AECL
TECHNOLOGIES INC.

March 31, 2004

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

Attention: Ms. B. Sosa
Project Manager, ACR

Reference:

1. Letter J. Kim to V. Langman, "Requests for Additional Information – ACR-700 Pre-Application Class 1 Pressure Boundary Design and Materials Review of Fuel Channels and On-Power Fueling", March 19, 2004.

Re: **Response to NRC's Requests for Additional Information (RAIs) #5 on Class 1 Pressure Boundary Design and Materials Review of Fuel Channels and On-Power Fueling**

In response to NRC's request (Reference 1) and in support of the NRC's pre-application review of the ACR (i.e., specifically focus topics # 1 – Class 1 Pressure Boundary Design, and # 8 – On-Power Fueling), attachment 1 provides AECL's responses to NRC staff requests for additional information on Class 1 Pressure Boundary Design and Materials Review of Fuel Channels and On-Power Fueling.

If you have any questions on this letter and/or the enclosed material please contact the undersigned at (905) 823-9060 extension 6543.

Yours sincerely,

Vince Lang

Vince J. Langman
ACR Licensing Manager

/Attachment:

1. Response to NRC's Requests for Additional Information (RAIs) on Class 1 Pressure Boundary Design and Materials Review of Fuel Channels and On-Power Fueling

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P | Sandstone

Attachment 1
(Letter V. Langman to B. Sosa, "Response to NRC's Requests for Additional Information (RAIs) #5 on Class 1 Pressure Boundary Design and Materials Review of Fuel Channels and On-Power Fueling", March 31, 2004)

Response to NRC's Requests for Additional Information (RAIs) #3 on PRA Quality

AECL's responses to NRC's requests for additional information on Class 1 Pressure Boundary Design and Materials Review of Fuel Channels and On-Power Fueling are provided in italic fonts following each of the NRC's questions as follows:

The following questions and comments were generated in support of the pre-application review of the Class 1 Pressure Boundary Design and materials. The following additional information is required for use in the Safety Assessment Report:

A. Class 1 Pressure Boundary Design

91. AECL requested a staff acceptance of certain aspects of the ACR-700 pressure boundary design. AECL should define the extent of the reactor coolant pressure boundary and discuss its compliance with the definition of the reactor coolant pressure boundary provided in 10 CFR 50.2. AECL should provide this information for the case where the fueling machine is attached and the case where the fueling machine is not attached. AECL should identify any pressure boundary components that meet the definition of the reactor coolant pressure boundary provided in 10 CFR 50.2 and are not designated Class 1. AECL should provide the basis for the classification of the components not designated Class 1.

    AECL Response:

    The extent of the reactor coolant pressure boundary (RCPB) can be defined with reference to the attached Figures 91-1 and 91-2. The flow sheet in Figure 91-1 shows all components that make up the pressure boundary when the fueling machine is not attached. Due to differences in Canadian and US regulations regarding Code Classifications, some of the code classifications for adjoining systems are under review. As a result some of the valve classifications shown in the figure for the second valve forming the limit of the boundary according to US regulations may change. The Figure should be used for defining the limits of the Class 1 boundary.

    All the components within this boundary are designed to the rules of ASME Section III Division 1. With the exception of fuel channel components, all the materials used to form the pressure boundary are ASME accepted materials for Class 1 application and are used within their limits of applicability.

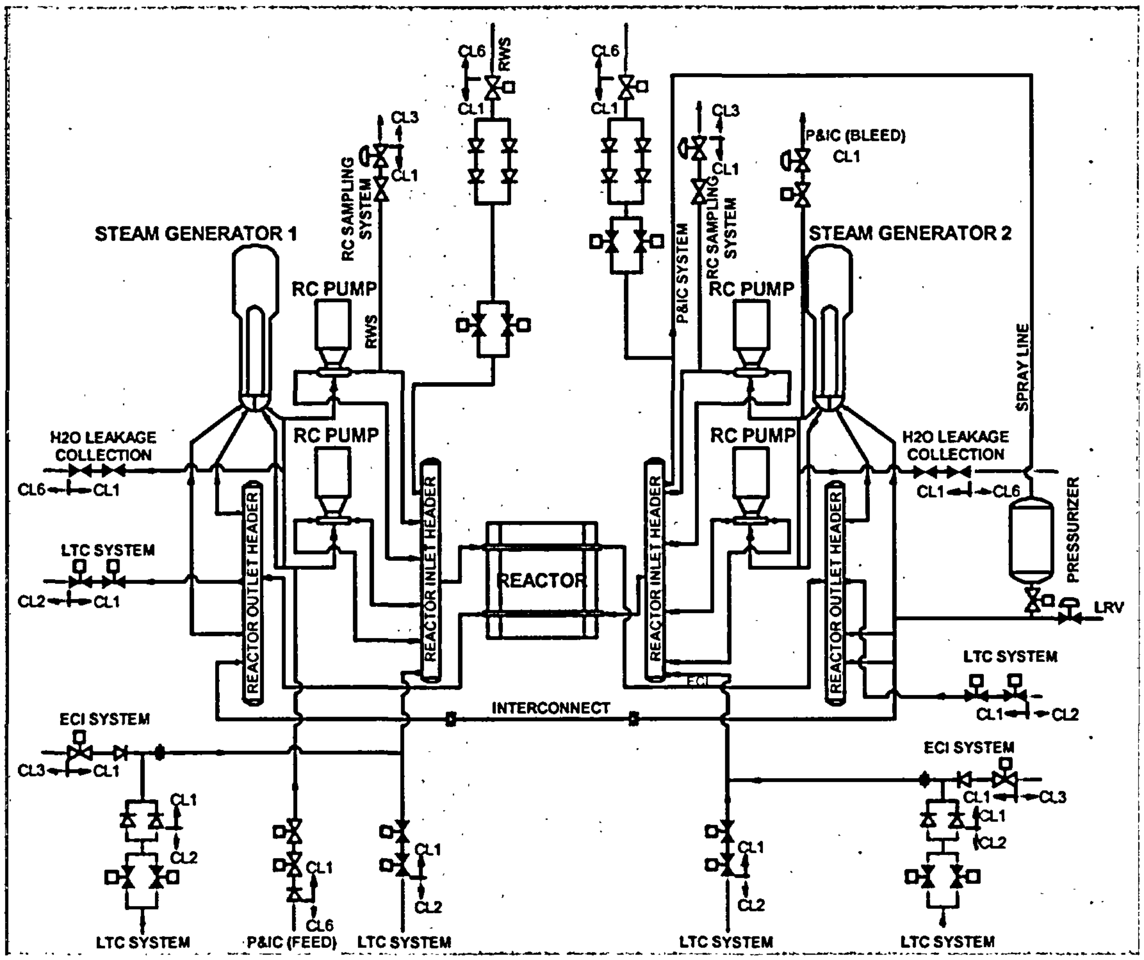


Figure 91-1: Flow sheet showing the limits of the Class 1 Reactor Coolant Pressure Boundary

Figure 91- 2 shows the RCPB for the Fueling Machine that becomes part of the RCPB when the fueling machine is attached to the fuel channel. All materials used in the fueling machine Class 1 pressure boundary are ASME –accepted Class 1 materials. All Class 1 components of the Fueling Machine are designed to the NB Rules of ASME Section III Division 1. Small diameter penetrations of the boundary using threaded joints have been qualified as has the use of Swagelok fittings for small bore tubing. Parts of the fueling machine pressure boundary not designated as Class 1, are acceptable according to the definition of the RCPB provided by 10 CFR 50.55a(c). Other features of the fueling machine that are not covered by ASME rules have been addressed with additional rules that supplement ASME.

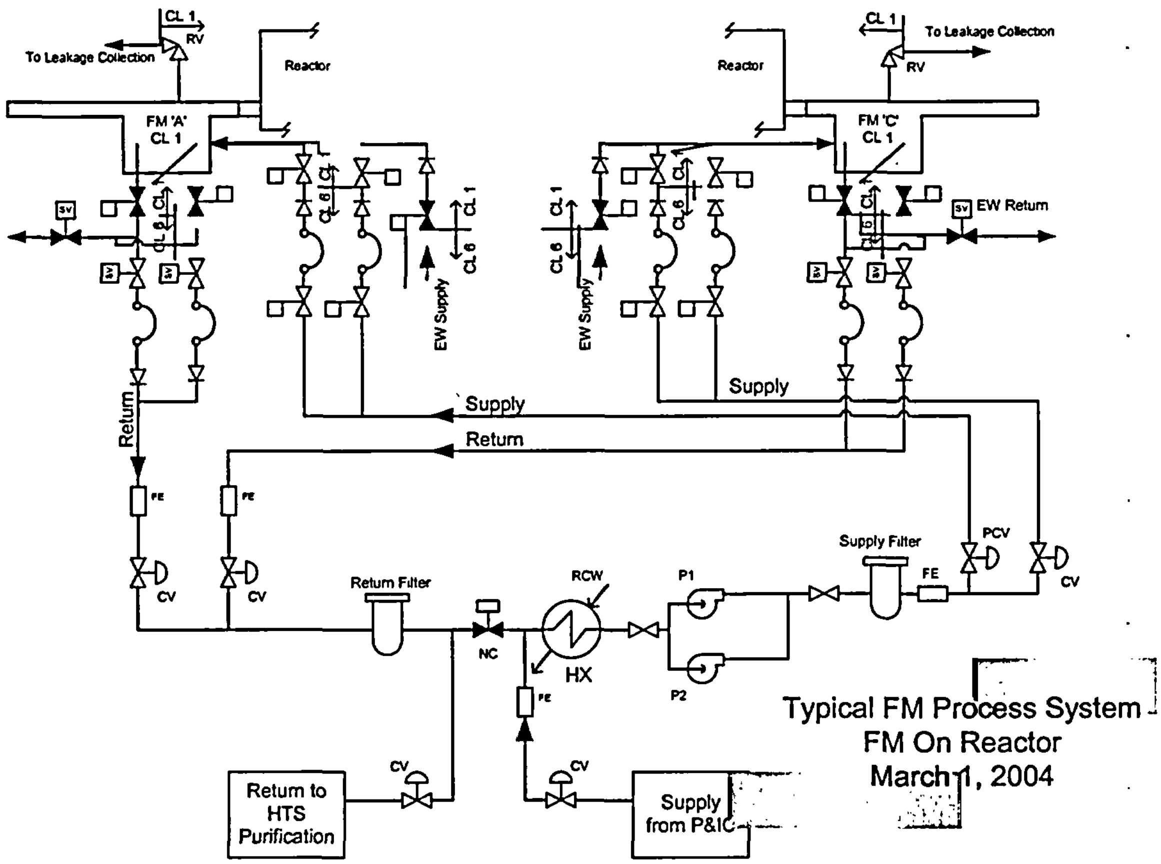


Figure 91-2: Flowsheet for the fueling machine process system and boundaries.

The fuel channels are unique CANDU components that have required the development of specific Canadian Standards to supplement the ASME code for the design of a fuel channel reactor. The components of the fuel channel that form the reactor coolant pressure boundary are the removable closure plugs, the end fittings and the pressure tube. The end fitting is classified as a Class 1 component in accordance with CAN/CSA N285.0 that references ASME Section III, NB. The closure plug, pressure tube and the rolled joint between the pressure tube and end fitting are all classified as Class 1C according to Canadian designation and have been designed to comply with CAN/CSA Standard N285.2. They have been designed using the rules of ASME Section III NB-3200 using both ASME materials (closure plug) and CANDU-specific materials. The rolled joint between the pressure tube and the end fitting is designed according to the rules of NB-3200. It is a non-integral joint and meets the requirements for non-integral joints as specified by ASME NB-3227.3.

Only the materials of the fuel channel are not listed as Class 1 materials according to ASME Section II Part D.

- *The pressure tube material has an ASTM designation (ASTM B353, UNS R60901) and the design stress intensity limits (S_m) are defined by the CSA Standard N285.6.1 (and are listed in Table 5-1(a) of Reference 91-1).*

- *The end fitting material is modified AISI Type 403 Martensitic Stainless Steel. Details of the fabrication of the end fittings are provided in Section 15.2 of Reference 91-1. The design stress intensity limits for the end fitting material are shown in Table 5-1(a) of Reference 91-1.*

- *The pressure boundary material of the closure plug is currently under review and will be an ASME accepted material.*

Reference: 91-1. The Technology of CANDU Fuel Channels, AECL Report 108-31100-LS-001, August 2003

92. AECL should clearly define which aspects of the Class 1 pressure boundary design meet ASME Code requirements and which aspects of the Class 1 pressure boundary design do not meet the ASME Code requirements. AECL should identify the specific paragraphs of the ASME Code that can not be met and identify the proposed alternative design criteria that will be used for the ACR-700 design. AECL should discuss how the alternative design criteria satisfy 10 CFR 50.55a(a)(3).

