ferritic materials given in Appendix G to 10 CFR Part 50) of a nuclear power reactor's RCPB meet the fracture toughness requirements of ASME Code Section III, Division 1, this requirement applies to all such materials incorporated into the ACR-700 design. The regulatory provisions of 10 CFR 50.60, inasmuch as they invoke these requirements in Appendix G to 10 CFR Part 50 would also apply to the ACR-700 design review.

AECL Response

See AECL's response to issue 22 in this document.

24. **Section 1.7.3, Page 1-19, 10 CFR 50.60, 10 CFR 50.61, and Appendices G and H to 10 CFR Part 50**

Therefore, based upon the requirements specified in GDC 14, 15, 30, 31, and 32, the staff has identified a need to develop appropriate review guidance and requirements related to maintaining the integrity of these components in light of their service environment. For the zirconium-alloy PTs, this review guidance should be integrated with guidance regarding the evaluation of other known potential degradation mechanisms (e.g., DHC) which may not require exposure to significant radiation fields to become active. For the 403 SS end fittings, the effect of the thermal environment should be considered, along with the existing radiation fields, when evaluating potential material property changes. The staff is considering the definition of a material surveillance program similar to that described in Appendix H to 10 CFR Part 50 but which may include the periodic testing of zirconium-alloy PTs and 403 SS PT end fittings that have been removed from service in lieu of surveillance specimens. In addition, the staff is also considering operating limits for these components, similar in intent to those specified in Sections IV.A.1 and IV.A.2 of Appendix G to 10 CFR Part 50, as well as protection from other design-basis events (DBEs), like those addressed in 10 CFR 50.61 for typical LWR vessels. The emphasis of the staff's review guidance, with respect to the protection of the PTs and PT end fittings from DBEs beyond normal operation and AOOs, should be to achieve adequate assurance that the probability of consequential multiple PT/end fitting ruptures, to the extent that a beyond-design-basis condition is achieved, is extremely low.

AECL Response

The material surveillance program proposed by AECL is described in 108US-30000-LS-001 Section 5 and 108US-31110-LS-001 Section 7.

25. **Section 1.7.3, Page 1-19, 10 CFR 50.60, 10 CFR 50.61, and Appendices G and H to 10 CFR Part 50**

It is anticipated that, like the material surveillance program discussed above for the zirconium-alloy PTs, a material surveillance program for the Zircaloy-4 calandria tubes would have an intent similar to a material surveillance program for typical LWR vessels, as described in Appendix H to 10 CFR Part 50.

AECL Response

The calandria tubes are not part of the reactor coolant pressure boundary and the material is not ferritic. The calandria tubes therefore do not fall within the intent of Appendix H, hence a material surveillance program is not required. Furthermore, any calandria tube leaks, as well as pressure tube leaks, will be detected by the on-line monitoring of the annulus gas system for moisture.

26. **Section 1.9.2, Page 1-20, Reduction in Creep Life from Environment Effects**

The FCTR describes extensive creep testing, including thermal creep, irradiation creep, and irradiation growth. This testing relates to maintaining the integrity of the RCPB and, hence, to ensuring that various regulations, including Appendix A to 10 CFR Part 50, are met. In this regard, the staff is concerned with the lack of available test data that compares creep behavior of the component materials in typical heavy water (including impurities, such as ppb to ppm levels of oxygen) to that in typical light water (including oxygen and other impurities) and in air. The coolant environment (including impurities) could reduce the creep life of components, as has been observed for fatigue life (see the discussion on fatigue in Section 1.4 of this report).

AECL Response

Irradiation creep and corrosion experiments have been conducted both in light and heavy water. There is no observable differences in either creep or corrosion properties (Section 6.4 of 108US-31110-LS-001) in these medium. Examination of removed pressure tubes from operating CANDU reactors shows that, other than surface oxidation, there is no indication any adverse effect would reduce the creep life of the pressure tube in a water or an annulus gas environment. Although the oxide layer can crack, the cracks do not extend to the metal, and they do not induce cracking in the metal. Creep ductility is discussed in Section 6.3 of 108US-31110-LS-001.

27. **Section 1.9.2, Page 1-20, Reduction in Creep Life from Environment Effects**

During the design certification, AECL will need to provide the technical bases to demonstrate that analyses and testing have adequately considered the environmental effects on creep design noted above.

AECL Response

See AECL's response to issue 26 in this document.

28. **Section 1.9.3.3, Page 1-22, Hydrides at Inlet and Outlet in Pressure Tube**

AECL will need to identify those areas where the hydrogen concentration does exceed the solubility limit, and will need to demonstrate that DHC will not occur in those locations. AECL will need to demonstrate during the design certification review that normal operating stresses (e.g., channel vibration, startup, and shutdown), transient stresses, and DBA stresses will not exceed the DHC threshold stress. This is discussed in more detail below.

