AECL
TECHNOLOGIES INC.

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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.J. Langman, "Requests for Additional Information – CATHENA Code for ACR-700 Application", May 14, 2004.

Re: **Response to the NRC's Requests for Additional Information (RAIs) on the CATHENA Computer Code (Non-proprietary Version)**

In response to an NRC staff request (Reference 1), Attachment 1 provides the non-proprietary version of AECL's responses to NRC staff requests for additional information on the CATHENA computer code.

The proprietary version of AECL's responses to these RAIs is submitted under a separate cover along with the reports listed in Table 1 of Attachment 2, which are also proprietary.

If you have any questions on this letter and/or the enclosed material please contact me at (301) 332-9152.

Yours sincerely,

Glenn Archinoff
Manager ACR Licensing

D070

/Attachments:

1. Response to NRC's Requests for Additional Information (RAIs) on CATHENA
2. List of Enclosures and References in AECL's Responses to the NRC's Request for Additional Information on CATHENA

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Attachment 1
Response to NRC's Requests for Additional Information (RAIs) on CATHENA

(Letter G. Archinoff to B. Sosa, "Response to the NRC's Requests for Additional Information (RAIs) on the CATHENA Computer Code", December 24, 2004)

AECL's responses to the NRC staff's requests for additional information on the thermal hydraulics code CATHENA are provided in italic fonts following each of the NRC's questions as follows:

The following questions and comments were generated to determine if the CATHENA code as it presently exists is able to adequately model ACR-700 transients and accidents or if additional code modifications and validations are required:

131. Draft Regulatory Guide DG-1120 "Transient and Accident Methods," Regulatory Position 1 provides 20 steps for a process of evaluation model development and assessment. These elements discuss how computer codes will be assessed for adequacy for specific applications, describes their usage with other computer codes and their qualification for the specific applications for which they will be used. Please address each of these 20 steps for use of the CATHENA computer code for ACR-700 safety analysis.

AECL Response
A response to this RAI will be provided by April 2005.

132. Step 2 to Regulatory Position 1 of DG-1120 discusses figures of merit which are the quantitative standards of acceptance that are used to define acceptable answers for safety analysis. For ECCS analysis, five specific criteria described in 10CFR50.46 must be met for LOCA analysis. Please include in your response if these five criteria for LOCA will be met for ACR-700 analyses using CATHENA, if not please provide the criteria that will be used and provide the technical as well as the regulatory basis for acceptance.

AECL Response
A response to this RAI will be provided by April 2005.

133. For LOCA and non-LOCA design basis transient and accident analysis, criteria for acceptance that are used by the NRC staff are found in NUREG-0800, "Standard Review Plan for Review of Safety Analysis Reports for Nuclear Power Plants." Where applicable, please indicate for each transient and accident category listed in Chapter 15 of NUREG-0800 for which CATHENA will be used, whether or not the acceptance criteria used by the staff will be met for ACR-700. If the NUREG-0800 acceptance criteria will not be met, please provide the criteria that will be used and provide the technical as well as the regulatory basis for acceptance. For events not found in NUREG-0800 for which CATHENA will be utilized in safety analyses for

ACR-700, please provide the acceptance criteria to be used and justify the technical as well as the regulatory basis for acceptance of analyses for these events.

AECL Response
A response to this RAI will be provided by April 2005.

134. Appendix B to 10CFR50 describes NRC requirements regarding quality assurance for nuclear power plants. Please provide descriptions of how the CATHENA computer code meets these requirements. Document COG-00-201 "CATHENA Quality Assurance Plan" is described as including the quality assurance procedures for CATHENA development, maintenance, verification and validation. Please provide the latest version of this document. See Regulatory Position 2 of DG-1120.

AECL Response
A response to this RAI will be provided by April 2005.

135. Regulatory Position 3 of DG-1120 deals with documentation. Please provide documentation for ACR-700 CATHENA analysis in the following areas:

<u>Requirements for Code Capability</u>
The NRC staff plans to review CATHENA only for specified ACR-700 applications. Please provide a list of the proposed uses of CATHENA in the licensing process of ACR-700 for which you seek NRC staff review and approval. For each application of CATHENA for ACR-700 analysis, please identify the section in the PIRT that addresses that usage.

AECL Response
A response to this RAI will be provided by April 2005.

<u>Methodology</u>
Please provide methodology documentation for use of CATHENA in ACR-700 analysis as described in the draft regulatory guide. You should include noding diagrams as well as the selection of input options for LOCA analysis as well as non-LOCA transients and accidents and justify the selection of each option chosen.

For LOCA analyses 10CFR50.46 provides the option of using one of two acceptable approaches. The first acceptable method is described in Appendix K to 10CFR50. The second method provides for a realistic approach with allowance for calculation uncertainty. Please identify the approach that will be utilized to analyze LOCAs for ACR-700 and discuss when the uncertainty analyses and supporting material required by 10CFR50.46 will be submitted. If the Appendix K approach will be followed, itemize how CATHENA will meet each of the Appendix K requirements. If another approach is taken for performing LOCA analysis other than those discussed in 10CFR50.46, please provide the technical as well as the regulatory basis for acceptance of this methodology.

AECL Response
A response to this RAI will be provided by April 2005.

Code Descriptive Manual
The NRC staff has been provided a theory manual for CATHENA Mod-3.5c. We understand that the ACR-700 will be analyzed for the DCD using CATHENA Mod-3.5d. Please provide appropriate modifications to the theory manual for all changes made to CATHENA to produce the new code version.

AECL Response
A response to this RAI will be provided by April 2005.

User Manual and User Guidelines
The NRC staff has been provided a user manual and user guidelines for CATHENA Mod-3.5c. We understand that the ACR-700 will be analyzed for the DCD using CATHENA Mod-3.5d. Please provide appropriate modifications to the user manual and user guidelines for all changes made to CATHENA to produce the new code version.

AECL Response
User manuals for CATHENA Mod-3.5d/Rev0 are COG reports COG-01-200 and COG-01-209. These reports are available and have been provided in PDF format on the enclosed CD-ROM. The PDF files are labeled:

- *"COG-01-200 – Cathena Input Manual MOD35d Rev0.pdf", and*
- *"COG-01-209 – GENHTP Input Manual MOD35d Rev0.pdf".*

Scaling Reports
Please provide scaling reports for the test facilities used in the CATHENA validation as discussed in the draft regulatory guide.

AECL Response
An updated scaling report using state-of-the-art methodologies and covering the RD-14M facility scaling for the Blowdown phase of the Large LOCA scenario will be provided in February 2005. The scaling reports covering the ECC phase and the Long Term Cooling phase will be provided in June 2005, and December 2005, respectively.

Assessment Reports
Step 4 of Regulatory Position 1 to DG-1120 deals with the development of phenomena identification and ranking tables (PIRT) for the various applications for the computer code. The PIRT provides a means of determining those processes and phenomena for which code assessment should be demonstrated. Please provide PIRTs for all uses of CATHENA for ACR safety analysis. Provide the qualifications of the PIRT panel members for the various applications of CATHENA.

The AECL has provided CATHENA validation reports for Mod-3.5c of the code. We understand that additional code validations have and will be preformed including ACR-700 specific validation. Please identify how this assessment addresses the various phenomena identified in the PIRT for all the applications of CATHENA for ACR-700 safety analysis.

A PIRT panel was assembled by NRC to identify significant thermal/hydraulic phenomena for ACR-700 safety analysis. AECL made various presentations to the NRC PIRT panel and provided supplementary material. Among the supplementary material was report 108US-03500-LS-001, "PIRT for Critical Header Break LOCA in ACR-700." The PIRT ranked processes expected during a Critical Header Break LOCA as high (H), medium (M) or low (L). Please provide a tabulation of how CATHENA was assessed or validated as adequate to model the processes identified in the PIRT commensurate with their ranked importance.

AECL Response
A response to this RAI will be provided by April 2005.

Uncertainty Analysis Reports
Please provide documentation of any uncertainty analysis performed for use of CATHENA for ACR-700 analysis.

AECL Response
A response to this RAI will be provided by April 2005.

Questions Relating to the CATHENA Theoretical Manual COG-00-008

Chapter 2. Conservation Equations

136. In assessing the quality of results from thermal/hydraulic computer codes, the ability of the code to conserve mass and energy over the course of long term transients and accidents is important. This is accomplished by comparing the total mass and energy within the reactor system to the integrated incoming mass and energy flow. The mass or energy that is lost or gained in the system is the mass or energy error. For the limiting small and large breaks LOCA events that will be analyzed for the design basis of ACR-700, please provide in graphic form the mass and energy errors in the CATHENA analyses. Please discuss the significance of the errors on the calculated results for ACR-700 safety analysis.

AECL Response
A response to this RAI will be provided by April 2005.

Chapter 3. Flow Regime

137. The flow regime maps used by CATHENA appear to be similar to those employed by the oil industry for pipe line oil-gas mixture flow. These maps are not based on pipes containing heat addition where the fluid can be highly non-equilibrium, particularly in the fuel channels.

Please justify the applicability of the flow regime maps to heated channels containing saturated and superheated fluid conditions that might occur at ACR-700.

AECL Response
A response to this RAI will be provided by April 2005.