AECL Response:

As indicated in the response to RAI 91 above, the entire reactor coolant pressure boundary is designed to the rules of ASME, only the material selections for the fuel channel components are not covered within the ASME code but rather in accordance with CAN/CSA N285.6. Therefore, the specific paragraph of the ASME code that cannot be met is: NB-2121 Permitted Material Specifications

The proposed alternative design criteria to be used for the fuel channel materials are those of the Canadian Standard CAN/CSA N285.6. Use of these material specifications will satisfy the requirements of 10 CFR 50.55(a)(3).

93. AECL should discuss the load combinations and associated stress limits used for the design of ASME Class 1 pressure boundary components and compare them with the staff guidance provided in Standard Review Plan (NUREG-0800) Section 3.9.3. AECL should also discuss any special limits that will be used for the rolled joint design.

AECL Response:

Load combinations for the design of ASME Class 1 pressure boundary components and piping will be per SRP guidance and will follow the ASME prescribed stress limits. RCPB detailed analysis is performed per ASME Section III, NB-3000.

For supplied equipment such as steam generators, pressurizer, condenser, headers, pumps, valves, etc,

- *AECL prepares specifications per ASME to meet ASME design criteria*
- *Designed / Manufactured per Section III, Class 1 Rules of ASME*

For the fuel channel and fueling machine, analysis is performed per ASME Section III, NB-3000. ASME special stress limits in NB-3227.3 applicable to progressive distortion of non-integral connections will be satisfied for the pressure tube / end fitting rolled joint design. Additionally, testing limits of NB-3226 will be satisfied.

For feeder pipes and other piping, analysis is per ASME Code 2001 Edition, Section III, NB-3600. Alternative method of reversing dynamic load appropriately modified per NRC proposed rules (FR Vol 69, No. 4, 2004 Jan 7) would be used for the seismic design of piping.

The table provided below shows the load combinations and associated stress limits for the Class 1 pressure boundary components and piping. Additionally, testing limits of NB-3226 will be satisfied.

Plant Event	*Operating Condition*	*Loading Combination*	*Service Limits*
Normal Operation	*Normal*	*Sustained Loads*	*A*
*Plant/System Operating Transients (SOT) **	*Upset*	*Sustained Loads + SOT*	*B*
Small LOCA	*Emergency*	*Sustained Loads + DBPB*	*C*
MS/FWPB	*Faulted*	*Sustained Loads + MS/FWPB*	*D*
SSE	*Faulted*	*Sustained Loads + SSE*	*D*
LOCA	*Faulted*	*Sustained Loads + LOCA*	*D*
LOCA + SSE	*Faulted*	*Sustained Loads + LOCA + SSE*	*D*

(OBE equivalent considered for fatigue calculations)*

94. AECL should provide additional information regarding those aspects of the pressure tube design that will be qualified by testing. Specifically, AECL should identify the test procedures, indicate the number of tests performed, and discuss how these tests assure that all design conditions have been bounded. AECL should also discuss how irradiation creep behavior is bounded by the testing.

    AECL Response:

    A limited qualification program for the ACR fuel channel will build upon the extensive experience of fuel channel performance as observed in currently operating CANDU reactors and surveillance testing of pressure tubes removed from reactors. The specification of the material for the ACR pressure tubes is within the range of previous experience. As indicated in the responses to the RAIs 91 and 92 in the foregoing, the fuel channel is designed using ASME Section III NB rules using the CANDU-specific *materials defined by Canadian Standards. The intent of the qualification program is to provide assurance that the performance of the ACR Fuel Channel under ACR conditions will meet the Design Requirements.*

For the Fuel Channel, a detailed program for the design and testing has been developed and documented in a Work Activity Plan. However, at this time, details of all the specific testing requirements have not been identified. The elements of the overall channel design include design and qualification of the following components:

Pressure tube
Rolled Joint
End Fitting Assembly
Restraint / Annulus Seal
Supports (Annulus spacers and bearings)
Calandria tube

For each item, the plan identifies all the activities required, the required inputs for the activity to proceed, the outputs of the activity (documentation), the applicable procedures, the AECL group (Branch) responsible for the output and the requirements for verification of the outputs including the type of verification required, the procedure to be applied for the verification and the responsibility for verification.

The qualification testing of each component will be focused on those aspects of performance that are not easily amenable to predictions of performance by analysis only. For example, the pull-out strength and leak rate testing of the rolled joint are both subject to qualification testing. Although the strength of the rolled joint can be predicted by modeling, there are sufficient uncertainties in the modeling that testing is judged to be required to demonstrate the pull-out strength on a number of rolled joints fabricated to the specifications. The leak rates of the joint cannot be modeled at this time and the tests are required to demonstrate the leak tightness of the joint both before and after thermal cycling of the joint.

For the qualification of the pressure tube design, a detailed Inspection and Test Plan has been prepared and is being implemented. The scope of the test plan is focused on the testing of prototype tubes to demonstrate that the tubes, shown to have met Technical Specifications that include requirements for chemical composition and strength, have the expected behavior in a range of parameters not specified in the Technical Specifications but expected to result from the choices of manufacturing parameters such as extrusion temperature, extrusion ratio, and amount of cold work. The evaluations include the following:

Longitudinal tensile tests to establish the tensile properties (Yield stress, UTS and Elongation) over the complete temperature range of reactor operation.
Transverse tensile tests
Microstructural evaluation including:

Crystallographic texture measurement
Dislocation density measurement
Grain morphology evaluation
Fracture toughness
Delayed Hydride Crack Velocity[1] *(DHCV) at ACR operating temperatures – this can only be measured after adding hydrogen to the material*
K_{IH} *– the threshold stress intensity factor required for Delayed Hydride Cracking – again on specimens with added hydrogen*
Tests of Irradiated Material[2]
Tensile, fracture toughness, DHCV, K_{IH}

These evaluations will demonstrate that the material is essentially the same as material previously used in CANDU 6 pressure tubes.

95. AECL should discuss compliance of the ACR-700 Class 1 pressure boundary design with the General Design Criteria (GDC) in Appendix A to 10 CFR Part 50. Specifically, discuss how the Class 1 pressure boundary design satisfies GDC 4, GDC 14 and GDC 15.

AECL Response:

<u>Compliance with GDC4 of Reactor Coolant System</u>

The primary system is designed, fabricated, and erected in accordance with ASME Section III, Class 1. This ensures that conditions including normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents have been considered in the design. Primary system materials beyond the scope of ASME Section II will be considered via 50.55a(a)(3). These materials have been proven by testing and experience to be compatible with, and suitable for, their environmental conditions.
ACR design considers dynamic effects by performing appropriate analyses and providing barriers and separations as necessary for the feeders and piping. The pressure tube is contained within a calandria tube so the effect of pressure tube rupture is not propagated to other pressure tubes.

The use of leak-before-break on portions of the primary system will be the subject of separate discussions. The ACR reactor assembly is designed for dynamic effects of a pressure tube and feeder ruptures.

1 Crack growth rate
2 The irradiations will be relatively short-term irradiations designed to provide measures of all these parameters once the initial, transient, irradiation response has occurred. For these parameters, previous testing of pressure tube material has demonstrated that most of the change expected to the end of life occurs with the initial transient response. The materials will be irradiated at ACR temperatures in pressurized light water coolant with appropriate chemistry control.

Compliance with GDC 4 by the Fueling Machine

The fueling machine is designed, fabricated, and erected in accordance with ASME Section III, Class 1. This ensures that conditions including normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents have been considered in the design. The fueling machine is a robust piece of equipment and will not be affected by feeder ruptures. The design of feeders includes barriers, supports, etc. to preclude any damage.
The analysis of the end fittings, the only high-energy component in the fueling machine vicinity, is performed to the relevant level D conditions and it shows a significant margin from the allowable. Therefore the effects on fueling machine of end fitting failures are not included.

Compliance with GDC 14 by the Reactor Coolant System

The primary system is designed, fabricated, and erected in accordance with ASME Section III, Class 1. Primary system materials beyond the scope of ASME Section II will be considered via 50.55a(a)(3).
Components are designed for accessibility. Preservice inspection and testing will be in accordance with ASME Section XI. For in-service inspection, a risk-informed inspection program similar to that of ASME Code Case N-578 will be proposed for the feeder tubes. A sampling program based on Canadian experience with monitoring of pressure tube performance will be proposed.
Canadian pressure tube experience has documented two pressure tube ruptures in 400 reactor-years of operation. Corrective actions have been taken. Continuous monitoring of the annulus gas system for humidity ensures that ruptures can be mitigated before they occur. Inspection includes 100% volumetric examination of pressure tubes prior to servicee and in-service monitoring to detect potential generic degradation. There have been no pressure tube leaks since 1986 using improved manufacturing and inspection processes.

Compliance with GDC 14 by the Fueling Machine

The fueling machine pressure boundary is designed, fabricated, and erected in accordance with ASME Section III, Class 1.
Components are designed for accessibility. Preservice inspection and testing will be in accordance with ASME Section XI. In-service inspection will follow the ASME Section XI rules for Section III vessels.
Fueling machine (FM) includes safety lock to prevent unintentional release from fuel channel. FM design has two independent and diverse interlocks to prevent the FM from accidentally unclamping from the fuel channel. One of the interlocks is a mechanical device actuated by reactor pressure. Non-ASME components present small cross-sectional areas that have very large margins of safety and would not result in challenges to ECI (10 CFR 50.55a(c)(2)(i)).

Compliance with GDC 15 by the Reactor Coolant System

The primary system is designed, fabricated, and erected in accordance with ASME Section III, Class 1. Primary system materials beyond the scope of ASME Section II will be considered via 50.55a(a)(3). The properties of these materials, which are certified by testing, are used to derive conservative allowable stress and load combinations according to ASME Section III rules. The resultant design ensures that the RCS pressure boundary design conditions are never exceeded for any anticipated operational occurrence. Design criteria for auxiliary, control, and protection systems to be addressed in other discussions.

Compliance with GDC 15 by the Fueling Machine

The fueling machine pressure boundary is designed, fabricated, and erected in accordance with ASME Section III, Class 1. The design of the FM includes an analysis of all credible operating occurrences to confirm a low probability of failure of the RCS pressure boundary within the fueling machine and that the design conditions are not exceeded. Design criteria for auxiliary control and protection systems to be addressed in other discussions.

B. 10 CFR 50, Appendix A, General Design Criteria (GDC)

96. The NRC staff requests that AECL discuss compliance of the ACR-700 Class 1 and 1C pressure boundary design with the General Design Criteria (GDC) in Appendix A to 10 CFR Part 50. Specifically, discuss how the Class 1 and 1C pressure boundary design satisfies GDC 1, 4, 14,15, 30, 31, and 32 from the materials engineering point of view.

    The following is a discussion of compliance with the GDC 14 for the area of materials engineering. This discussion does not reflect an agency position but is merely a discussion of GDC 14 for the purpose of illustrating a method for AECL to improve its understanding of the GDC. Compliance with the GDC is discussed throughout the Standard Review Plan, which is contained in NUREG-0800. In some SRP sections, there are specific statements of what constitutes compliance with GDC 14 while in other cases, GDC 14 is combined with other GDC including GDC 1, 4, 15, 30, and 31. This combining of GDC requirements was probably performed since the criteria overlap each other in many cases. These GDC are contained in Appendix A to Part 50 of Title 10 of the Code of Federal Regulations. The requirement for these GDC are as follows:

    > GDC 1: Structures, systems, and components (SSCs) important to safety shall be designed, fabricated, erected and tested to quality standards. Where generally accepted codes and standards are used, they shall be evaluated to determine their adequacy and sufficiency to assure a quality product in keeping with the required safety function. A quality assurance program shall be established to ensure these SSC will perform their safety function.