AECL Response

This subject is addressed in AECL report 108US-31110-LS-001 Section 6.4 to 6.7.

29. **Section 1.9.3.3, Page 1-23, Hydrides at Inlet and Outlet in Pressure Tube**

Operating and transient stresses, such as those associated with startup and shutdown, are generally low. The CAN/CSA-N285.2 Standard requires that the maximum tensile stresses under design level A (operating) and level B (upset), plus the maximum initial residual tensile stress, will not exceed 67 percent of the tensile stress required to initiate DHC in laboratory tests of unnotched specimens (i.e., 450 MPa based upon tests of irradiated specimens). The staff is unclear about the limits under level C and D conditions.

AECL Response

Level C and Level D service limits are similar to those in ASME. Slow crack growth mechanisms such as Delayed Hydride Cracking are unlikely to play any significant role in fracture during a limited number of level C and D events.

30. **Section 1.9.3.3, Page 1-23, Hydrides at Inlet and Outlet in Pressure Tube**

The staff is concerned that the hydrogen concentration in the PT which exceeds the hydrogen solubility will move inboard of the compressive region after long reactor operating times, leading to the potential for shearing of the PT by DHC just inboard of the end fittings. For example, during a seismic event with the fueling machine attached, the staff can envision peak bending moments and consequent tensile stresses in the PT near the inboard end of the end fitting. During the design certification phase, the staff will evaluate the AECL model and CAN/CSA-N285.2 analysis to determine whether meeting the CAN/CSA Code satisfies these staff concerns regarding DHC of the PTs.

AECL Response

For earthquake loads, slow crack growth mechanisms are unlikely to play a significant role in fracture. See AECL's response to issue 29 in this document.

31. **Section 1.9.4, Page 1-24, Safety Margins**

The staff understands that information exists or is being assembled on the different degradation processes and modes at the higher temperatures and in the appropriate coolant environments. However, the staff considers this general area of safety margins to be an open issue. This issue relates to maintaining the integrity of the RCPB and, hence, to ensuring that various regulations, including Appendix A to 10 CFR Part 50, are met. During the design certification, the staff will need to understand how AECL determines margin to failure to account for uncertainties. For example, Appendix 5 to Section II of the ASME Code requires creep rate and creep rupture strength data at 50 °C (90 °F) intervals to 50 °C (90 °F) above the maximum intended use temperature. AECL has no plans to conduct creep at design temperature plus 50 °C (90 °F). AECL will need to present the technical basis to demonstrate that there is adequate safety margin to account for uncertainties in creep analysis, such as heat-to-heat variability, data scatter, and model uncertainties.

AECL Response

As discussed in AECL report 108US-31110-LS-001 Section 6.3, creep rupture is not expected to be a concern. Irradiation creep and growth is believed to be superplastic, which means that very large strain will occur before necking and fracture. Past inspections of removed pressure tubes have not indicated any necking, which would indicate any creep rupture problems. Further details are available in report 108US-31110-LS-001 Section 7 and Appendix A.

32. **Section 1.9.8, Page 1-26, Garter Spring Control**

The issue of potential DHC at garter spring contact locations relates to maintaining the integrity of the RCPB and, hence, to ensuring that various regulations, including Appendix A to 10 CFR Part 50 are met. During the design certification, the staff will evaluate the AECL analysis of the PT temperature at locations where garter spring spacers contact the PT to ensure that DHC at these locations in not a concern.

AECL Response

This issue is addressed in report 108US-31110-LS-001 Section 6.5.

ATTACHMENT 2
(Letter G. Archinoff to A. Cubbage, "Submittal of Report 108US-30000-LS-001 on Reactor Coolant Pressure Boundary", March 24, 2005)

AECL's Response to Section 5 of the NRC's Pre-Application Safety Assessment Report (PASAR) for the ACR-700 - Canadian Design Codes and Quality Assurance Standards

This attachment includes AECL's responses to specific issues identified in Section 5 (Focus Topic 6 – Canadian Design Codes and Quality Assurance Standards) of the NRC's Pre-Application Safety Assessment Report (PASAR) for the ACR-700. Items addressed herein are those PASAR items and issues that are related to reports "Codes, Standards and Acceptance Criteria for ACR-700 Reactor Coolant Pressure Boundary and On-Line Fueling", 108US-30000-LS-001 and "ACR-700 Pressure Tubes Integrity", 108US-31110-LS-001[2].