138. Experience in application of RELAP5 to the N-Reactor showed that when ECC water entered hot horizontal fuel channels, the high rates of steam generation tended to force steam back toward the inlet pipes creating a slugging or chugging motion that further inhibited the rapid entrance of additional liquid in the fuel channels. For the same reasons one would expect a highly oscillatory behavior with slugging and chugging at ACR-700 particularly when the fuel channel is heated and refilled. In fact the NRC staff analyses using CATHENA has observed oscillatory channel flow in the recovery from a critical inlet header break. The flow regime maps in CATHENA do not appear to address the oscillatory slugging/chugging behavior where the flow continually reverses for some period of time. Please address the ability of the code to model this behavior.

AECL Response
A response to this RAI will be provided by April 2005.

139. Prediction of limiting conditions for countercurrent flow of steam and water is significant for ACR-700 since following a loss of coolant accident ECC water that is injected into the inlet headers must flow against the rising steam within the feeder pipes to reach the fuel channels. At the flooding limit separated flow will no longer occur so that any incoming ECC water will be carried out with the rising steam. CATHENA uses weighing factors to provide a smooth transition between countercurrent separated flow and mixed concurrent flow. For horizontal flow such as would occur within the fuel channels, CATHENA determines the flooding limit using the correlation of Ardron and Banerjee. For inclined and vertical flow such as would occur in the feeder tubes, the flooding limit is determined using a modification by Popov and Rohatgi to the Ishii entrainment criterion. Flooding behavior can be quite different depending on whether the liquid phase is subcooled or saturated. Please discuss the conditions that would occur within the fuel channels and feeder tubes in the recovery phase following a LOCA and justify that the flooding correlations within CATHENA are valid for these conditions. Include fluid conditions as well as size and geometry conditions.

AECL Response
A response to this RAI will be provided by April 2005.

Chapter 4. Constitutive Relations

140. Section 4.4.1.3.6 describes the crept pressure tube Friedel two phase friction model. On page 4-17 it is stated that "At present, the dependence of the two-phase multiplier on void fraction is not certain." Please discuss the experimental data base for the Friedel two phase

friction model. Quantify the uncertainty in the model and provide analyses showing the sensitivity of CATHENA results to the uncertainty in the model.

AECL Response
The MOD-3.5c/Rev 0 Theoretical Manual did not have this detail included for the Friedel two-phase friction model. The experimental database for the Friedel two-phase friction model as well as a discussion of the uncertainty in the model can be found in reference [1]. The sensitivity of CATHENA results to the uncertainty in the Friedel two-phase multiplier will be included in code validation results. The updated documentation to include the required level of detail for the Friedel two-phase friction model will be included in the MOD-3.5d Theoretical Manual.

References:
1. *C.W. Snoek and L.K.H. Leung, "A Model for Predicting Diabatic Pressure Drops in Multi-Element Fuel Channels", Nuclear Engineering and Design, 110 (1989) 299-312.*

141. Section 4.4.2.1 discusses methodology for computing two phase frictional pressure drop within horizontal channels for stratified flow. Please discuss the experimental data base for this model and provide justification for use of this model for the horizontal fuel channels of ACR-700.

AECL Response
The wall shear modeling for stratified flow conditions within the first few paragraphs of Section 4.4.2-1 and equation 4.4-78 is derived on the basis of geometric considerations within a horizontal pipe or channel. No experimental data was used in the derivation. Equation 4.4-78 is defined on the basis of a circular pipe geometry. For other available channel types, (i.e., those containing 7-element RD-14M electrical heaters, 37-element CANDU bundles, or 43-element CANFLEX bundles appropriate to the ACR-700) these relations are replaced by tabulated geometric data functions. These tabulated functions define the phase contact fractions, A_{kw}, and phase contact diameters, D_{kw}, based on geometric considerations.

The separate pipe modeling (i.e., the calculation wall shear for each individual phase based on geometric contact fractions) is a common approach adopted in two-fluid models such as those used in RELAP and TRAC [1,2]. The pressure drop during stratified flow conditions is extremely small and is dominated by wall and interface shear within the gas phase. Reference [3] provides the only known available experimental data available for stratified conditions within channels containing bundles. These experiments were performed for 7-element bundles which have primary hydraulic parameters (hydraulic diameters, surface area per unit volume) similar to those in both CANFLEX and 37-element CANDU bundles.

The later paragraphs in Section 4.4.2.1 (starting on page 4-21) are restricted to circular pipes during countercurrent flow when the liquid Froude number is greater than unity (i.e., equivalent to supercritical liquid flow in open channel flows). The justification for these relations relies on comparisons with experimental data (both air-water and steam water) for countercurrent flooding downstream of elbows in CANDU feeder piping.

References:

1. *Kowalski, J.E., 1987, "Wall and interfacial shear stress in stratified flow in a horizontal pipe", AIChE Journal Vol. 33, (2), 274–281.*
2. *Ransom, V.H. et al., 1982, "RELAP5/MOD1 code manual Volume 1: system models and numerical methods", NUREG/CR-1826, EGG-2070.*
3. *Rohatgi, U.S., J. Jo, and L. Neymotin, 1982, "Constitutive relations in TRAC-PD2", NUREG/CR-3073.*

142. Section 4.6.1.1 states that for superheated liquid, large numerical constants are utilized in calculation of interfacial heat transfer to ensure that the liquid does not significantly deviate from saturation. Please discuss the conservatism of this assumption for the various accident conditions analyzed by CATHENA for ACR-700. Are there circumstances when a sudden depressurization is analyzed when the rigorous treatment of superheated water might affect the result?

AECL Response

Background:

For superheated liquid (also referred to as meta-stable liquid state), where $h_f > h_f^{sat}$, CATHENA uses the following equation to calculate the liquid-to-interface heat transfer:

[ ]

The large numerical constant used in the first term in the above equation along with the addition of the second term ensures that the liquid temperature does not remain significantly superheated ($T_f > T^{sat}$) for extended periods.

Explanation:

The CATHENA meta-stable vapor generation model assumes that the transport of latent heat through the liquid to the interface is the dominant process limiting vapor generation. Although a detailed model for a time delay related to bubble formation kinetics or the activation of heterogeneous nucleation sites is not considered, a time delay for vapor generation will result during a CATHENA simulation of a depressurization transient. Under single-phase superheated liquid conditions, an initial void fraction of $\alpha_g = 10^{-6}$ is used as the basis for the initial interface heat transfer and void generation rate. For slower pressure transients (e.g., dp/dt < 3 MPa/s) the first term dominates, whereas for faster pressure transients the second term tends to dominate ensuring the liquid superheated is limited to approximately 50 °C. This modeling also ensures that the liquid *properties remain within the limits of the thermodynamic property functions used in CATHENA.*

The vapor generation time delay resulting from the use of this interface heat transfer model was most visibly demonstrated in validation against the Edward's Pipe Blowdown Tests [1, 2] at 5.6, 7.0 and 10.4 MPa, with initial subcooling in the range 30-40°C. In all cases reported in [2, Section 5.2], the predicted arrival time of the decompression wave at the measurement locations was within 0.5 ms of the measured time, indicating that the sonic velocity in the initial liquid state was accurately predicted [2, Section 5.2]. The saturation undershoot observed in these simulations provides indirect validation of the heat transfer modeling approach adopted. Further indirect validation is provided in the validation of CATHENA MOD-3.5c for coolant voiding against RD-14 and RD-14M experiments for small and large breaks [2, Section 5.2]. In large break RD-14 and RD-14M tests (e.g., B8703 and B9111), the predicted coolant voiding rates at both the inlet and outlets of all channels were typically within the scatter in the experimental measurement. An uncertainty analysis was conducted [2, Section 5.2] to assess the impact of uncertainties in the interfacial heat transfer and the results show that predicted channel voiding rates were not significantly affected by uncertainties in the interfacial heat transfer coefficient.

For ACR-700 simulations, the assumption that the liquid temperature does not significantly exceed T^{sat} for extended periods does not have significant impact on the simulation behavior. For example, for the 100% inlet header break scenario, significant initial superheat due to sudden depressurization is expected. Inlet header breaks are expected to have the highest break flow rates due to higher coolant density and higher header pressures and therefore provide the largest depressurization rates. The steepest slope in the depressurization curve occurs in the first 0.005s following the opening of the break. The average depressurization rate in this period is about 1000 MPa/s. In Edward's blowdown tests, CATHENA has been shown to consistently match the measured depressurization and the flashing (void generation) rates at even higher rates of depressurization. For example, in the 7.0MPa blowdown tests [1], the measured initial depressurization rate of 6000 MPa/s has been simulated within the measurement uncertainty. Initial void generation rate at all measuring stations has also been simulated within the measurement uncertainty. Since the largest ACR-700 depressurization rates do not exceed those that are measured in the Edwards tests, current CATHENA meta-stable void generation rate methodology is considered to be applicable to ACR scenarios where meta-stable liquid conditions occur.

References:

1. *Edwards, A.R. and O'Brian, T.P., "Studies of Phenomena Connected With the Depressurization of Water Reactors", UKAEA Report, Journal of the British Nuclear Society, Volume 9 (1970)*

2. *X.M. Huang and J.P. Mallory, "Validation of CATHENA MOD-3.5c for Break Discharge Characteristics - Overview Report", Atomic Energy of Canada Ltd, RC-2339 (2000).*

143. Equation 4.6-48 provides the inter-phase heat transfer coefficient for the "piston flow regime." Please discuss how the values for "segment length" and "conduction length" are determined. What experimental data have been used to confirm these values?