GDC 4: SSC important to safety shall be designed to accommodate the effects of and be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. These SSCs shall be appropriately protected against dynamic effects (except for pipe ruptures if the Commission reviews and approves analyses that indicates that the probability of pipe rupture is extremely low).

GDC 14: The RCPB shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

GDC 15: The reactor coolant system (RCS) shall be designed with margin to assure the design conditions of the reactor coolant pressure boundary (RCPB) are not exceeded during normal operation.

GDC 30: The RCPB shall be designed, fabricated, erected, and tested to highest standards practical and means should be provided for detecting, and to the extent practical, locating leakage.

GDC 31: The RCPB shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions the boundary behaves in a nonbrittle manner and the probability of rapidly propagating fracture is minimized. The design shall reflect operating conditions and uncertainties in determining material properties, the effects of irradiation on material properties, residual, steady state, and transient stresses, and size of flaws.

GDC 32: The RCPB shall be designed to permit periodic inspection and testing of important areas and features to assess their structural and leaktight integrity and an appropriate material surveillance program for the reactor pressure vessel.

Based on a review of SRPs in the materials engineering area, information provided to demonstrate that the ACR-700 design satisfies GDC-14 from the material engineering point of view would need to address at least the following statements.

a The applicant must demonstrate that the materials selected satisfy Appendix I of Section III of the ASME Code and Parts A, B, C and D of Section II of the Code.

The applicant must demonstrate that the components are constructed in accordance with and receive the fabrication inspections required by Section III of the ASME Code.

b The applicant should indicate that the yield strength of cold-worked austenitic stainless steel will not exceed 90,000 psi. The applicant should also meet the guidelines of Regulatory Guide (RG) 1.85 if using materials of construction that are approved for use in ASME code cases. RGs provide an acceptable method to meet the regulations, but are not themselves requirements.

c The applicant must demonstrate they will meet the requirements of Appendix G of 10 CFR Part 50 so as to ensure adequate fracture toughness.

d RG 1.36 should be followed to demonstrate the compatibility of austenitic stainless steel with thermal insulation.

e RG 1.44, "Control of the Use of Sensitized Stainless Steel" and RG 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Component of Water Cooled Nuclear Plants," should be followed.

f Welding of austenitic stainless steels in the RCPB should be in accordance with the recommendations of RG 1.31, RG 1.34, and RG 1.71. These controls provide reasonable assurance that welded components of austenitic stainless steel will not develop micro fissures during welding and will have high structural integrity.

g For steam generators, crevices between the tubesheet and the inserted tube will be minimal. The tube support plates will be manufactured from ferritic stainless steel and promote high velocity flow along the tube (i.e., they are not circular drilled holes).

h For materials not specified in the ASME Code (or deviations from a listed specification in the Code), their suitability is evaluated on the basis of data submitted in accordance with the requirements of Section III, Appendix IV-1400 and 10 CFR Part 50, Appendix G. These data must include information on mechanical properties, weldability, and physical changes of the material.

AECL Response:

a *As indicated in a previous response, with the exception of components of the fuel channel, all other materials forming the reactor coolant pressure boundary satisfy the requirements of Appendix 1 of Section III of the ASME Code and the requirements of Parts A,B,C and D of Section II of the Code.*

b *Raw austenitic stainless steel material is supplied in solution-heat-treated condition to ensure non-sensitized condition in the material. Cold worked austenitic stainless steel with 0.2 percent yield strength greater than 90,000 psi (620 MPa) is not used for RCPB components to reduce probability of stress corrosion cracking.*

c *The requirements of Appendix G of 10 CFR 50 are met for the ferritic components of the RCPB by meeting the ASME code requirements. The additional aspects of 10 CFR 50 Appendix G applicable to Reactor Pressure Vessels are not applicable to the CANDU design. The fracture toughness of fuel channel components is discussed in some detail in Section 10.4 of the FCTR. The high fracture toughness of ACR pressure tubes manufactured with practices to achieve low impurity concentrations is expected to be maintained throughout the reactor life.*

d *There is no insulation material directly in contact with the stainless steel components of the the RCPB. The austenitic stainless steel feeders are within an insulated cabinet. The practice with respect to stainless steel components in contact with insulation for other systems is the following:*

- *To prevent external SCC of austenitic stainless steel, all insulation material is tested in accordance with the ASTM C 692 Standard and the chemical analysis of the insulation material meets the requirements of ASTM C 795 Standard to assure that the leachable concentration of chloride, fluoride, sodium and silicate are within the acceptable region of Figure 1 of Regulatory Guide 1.36.*

- *All austenitic stainless steel pipe is painted with Thurmalox 70 paint. This silicone-based paint acts as a barrier, preventing contact between pipe surface and chloride and fluoride contamination that may be leached out of the insulation.*

e *Regulatory Guides 1.36 and 1.44 regarding stainless steel use and cleaning will be met. Austenitic stainless steel components are protected against contaminants that can cause stress corrosion cracking during, fabrication, shipment, storage, construction, and testing. The low carbon variant ('L' grade with a carbon content of < 0.03 wt%) of the un-stabilized austenitic stainless steels is specified to avoid concerns with sensitization. The feeder pipe material (SA-312 Type*

TP316N) is specified with carbon content to meet the L grade requirement, and to avoid concerns with sensitization. All RCPB austenitic stainless steels are screened in accordance with ASTM A 262 standard to ensure non-susceptibility to stress corrosion cracking.

f *All austenitic stainless steel filler metal used for welding of RCPB components meet the requirements of NB-2340 of the ASME BPVC, Section III, Division I and conform to the relevant specifications in ASME BPVC, Section II, Part C which assures control of weld metal ferrite content to Ferrite Number 5-20 range.*

All welding is performed according to the requirements of Articles NB-2400 and NB-4300 of ASME BPVC, Section III, Division I. Only low hydrogen filler material is used. The manufacturer's filler metal control and handling procedures are reviewed for compliance with the recommendation of the Appendix to SFA-5.1 of ASME BPVC, Section II, Part C, regarding prevention of moisture pick-up.

Austenitic stainless steel welds and repair welds to weld or base metal, which are exposed to system fluid, are solution annealed, unless the base metal and filler metal are both the low carbon variants. If solution annealing is impossible or impractical, a low heat input welding process with restricted low interpass temperature is used.

Regulatory Guide 1.34 is not applicable. The majority of joints in the RCPB are not amenable to the electroslag welding process, and the process is therefore not specified.

g *AECL is aware of all the major issues with respect to steam generator performance. For example, AECL participates in the NRC Tube Integrity Project for Steam Generators. In terms of material selection and design, steam generators for the current CANDU 6 plants have ferritic stainless steel support structures in the form of lattice bar supports. Incoloy 800 tubing is specified for CANDU 6 plants and the performance of this material in the existing plants has been excellent after twenty years of service. ACR steam generators will use best practices for design and will meet NRC SRP requirements. The current tubing material is specified as Incoloy 800 with a customer option to choose Alloy 690.*

h *The materials not specified by the ASME code are the materials used in the fuel channel components including the pressure tube and the end fitting materials. The material requirements for these components are specified by CAN/CSA N285.6 and the allowable stress intensities are defined based upon testing and experience. There are no welds in the fuel channel pressure boundary and weldability is not an issue for use of these materials. Physical changes of the* *material under irradiation will be demonstrated by testing as discussed elsewhere in these responses.*

C. Inservice Inspection

97. The Technology Report, Chapter 11, Inspection and Monitoring of CANDU Fuel Channels, addresses the Periodic Inspection Program (PIP) which involves inspection of a relatively small sample of tubes on a recurring basis to ensure identification of generic problems. Table 11-1 provides a summary of inspection requirements. The governing standard for periodic inspection is CSA-285.4-94. This standard appears to contain considerably less detail regarding inservice inspection than the ASME Code, Section XI, and lacks sufficient detail for staff's review of the PIP. For example, Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components, contains general requirements for preservice and inservice inspection on scope and responsibility, examination and inspection, standards for examination and evaluation, repair/replacement activities, system pressure tests, and records and reports. The sections on examination include component locations to be inspected, inspection areas or volumes, methods, frequency, acceptance criteria, and qualifications of examination personnel. Flaw evaluation criteria are contained in Section XI as requirements. Appendix I to Section XI further specifies the examination techniques to be used. For example, Appendix I requires that certain components be examined in accordance with the requirements of Appendix VIII to Section XI. Components covered by Appendix VIII must be examined by performance demonstration techniques that apply to the total examination system - equipment, procedures, and personnel. Please explain the extent to which the ASME Code, Section XI, applies to the components in the Code Class 1 and 1C pressure boundary. For those components not covered by the preservice and inservice inspection rules of ASME, Section XI, provide a reconciliation summary between CSA-285.4 and other AECL documents, as applicable, and the ASME Code, Section XI. The reconciliation summary should address the areas covered by Section XI as addressed above.

    AECL Response:

    Inspection of the reactor pressure boundary components in ACR will be carried out in accordance with ASME Section XI. Some of this inspection is expected to be risk-informed. There are two areas where such alternative inspection approaches are required due to the design and materials of the ACR. These are the feeder pipes and the fuel channels.

    The feeder pipes are austenitic stainless steel (ASME SA 312 TP 316N) with a size range from 2.5" to 3.5". With two feeder pipes per channel and a number of welds in each feeder pipe, there is a need to reduce the scope of the inspection defined by ASME

Section XI. AECL proposes that a Risk-Informed approach to feeder inspection be applied. Details of the examination program and the supporting data will be provided in the application for ACR design certification.

For fuel channel inspection, AECL proposes that an inspection program similar to that in the revised, 4th Edition of the CAN/CSA N285.4 standard be applied to ACR. CAN/CSA N285.4-94 is the current inspection standard for CANDU reactors. The fourth edition has recently been approved by a committee vote and is now in final editing. The revised standard contains significant changes in the areas of fuel channels, feeder pipes, and steam generator inspection – particularly in inspection sample size.

CAN/CSA N285.4 may initially appear to lack detail; however, it should be noted that for inspection of welded components and systems it basically calls up ASME Sections III, V, and XI. The same applies to mechanical couplings. Table 1 of CAN/CSA N285.4 summarizes test methods for various components, acceptance criteria, and cross-references to ASME Section XI tables. A detailed examination of CAN/CSA N285.4 will generally reveal that CANDU inspection requirements don't differ that much from ASME requirements especially in the areas of inspection methods, records and reports, acceptance criteria and qualification of inspection personnel. For the specific case of fuel channels, the following are specified in CAN/CSA N285.4-94:

- *Inspection location – the fraction of high-power channels that have to be included in the inspection sample is specified (Clauses 12.3.2 and 12.4.2).*
- *Inspection area - the entire volume of the pressure tube, including both rolled joints, has to be examined for flaws; the maximum interval between dimensional measurements like diameter and wall thickness is also specified (Clause 12.3.3).*
- *Permissible inspection methods are identified (Clause 12.5).*
- *Inspection frequency is clearly specified (Clause 12.4.3).*
- *Unconditional acceptance criteria are listed for all specified tests (Clause 12.7.2).*
- *Qualification of inspection personnel is addressed (Clause 12.6).*

CANDU pressure tubes are really a very simple component to inspect compared to the complex geometry and numerous different weld configurations found in PWR and BWR pressure vessels. By contrast, pressure tubes are straight, thin-walled, seamless tubes manufactured from vacuum melted material; they contain no welds. They are subject to stringent manufacturing inspection requirements.

The end fittings are generally not inspected in service. There are no known SCC mechanisms operating in the CANDU reactor primary water chemistry environment and there have been no observations of any erosion/corrosion issues in end fittings of channels removed from service. Each end fitting is a single forging and is subjected to volumetric inspection when manufactured.

Closure plugs have not been subject to volumetric inspection in the past. There are no welds in the pressure boundary and no plausible degradation mechanisms have been identified.

Note that detailed evaluation criteria for pressure tube inspection results have been in place since 1990. They have recently been adopted in committee as a new CSA standard, CSA N285.8, "Technical Requirements for In-service Evaluation of Zirconium Alloy Pressure Tubes in CANDU Reactors". This standard has been published as an industry document until the formal publication of the CSA Standard can be achieved.