1. **Section 5.2, Page 5-2, Regulatory Issues – Design Codes**

   If Canadian codes and standards are used for design, the design requirements must be compatible with construction practices. It is unclear to what extent AECL will propose Canadian codes and standards for the construction, fabrication, inspection, and testing of plant SSCs and whether they will be compatible with U.S. design codes and standards.

   AECL Response

   The ACR-700 design will comply with the US codes and standards, where they are applicable. In the very few cases where US standards do not address ACR-specific features (e.g. fuel channels and fuel channel inspections), it is proposed that the appropriate parts of Canadian standards be directly included in Tier 2 (i.e. not "codified").

2. **Section 5.2, Page 5-2, Regulatory Issues – Design Codes**

   However, AECL may propose alternatives to the design requirements of 10 CFR 50.55a, pursuant to 10 CFR 50.55a(a)(3). These alternatives may use some or all of the provisions in Canadian codes and standards. In proposing alternatives, AECL must submit the alternative, including the proposed Canadian code and standard, to the NRC and describe which portions of the ASME Code or IEEE standard it will not meet, as well as which provision(s) in the Canadian code and standard it proposes to use in lieu of meeting the ASME Code or IEEE standard. AECL must demonstrate that the alternative provides an acceptable level of safety, or that compliance with the specified ASME Code requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

2 AECL Report "ACR-700 Pressure Tubes Integrity", 108US-31110-LS-001 is designated PROTECTED-Proprietary and is submitted to the NRC under separate cover.

AECL Response

In the cases where the appropriate parts of Canadian standards are proposed, the DCD Tier 2 material will demonstrate that the alternative provides an acceptable level of safety.

3. **Section 5.2, Page 5-2, Regulatory Issues – Design Codes**

   When an ACR-700 system or component is found to be within the scope of the ASME Code or IEEE standard, but the requirements in the ASME Code or IEEE standard do not apply, might not exist, or might be insufficient to address the particular aspect of the ACR-700 design, then AECL should provide the applicable portions of the Canadian code or standard that it used for the design, including any specific technical design methods and acceptance criteria it plans to use. AECL should provide an evaluation of the acceptability of the system or component in conjunction with its design certification application.

   AECL Response

   Where the requirements of the ASME code or IEEE standards do not apply, the appropriate portions of a Canadian code or standard will be included in Tier 2 of the DCD.

4. **Section 5.2, Page 5-3, Regulatory Issues – Design Codes**

   A Canadian code, standard, or design method might be found to be within the scope of 10 CFR Part 50, other than 10 CFR 50.55a, but not meet the specified U.S. regulatory requirement. If such exceptions to the regulations exist, then AECL must submit to the NRC a request for an exemption pursuant to 10 CFR 50.12 in conjunction with its design certification application.

   AECL Response

   Although there is no known requirement for an exemption at this time, this will be done where necessary during the DCD preparation.

5. **Section 5.2, Page 5-3, Regulatory Issues – Design Codes**

   When submitting the design certification application for the ACR-700, AECL should identify the Canadian equivalent codes and standards to those U.S. codes and standards that are referenced in NRC guidance documents, submit those Canadian codes and standards to the NRC, describe how the Canadian code and standard deviates from NRC guidelines, and justify why the Canadian code or standard is acceptable.

   AECL Response

   Deviations from US codes and standards will be identified and a justification as to why they are acceptable will be provided at the time of DCD submission.

6. **Section 5.2, Page 5-3, Regulatory Issues – Design Codes**

Numerous codes and standards are used in nuclear power plant designs that are not required by regulations nor referenced in NRC guidance documents. Because no regulatory requirements or guidance exist, the Canadian equivalent to these codes and standards may be used without the need for NRC staff review and approval. The applicant should identify these Canadian codes and standards in design and procurement specifications, as well as in construction and installation specifications.

AECL Response

The Canadian design, procurement, construction, and installation codes and standards not required by NRC regulations will be identified in design and procurement specifications. However, for work done in the US, comparable US codes and standards will also be identified and used to the maximum extent possible.

7. **Section 5.4, Page 5-4, Technical Issues – Design Codes**

For the design of safety-related piping systems in the ACR-700, AECL proposed to use the 2001 edition up to and including the 2003 addenda of ASME Code, Section III. It should be noted that the NRC staff has not yet incorporated by reference the 2001 edition up to and including the 2003 addenda into 10 CFR 50.55a. The proposed rule issued in the Federal Register (69 FR 879) included several modifications and limitations on the use of the 2001 edition up to and including the 2003 addenda for piping seismic design in paragraph NB-3000 of the ASME Code. AECL must meet any modifications and limitations on the 2001 edition up to and including the 2003 addenda related to piping design when the final rule is issued later this year.