AECL Response
The piston flow regime liquid inter-phase heat transfer coefficient value was not derived experimentally. The form of the generalized heat transfer coefficient (defined in Equation 4.6-48) is taken as the conduction limit for stagnant conditions where the interface is assumed to span the pipe cross-section over zero axial distance.

The value of $1/D_z$, where D_z is the local hydraulic node length, represents the interface area per unit volume for this assumed limiting interfacial geometry. The conduction length, K_L of 5-mm is much smaller than typical nodalization dimensions used (typically axial nodalizations are above 0.5 m). A user may also assess the impact of this modeling through access to the uncertainty analysis parameters in CATHENA.

Confirmation of this approach has been provided through validation of feeder-channel during quench-refill [1] when refill piston fronts tend to be evident.

References:
1. *R. Kouyoumdjian and J.P. Mallory, 2001, "Validation of CATHENA MOD-3.5c/Rev 0 for Countercurrent Flow – Overview Report", Atomic Energy of Canada Ltd Report, RC-2475.*

144. Section 4.8 describes the use of empirical spacial dependant velocity and void fraction coefficients in CATHENA over the cross section of a conduit. Please discuss how the spacial dependant models are utilized for ACR-700 safety analysis. If the models are utilized for ACR-700 safety analysis, please describe the validation of the coefficients by comparison to experimental data. Provide the impact on safety analyses of the uncertainty in the coefficients based on the validation results.

AECL Response
A response to this RAI will be provided by April 2005.

145. The level swell model in CATHENA is described in section 4.9. The model is stated to be important for determining flow regime as well as heat transfer within the horizontal fuel channels. Please provide the following information concerning the level swell model:

145a. The model is stated to be fully described in papers by P.P. Revelis and M. E. Lavack. Please provide these papers.

145b. The discussions in Section 4.9 appear to relate to rectangular flow geometries. Justify that the model is adequate for the determining two-phase level within circular fuel channels containing ACR-CANFLEX fuel bundles.
145c. Validation of the level swell model is discussed in RC-2240 "Validation Plan for CATHENA Mod-3.5c" and RC-2701 "CATHENA Mod-3.5c/Rev 0 Systems Thermal/hydraulic Validation Manual." These documents describe comparison of CATHENA results with level swell data from large vertical tanks. Justify that the level swell model has been adequately validated for level swell within circular fuel channels containing CANFLEX fuel bundles.
145d. The CATHENA theory manual indicates that the level swell model is available for use with any horizontal pipe. The CATHENA input manual, COG-00-324 states that the level swell model is available only for 37- and 7- element horizontal channels with vapor generation. Please discuss how the level swell model will be applied for ACR-700 analysis.
145e. Accurate determination of two phase level will be important for determining the void fraction of the fluid entering the feeder pipes from the headers during LOCA analysis. Discuss how this will be accomplished in CATHENA analyses for ACR-700.
145f. What provision is made for accounting for level swell in vertical stacks of CATHENA nodes such as the modeling of the ACR-700 steam generators? How is layering of a two-phase mixture and pure steam in vertical stacks containing multiple CATHENA nodes prevented?

AECL Response
Background:
The CATHENA level swell model calculates a two-phase level in a horizontal pipe component from the collapsed liquid level and heat input (i.e., vapor generation) within the lower two-phase region. The model can only be specified for horizontal pipe components and is available for all channel types (including 43-element CANFLEX). From the question, it appears that the CATHENA level swell model has been confused with the level swell model in codes like RELAP where levels in pipe components (and in particular for vertical stacks of nodes composing for example the reactor core) are tracked and phase separation is enforced if some level criterion is met. In PWR applications this level tracking was developed primarily as part of tracking the quench-front progression in a vertical reactor core. In contrast for ACR-700 applications, the quench front progression is tracked by the fraction of the bundle submerged as defined by the liquid fraction under stratified flow conditions. As a result, for ACR-700 no level tracking methodology is required.

The CATHENA level swell model is not being applied in ACR-700 analyses. The impact of not using the level swell model in the analyses is a conservative calculation of the fuel cladding temperatures. The fuel cladding surface area exposed to steam cooling during stratified flow conditions is defined by the fluid level in the pipe or channel. If the level swell model is not used this fluid level is defined by the collapsed level rather than the higher two-phase swelled level. As a result, more cladding surface area will experience the lower heat transfer conditions of vapor cooling if the level swell model is not used.

145a. The following requested references are provided in PDF format:

- *P.P. Revelis, S. Pereira, N.U. Aydemir, B.N. Hanna, "A Model for Level Swell in Horizontal Pipes", Atomic Energy of Canada Limited Report, COG-92-410.*
- *M.E. Lavack, "Level Swell Implementation in CATHENA MOD-3.5b/Rev 0", Atomic Energy of Canada Limited Report, RC-1640 (1996).*

145b. The documentation of the level swell model contained in the MOD-3.5c/Rev 0 Theoretical Manual is a simplification of the model and its implementation in the code. The MOD-3.5d Theoretical Manual will include a more detailed explanation of the level swell model.

145c. The statement "Validation of the level swell model is discussed in RC-2240 "Validation Plan for CATHENA Mod-3.5c" and RC-2701 "CATHENA Mod-3.5c/Rev 0 Systems Thermal/hydraulic Validation Manual" *is incorrect. No validation of the "level swell model" was discussed in either report. These reports describe the validation simulations performed for the "Level Swell and Void Holdup Phenomena" as defined in the validation matrix.*

145d. The statement in CATHENA MOD-3.5c/Rev 0 Input Reference (COG-00-326) that the application of the swell model is limited to 7- and 37-element channels is incorrect. This statement was also made in the CATHENA MOD-3.5d/Rev 0 Input Reference (COG-01-200) but will be revised in a later revision of the manual.

145e. The determination of levels in headers and their impact on analysis will be addressed separately with other header modeling related questions.

145f. The validation for level swell in vertical stacks of nodes such as steam generators was discussed in Reference 1 (Section 3.4 Level Swell and Void Holdup). Level swell or vapor holdup in vertical pipe components (e.g., represented through a stack of CATHENA nodes) is calculated by the two-fluid model within CATHENA. Layering of two-phase mixtures and pure steam in vertical components containing multiple CATHENA nodes has not been observed in CATHENA simulations. This is primarily because CATHENA does not track levels and enforce phase separation at levels. As a result, no algorithm to prevent the occurrence of layering has been included in CATHENA.

References:

1. "CATHENA MOD-3.5c/Rev 0 Systems Thermalhydraulic Validation Manual", RC-2701

146. The connections for the small diameter inlet feeder pipes are located radially on the side of the fuel channels. Thus, following an event where the channel voids (i.e., a LOCA), upon reflooding when the channel begins to fill with ECC injection, the water level in the fuel channel will increase. The fuel channel liquid level will increase until the liquid level reaches the outlet elevation which is on the side of the fuel channel end cap. Thus, ECC water will flow into a

channel to roughly the mid plane flowing along the bottom of the channel and exiting at the mid plane at the channel outlet. In this condition, any additional water added to the channel will compress the steam into the upper vapor space since steam cannot exit the fuel bundle (the water level is above the channel outlet and inlet pipes). Sufficient turbulence and mixing at the liquid-steam interface might not occur to condense the steam in the upper region. Under these conditions, the steam phase would superheat (in a piston effect) and create the potential for a long term exposure of the rods in the top of the channel to steam cooling at high temperature. At this condition the upper fuel elements might remain elevated in temperature for oxidation to approach high levels for an extended period. Please clarify how the fuel channels are cooled following a LOCA under these conditions. What experiments were performed to investigate this phenomenon? Compare the orientation of the feeder tube to the fuel channel of the test facility to those of ACR-700.

AECL Response
A response to this RAI will be provided by April 2005.

Chapter 5. Heat Transfer Modeling

147. Sections 5.2.1.1, 5.2.1.2 and 5.2.1.3 discuss axial integration of the heat flux between solid boundaries and the fluid contained within. Linear smoothing of temperature within adjacent heat structures and use of temperature profiles determined from quench front progression are discussed. What provisions are included to ensure that an energy balance is maintained within the heat structures? What checks are made by the code to ensure that energy is conserved for each heat structure using these models?

AECL Response
Linear interpolation of the wall temperature is only performed within an individual GENHTP model used to simulate solid components in a CATHENA idealization. Each "heat structure", which may be composed of a number of axial nodes, is treated independently. No interpolation between adjacent solid conduction models is performed for any of the CATHENA heat flux integration methods. For ACR-700 analysis no tracking of quench fronts in the heat flux integration method is performed in the heat transfer integration methods used.

Within the solid model, the energy conservation equation (i.e., the heat conduction solution) is solved using a numerical method that provides energy conservation. The validation of the conduction calculation provided in Reference 1 (Section 3.6) indicates that the numerical conduction calculation converges to the analytical solution of the solid energy conservation equation. To ensure energy conservation between solid and the attached thermal hydraulic models, an energy conservation feedback algorithm is used. This conservation algorithm is described in Reference 2 (Section 6.4.1).