The latest edition of CAN/CSA N285.4 contains general requirements for performance demonstration (PD) of inspection equipment/procedures/personnel. For carbon-steel weld inspection and steam generator tube inspection this requirement has usually been satisfied by adoption of European or EPRI methodologies for PD. For unique CANDU components like feeders and fuel channels the main PD activities to date have concerned qualification of new and/or unproven test methods. PD of test methods/equipment that have been in use for decades in some cases has received differing amounts of attention depending on the inspection equipment and the organization operating it. In the case of the AECL Fuel Channel Inspection System (AFCIS), formal demonstrations with independent witnesses have been conducted on all system measurement methods. A total of 35 different tests were performed and are documented in proprietary AECL reports.

98. Certain ASME Code requirements, including requirements in Section XI, are endorsed by reference in NRC regulations. This action makes these code requirements have the force of any other regulation. AECL will utilize the ASME Code, CSA standards, and possibly other AECL documents for design and inservice inspection of the ACR-700. Related to the preceding question, the NRC will evaluate the acceptability of the documents that supplement the ASME Code. The NRC will also need to address how the supplemental documents will be incorporated into the regulatory framework as requirements. The preceding is for AECL's information and AECL is not expected to respond on this topic.

99. General Design Criterion (GDC) 32 of Appendix A to 10 CFR Part 50 requires, in part, that components which are part of the reactor coolant pressure boundary shall be designed to permit periodic inspection and testing. In addition, the ASME Code and GDC 1 require that other safety significant components be inspected and/or tested. In the ACR 700 design, it appears that certain components may be inaccessible for either post-fabrication inspection or inservice inspection (e.g., calandria tubes and lattice tubes). If there are components that can not be reliably inspected/tested, please identify these components and discuss what programs are in place to ensure these components are acceptable for initial and continued service. For example, is there a program (or requirement in the Code) to periodically verify that a component which is not accessible for inspection/testing is performing as expected (e.g., through destructive examination of

that component). It is noted that for several components (e.g., calandria tubes) information pertaining to the inspectability was provided; however, all components did not appear to be addressed.

AECL Response:

Both calandria tubes and lattice tubes are accessible for post-fabrication inspection. It is only after the pressure tubes are installed that these components become "inaccessible" unless one removes pressure tubes first. (Large-scale visual inspection of calandria tubes from inside the calandria vessel is possible, in principle, through the viewing port on the reactivity mechanism deck). However, note that neither calandria nor lattice tubes form part of the reactor coolant pressure boundary but are supporting structures for the pressure boundary components.

The lattice tubes, calandria tubes, pressure tubes, and rolled joints are all monitored continuously for leakage during service by the annulus gas system. This high-sensitivity leak detection system can detect leakage rates of grams/hour while the type of leak can be determined by isotopic analysis of the leaking fluid (i.e., lattice tube – light water; calandria tube and its rolled joints to the tubesheets – moderator; pressure tube and its rolled joints – primary coolant).

100. It appears that there is a code requirement for monitoring the fracture toughness and delayed hydride cracking velocity for pressure tubes and that this requirement only applies to the CANDU lead unit with a particular type of pressure tube alloy (section 11.2.4.5). Given that there are many factors that affect these material properties, discuss the technical basis for limiting this requirement to a lead plant (operating experience in light water reactors has indicated that the "lead plant" is not always the first plant to experience an issue).

Please discuss any other situations in which the concept of a lead plant is applied in determining the inspection and testing requirements. In general, the NRC's regulations do not recognize the concept of a lead plant. For example, the requirements associated with a design certification would apply equally to all combined operating license holders of the certified design.

AECL Response:

The surveillance requirement with respect to the lead unit is based upon potential changes in material performance due primarily to effects of neutron irradiation under stress at operating temperature. Since all pressure tubes will experience a very similar irradiation/temperature/pressure environment, removal of tubes from a unit leading in operating time is appropriate. In Canada for current generation CANDUs, utilities have generally shared costs of doing such surveillance inspections through cooperative *agreements. The improved manufacturing methods implemented over a number of years have resulted in more uniform pressure tube properties, especially with respect to initial fracture toughness and this is expected to translate into more uniform performance after irradiation. For ACR, the expectation is that pressure tubes manufactured to the same specification by the same manufacturer would behave under irradiation in a similar manner.*

Aspects of pressure tube integrity performance associated with water chemistry effects that could potentially vary somewhat from one plant to another are monitored through measurements of hydrogen isotope concentration in those tubes subject to in-service inspection for each plant.

101. In section 11.3, a discussion of your multifunctional AECL fuel channel inspection system (AFCIS) is provided. Discuss the criteria by which this system was qualified. For example, discuss how the AFCIS was qualified to measure the pressure tube to calandria tube gap and how the uncertainties in these measurements are accounted for in determining the acceptance criteria and inspection frequency.

AECL Response:

The original performance demonstration (PD) of AFCIS was done to satisfy 35 criteria or performance targets that were established jointly by AECL and the first customer and their regulators. As such, the test program represented the requirements of the utility, the nuclear regulator, the boiler and pressure vessel inspection organization, and AECL. The entire test program was witnessed by representatives from these groups. A subsequent PD was performed to satisfy the requirements a second utility that operates both PWR and CANDU reactors. That demonstration was essentially a subset of the first PD; tests were witnessed by 11 representatives from various parts of the utility organization.

For the case of pressure-to-calandria tube gap, the PD requirement was to measure the gap on a full-scale fuel channel mock-up to within +/- 1 mm of a known value. The mock-up has facilities to vary gap over the full range of 0 mm to 17 mm. The independently verified AFCIS measurement was within 0.4 mm of the actual gap value.

The simplest acceptance criteria for gap measurements on in-service fuel channels is that there be no contact by the end of the next inspection interval (CAN/CSA N285.4-94, Clause 12.7.2.3). The prediction of gap at the next inspection naturally should address any implications of measurement uncertainty.

Inspection frequency would normally default to that specified in CAN/CSA N285.4 if there will be no contact by the end of the next inspection interval. If lack of contact over the next inspection interval can not be assured, then a number of possible approaches *have been used in Canada in the past, including all or some combination of the following:*

- *a reduced interval to the next inspection;*
- *further inspections;*
- *demonstration that pressure tube hydrogen content is and will remain sufficiently low to preclude the possibility of hydride blister formation even if there were pressure-to-calandria tube contact;*
- *remedial action to remove the possibility of contact.*

102. In section 11.1 you indicate that your inservice inspection program tends to be reactor specific and is generally directed by the reactor operator with concurrence of the regulatory authority. Please discuss this process since it appears to significantly differ from U.S. practice.

AECL Response:

The terminology for inspection is different within Canadian and US Codes and Standards. The Canadian equivalent of the US In-Service Inspection (ISI) is what is termed Periodic Inspection. These are inspections mandated by the standards to address specific inspection scopes and inspection intervals. The term "In-service Inspection" in Canada is used to denote those inspections that are the result of periodic inspection findings that indicate conditions exist within a particular reactor that require more inspection information in order to continue to be able to demonstrate fitness-for-service of the reactor components over the longer term. These findings may lead to additional inspections of reactors of either the same design or having had an operating history associated with the adverse inspection results. These additional inspections are addressed by the so-called "In-service" Inspection Programs for the specific reactors affected or potentially affected. In the US, additional inspections have been mandated by the NRC for specific reactor designs when adverse events occur in a single plant. In Canada, such additional inspections would be made part of In-service Inspection Programs but each program would be designed by a utility and accepted by the regulator. In some cases, utilities work cooperatively to provide a uniform approach in dealing with the inspection issue.

D. Code Classification

103. To understand the appropriate design and inspection requirements for structures, systems, and components in the ACR-700 design, it is necessary to understand their Code classification. Please provide a summary of the classification of the major SCCs in the ACR-700 design (with emphasis on the components making up the fuel channel design.)

AECL Response:

See responses to questions 91 and 92.

E. Degradation Mechanisms and Related Inspection and Monitoring

104. The Fuel Channel Technology Report (FCTR) states that "the pressure tube to end fitting rolled joints in a CANDU reactor have never come apart nor allowed excessive leakage of the reactor's coolant" (p. 3-3). Please describe quantitatively the amount of leakage that has occurred at these joints in relation to any applicable fitness for service guidelines. How was the leakage detected, and how was it addressed? What was the cause of the leakage? Could this leakage have caused long-term degradation of the structural integrity of the joint or any other components? Please discuss any efforts being undertaken to ensure leak tight joint integrity given the higher temperature and fluence and geometry changes in the ACR-700 relative to earlier reactor designs.

AECL Response:

During normal operation of a CANDU reactor, the circulating gas in the annulus gas system is monitored for moisture. The dew point of the gas in the system after a purge (venting and filling until the dew point is reduced) is typically in the range of –30 to –40°C. During normal operation, over a period of days, the dew point rises primarily due to two effects – 1) the ingress of water vapour into the system through the rolled joints and 2) the production of water vapour through the oxidation of deuterium gas also entering the annulus gas system by diffusion through the rolled joint and probably through the end-fitting material. The sensitivity of the leak detection capability decreases as the dew point increases and the limiting dew point allowed prior to a system purge is defined. The limit is normally in the range of –10°C to 0°C. Typically, the time interval between purges when there is no abnormal leakage in the system is of the order of a week or so. This corresponds to a total leak rate for all the rolled joints in the system of the order of 0.1 g/h (grams per hour). Very small leaks can be both detected using this system of dew point monitoring and localized to a small number of channels in the reactor. The annulus gas can be sampled and the condensed water characterized by chemical analysis to determine the source of the leak since the system interfaces with the *reactor coolant pressure boundary as well as the moderator and the end-shield cooling systems.*

There has been only one recorded incident in which an abnormally rising dew point has been associated with a rolled joint leak. (Other incidents of leaks associated with rolled joints occurred early in the history of CANDU and were associated with cracks in the pressure tube adjacent to the rolled joints and were not leakages of the rolled joint itself.) In the incident of rolled joint leakage, a persistently rising dew point developed over a number of months to a maximum leak rate of 4 g/h (grams per hour) of heavy water. The leakage was localized to a small number of channels and ultrasonic inspection of these channels identified the leaking joint. The leak originated at a very shallow, linear surface flaw on the external surface of the pressure tube in the rolled joint area that had oxidized during service and subsequently produced the leak. The leak rate always remained very low and there was no detectable effect of this leakage on any component.

Since any acceptable leak rate (i.e., a leak rate that does not elicit an operator search for the leak) is extremely small (small fractions of a gram per hour), there is no realistic possibility of erosion producing any structural damage to the joint nor to any other components.

ACR rolled joints are under development and must be shown by qualification testing to have similar, very low leak rates and be virtually leak-tight.

105. The Fuel Channel Technology Report (FCTR) describes the enhanced deuterium/hydrogen ingress in the rolled joint due to galvanic corrosion between the dissimilar end fitting and pressure tube materials (p. 9-10). If this fitting creates a galvanic corrosion cell, then the generation of hydrogen is a cathodic reaction that must be coupled to an anodic (corrosion) reaction (or reactions). The FCTR indicates the oxidation of zirconium to zirconium oxide is the anodic process for uncoupled zirconium alloy in the coolant environment. However, the predominant anodic reactions in a galvanic corrosion cell may not be the same as for the uncoupled alloys in the same environment. Since the rate of the cathodic reaction(s) must be equivalent to the rate of the corrosion reaction(s), enhancing the rate of hydrogen generation due to galvanic coupling would increase the rate of the corresponding corrosion reaction. Corrosion reactions occurring in a localized area can cause rapid penetration. What are the corrosion reactions that form the anodic part of the galvanic corrosion cell, and where do they occur? Do these galvanically driven corrosion reactions have any direct effect, such as thinning, that degrades the joint or any other components? What is the effect, if any, on the end fitting material (i.e., martensitic stainless steel) from hydrogen ingress? Is the corrosion and/or hydrogen ingress monitored in any way?

AECL Response:

In the FCTR, Figure 9-11 shows the main routes for hydrogen to enter the pressure tube in the rolled joint region. The routes are:

a. *Hydrogen uptake due to corrosion of the inner surface (i.e., 'waterside') of the pressure tube. This is the same process as for the body of the tube where typically 2 to 8 percent of the 'corrosion-freed' hydrogen enters the metal;*
b. *Hydrogen uptake due to corrosion of the pressure tube in the crevice exposed to the coolant; and,*
c. *Transfer of hydrogen from the 403 SS end fitting to the pressure tube.*

Since not all hydrogen generated enters the pressure tube, there is not a one-to-one correlation between the amount of hydrogen in the pressure tube and the amount of corrosion required to generate the hydrogen.