AECL Response

Safety-related piping design will comply with the ASME Code Edition and addenda as specifically incorporated by reference by the NRC in 10CFR50.55a(b)(1).

ATTACHMENT 3
(Letter G. Archinoff to A. Cubbage, "Submittal of Report 108US-30000-LS-001 on Reactor Coolant Pressure Boundary", March 24, 2005)

AECL's Response to Section 7 of the NRC's Pre-Application Safety Assessment Report (PASAR) for the ACR-700 - On-Power Fueling

This attachment includes AECL's responses to specific issues identified in Section 7 (Focus Topic 8 – On-Power Fueling) of the NRC's Pre-Application Safety Assessment Report (PASAR) for the ACR-700. Items addressed herein are those PASAR items and issues that are related to reports "Codes, Standards and Acceptance Criteria for ACR-700 Reactor Coolant Pressure Boundary and On-Line Fueling", 108US-30000-LS-001 and "ACR-700 Pressure Tubes Integrity", 108US-31110-LS-001[3].

1. **Section 7.4.1, Page 7-13, Use of Canadian Codes and Standards**

   Design certification applicants must meet the requirements of the ASME Codes as specified in 10 CFR 50.55a. Consequently, if the ACR-700 design does not satisfy certain ASME Code rules, the applicant must explain why the ASME Code rules cannot be satisfied, and either request an exemption from the regulations pursuant to 10 CFR 50.12, or request relief from the code rules by proposing alternatives to the ASME Code pursuant to 10 CFR 50.55a(a)(3). For those cases where no ASME Code rules exist, or where they are insufficient to address a portion of the ACR-700 design, the applicant must submit the alternative code rules to be used, including the technical basis for the acceptability of the proposed alternative rules, for NRC review and approval. Although Section 5.2 of ACR 108-35000-LS-001 identifies the alternative codes relevant to the design of the on-power fueling system components, it does not provide the required technical justification for acceptability, pursuant to 10 CFR 50.55a(a)(3)(i). Technical justification of these alternative codes is necessary, because the NRC staff will need to incorporate the use of these supplemental codes and standards into the regulatory framework as requirements in the ACR-700 design certification rule.

   AECL Response

   This is addressed in AECL Report, "Codes, Standards and Acceptance Criteria for ACR-700 Reactor Coolant Pressure Boundary and On-Line Fueling", 108US-30000-LS-001. This is also addressed in AECL's response to Item 18 in Attachment 1.

2. **Section 7.4.3, Page 7-14, Classification of System Components**

   During the detailed review of the ACR-700 design for certification, in order to determine that the safety systems and related components have been adequately classified, the staff will need to determine that AECL has verified that all of the

3 AECL Report "ACR-700 Pressure Tubes Integrity", 108US-31110-LS-001 is designated PROTECTED-Proprietary and is submitted to the NRC under separate cover.

necessary ASME Code, Section III, and other requirements are met, either directly or as a result of meeting requirements equivalent to RG 1.26 guidance in all important aspects. On a case-by-case basis, where it is determined that ASME Code, Section III, requirements apply, but certain components do not meet such requirements, the staff may authorize alternatives to these requirements in accordance with 10 CFR 50.55a(a)(3).

AECL Response

Classification of system components is addressed in AECL Report, "Codes, Standards and Acceptance Criteria for ACR-700 Reactor Coolant Pressure Boundary and On-Line Fueling", 108US-30000-LS-001.

3. **Section 7.4.3, Page 7-14, Classification of System Components**

In general, the proposed safety system classification appears to be appropriate, with one major exception identified with the currently available level of detail. For some systems interfacing with the RCS, there is only one Class 1 isolation valve at each of the interfacing system class boundaries. Two Class 1 isolation valves in series are required at these interfaces, in accordance with 10 CFR 50.50a(c), in order to ensure the integrity of the Class 1 RCS and to prevent possible overpressurization of lower pressure systems.

AECL Response

Double isolation will be provided where required. This is addressed in AECL Report, "Codes, Standards and Acceptance Criteria for ACR-700 Reactor Coolant Pressure Boundary and On-Line Fueling", 108US-30000-LS-001, as shown in Figure 2-3. This is also addressed in AECL's response to the PASAR issue identified in Attachment 1, item 3.

4. **Table 7-2, Page 7-20, Analysis of Failure Modes and Operating Experience for On-Power Fueling**

Loss of cooling to fuel while in the fueling machine head (while machine is away from reactor face).

AECL Response

The details of the fueling machine water supply are covered in 108US-30000-LS-001 Sections 2.2.5, 3.4.2 and 5.2.4.