References:
1. *"CATHENA MOD-3.5c/Rev 0 Systems Thermalhydraulic Validation Manual", RC-2701*
2. *"CATHENA MOD-3.5c/Rev 0 Theoretical Manual", COG-00-008*

148. On page 5-14 the location of temperatures used in the "Quench Inferred Temperature Method" are calculated. The location of the temperatures where nucleate boiling, critical heat flux and stable film boiling first occur are functions of the total boiling length which is a user input. Discuss how this boiling length is determined in such a manner so as to be conservative for all conditions of flow including flow reversals, pressure and temperature such as might be encountered in a transient or accident analysis for ACR-700 using CATHENA.

AECL Response
The quench inferred heat flux integration algorithm discussed is not being applied in ACR-700 analyses. The reason the quench inferred temperature method is not adopted for ACR-700 analysis is because use of the method does not allow output of an unambiguous indication of the occurrence or margin to CHF. This heat flux integration has been retained in CATHENA for compatibility with older simulations where the margin to CHF was not a primary consideration. As a result further discussions of the quench inferred temperature heat flux integration method are not relevant to ACR-700 analyses.

149. Section 5.2.2 describes how the surface area of a heat structure that is exposed to the bulk vapor phase is determined for mixed flow regimes (dispersed-bubble, slug, plug, churn, churn-turbulent, intermittent and disperse-droplet flow). Justify that this model is valid for all mixed flow regimens and all heat structure shapes (slab, pipe wall, tube bundle, etc.) that will be evaluated for ACR-700. How has this model been validated?

AECL Response
Background:
For mixed flow regime, the contact area fraction for surface (n) is calculated from the following expression:

[ ]

where W_G and W_X are functions of vapor mass flux and quality, respectively. The above equation is intended to capture the effect of droplet impingement at high velocities while at the same time providing a smooth approach to single-phase steam conditions.

Validation:
Although CATHENA assumes that all heat structures are circular, a non-circular geometry can also be modeled with appropriate definitions of equivalent thermal mass and heat transfer area. The application of the above contact area relation in mixed flow

regimes can only be validated indirectly since direct wall-to-fluid contact areas are not experimentally available. Indirect validation of the heat transfer has been performed in a number of validation exercises involving circular geometries including bundles. There has been no attempt to validate the applicability of the above relation for non-circular geometries since ACR-700 components with significant heat generation are circular (i.e., horizontal fuel pins). Similarly, the refill phase of a large break LOCA is governed by heat transfer processes in the feeders and fuel channels that are circular as well.

The use of the above expression for steam contact fraction (in mixed flow) has indirectly been validated against RD-14M or CWIT blowdown/refill experiments.

The following validation exercise (provided on enclosed CD-ROM) contains an RD-14M critical break experiment (test B9902) where the horizontal fuel channel void fraction varies from zero to very large values as the forward flow stagnates, the flow regime varying between single-phase liquid to stratified gas-liquid flow:

[1] T.V. Sanderson and D.J. Wallace, "Validation of CATHENA MOD-3.5c for Coolant Voiding - RD-14M Critical Break Test B9902", Atomic Energy of Canada Ltd Report, RC-2341 (2000).

The following validation exercise covers flow regimes in the heated channels from single-phase to separated (horizontal stratified) flow regimes and is also considered to be an indirect validation of the contact area fractions for the gas phase:

[2] J.E. Kowalski and X.M. Huang, "Validation of CATHENA MOD-3.5c for Quench/Rewet Characteristics - CWIT Channel/Feeder Refill Tests", Atomic Energy of Canada Ltd Report, RC-2466 (2001).

150. Section 5.2.4 describes the CATHENA fin model. This model is stated be applicable to single-phase flow only. Is the fin model to be used in the safety analysis of ACR-700?

AECL Response
The ACR-700 fuel does not have fins. Appendages to the ACR-700 fuel, such as bearing pads and spacers are not incorporated in the design for heat transfer purposes, and have a negligible effect on the heat transfer from the fuel. Therefore, the fin model is not being used in ACR-700 analysis.

151. Section 5.2.5 describes the Entry Length Model by which heat transfer coefficients are modified because of closeness to upstream flow obstructions. What validation has been performed for this model for use in ACR-700 analysis? The model uses a user input quality. How is that quality determined? The text states that no checks are provided in the code to ensure that the model is not used for flow conditions for which it is not applicable. What assurances are there that the model will be used correctly?

AECL Response
The model has been used in some validation exercises and a typical example of its use is for locations such as cooling jackets where the flow enters perpendicular to the primary flow direction in the jacket. For these applications, the flow is defined directly from the experimental conditions. For validation exercises where this or other non-standard input models are used, justification for the use of the model and model parameters is included in the validation exercise documentation.

Since the use of the model in validation has been for experiment-specific purposes, it is not being applied to ACR-700 analysis. The main instance in which an upstream obstruction could affect heat transfer in the ACR-700 design is the effect of the fuel support plug on the heat transfer from the upstream fuel bundle. This has a negligible significance in safety analysis because the upstream bundle power is much lower than that of the bundles near the axial center of the core.

152. Section 5.3.2 states that CATHENA can calculate direct contact heat transfer between the fuel bundle bearing pads in contact with the pressure tube, fuel pin contact with the pressure tube as a result of bundle slumping, and pressure tube contact with the calandria tube as a result of pressure tube ballooning. The contact conductance is supplied by the user. For each type of direct contact calculation for which this model will be used in ACR-700 analysis discuss how the conductance is determined for inputting into the code.

AECL Response
CATHENA can be used to calculate direct contact heat transfer between two solid surfaces. However, the direct contact heat transfer between the fuel bundle bearing pads and the pressure tube, and between the fuel pin and pressure tube as a result of bundle slumping are not modeled using CATHENA for the ACR-700 analysis. AECL R&D work indicates that these two types of contact have no significant effect on the heat transfer from the bundle to the pressure tube and on the fuel channel integrity.

Direct contact heat transfer between the pressure tube and calandria tube as a result of pressure tube ballooning is not expected to occur. In the ACR-700 analysis, the CATHENA deformation model is used to monitor local hoop strains to ensure that pressure tube strain is limited.

153. Sections 5.3.3 describes two pressure tube deformation models. Please provide the following information concerning these models.

153a. Describe the transients and accident scenarios for which each of these deformation models will be utilized. Identify in each case whether the transient or accident is part of the design basis or beyond design basis.

AECL Response
A response to this RAI will be provided by April 2005.

153b. The pressure tube expansion models that are applied after first contact with the calandria tube are discussed in Section 5.3.3.2.5. These discussions include a statement that the ring deformation model used in these calculations in not analytically valid and a statement that the effect of pressure tube ballooning on thermal/hydraulics or heat transfer not included. Justify that it is appropriate to use these models for ACR-700 safety analysis.

AECL Response
In the ACR-700 design, pressure tube ballooning into contact with the associated calandria tube is not expected to occur. Hence, neither the circular nor the non-circular (ring) pressure deformation model is used to assess the effect of pressure tube ballooning on thermal/hydraulics or heat transfer.

154. Section 5.3.4 describes the calandria tube deformation model. Please describe the transients and accident scenarios in which the calandria tube creep-strain-rupture model will be utilized for ACR-700. Identify in each case whether the transient or accident is part of the design basis or beyond design basis.

AECL Response
In the ACR-700 analysis, the calandria tube deformation model is not used. This model is only applicable in the case of pressure tube / calandria tube ballooning contact (full contact). However, in ACR-700 analysis, pressure tube / calandria tube ballooning contact is not expected to occur.

155. Section 5.4.3 describes the treatment of heat sources within the heat structures of CATHENA. Sources of heat are described as the heat generated by the fuel pellet and heat generated as a result of the zirconium-steam reaction at high temperatures. Options for specifying the heat generation history are stated to be user input or the point reactor kinetics model. The point kinetics model is described in Section 7.15.6. For analysis of ACR-700 please provide the following information:

155a. We understand that CATHENA has the ability of be coupled to three-dimensional neutronics computer codes for computation of reactor power. Provide the details of how this is accomplished and how the resulting heat generation is added to the associated CATHENA heat structures. Specify which design basis accidents and transients for ACR-700 will be analyzed using point-kinetics and which will be analyzed using the more detailed methodology.

AECL Response
A response to this RAI will be provided by April 2005.

155b. Some of the structures surrounding the fuel pins may be subject to heating by gamma rays generated in the fuel. Discuss how gamma ray heating is considered by the code.

AECL Response
A response to this RAI will be provided by April 2005.

155c. Describe the models that will be used to calculate the decay heat generation. How will these models be made conservative? Provide your answer for both LOCA and non-LOCA conditions. Will the requirements of 10CFR50 Appendix K be met concerning decay heat? If you propose to use the 1979 or the 1994 ANS standards to calculate decay heat, please address the concerns discussed in NRC Information Notice 96-39. Please justify that the decay heat model which you will use is applicable to the ACR with slightly enriched fuel, light water coolant and heavy water moderation.

AECL Response
A response to this RAI will be provided by April 2005.

155d. The ACR-700 will use slightly enriched fuel by which more fissions will occur in uranium compared to a standard CANDU reactor which uses natural uranium so that more plutonium fission occurs. Since uranium fission products have a higher power release than those from plutonium, discuss how the decay heat model will be implemented for conservative prediction of decay heat for ACR-700.