Hydrogen transferred from the end fitting (into the pressure tube) results from the cathodic reduction of H_2O on the stainless steel surface. The balancing anodic reaction can take place on either the end fitting or the pressure tube dependent upon the condition of each surface (i.e., passivated by oxide or not).

Crevice corrosion can occur in occluded regions having a flat and shallow geometry where the transport of reactants is limited. Upon breakdown of the surface oxide (inside the crevice), the anodic reaction [1] (shown below) occurs inside the crevice and the cathodic reaction [2] occurs just outside the crevice. The dissolution (i.e., reaction [1]) of the metal is followed by hydrolysis with water to form acidic conditions (and a deposit of ZrO_2) inside the crevice [3]. The following example (see Figure 105-1) is constructed for Zr, however, the same process may also take place on the 403SS upon breakdown of its surface oxide.

$$Zr \rightarrow Zr^{4+} + 4e^- \qquad [1]$$

$$4H_2O + 4e^- \rightarrow 4H + 4OH^- \qquad [2]$$

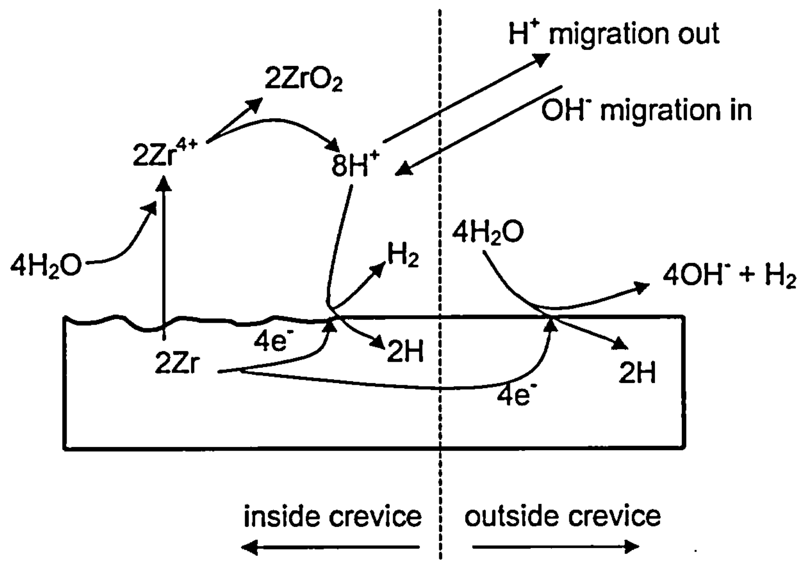
$$Zr^{4+} + 2H_2O \rightarrow ZrO_2 + 4H^+ \qquad [3]$$

The acid produced in the crevice by [3] can lead to further oxidation of the metal as a cathodic reaction (shown as [4] but similar to [2]) is now possible inside the crevice,

$$H^+ + e^- \rightarrow H \qquad [4]$$

However, in order to support the acidification of the crevice, the migration of anions into the crevice is required to maintain charge balance (electroneutrality) in solution. The dominant anion in the mildly alkaline coolant is the hydroxide (OH^-) ion and its *migration into the crevice results in neutralization. The protective oxide is re-established and the crevice corrosion ceases.*

This is not the 'typical crevice' associated with 'rapid penetration' as the coolant is relatively free of 'aggressive anions' (for example chloride, Cl^-). In the presence of Cl^- (i.e., replace OH^- with Cl^- in the figure below) the corrosivity of solution within the crevice increases allowing crevice corrosion to propagate. Rolled joint assemblies removed from reactors have not indicated that any aggressive corrosion conditions are occurring. Figure 105-2 shows surfaces of both the end fitting and pressure tube in an inlet rolled joint removed from service after approximately 100,000 hours of operation. Some slight corrosion is evident but nothing indicative of an integrity issue.



Note: Amount of corrosion-freed hydrogen entering metal is not to scale.

Figure 105-1: A schematic illustrating reactions occurring inside and outside a crevice on zirconium alloys

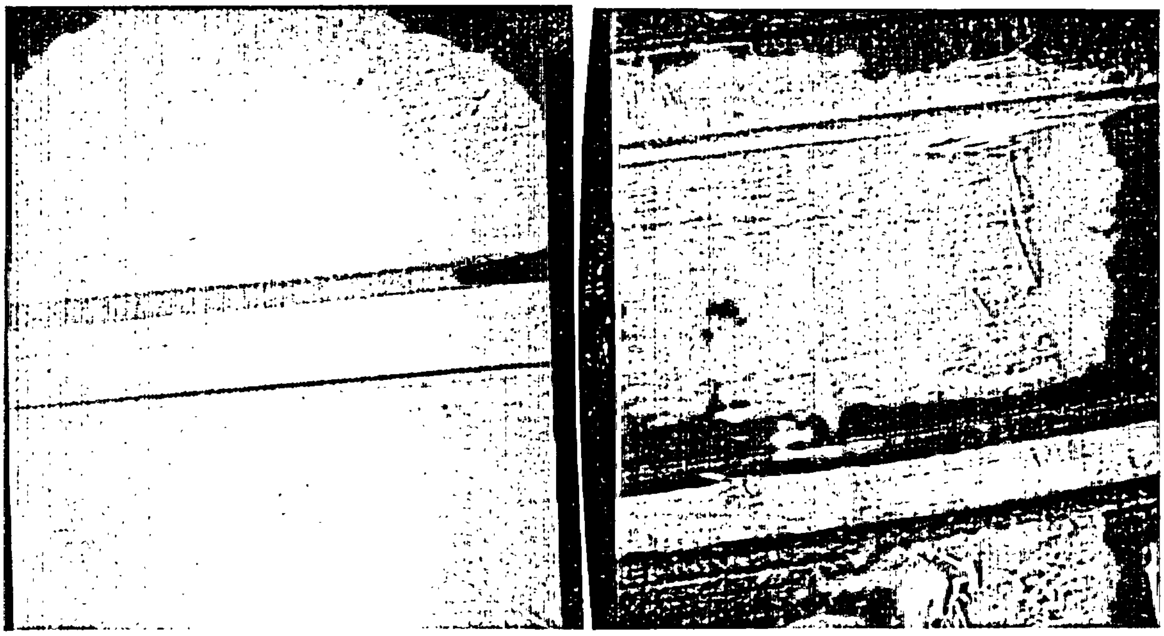


Figure 105-2: Showing the end fitting in-board groove and the in-board most ribs resulting from joint assembly after approximately 100,000 hours of operation at a channel inlet. The white spot at the bottom right is an identifying mark used by the examiners.

106. In the Fuel Channel Technology Report (FCTR), potential degradation mechanisms have been identified for various components. In addition, operating experience associated with some components was provided (e.g., pressure tube leakage and ruptures). Please provide a summary table that indicates the following for each component in the fuel channel

a) potential degradation mechanisms

b) if the mechanism has been observed in a plant, a discussion (or reference) to pertinent operating experience including a discussion on (or reference to) the severity of the degradation and whether it was within expectations given the service life (if not, a discussion of the corrective actions)

c) the inspection or monitoring requirements for this mechanism (or a reference to where the requirements are listed). If no inspection or monitoring is performed (or if the requirements are not in the Code or the plant technical specifications), the basis (or reference to the basis) for this approach.

d) the qualification data or qualification criteria (or reference to these data or criteria) supporting this inspection or monitoring method

e) the basis for the flaw acceptance criteria (including a listing of what the safety factors associated with this criteria are) including a discussion of whether the acceptance criteria accounts for continued degradation between inspections

f) when the scope of inspection or frequency of monitoring would be increased (e.g., if the amount of sag expected at a specific time of life is more than predicted, is the scope expanded).

g) the basis for the inspection frequency (or reference to the basis)

AECL Response: *To be provided by April 15, 2004.*

107. The FCTR indicates that Alloy X-750 is used in the garter springs. As mentioned in section 17.5 of the FCTR, Alloy X-750 is susceptible to experienced primary water stress corrosion cracking. Please discuss AECL investigations into the potential for the background level of moisture in the annular space to cause degradation of the garter spring material or any other materials in this environment such as the bearings or the bellows.

AECL Response:

The limits on the moisture content of the carbon dioxide annulus gas during normal operation (including shutdown) are low (upper limit of dew points in the range of –10°C to 0°C. Under these conditions, there is no possibility of the formation of liquid water within the annulus. Therefore, SCC in this environment is deemed to be not possible as the SCC mechanism requires the presence of a conducting electrolyte on the surface of the material. In the case of water ingress into the system through leakage, the design of the ACR annulus system would allow only four channel annuli to be flooded if a leak developed. Channels known to have had flooded annuli during a leakage event would be added to future in-service inspection requirements in order to confirm that the flooding had not resulted in any garter spring degradation.

Degradation of bearings could also occur in the case of flooded channel annuli but not otherwise. Bearing degradation could potentially lead to constraints on channel elongation and thermal movement for which the bearings were designed. This would likely be detected by measurements of channel elongation which would indicate abnormal behavior for channels with bearing deterioration. An elongation inspection program for channels having had flooded annuli would be required.

No deterioration of bellows components is expected under any operating conditions.

108. The ACR-700 pressure tube end fitting is designed from a modified Type 403 stainless steel with an Inconel alloy 625 feeder elbow attached by a weld. A feeder tube is welded to the outlet side of the feeder elbow. The staff is not clear regarding the material or materials that make up these feeder tubes between the feeder elbow and the headers. Please provide information on the materials specifications for the tube material(s). Please address the potential for degradation, such as erosion corrosion or stress corrosion cracking, to occur in any locations in these elbows and tubes. Please include the results of operating experience and any additional research performed to account for differences in design and/or environmental conditions from existing CANDU reactors.

In addition, please address programs that will be implemented to inspect or monitor the condition of these elbows and tubes and the basis for these programs.

AECL Response:

A design change has been implemented. This design change substitutes a bolted, gasketed seal between the end fitting and a feeder elbow end component. The feeder material has been selected as SA- 312 Type 316 N with additional requirements to limit the carbon concentration to 0.03%. The austenitic material has been selected specifically to address issues of flow accelerated corrosion of carbon steel (SA 106 Gr B) in some CANDU 6 reactors. A design change for the CANDU 6 reactors was implemented for recent construction to address flow accelerated corrosion through the addition of a minimum chromium content requirement to the SA-106 B specification. Although this has been demonstrated to be effective in significantly reducing flow accelerated corrosion under CANDU 6 temperature and coolant chemistry conditions, the increased temperatures and flow velocities in ACR feeders have led AECL to specify austenitic stainless steel for protection against flow accelerated corrosion. Fabrication requirements for the feeders will ensure that residual stresses are maintained at a low level and that welds are not sensitized. Details of the elbow bolted connection continue to be worked out within the design process.

The feeders require an alternative inspection program that will be provided with the application for design certification. The intent is to use a risk-informed process.

109. In section 9.2.3, information pertaining to the possible blocking of the "pig tails" in the annulus gas system was provided. To limit this blocking, oxygen is now maintained in the annulus gas system. Please discuss whether the addition of oxygen has been successful in limiting the blocking of the system. Also discuss how it was determined that the pig-tails were becoming blocked. Discuss whether blocking of the pig tails (or annulus gas system) can go undetected during operation and the implications of this.

AECL Response:

Blocking of annulus gas "pig tails" occurred in a small number of plants. Some plants operating without oxygen addition have not had any occurrences of blockage. Plants in which blockage occurred have had no blockages since oxygen addition was implemented.

Blockages were detected by monitoring of flow rotameters for each annulus gas string.

Detection of blockage could potentially go undetected for the period between observations of the rotameters on the annulus strings. There is not a uniform practice for monitoring of these rotameters within the CANDU utilities. Blockages would result in a delayed response time to leakage within the system. The blockage would be cleared when the pressure in the blocked annulus rose due to the leakage.

110. Many of the plots of corrosion rates (or oxide thickness) show a large amount of scatter. Conceptually discuss how the design of the various components accounts for the scatter in these correlations to ensure that the component will maintain adequate margin against failure (under normal operation and postulated accident conditions) at the end of the operating life of these components (or at the most limiting time in life given that some factors may "strengthen" a material whereas others "weaken" a material). For example, are 95-percent upper prediction limits (or bounding estimates) used on all inputs of the analyses?