AECL Response
Solid heat transfer models in CATHENA are flexible and allow the user to define the heat generation and its distribution within the model. The user defined heat generation can be linked to a system control model that could define for example, heat generation resulting from γ-heating or the heat generation derived from an external kinetics calculation. If the internal point kinetics model is being used, the user may specify a decay heat curve, which then may be applied to one or more solid heat transfer models. The only restriction is that only one heat generation mechanism may be specified for each solid region modeled. Note, however that a solid model may be composed of a number of solid regions each with their own material property and heat generation specification. As a result, the implementation of heat generation, including decay heating, can be performed either through the CATHENA point kinetics model or defined by the user through the system control models. The implementation choices are defined by the user based on the analysis requirements.

A conservative decay heat curve will be applied for ACR-700 safety analysis. The curve and its conservatism will be established by a detailed analysis of the decay heat characteristics of ACR-700 fuel at various burnups.

155e. Heat generation from zirconium-water reaction is calculated using the equation of Prowse and Vandenberghe. Discuss the conservatism of this equation for reactor safety analysis. Provide a comparison of the results from the Prowse and Vandenberghe equation with those of the Baker and Just equation which is required to be used for LOCA analysis by Appendix K to 10CFR50. Provide this information for the limiting design basis LOCA analyzed for ACR-700.

AECL Response
A response to this RAI will be provided by April 2005.

156. Section 5.5 describes the heat transfer feedback effects for changes in fuel channel geometry due to pressure-tube ballooning. The discussions do not include the heat transfer and flow blockage effects from fuel element cladding ballooning such as might occur if fuel elements were overheated in a depressurized fuel channel. Please describe how these phenomena are determined, how they are included in your evaluation models, and how they have been experimentally validated.

AECL Response
A response to this RAI will be provided by April 2005.

157. Section 5.5.2 describes modification of the radiation heat transfer model to account for temperature and geometry changes within the fuel channels. The methodology discusses how specific radiation heat transfer matrixes are input into CATHENA to account for different conditions of emissivities, fuel channel creep and fuel bundle geometry. The examples are for 37 element fuel assemblies. Please discuss how the matrix values will be obtained for ACR-700 fuel. Discuss which analyses of transients and accidents these models will be applied to. Identify which of the postulated events is beyond the design basis.

AECL Response
The view factor matrix values required for the radiation heat transfer calculation are defined by the stand-alone CATHENA pre-processor program MATRIX. The input to the MATRIX program defines the geometry of the bundle through a specification of the location of each pin surface. This calculation is not specific to 37-element geometries, however this has been a common application and therefore appears as a typical example. The MATRIX program is fully capable of calculating the view factor matrix required in the radiation heat transfer calculation for the ACR-700 fuel from code user specification of the geometry of the fuel bundle and the surrounding pressure tube.

The MATRIX-1.05 pre-processor utility is documented in "MATRIX-1.05 A Stand-Alone Preprocessor Utility for CATHENA Users", J.B. Hedley, Atomic Energy of Canada Limited Report, COG-99-232. This report is provided, in PDF format, on the enclosed CD-ROM.

The transients and accidents that will use these models will be identified by April 2005.

158. The radiation models discussed in Section 5.5.2 appear to be valid only for a voided fuel channel. For fuel channels that are partially filled with liquid please discuss how radiation heat transfer will be calculated for the fuel elements above the liquid surface to the surroundings including the liquid surface.

AECL Response
The CATHENA radiation model transfers heat only between solid models. As a consequence, radiative heat transfer in an ACR-700 channel is modeled between various fuel pins, or between fuel pins and the pressure tube walls. Radiation heat transfer to a liquid partially filling the fuel channel from the pins above it is not directly modeled. With the assumption of transparent medium, radiation heat transfer from upper fuel to the liquid is indirect, in that the radiative heat flux received by the submerged fuel pins and pressure tube portions will subsequently be transferred to the liquid by conduction and convection.

For ACR-700 applications, the radiation heat transfer approach is conservative since the fuel pin temperatures above the surface of the liquid will tend to be overestimated since they are transferring heat to fuel pins that are hotter than the local liquid saturation temperature.

Chapter 6. Numerical Methods

159. Section 6.3 discusses how temperature distributions within fuel pins and piping walls are calculated. In determining heat transfer from the fuel pins, local fluid conditions within the coolant channels are important. As the fuel channels age the channel walls may creep in the radial direction causing mal-distribution of coolant about the fuel pins. Discuss how the effect of radial creep will be considered in the calculation of fuel pin heat transfer. Consider all heat transfer regimes that the fuel pin will experience during design basis transients and accidents.

AECL Response
The heat-transfer regimes covered in the design basis transients and accidents include nucleate boiling, film boiling, and superheated steam cooling. The impact of nucleate-boiling heat-transfer calculations on clad-temperature integrity is small (due to mainly low associated clad temperatures). Therefore, the coolant distribution of a bundle inside the channel with radial creep does not have a strong impact on nucleate boiling temperature.

The current approach to determine the coolant-to-clad heat-transfer coefficient at film boiling applies the lower-bound calculations for maximum clad temperature in the bundle at cross-sectional-average flow conditions. The post-dryout correlation bounds all full-scale bundle data on minimum post-dryout heat-transfer coefficient (or maximum post-dryout clad temperature) obtained at the CANDU 6 conditions of interest for uncrept and 5.1% crept channels of relevant profiles (similar to ACR-700 conditions except for the slight increase in ACR-700 pressure, 12.5 MPa as compared to 11 MPa in the full-scale bundle test). Therefore, the correlation has included the effect of flow and enthalpy distributions on post-dryout heat transfer within the bundle in uncrept and crept channels. The correlation is expressed in terms of a ratio of heat-transfer coefficient, with reference to the CHF and fully developed post-dryout conditions. The CHF *prediction method includes a modification to account for the creep effect. Channel creep has little effect on the ratio of heat-transfer coefficient.*

At conditions and scenarios where superheated-steam cooling is encountered, the flow and enthalpy distributions in the bundle are anticipated to be relatively uniform. Therefore, the effect of creep on heat-transfer coefficient from fuel pins to coolant is small.

160. For some events, particularly for the cases when the top of the fuel channel boundary bows out due to heating, there will be a higher flow at the top and a lower fluid velocity at the bottom of the channel. A multidimensional calculation may show some localized regions of low flow near the boundaries where the hot rods are located that produce CHF earlier than that for the one-dimensional calculation. Please address the applicability of the channel average CHF approach to capture 3-D effects.

AECL Response
The specified scenario is similar to the channel creep effect where the pressure tube expands resulting in a large by-pass flow at the top subchannels between the pressure tube and bundle. The current CHF prediction method was derived with full-scale CANFLEX bundle data obtained at conditions similar to ACR-700 range of interest (test pressure up to 11 MPa as compared to ACR-700 pressure of 12.5 MPa and test mass-flow rate up to 25 kg/s as compared to ACR-700 mass flow rate of 26 kg/s) for uncrept, 3.3% and 5.1% crept channels of relevant profiles. The effect of pressure from 11 MPa to 12.5 MPa is relatively minor on CHF. Based on these data, the CHF prediction method has included the effect of flow and enthalpy distributions on CHF within the bundle in uncrept and crept channels. Confirmatory data will be obtained with the ACR-700 bundle string at relevant conditions of interest.

161. Section 6.4.3 describes the stratified steam bubble model in CATHENA. In using this model at very low flows, a temperature gradient in the steam space can be determined. The temperature gradient can be used in heat transfer calculations. This model would appear to be particularly useful in evaluating fuel pin heatup within a partially drained fuel channel. Please provide the following information concerning this model:

161a. Comparisons with experiment data are referred to first for determining the XL length where entry effects are no longer important and second for comparison with CATHENA temperatures with and without the steam bubble model. Please provide this data comparison. Discuss the source of the data and justify that it is appropriate for evaluation of ACR-700 fuel channels.

AECL Response
For ACR-700 analysis the application of the steam bubble model is being considered only for the simulation of the long term cooling phase of a small LOCA. In this case, the application of the steam bubble model will be used to assess the uncertainty in fuel clad

temperatures in response to possible thermal stratification in the fuel channel. Under all other conditions, thermal stratification in the channel is not considered to contribute significantly to fuel clad temperature variations since the driving forces and as a result vapor velocities are expected to be larger than those where free convection would dominate. For example, for vapor flows exceeding approximately 1.3 m/s no significant thermal stratification is expected. This limit for a free-convection threshold can be determined from

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with a ΔT of 1000°C between the liquid and the steam given a half filled ACR-700 channel height. This temperature difference is at the high end of what could be expected in any analysis where water was available in a channel. Note that the vapor velocity threshold decreases with decreasing temperature differences. That is, for lower vapor–to-liquid temperature differences thermal stratification is even less significant. In addition, this threshold does not change significantly over a large pressure range (200 kPa to 14 MPa).

Experimental comparisons for the steam bubble model are presented in the PDF file labeled "Steam-bubble-application-Appendix.pdf", contained on the enclosed CD-ROM. The results, which are given in Figures I1 and I2 of the pdf file, were used to determine the threshold value for $Gr^*_l/(Re_l)^2$ where the model is applied or turned-off. The experimental tests (also referred to as "Boil-off Tests S-1-2") were conducted primarily to study the phenomena of pressure tube heat-up, pressure tube deformation and pressure tube/calandria tube contact as well as post solid-solid contact heat transfer in a CANDU channel. Only the pressure tube heat-up is applicable to ACR-700 channels because of the thicker and therefore stronger pressure tube.

Although these tests were performed with a 37-element fuel element simulator, the behavior in a 43-element ACR-700 fuel bundle will be similar since both geometries have similar flow areas, channel diameters and gaps between fuel element surfaces. Also the ACR-700 bundle has a flatter radial power profile than the 37-element CANDU bundle. Therefore a lower power is proportionally generated in the higher elevation part of the pin bundle exposed to steam cooling. As a result, free convective (buoyancy driven) flows would be lower in the ACR-700 bundle and thermal stratification will be even less likely.

161b. Once stratified conditions are determined to be present in the steam space, then the temperature in the steam space is determined to vary linearly with height between Tsat and Tmax. The determination of Tmax is not clear. Please describe how Tmax is determined. How has the Tmax model been validated to be accurate?

AECL Response

When the phases separate (under low flow conditions) the top pin is the location where the steam will be hottest if thermal stratification is significant. Thus, in the steam bubble model, the value of T_{MAX} for the gas phase is assumed to be the fuel-pin cladding temperature immediately below the inside surface of the pressure tube (i.e., top pin in the horizontal fuel bundle). For the liquid phase, the maximum temperature is assumed to be the saturation temperature (T_{SAT}) corresponding to local fluid pressure based on the assumption that non-equilibrium conditions at the interface cannot be maintained for the time-scale of interest to these calculations.

161c. It is indicated that the steam bubble stratification model cannot be used if the "quench inferred temperature distribution" is used for the fuel channel. Please justify that a fuel channel that is partially drained and subsequently reflooded can be adequately evaluated without making use of both of these models.

AECL Response
Neither the steam bubble stratification model nor quench inferred temperature distribution model are needed to adequately predict the refill behavior of a partially drained and subsequently reflooded fuel channel.
1) The steam bubble stratification model has the largest effect under stagnant conditions. Since the fluid conditions during reflood are inherently violent (vapor generation, spattering etc.) the effect of thermal stratification would not be as important as the other heat transfer processes.
2) It is not necessary to use the quench inferred temperature distribution wall-to-fluid heat flux integration method to predict the reflood behavior of a channel. Any of the heat flux integration methods will predict this phenomenon. The quench inferred heat transfer integration algorithm has been retained primarily for compatibility with transient simulations performed with prior code versions.

162. If the stratified steam bubble model is not used, for the case of a fuel channel which may have lost water during a LOCA event, CATHENA would represent the steam region with a single average temperature and the liquid with a separate single temperature. For a partially filled channel the steam may be stratified so that the temperature at the top of the fuel channel may be elevated in comparison to the average. Please discuss how this effect will be accounted for in ACR-700 safety analyses. Consider the effect of temperature gradient in the steam space on heat transfer from the exposed fuel elements and to the heat structure nodes of the fuel channel wall.

AECL Response
A response to this RAI will be provided by April 2005.

7. Component models

163. Section 7.2 discusses CATHENA component models for evaluating the effect on momentum from sudden area changes. Please provide the following information concerning these models.

163a. Discuss how gradual area changes are treated such as flow through a venturi.

AECL Response
CATHENA does not include a venturi model. This is primarily because none of the CANDU related experimental facilities have ever included venturi components. Although CATHENA does not provide a direct means of defining the cross section area as a function of position in a pipe component (outside of the crept pressure-tube model that is only applicable to the fuel channels), components with gradual area changes can be represented using a number of pipe components linked with junction resistance models.

For ACR-700 analysis, venturis are modeled as a short pipe component whose hydraulic diameter and flow area are those at the throat of the venturi. Entrance and exit losses due to the sudden area change in the model are overridden using JUNCTION RESISTANCE models. A minor loss coefficient applied to the pipe component is tuned to give the correct overall pressure loss of the venturi. For sonic flow, choking is modeled based on the pipe component flow area.

163b. It is stated that across area changes the phase densities are assumed to be unchanged. Since the phase densities actually will change and will provide a reversible pressure effect across the area change, you should justify that neglecting this effect provides for conservative analytical results.

AECL Response
It is correct that the assumption of constant density across an area change will lead to omission of a reversible pressure effect. It can be shown that an increase in density will lead to a pressure gain whereas a decrease would lead to a pressure loss. We call this a reversible gain (or loss) since the change in pressure is purely due to acceleration or de-acceleration of the fluid and the pressure change can be reversed should the fluid be brought back to its initial flow velocity through either an area or a density change. The magnitude of this term can be evaluated in a conduit with an abrupt change in density in addition to a change in area. Applying the energy equation with zero loss (that is, the Bernoulli equation), it can be shown that the reversible pressure change is:

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where $\rho_2 = \rho_1 + \varepsilon$, $\Delta P = P_2 - P_1$ and v is the flow velocity. The above equation is valid for small density changes, that is, ε is small.

For CANDU/ACR-700 design, one of the largest area changes occurs at connections to headers. Typical values of reversible pressure losses arising from density variations at connections to inlet and outlet headers as well as channel to feeder connections are given in the table below under nominal operating conditions:

Connection	$(\rho_1/\rho_2)^2$ gas	$(\rho_1/\rho_2)^2$ liq
Outlet header to outlet feeder	*0.99916*	*1.000000*
Inlet header to inlet feeder	*No gas*	*1.000039*
Fuel channel to end-fitting	*1.00327*	*1.000000*

As ε increases, the above equation will not hold and calculation of reversible losses will also be not accurate since the Bernoulli equation is a special case of an energy integral for incompressible flows. In recognition of that, CATHENA also has an option to override the above reversible loss calculation where the actual momentum flux terms are written at each face of the momentum control volume. In this calculation, density changes on either side of the area change are taken into account.

Extensive experience with CATHENA shows that small inaccuracies in the calculation of reversible losses (should they be invoked) do not influence calculations. For example, in the simulation of a large break LOCA, selecting reversible losses does not influence the refill and subsequent behavior of the reactor system since the associated terms are quite small in comparison to frictional losses.

163c. It is stated that the reversible pressure losses from area changes can be included or not as a user option. Are the reversible pressure losses included in the ACR-700 CATHENA model. If not, please justify their omission in particular for sudden area changes such as for the feeder pipe connections, pressurizer surge line, accumulator lines and relief and safety valves.

AECL Response
A response to this RAI will be provided by April 2005.

163d. Equations 7.2-6 and 7.2-7 provide the pressure losses across an area change for each of the two phases passing through the area change. After passing through an area change using the equations, each phase will be at a different pressure. Is this a valid state for pipe flow including mixed flow regimes? Please explain your response.

AECL Response
It is possible for a two-phase mixture to have unequal phase pressures due to effects arising from drag between the phases (mixed flow) or variation of hydrostatic head in a conduit containing a well defined level (such as in separated flow in a horizontal channel).

CATHENA is a two-pressure code and the finite-difference equations have been formulated in terms of two pressure fields for each phase. A similar approach has been taken in other system codes, such as CATHARE. In the current formulation, the finite-difference matrix terms are written in terms of gas-phase pressure only. The additional closure relations that define the liquid phase pressure are provided by phase-to-interphase pressure difference terms as explained in Section 4.3 and Appendix C through Equations C-1 to C-5. Constitutive laws defining these relationships are given in Section 4.3 for each flow regime modeled. There is no physical requirement that the phase pressures should be equal in mixed flow regimes.

164. The Accumulator tank model is described in Section 7.3. Please provide the following additional information concerning the conservatism of using this model for the safety analysis of ACR-700.

164a. The model does not include the effects of momentum in computing the flow exiting the accumulators. Please justify the conservatism of not including momentum effects in the accumulator model for ACR-700.

AECL Response
The Accumulator model is not being applied to ACR-700 analyses. The CATHENA generalized tank model is being used for components such as tanks (pressurizer, ECC accumulator, etc). In these cases the tank model is meant to represent relatively large volumes and diameters in which the level and contents move slowly relative to the velocities in the attached pipes. For these applications, the bulk momentum of the liquid in the tank has a negligible effect on the flow entering or exiting the tank. The neglect of momentum with the tank results in a conservative calculation of liquid ECC flow, since liquid motion in the tank adds to the exit momentum (effectively increasing the tank static pressure).

Note that the pressure changes resulting from the abrupt area change at the boundary between the tank and any attached pipe components is included separately (through appropriate junction resistance models) in the CATHENA idealization.

164b. Cover gas expansion is calculated using a polytropic coefficient that is assumed to remain constant over the evaluation. For very large breaks use of the default value which is for isentropic expansion would be appropriate. For smaller breaks the coefficient would approach unity. Please describe and justify how the polytropic gas coefficient is determined for ACR-700 safety analysis of various postulated break sizes.

AECL Response
A response to this RAI will be provided by April 2005.

164c. A facility specific accumulator model is provided for the RD-14 test facility which includes features not included in the generic accumulator model that will be used for analysis of the ACR-700. Considering the difference in the CATHENA accumulator models that will be used for ACR-700 data and that which were used to qualify the code using experimental data, please justify that code validation using the facility specific model is valid for ACR-700.

AECL Response
A response to this RAI will be provided by April 2005.

165. The adjacent-node mixing model described in Section 7.4 is used by the code as default to describe thermal mixing between adjacent nodes in pipes. Please provide the following information concerning this model.

165a. Please describe implementation of the model for ACR-700. Justify that for each usage the model has been benchmarked against appropriate data. For example, consider low flow or no flow conditions in a fuel channel. Justify that the model correlations have been validated using data typical of ACR-700 fuel bundle geometry.