In addition, for several components there are a number of degradation mechanisms that potentially can affect their integrity. Discuss how the effects of these mechanisms are combined in assessing whether adequate design margins are maintained at the end of life (i.e., has the pressure tube been analyzed for the worst case sag, the worst case corrosion, etc, to ensure that at the end of the operating life (or the most limiting time in life) the pressure tube will still have margin to failure (as specified in the Code or in the design rules) under normal operating and postulated accident conditions).

AECL Response:

In component design, for issues likely to affect plant life, predictions of performance are generally based upon bounding estimates (95% prediction limits at 50% confidence). Wall thinning results from both dimensional changes due to irradiation deformation and from corrosion. The strength of the pressure tube at the end of life for ACR uses upper bound corrosion rates and upper bound wall thinning rates to produce the thinnest wall and compares resulting stresses with the allowable stress intensities for all loading conditions (i.e. Levels A,B,C andD) based upon the material allowable stress intensity (S_m) and neglecting strengthening due to irradiation and due to anisotropy (S_m is based upon axial strength measurements but tubes are stronger in the circumferential *direction). In the case of ACR wall thickness, there will be additional margin on strength due to the need to limit the rate of diametral creep due to irradiation.*

For ACR, sag is unlikely to present a life limit as the sag rates will be low for the strong calandria tube and there is a large pressure tube to calandria tube initial gap.

111. Discuss your policy for removing data from correlations. Are all data retained unless they are determined to be from an invalid test (which must be distinguished from a test that did not match expectations)?

AECL Response:

Data that lie outside correlations would generally be removed only if it could be determined with some confidence that they were not valid. Such data would be carefully scrutinized and the potential reasons for the results would be evaluated. Such data could result in a non-conformance report and, if required, a resulting root cause analysis. However, AECL does not have a documented company policy on the specific issue of removing data from correlations.

112. The staff will need to review the research programs which support extending qualification of the component materials from existing CANDU operating conditions to ACR-700 operating conditions for each of the degradation mechanisms. Please discuss the status of these programs and the possibility of scheduling a separate meeting to discuss the basis for these programs and available results.

AECL Response:

The program of material testing in support of ACR has been defined and is being implemented. As indicated in other responses, some aspects of degradation have been under investigation for some time through tests of current CANDU pressure tube material by irradiation at ACR temperatures. This includes corrosion and deuterium ingress testing in the Halden reactor. More recently, DHC crack growth rate tests of material taken from pressure tubes irradiated in current CANDU reactors has been tested at ACR temperatures with added hydrogen.

Prototype ACR pressure tubes have been successfully produced and have been characterized in terms of their tensile properties, microstructures and crystallographic textures. These characteristics of the prototype tubes place these tubes within the range of characteristics of Zr 2.5Nb material in use in the current CANDUs. Testing of material from these prototype tubes for corrosion and mechanical properties following irradiation has not yet started and most of these planned tests are of significant duration (several years).

A separate meeting to discuss results of testing at ACR temperatures, characterization of prototype tubes as well as the basis of and plans for additional testing can certainly be arranged.

113. The Fuel Channel Technology Report (FCTR) describes long-term thermal fatigue tests of the rolled joints in a water loop followed by helium leak test and hot pullout test. The FCTR does not address fatigue of the Fuel Channel base metal (Zr-2.5Nb) or of the End Fitting base metal (403 SS) in the ACR-700 water coolant including impurities under operating temperatures and stresses. Research for LWR materials has shown that the environment can have a significant effect on the fatigue and SCC resistence of austenitic and ferritic steels. In pure water with oxygen levels in the parts per billion (ppb) range (similar to oxygen levels in PWRs), the fatigue life of austenitic stainless steel components is reduced relative to that in air, while there is no significant reduction to fatigue life of austenitic steel for oxygen levels in the parts per million (ppm) range (similar to BWRs). On the other hand, for oxygen levels in the ppm range, while the fatigue life is not reduced compared to that in air, the susceptibility of these materials to stress corrosion cracking (SCC) is increased. For ferritic steels there is a reduction in fatigue life when oxygen levels are in the ppm range, but no reduction in fatigue life at the ppb level. Ferritics are more resistant to SCC, but less resistant to erosion corrosion. Therefore, it is important to know how ACR-700 materials behave in their operating environment so that long term integrity and safety margins can be evaluated.

What information is available to support the design basis relative to the effect of the ACR-700 coolant environment, including impurities, on component fatigue life and SCC susceptibility of pressure boundary materials? In the absence of experimental information, what justification, or accommodation, is made for the fatigue design curve/method used (such as those derived from test data in air)?

AECL Response: *To be provided by April 15, 2004.*

114. The FCTR describes extensive creep testing including thermal creep, irradiation creep, and irradiation growth, however it is not clear whether the tests were conducted with the creep specimens in contact with typical reactor water including impurities. The coolant environment (including impurities) could reduce the creep life of components as observed also for fatigue life described in the previous question. Was creep testing on component materials performed in typical reactor water containing impurities? In the absence of experimental information, what justification, or accommodation, is made for the creep design curve/method used (such as those derived from test data in air)?

AECL Response:

Creep testing has been carried out under a variety of conditions. Bent-beam stress relaxation tests of pressure tube material have been carried out under CANDU water chemistry conditions (in light water) but most creep tests are not carried out under those conditions. Long-term creep ductility tests at CANDU 6 operating temperatures are based upon small capsule tests of Zr2.5Nb material manufactured as a small diameter tube having a pressure tube microstructure. These capsules have been irradiated in NaK in order to provide adequate heat transfer to ensure specimen temperature uniformity and without the need for a pressurized loop for high flux neutron irradiation. There are no plans to carry out creep tests to end-of-life fluences at ACR temperatures as such tests would require many years of irradiation even in a high flux reactor and the available information, as indicated in the FCTR, provides support for creep ductility with significant margin beyond the deformations expected by the end-of-life in ACR. Rather, creep capsule tests will be irradiated to provide the information required to confirm the steady state creep rates under ACR conditions. These capsules will also be irradiated in NaK.

As indicated in the FCTR, the design equations for pressure tube deformation are based upon in-reactor material testing with validation based upon comparison with actual pressure tube performance in reactors. For the ACR, most of the channel will be operating at temperatures for which the creep behavior is well known as it is within the range of current operation (CANDU 6 operates with outlet temperatures of 310°C). The objective of the additional testing program is to confirm the expected behavior for the regions of the ACR pressure tubes at higher temperatures.

Examination of removed pressure tubes from operating CANDU reactors shows that other than surface oxidation, there is no indication that any other adverse effect would reduce the creep life of the pressure tube in a water environment. Although the oxide layer can be cracked, the cracks do not extend to the metal, and they do not induce cracking in the metal.

115. The FCTR discusses the gas side corrosion and hydrogen ingress in terms of maintaining an oxidizing, dry environment. Have creep, fatigue, and SCC testing been conducted in this environment of carbon dioxide, hydrogen, oxygen, water, and other possible impurities to evaluate the effect on pressure tube, end-fitting, and calandria tube life under operating temperatures and stresses?

AECL Response:

No. There has been no indication that the environment in the annulus gas has had any effect on the integrity of any of these components. The hydrogen level is maintained very low by the presence of the oxygen that acts to scavenge the hydrogen in the annulus gas.

Hydrogen entering the pressure tube from the annulus gas side would be taken into account through inspection and surveillance but would be indistinguishable from the hydrogen entering the tube from the waterside due to corrosion.

The stresses in the pressure tube are generally quite low except near the rolled joint. where they are limited to a fraction of the stress required to initiate DHC at a smooth surface.

116. What materials and which components have a fatigue or creep analysis for design? What are the environmental and stress conditions surrounding those components?

AECL Response:

ASME rules for fatigue analysis are applied to all Class 1 components including the pressure tube and the end fitting.

117. CANDU reactors use heavy water as a coolant. The proposed coolant for the ACR-700 is light water, however heavy water will still be used as a moderator. In this regard, is there test data available that compares fatigue, creep, corrosion, SCC, and DHC behavior of the component materials in typical heavy water (including impurities) to that in typical light water (including impurities) and in air where appropriate? In the absence of experimental data what assumptions are made with respect to the materials behavior in the particular environment versus the available data? Explain and justify the assumptions.

AECL Response:

Impurity levels in the coolant will be maintained at very low levels through the purification system and are not expected to be different in type or quantity in the light water ACR coolant compared to the current heavy water coolant. Current CANDU 6 plants operate with very low impurity levels. Species such as Cl, F, are generally below the detection limit (typically 5 ppb; Sulphate is not routinely monitored in CANDU 6 but it will be monitored in ACR. In CANDU 6, total organic carbon levels in the reactor coolant are in the several hundred ppb range and some ammonia is also present. Neither of these latter species are deemed to be potentially harmful.

Light and heavy water are very similar from a chemical viewpoint. Differences between light water and heavy water in terms of corrosion and ingress of hydrogen isotope for Zr 2.5Nb materials are within uncertainties of the experimental determinations in general. It could be expected that differences in behavior between hydrogen and deuterium could appear in factors dependent upon diffusion rates and that any such differences would be *of the order of the ratio of the square roots of the isotopic masses, i.e. √2 (1.41). DHC testing, historically, has been carried out using added hydrogen (^1H) instead of deuterium and hence the data available already meet ACR requirements in terms of use of the relevant isotope of hydrogen.*

118. Hydrides form in the outer surface of pressure tubes that come into contact with calandria tubes because of hydrogen diffusion to the lower temperature at the contact point and the dependence of the hydrogen solubility limit on temperature. Hydrides are undesirable because of the potential for DHC. Is there data on the temperature of the pressure tubes at the point of contact with the garter springs? If not, is there analysis of the pressure tube temperature at locations where garter springs contact both the calandria tube and the pressure tube, taking into account the temperature difference between the two tubes and the thermal conductivity of the garter spring?

AECL Response:

There are no measurements of the temperature at the contact point of the pressure tube with the garter spring spacer. The temperature is calculated by analysis. The spacer is heated by gamma radiation that must be taken into account when the reactor is operating. In general, the temperature reduction due to the garter spring is very localized since the overall contact consists of a number of very small contacts. For ACR, the design of the spacer has not been completed and the thermal analysis has therefore not been done. The objective of the design will be to minimize the temperature reduction in the pressure tube at the points of contact with the calandria tube so that there is confidence that hydrides cannot be precipitating in the pressure tube during normal operating conditions at the garter spring contacts. Due to pressure tube elongation with operating time, the position of the contact on the pressure tube changes continuously with time. Therefore there will not be any buildup of hydride at the contact spot. Such buildup could potentially occur during shutdown-start up cycles due to hysteresis in the solubility behavior of hydrogen in pressure tube material if the TSSD were to be exceeded in the tube at the garter spring contact location and if the location were to remain fixed. It is important to point out that exceeding the TSSD solubility limit at the contact location is not a sufficient condition to result in hydrides precipitating at that location.

119. The increase in the maximum operating temperature of the ACR-700 to 325□C will increase the kinetics of many of the materials degradation processes and modes. Does data exist on the different degradation processes and modes at the higher temperatures and in the appropriate coolant environments? If not, has existing data been extrapolated to account for the effects of increased kinetics, such as increased crack growth rates and reductions in crack initiation times? What were the assumptions and bases for the extrapolations? Are there adequate safety margins remaining at these higher temperatures?

AECL Response:

Testing has been carried out at temperatures relevant to ACR outlet temperatures on a number of factors that must be taken into account in assessing potential degradation.

Such tests include corrosion and hydrogen isotope uptake testing in a heavy water loop in the Halden high flux reactor in Norway. Heavy water has been used because it improves the ability to measure relatively small changes in concentration attributable to the corrosion. Testing has been carried out at 325°C and a range of parametric tests have been carried out to aid in improvement to the corrosion and hydrogen ingress models. ACR prototype pressure tube material corrosion tests under irradiation will begin this year and are expected to last for at least three years for a total time under full power irradiation of 560 days. The amount of corrosion expected in these tests is small as the total wall loss due to corrosion over a 30 year ACR pressure tube life is predicted to be approximately 200 microns. The deuterium uptake in theses tests will be measured to to show that expected benefits of slight pressure tube chemistry changes to reduce hydrogen isotope ingress due to corrosion have been achieved.