AECL Response
A response to this RAI will be provided by April 2005.

165b. Nodal computer codes such as CATHENA artificially mix fluid between adjacent fluid nodes since the average of the properties in the upstream node is passed to the downstream node instead of those at the interface (numerical diffusion). Please justify that code errors produced by numerical diffusion are not increased as a result of the adjacent node mixing model. Please justify that energy is conserved using this model.

AECL Response
Diffusion: *The numerical diffusion of enthalpy due to upwinding is not increased by the adjacent node mixing model since this model is only applied for zero or nearly zero flow conditions. This is shown below following a brief explanation to the background of the model.*

Phasic energy equations in CATHENA do not contain axial conduction terms since they are usually very small compared to convective transport of energy. Under certain conditions, however, transport of energy due to molecular diffusion or diffusion resulting from turbulent fluctuations can become significant. In the absence of diffusive transport mechanisms, and no significant pressure gradients, a dead-end branch may attain temperatures significantly different than the adjacent node that is attached to it.

The diffusive processes in the adjacent-node mixing model have been modeled using well-known and generally accepted correlations available in open literature [1,2]. The contribution of smearing (via diffusion) of an enthalpy front is negligible when the fluid is in motion since the model is applied only for very low flows. The application of the model is based on the ratio of free convective forces to forced convection, characterized by the inequality $Gr/Re^2>1$. For ACR-700 nominal operating conditions, Re number in *the heated channels is approximately at the order of 10^5 and in the feeders about 10^6-10^7 based on CATHENA computed velocities, density and feeder/channel hydraulic diameters. Grashof number varies according to $1.3x10^{11}\ \Delta T\ \delta^3$, where ΔT is the temperature difference between two node centres. For heated channels temperature varies from inlet to outlet between 282 °C to 329 °C. In inlet or outlet feeders, there is no appreciable temperature variation within the fluid since there is no significant heat addition or extraction. Thus, in the feeders the temperature can be considered to remain constant[1] since there is no appreciable axial temperature gradient. Based on the above, it can be shown that Grashof number within the channels will be at the order of $1.3 \times 10^{11} \times 10^{-6} \approx 1.3 \times 10^5$. The criterion $Gr/Re^2>1$ can now be numerically evaluated as $1.3 \times 10^5/(10^5 - 10^7)^2 > 1$, and therefore the model is automatically turned off by the code. As the above example illustrates, adjacent node diffusive mixing is not applied when there is flow in the system. An order of magnitude analysis would demonstrate that in fact this is true whenever there is flow in a piping system. As intended, the model is applied when flow velocities are zero or nearly zero. It follows that numerical diffusion in the energy equation cannot be increased by the application of this model. Further more, the model is applied at near-zero flow conditions and the resulting diffusion is one that is governed by physical processes.*

<u>Energy Conservation:</u> The adjacent mixing model is implemented such that energy is conserved as heat diffuses from one node to the neighbor node. Energy conservation is assured since heat is transported using the same link velocity and the same enthalpy gradient. In the numerical implementation identical coefficient terms, in Section 7.4 of Reference 3, are defined on either side of a link between the two nodes. As a result, energy conservation is ensured since what is added to one node is subtracted from its neighbor and only the energy distribution in the network is influenced.

References:
1. *Mac Gregor, R.K. and Emery, A.P., "Free Convection Through vertical Plane Layers: Moderate and high Prandtl number Fluids", J. Heat Transfer, Vol. 91, p.391 (1969)*
2. *Hollands, K.G.T., Raithby, G.D. and Konicek, L., "Correlation Equations for Free Convection Heat Transfer in Horizontal Layers of Air and Water", Int. J. Heat and Mass Transfer, Vol. 18, p.879 (1975)*
3. *"CATHENA MOD-3.5c/Rev 0 Theoretical Manual", COG-00-008*

166. The Groeneveld table lookup critical heat flux (Section A.2.2.6) which we understand is the default model utilizes a boiling length multiplier. The boiling length multiplier is stated to be applicable only to unidirectional flow for positive flow down a channel. Please describe what is done for flow reversals within fuel channels and justify that the results will be conservative for ACR-700 safety analysis.

1. *When temperature is constant, Gr number is zero and no diffusion of heat is computed by the model.*

AECL Response
The CANFLEX Mk IV CHF tables [1] for bundles are based on a boiling length average definition of heat flux. Since CATHENA uses the local heat flux to determine heat transfer conditions, a boiling-length average multiplier is required to convert the values obtained from the tables to a local heat flux value. The resulting CHF value is defined as a multiplier $K_{BLA/local}$ on the CANFLEX Mk IV table value and is defined as follows:

[ ]

Two approaches to account for the difference between the boiling-length-average (BLA) values defined in the CANFLEX Mk-IV table and the local CHF value required within the code are provided. The first approach, as documented in Reference 2, uses the integral of the heat flux over the boiling length defined by the local of the Onset of Significant Void (OSV) in the channel.

[ ]

This approach is limited to unidirectional flow conditions where a location of the OSV point can be defined. If the simulation conditions do not meet these restrictions the user is notified in the simulation files. The user is instructed through the code documentation that the above approach is to be used for transients where a slow approach to CHF and unidirectional flow are expected.

For simulations where a rapid approach to CHF is anticipated (for example large-LOCA), an alternative local definition for $K_{BLA/local}$ that is independent of flow direction is defined. The local definition for the boiling-length-average multiplier is a function of the axial derivative of the heat flux given,

[ ]

In steady-state application, both approaches (integral and local) provide similar uncertainties [1].

The CHF tables provide a best-estimate of the first occurrence of CHF over the bundle cross-section within the uncertainty in the table values defined in Reference 1. Conservatism in application of the CHF model is ensured by applying a user-defined conservatism factor to the CHF value taking into account the uncertainty in the CHF tables.

References:

1. *Leung, L.H.K., S.K. Yang, Y.J. Guo and W.W.R. Inch, "A Look-up Table of Critical Heat Flux for the CANFLEX Mk-IV in Crept and Uncrept Channels", Atomic Energy of Canada Limited Report, CANFLEX-161, FFC-FCT-383, 2001.*
2. *"CATHENA MOD-3.5c/Rev 0 Theoretical Manual", COG-00-008*

167. The Groeneveld table lookup critical heat flux (Section A.2.2.6) includes tables for predicting the CHF within pipes. The range of validity is stated to be for vertical tubes that are 8 mm in diameter. Correction factors are provided for pipes of larger diameter and for non vertical orientation. Please describe the basis and validation of the correction factors and justify they produce results that are conservative for ACR-700 safety analysis.

AECL Response
The CHF look-up table for bundle was based on the tube-CHF table, and hence incorporated the parametric trend of tube CHF over the wide range of flow conditions. Modifications have been introduced to account for the enthalpy imbalance between bundles and tubes, diameter effect, and channel orientation effect for the conversion of the tube CHF table to bundles. Brief descriptions of these modifications were presented in Leung et al. ("A Generalized Prediction Method for Critical Heat Flux in CANDU Fuel-Bundle Strings", Proceedings of the 11th International Heat Transfer Conference, Kyongju, Korea, Aug. 23-28, Vol. 6, pp. 15-20, 1998). After setting up the bundle tables using the analytical method, experimental CHF values for the CANFLEX bundle were incorporated into the table to improve the prediction accuracy. The CHF mechanism at ACR-700 flow conditions corresponds to liquid film dryout, where the clad-temperature rise beyond CHF is gradual and controllable. The predicted CHF from the bundle CHF table represents the initial occurrence of dryout at any location in the 6-m long bundle string. This is considered conservative as the heat-transfer rate remains high (almost the complete bundle string remains under nucleate boiling) and the clad temperature remains low and controllable for Loss of Regulation and Loss of Flow types of accident scenarios. Confirmatory data will be obtained with the ACR-700 bundle string at relevant conditions of interest.

168. Section 7.6 describes the Break component by which critical flow is calculated using models that are provided as options to the user. For analysis of ACR-700 please discuss how models are selected to ensure that conservative results are obtained. Include discussions for loss of coolant accidents, steam generator tube ruptures, safety/relief valve flow, steam line break and feedwater water line breaks. Conservative results should be considered those which minimize the margin between the code predictions and the "figures of merit" as discussed in Section 1.1.2 of DG-1120 for the event being analyzed.

AECL Response
A response to this RAI will be provided by April 2005.

169. Section 7.6 7.3 describes the criterion for transition between choked flow and subsonic flow. A critical pressure ratio of between 0.5 and 0.6 is assumed. Please justify the accuracy and the conservatism of using this criterion rather than checking against the sonic velocity and evaluating the throat pressure as criteria for the transition. Consider cases of reactor system breaks to the containment building as well as steam generator tube breaks to secondary system pressure.

AECL Response
A response to this RAI will be provided by April 2005.

170. Section 7.8 describes the "Delay Line Model." This model divides piping into segments for computing the progress of a temperature front flowing down a pipe. Please justify that energy is conserved using this model. We understand that flow reversals cannot be treated. Discuss the limitations of the model for rapidly changing flow or oscillating flow in the positive direction.

AECL Response
The "Delay Line Model" is not being applied to ACR-700 analyses. The feedwater train idealization will be sufficiently fine to capture transients due to transport delays.