Testing of ACR pressure tube rolled joints in a loop facility upgraded to ACR conditions will also be carried out. These tests will begin after the rolled joint development tests are completed this year. The objective of the tests will be to provide information on rolled joint corrosion and hydrogen isotope uptake performance in long tem tests (a minimum of three years) to validate the predictive model for rolled joint hydrogen ingress for ACR.

Deformation, in general, does not lead directly to any integrity issue. Deformation testing will be carried out at ACR temperatures to confirm the steady state creep rates.

Tests to measure characteristics of delayed hydride cracking in irradiated pressure tube material from current reactors at ACR operating temperatures have begun. These tests will be supplemented by tests of prototype ACR material in both the unirradiated and irradiated material conditions. Conservative predictions of crack growth behavior for ACR conditions using extrapolations of irradiated material DHC crack growth rates indicates that, even in conditions under which crack growth could occur at operating temperatures (which are not expected during the pressure tube lifetime), there would be sufficient time available for the operator to detect a leak and safely shut down and depressurize the reactor to avoid pressure tube rupture. The extrapolations of crack growth rate are based upon Arrhenius temperature behavior with well-characterized activation enthalpies. No credit has been assumed for a potential reduction in crack growth rate due to higher temperature operation.

120. DHC requires hydride precipitation and a tensile stress above a certain threshold. Figure 9-2 of the FCTR shows the deuterium concentration profile along a pressure tube

in a CANDU reactor after 14 EFPY of operation. Superimposing the hydrogen solubility limit (terminal solid solubility - TSS) on this curve indicates hydrides are present at the inlet and outlet of the pressure tube. This is also shown in Figure 9-14. Please provide an analysis or Figure that shows the normal operating, transients, and design basis accident stresses relative to the DHC threshold stress superimposed on the deuterium concentration profile. Include operating stresses such as channel vibration during at-power operation and start-up and shutdown stresses.

AECL Response:

Figure 9-2 of the FCTR is a plot of measured ***deuterium*** *concentration in the pressure tube along its length. In order to compare hydrogen isotope concentrations with the solubility limit, it is AECL's normal practice to add half the deuterium concentration (by weight) with the hydrogen concentration to arrive at a "hydrogen equivalent" concentration that can then be compared with the hydrogen solubility limit. If the hydrogen concentration in the tube in Figure 9-2 were 10 ppm, then at the inlet end, a deuterium concentration of approximately 40 ppm is required for the total hydrogen isotope concentration to exceed the solubility limit at the operating temperature. This concentration is exceeded only over a short distance from the end of the tube, and in this case, is mostly or entirely within the compressive stress zone of the rolled joint. Similarly, at the outlet end, the solubility limit is not exceeded at operating temperature.*
Figure 9-14 of the FCTR shows a similar situation since the compressive region of the rolled joint extends from the end of the pressure tube to the burnish mark indicated in the upper portion of the figure.

Notwithstanding the explanation in the foregoing, the intent of the fuel channel design for ACR is to maintain the concentrations of hydrogen below the solubility limit at operating conditions for the life of the pressure tubes for all parts of the pressure tube between the compressive regions of the rolled joints. This would imply that at operating temperatures, independent of the stress state, DHC could not occur.

For an unflawed tube, there will be no operating transients that result in a stress in the pressure tube in excess of the stress required to initiate DHC at a smooth surface. Since all operating transients have not yet been finalized for ACR, specific stresses cannot be provided. Nevertheless, there is confidence in the preceding statement as 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 shall not exceed 67% of the tensile stress required to initiate DHC in laboratory tests of unnotched specimens (450 MPa based upon tests of irradiated specimens). This requirement will be met by the ACR design.

121. Please provide the technical basis and any unpublished data to support the relationship in section 12.2.1.3 of the fuel channel technology report that the lower bound of K_{IH} = 4.5 MPa(m)^0.5.

Please provide data and analyses demonstrating similarities and differences between hydriding and deuteriding in the Zr-2.5Nb pressure tube material.

AECL Response: *To be provided by April 15, 2004.*

F. Application of Leak-Before-Break

Regulatory Structure

[1] 10 CFR Part 50, Appendix A - General Design Criterion 14 - Reactor Coolant Pressure Boundary

[2] 10 CFR Part 50, Appendix A - General Design Criterion 4 - Environmental and Dynamic Effects Design Bases

[3] Federal Register, Volume 52, page 41283, "Modification of General Design Criterion 4 Requirements for Protection Against Dynamic Effects of Postulated Pipe Ruptures," October 27, 1987. (52 FR 41283)

[4] Draft Standard Review Plan (DSRP) Section 3.6.3

[5] NUREG-1061, Volume 3, "Report of the U.S. Nuclear Regulatory Commission Piping Review Committee, Evaluation of Potential for Pipe Breaks".

Analysis

GDC 14 establishes the overarching requirements for reactor coolant pressure boundary integrity. GDC 4 addresses specific concerns related to the dynamic effects of postulated piping ruptures, including pipe whip, jet impingement, etc., and provides a clause stating that such effects may be eliminated from a facility's design basis "when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis of the facility." The performance goal associated with the use of the term "extremely low" relates to having rupture frequencies of 10^{-6} of less per reactor year when all rupture locations are considered in the fluid piping system to which the "analyses" are being applied. This performance goal is clearly specified in 52 FR 41283.

In 52 FR 41283, the NRC also clearly laid out the fact that the "analyses" referred to in GDC-4 are those outlined in Chapter 5 of NUREG-1061, Volume 3 and called "Leak-Before-Break" (LBB) analyses. The information in NUREG-1061, Volume 3, Chapter 5 is consistent with that in DSRP 3.6.3.

NRC Staff Assessment of AECL Information

The NRC staff's assessment is summarized in the following points:

122. From a regulatory perspective, it is unclear what, if any, "credit" is taken in the design of the ACR-700 based on the analysis by AECL using leak-before-break principles. As noted above, in the U.S. regulatory structure credit for LBB approval is directly related to eliminating the dynamic effects of piping rupture from a plant's design and licensing basis. AECL must explain what credit is taken within the ACR-700 design for its analysis to leak-before-break principles, or alternatively, how would the design have to be modified if the NRC staff did not accept the AECL analysis.

AECL Response:

AECL is not seeking any credit for Leak Before Break under Draft NUREG 1061 Vol. 3 in the sense of eliminating the need to analyze pressure tube ruptures. Pressure tube ruptures are postulated and the dynamic effects of such rupture are evaluated as part of safety analyses. Annulus gas leak detection and response are used as a defence-in-depth. The calandria tube in ACR will have a low probability of consequential rupture in a pressure tube rupture event. Therefore, the fuel channel complies with GDC 4.

Leak-before-break may be applied to other parts of the reactor coolant piping system. Any such application will be supported by analysis consistent with the requirements of NUREG 1061 Vol.3.

123. The technical "sequence-of-events" analysis performed by AECL to demonstrate leak-before-break principles is not consistent with the formal LBB analysis endorsed in NRC NUREG-1061, Volume 3 or Draft SRP 3.6.3, although the underlying concept is similar. In the case of the formal LBB analysis of NRC NUREG-1061, Volume 3, there is an understanding that passing such an analysis is equivalent to demonstrating the probability of fluid system piping rupture would be on the order of 10^{-6} when all rupture locations are considered in the fluid system piping or portions thereof.

AECL must provide more technical information to explain how their "sequence of events" analysis is performed. Information on the performance of the leak detection system would be relevant. If, from question (1) above, the intent of their analysis using leak-before-break principles is to address GDC-4 issues, AECL must show how their analysis leads to a conclusion that it demonstrates the probability of fluid system piping rupture would be on the order of 10^{-6} when all rupture locations are considered in the fluid system piping or portions thereof. The NRC staff would have to endorse a subsequent policy change to accept the AECL analysis as a method for demonstrating compliance with the "dynamic effects exclusion clause" in GDC-4.

AECL Response:

Not applicable. See response to 122.

124. Beyond the design of the pressure tubes, is AECL using leak-before-break concepts to define what must be included (or may be excluded) from the ACR-700 design and licensing basis? For example, are pipe whip restraints installed to protect against the dynamic effects of postulated piping ruptures? This would be required under GDC-4 unless a formal LBB analysis supported eliminating the dynamic effects of piping rupture from the ACR-700 design and licensing basis.

AECL Response:

Not applicable. See response to 122.

G. Irradiated Materials

125. Please address the following topics. To the extent portions of these topics may be discussed in the fuel channel technology report, please refer to the applicable subsections and tables.

a) boundaries of concern for irradiation effects, i.e., components affected,

b) response to irradiation of the components within these boundaries; please also address synergistic effects, such as irradiation and delayed hydride cracking, and the effect on component response as a result of differences in component geometry and environmental conditions between operating CANDU reactors and the ACR-700,

c) monitoring/inservice inspection to preclude failure of components within these boundaries,

d) assumed failures of components within these boundaries, basis for failures assumed, and basis for consequential failures or lack thereof, and

e) significance of assumed failure.

AECL Response:

a) *For the reactor coolant pressure boundary, the components affected by irradiation are the pressure tube and the inboard end of the end fitting including the rolled joint region. In current CANDU reactors, the ends of the pressure tube are not fully irradiation hardened as the fast neutron flux is significantly reduced at the pressure tube ends. In ACR, the details of the fluxes at the rolled joint region are not yet available but similar reductions are expected. Moving outboard (away from the core) the neutron flux continues to fall and the effects are reduced.*

The supporting structures for the fuel channel are also subject to irradiation effects. These include the annulus spacers, the calandria tubes, the inboard ends of the lattice tube and the inboard bearing components.

Of course, all the structures within the core (reactivity mechanisms and their guide tubes, liquid injection shutdown system nozzles) as well as the calandria vessel are also subject to neutron irradiation. However, as these are not part of the coolant pressure boundary, they are not discussed further in this response.

b) *There are a number of effects of irradiation on the pressure tube. The effects of irradiation on changes in the mechanical properties of pressure tube material are discussed primarily in Section 10 of the FCTR with the supporting data being shown in the accompanying figures.*

Irradiation produces changes in the microstructure of the material on a fine scale. As discussed in Section 8.1 of the FCTR, irradiation produces point defects that migrate and agglomerate in a variety of ways. During the initial phase of irradiation, the principal changes taking place are the formation of small dislocation loops on specific crystallographic planes. The formation of these loops has two effects: there is a small volume change of the material; and, the mechanical properties are changed. These changes occur over relatively small neutron fluences corresponding to operating times of less than a year. Subsequent irradiation causes no further volume changes of the material, but the material continues to change dimensions due to creep and growth processes. The microstructure develops relatively slowly over time after the initial changes take place. Some of the microstructural / microchemical changes include precipitation of Nb rich particles within the α-phase of the alloy and redistribution of other chemical species (such as Fe) within the microstructure. Dislocation distributions also evolve with irradiation fluence and this evolution is dependant upon irradiation temperature. These microstructural changes are reflected in on-going, relatively small, changes to the mechanical properties.

Figure 10-6 of the FCTR shows the effects of irradiation on the transverse strength of the material. The initial transient is right up against the vertical axis in this figure. Figure 10-7 shows the changes in the transverse total elongation due to irradiation. The key change in tensile behavior is a trend to more localized deformation. Fractures in tensile tests remain ductile although the uniform deformation is reduced in extent.

The fracture toughness also changes with irradiation. The effects are discussed in Section 10.4 of the FCTR. Figure 10-16 shows the effects of irradiation on fracture toughness for two materials having differing chlorine concentrations. ACR pressure tubes will be produced using methods to achieve very low chlorine concentrations and fracture toughness will be maintained at a high level even after irradiation. However, it is important to note that even the higher chlorine material has sufficient fracture toughness in current pressure tubes. Critical cracks in current CANDUs, even for high-chlorine-content tubes, are large enough that there is confidence that a crack, potentially developing from a small flaw if all the requirements necessary for delayed hydride crack propagation were to be met, would leak and be detected allowing reactor shutdown and depressurization thus preventing tube rupture.

Irradiation also has an influence on the crack growth by the delayed hydride cracking (DHC) mechanism. If the conditions necessary for the DHC mechanism to operate are present, then cracks can grow somewhat faster in irradiated material than in unirradiated material. This is illustrated in the FCTR in Figure 12-7. The increase in DHC velocity at a single temperature as a function of neutron irradiation fluence is shown in Figure 125-1 below. The blue points correspond to material from a single tube whereas the other data is from a number of different tubes. It can be seen that there is, again, an initial transient increase in crack growth rate followed by a very slowly changing growth rate.

The mechanical properties of the zones of the end fitting subject to neutron irradiation also change with the irradiation. In particular, there is a shift (increase) in the fracture toughness transition temperature with neutron irradiation. This is discussed in the FCTR in Section 15.3.2. and is illustrated in Figures 15-1 and 15-2.

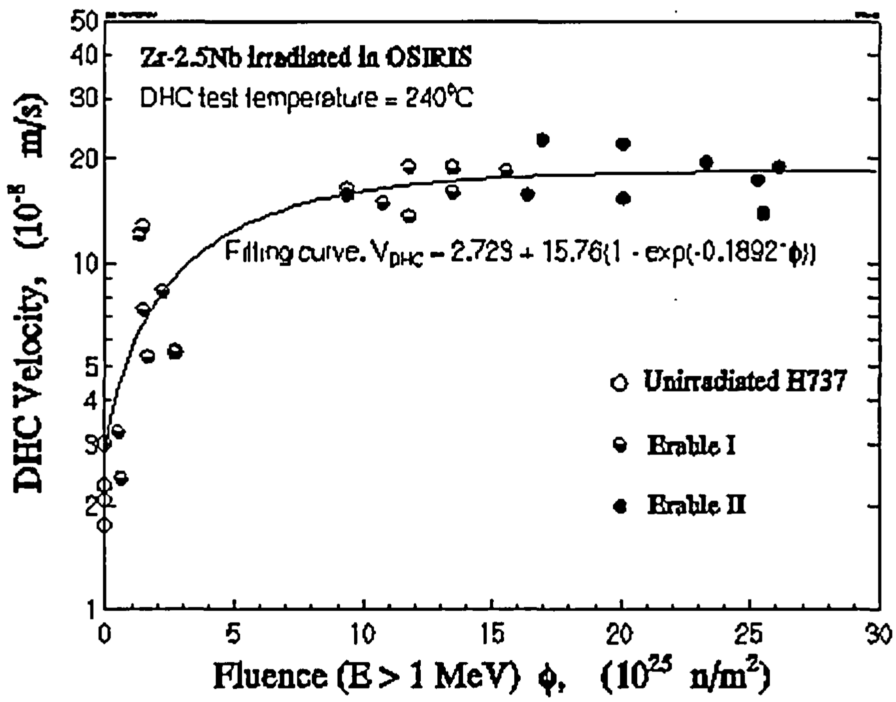


Figure 125-1: Showing the change in DHC velocity (crack growth rate) measured at 240°C with irradiation fluence. End-of-life fluence in ACR will be greater than shown here.

c) *The inspection and monitoring program for these components includes In-service Inspection of a limited number of pressure tubes to detect potential generic degradation together with continuous monitoring of the annulus gas that provides a defence-in-depth against crack growth to rupture of any unknown flaws. The inservice inspection results allow assessments of the potential for DHC to actively occur during reactor operation from unknown flaws. The Canadian Inspection Standard, CAN/CSA N285.4, also requires removal of pressure tubes to carry out testing of fracture toughness and DHC crack growth rate. Taken together, these provisions limit the potential for pressure tube rupture.*

d) *Assumed failures of these components include spontaneous pressure tube rupture with and without calandria tube failure as well as single channel events resulting in loss of flow in the channel. These latter, low-probability events can result in both pressure tube and calandria tube failure. More details of these latter events will be provided in a subsequent submission.*

For the case of a spontaneous pressure tube break, the annulus is rapidly filled with coolant and the calandria tube must withstand some degree of waterhammer loading. The annulus bellows at the channel ends will likely fail by over-pressurization and coolant will escape to the reactor vault. The calandria tube is *being designed to withstand full reactor coolant pressure (the flow from the channel is choked at the bearings in the lattice tube) until the reactor can be shut down and depressurized. Under these circumstances some high-stress creep of the calandria tube can occur. If the calandria tube fails due to waterhammer or due to high stress creep, some damage to surrounding channels can occur but they will not fail from impact of the broken channel. Any adjacent reactivity guide mechanisms could also be damaged. The calandria vessel will see a pulse of increased pressure if the calandria tube fails in this event. The vessel can be shown to sustain the event without failure. It is important to note that, even though failure of the calandria tube in the event of spontaneous pressure tube rupture is analysed. ACR calandria tubes are expected to have a low probability of failure in this event.*

e) *For the case in which a spontaneous pressure tube rupture does not fail the surrounding calandria tube, the consequences are relatively minor. The fuel would be expected to remain well cooled at all times. The loss of coolant through the annulus bellows would lead to Emergency Coolant Injection. Since damage to the core would be confined to a single channel, that channel would be replaced after the damaged fuel had been recovered.*

In the case of rupture of the calandria tube, a low probability event as indicated in the foregoing, the damage is obviously greater. Although the fuel would remain cooled, some fuel or fuel pieces could be ejected into the moderator. Calandria tube rupture could lead to guillotine rupture of the pressure tube, possible damage to the dual-stud restraint, potential ejection of one or both end fittings, and potential loss of moderator through the open lattice tubes. Emergency coolant injection would occur. Recovery from such an event would require a great deal of inspection and replacement of damaged core components.

H. On-Power Fueling

The design of the ACR must meet the NRC regulations which are contained in Title 10 of the Code of Federal Regulations (10 CFR). In addition to the regulations or requirements that are contained directly in 10 CFR, certain sections of the ASME Code, Division 1, are incorporated by reference in 10 CFR 50.55a. For example, design rules of Section III and inservice inspection rules of Section XI are incorporated by reference in 10 CFR 50.55a. The NRC has developed Regulatory Guides which describe methods acceptable to the NRC staff of implementing specific parts of the Commission's regulations, to delineate techniques used by the staff in evaluating specific problems or postulated accidents, or to provide guidance to applicants. These Regulatory Guides are not substitutes for regulations, and compliance with them is not required. Methods and solutions different from those set out in the Regulatory Guides are acceptable if they provide a basis for the findings requisite to the issuance of a design certification or license by the Commission. In addition to these Regulatory Guides, the NRC has a Standard Review Plan (SRP) which the NRC staff uses to review applications. This review plan frequently references the Regulatory Guides. The SRP is organized by closely related technical topics and the organization of applications for design certifications of light water reactors (LWRs) typically follow the format of the SRP.

Although the AECL report on "The Technology of On-Power Fueling" provides an excellent description of the function, operation, and design considerations of the fuel handling system, it does not provide a clear description of how the Fuel Handling system meets the NRC requirements. For example, in the area of materials engineering, the document did not describe what the material specifications were. As a result the staff could not ascertain whether the proposed design of this portion of the reactor coolant pressure boundary would meet the requirements of General Design Criteria 1 and 30 of 10 CFR 50 Appendix A. Similarly, it did not address whether the materials were compatible with the environment (both internal and external to the equipment), what the chemistry of the system would be, how the materials would be fabricated and processed, how welding would be performed and what the inservice inspection requirements would be. All of this information is necessary to ascertain whether the proposed design satisfies the requirements contained within 10 CFR 50, Appendix A. Furthermore, the design parameters of the system were not provided (e.g., design temperature, pressure, dimensions, etc.). The preceding is only a partial list of regulations in the area of materials engineering. A complete listing of regulations applicable to LWRs can be assembled from the SRP. A list of SRP sections in the area of materials and chemical engineering will be provided separately.

In many cases these issues could be addressed by indicating that the structure, system, or component would be fabricated, erected, tested, and inspected consistent with existing Regulatory Guidance and/or the Standard Review Plan. If methods different than those discussed in these regulatory guidance documents are used, their use must be justified. Given the materials of the fuel handling system and the water chemistry regimes are probably similar to those used in light water reactors, much of the guidance provided in existing NRC documents is appropriate for the ACR; however, for areas such as for inspecting the fuel handling equipment there may not be regulatory guidance for how to satisfy the NRC regulations. This will require detailed information from the applicant.

Despite the limitations of the report, the staff could ascertain that various technical and policy issues will need to be addressed by AECL. These issues are discussed below:

126. Use of Canadian Codes and Standards

Because of the design of the ACR, the Canadians have supplemented the rules for pressure retaining components contained within the ASME Boiler and Pressure Vessel Code since the ASME Code rules may not apply, may not exist, or may be insufficient to address the ACR design.

Applicants are required to follow the ASME Code as discussed in 10 CFR 50.55a. As a result, if the ACR-700 design is such that an ASME Code rule cannot be satisfied, the applicant will need to justify the reason the Code rule cannot be satisfied and request an exemption from the regulations pursuant to 10 CFR 50.12 or request NRC to approve an alternative to the ASME Code pursuant to 10 CFR 50.55a(a)(3)(i). For the cases where no ASME code rules exist or where they are insufficient to address a portion of the ACR design, the code rules to be used (including the technical basis for the standard) should be submitted for NRC review and approval. The NRC would review these supplemental codes to ensure they satisfy the NRC regulations pertaining to Materials Engineering (e.g., General Design Criteria 1, 4, 14, 15, 30, 31, and 32). For the fuel handling system, none of this information was provided. For the use of supplemental codes, the NRC will need to address how the supplemental codes will be incorporated into the regulatory framework as requirements (e.g., in the design certification rule).

AECL Response:

An outline presentation on our code use was provided as part of the presentations in our March 4th 2004 meeting at NRC. AECL is preparing a document detailing our supplementary rules for our pressure boundary related parts explaining the background for those rules including how we make use of similar methodologies to ASME code rules for those parts. This will be submitted to NRC this summer. The detailed justification for both those rules and the design will be provided on an ongoing basis both through the current focus topic discussion and as part of the actual DCD submission.

127. Inservice Inspection (ISI) Program

The inservice inspection requirements for the Fuel Handling system are not addressed in the ASME Code; however, a Canadian standard has been developed to address the inspection of this portion of the reactor coolant pressure boundary. Given that the inservice inspection requirements for the Fuel Handling system would not be covered through licensee implementation of 10 CFR 50.55a, the NRC will need to determine how the inspection requirements would be captured elsewhere such as in a rule (e.g., the design certification rule). and/or in the technical specifications.

AECL Response:

Justification of the inspection methodology and plan and their basis in past experience and risk analysis for ACR will be provided for fuel handling and our other systems as part of our licensing submission. As per our March 4th 2004 presentation we gave a general overview of our use of risk informed inspection.

128. Operating Experience

The operational history of the fuel handling system was provided. Although useful from an operational standpoint, a more detailed assessment of the inspection results of the fuel handling equipment will be needed to ascertain whether the inspection and maintenance requirements are adequate for ensuring the integrity of this portion of the reactor coolant system. For example, in section 5.3.2.1, "Fueling Machine On-Reactor", a description of several hose failures was provided; however, the corrective actions taken to address these occurrences was not provided (e.g., redesigning the component or increasing the inspection frequency in the Canadian Standards Association (CSA) standard). In addition a Fueling Machine heat exchanger failed (Section 5.3.2.5); however, the cause and corrective action were not provided. The intent of this information would be to verify the adequacy of the design and/or inspections for satisfying the requirements of 10 CFR 50, Appendix A.

AECL Response:

See response to 127.

129. Limiting conditions for operation

Although a description of the fuel handling system was provided, critical variables that must be satisfied for fueling to occur were not provided. From a materials engineering standpoint, critical variables may include limits on water chemistry and flaw acceptance limits in the fuel handling system.

AECL Response:

Fueling machines use main Reactor Coolant System water chemistry, with connections to the RCS pressure and inventory control and purification system for supply and return.

Inspection rules for our ongoing maintenance and inspections programs, much of which typically occurs during plant operation, will be provided as per answer 127 above.

130. Classification of System Components

The description of the on-power fueling system did not contain sufficient detail for the NRC staff to determine how various components were classified. That is, the NRC staff was unable to determine where the ASME Code classification boundaries have been established. The intent of this information would be to verify the adequacy of the Code classification boundaries with respect to NRC requirements. NRC Regulatory Guide 1.26, "Quality Group classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants", discusses requirements in this area.

AECL Response:
An overview of our classification was provided in our March 04, 2004, meeting. Further details are being provided this summer as discussed in the answer to question 126 above.