171. Section 7.9 describes the "Fisher Valve Model." Please discuss the use of Fisher valves in ACR-700 and their significance for safety analysis. If the valves are important for safety analysis, please discuss the range of the data on which the flow equations for the valves are based for both single and two-phase flow and compare these ranges with the conditions predicted for ACR-700.

AECL Response
The "Fisher Valve Model" is not being used in ACR-700 analysis. The specific Fisher Valve specifications incorporated in the model are not representative of ACR-700 valves modeled using CATHENA. In ACR-700 analysis the CATHENA controllable valves and check valves (section 7.19) are used.

172. Section 7.10 describes the "Generalized Discharge Model" by which critical flow is determined from basic principles. The model has been extended to include non-equilibrium terms based on the work or Ransom and Trapp. Our experience with the Ransom and Trapp critical flow model in RELAP5 is that critical flow is under predicted at low pressures. If the Ransom and Trapp model is used to calculate critical flow for ACR-700, justify that the model in CATHENA is accurate by comparison to low pressure two-phase critical flow data.

AECL Response
A response to this RAI will be provided by April 2005.

173. Equation 7.10-54 presents a constant "k" by which interfacial mass and heat transfer is derived from both equilibrium and non-equilibrium contributions. The constant is fit to match

experimental data. If this equation is to be used for ACR-700 analysis, please discuss how the value of the constant "k" was determined from experimental data.

AECL Response
A response to this RAI will be provided by April 2005.

174. Section 7.11 describes the generalized tank model (GTM). Please provide the following information concerning this model.

174a. Will the GTM be utilized to calculate maximum containment pressures and temperatures to establish the design basis for the building design and equipment qualification? If so, please provide the details of the options to be used, justify that these options are conservative and provide comparisons to appropriate experimental data. Provide comparisons of your methodology with the guidance of SRP 6.2.1.1.A.

AECL Response
A response to this RAI will be provided by April 2005.

174b. Will the GTM be utilized to calculate minimum containment pressures for use in emergency core cooling evaluations? If so, please provide the details of the options to be used, justify that these options are conservative and provide comparisons to appropriate experimental data. Provide comparisons of your methodology with the guidance of SRP 6.2.1.5.

AECL Response
A response to this RAI will be provided by April 2005.

174c. Will the GTM be utilized to determine NPSH for safety related equipment following an accident? If so, please demonstrate that the analysis meets the requirements of NRC GL 97-04.

AECL Response
A response to this RAI will be provided by April 2005.

174d. We understand that the GTM will be utilized to model the pressurizer for ACR-700. Please provide validation of the model for the pressure effects of in-surges and out-surges into the pressurizer as well as for the condensation efficiency of the pressurizer spray.

AECL Response
A response to this RAI will be provided by April 2005.

175. Section 7.15 describes the CATHENA point kinetics model. Will the point kinetics model be used to model ACR-700? If so then please describe and justify which options will be implemented. Under what conditions and for which transients the model will be utilized?

AECL Response
A response to this RAI will be provided by April 2005.

176. Since the ACR-700 will have a negative coefficient of reactivity for steam voids within the coolant channels, the NRC staff believes that it may be appropriate to utilize point kinetics to model certain transients and accidents for ACR-700. The staff would like to use point kinetics in audit calculations using RELAP5. Please provide the following data for ACR-700 which will be used in the RELAP5 point kinetics model: delayed neutron precursor yield and decay constants, scram reactivity as a function of time, reactivity as a function of coolant density and temperature, and reactivity as a function of fuel temperature. The heavy water moderator may be a source of delayed photo-neutrons. Describe how these photo-neutrons are included in a point kinetics model.

AECL Response
A response to this RAI will be provided by April 2005.

177. Section 7.16 describes the CATHENA pump model. Built-in models for 8 pump designs are described. ACR-700 pump characteristics are not included. Please discuss how the pump characteristics for ACR-700 will be determined and utilized in a conservative manner for safety analysis.

AECL Response
A response to this RAI will be provided by April 2005.

178. The CATHENA pump model description in Section 7.16 states homogenous flow is assumed through a pump and that this assumption is valid only for low void fractions. It is further stated that a pump model with a wider range of applicability would be desirable and will be incorporated when it is available.

178a. Please provide the schedule for developing the improved pump model. Discuss the need for such a model for ACR-700 safety analysis.

AECL Response
The development of a more fundamental pump model is part of the longer-term advanced thermal hydraulic code development project.

A more fundamental pump model is not required for ACR-700 analyses. Through the current code version input, the user has complete control of the pump behavior during a transient simulation. The user specifies pump head and torque versus flow for both single-phase and two-phase fully-degraded conditions for all four pump operational quadrants. In addition, the user defines the transition process from single-phase to fully degraded two-phase behavior. These modeling inputs allow the analyst all of the tools necessary for examination of nominal behavior as well as modeling uncertainties.

The RCS pump model used for ACR-700 analysis is a User Pump Model (section 7.16.10) which is based on manufacturer's data for the first quadrant. The model uses the same two-phase head multiplier and two-phase torque multiplier as the ANC pump (section 7.16.2). Since data for two-phase head and torque degradation in full-scale pumps is limited, uncertainties in the two-phase multipliers will be covered by sensitivity studies on these parameters.

178b. In the US reactor coolant pumps are tripped either automatically or by procedure when the reactor coolant becomes two phase. This is because under small break LOCA conditions the reactor system may become highly voided if the coolant pumps are permitted to remain operating. Delayed trip of the reactor coolant pumps while the reactor system is highly voided for certain break sizes has been determined to lead to core uncovery for an extended period of time. Please describe any studies applicable to the ACR-700 investigating the effects of pump trip on core uncovery during a LOCA.

AECL Response
A response to this RAI will be provided by April 2005.

179. Do the loop seal regions of the reactor coolant pumps trap water during blowdown and cause steam binding during reflood. What benchmarking has been done to justify loop seal clearing during small and large breaks? Is it important and if not, why not?

AECL Response
A response to this RAI will be provided by April 2005.

180. The CATHENA secondary-side separator model is discussed in Section 7.17.1. The model calculates the void fraction transported through the separation equipment as a function of user provided input. How will the user input be determined for steady state operation and for accident analysis? What is the experimental basis for these assumptions? Following a main steam line break what assumptions will be made for the separation equipment? How are these assumptions justified and how are they made conservative for 1) containment analysis and 2) for reactor system cooldown analysis?

AECL Response
A response to this RAI will be provided by April 2005.

181. CATHENA horizontal connector separation models are described in Section 7.17.2. These models provide for calculation the void fraction in the off-take pipe as a function of the water level within the upstream pipe. Modifications are provided to calculate steam and liquid pull-through for high velocities within the off-take pipe. Please provide the following information concerning this model.

181a. It is stated that the application of the liquid and vapor pull-through models has not been validated for CATHENA. Since entrainment at the entrance to feeder pipes may be important to determining voiding and refill of the headers and pressure tubes during a LOCA, please discuss how this validation will be accomplished. Please justify that any test data referred to is adequately scaled for ACR-700.

AECL Response
Background:
The importance of flow distribution in headers has been recognized by AECL and various programs have been put in place to enhance understanding of header phenomena. For example, the Large Scale Header Facility (LASH) was designed to perform experiments under conditions typical of those expected during postulated-accidents. Experiments were conducted in the facility between about 1985 and 1992. A variety of flow patterns and a wide range of feeder inlet void distributions has been detected under steady-state or transient conditions. A large number of data have been generated. An analysis of steady-state LASH tests can be found in [1]. Similarly, the RD-14M Component Characterization Facility (CCF) has been used to characterize flow and void distribution in an RD-14M inlet header [2]. A third facility, the Header Flow Visualization Test Facility (HFVF) was constructed to allow direct observation of the two-phase behavior in a manifold similar to a CANDU header-feeder system [3]. This later facility also provided analysis background for LASH instrumentation response since these facilities are of similar scale.

Entrainment and Vapor Pull-Through:
Vapor pull-through/liquid entrainment correlations developed in [1] have not extensively been tested, due mostly to the difficulties involved in defining a distinct water level in the headers. However, considering a collapsed water level in the header it has been shown that predictive capability may be incorporated in CATHENA to better estimate the feeder void fraction and flow distributions in the header. These correlations have been implemented in CATHENA and documented in the CATHENA Theory Manual. Implementation of vapor pull-through and liquid entrainment correlations in CATHENA has been verified against LASH data for multiple feeders and against steam-water data for single-branch tests performed at Idaho National Engineering Laboratory (INEL) [4]. This verification effort has been documented in [1] and favorable results have been demonstrated for both single-branch as well as multiple feeder configurations.

AECL has recently assembled a team with the objective of improving our understanding and modeling of header and multiple channel behavior. As part of this objective, existing data will be analysed and its applicability to ACR-700 will be determined. This will be followed by validation of CATHENA against selected tests to determine applicability of the code version to ACR-700 analysis.

Scaling:
All the above test facilities have been built and operated prior to AECL's undertaking of the ACR-700 design and licensing activities and therefore, scaling assessments for ACR-700 headers are planned. For example, LASH and HFVF facilities are full diameter but

half length Pickering-type headers. The header task team assembled by AECL will determine the applicability of this data to ACR-700 conditions. This team will determine further work that might be required to better understand scaling issues with respect to using these data.

References:

1. *J.E. Kowalski. and B.N. Hanna, "Studies of Two-Phase Flow Distribution in a CANDU-Type Header/Feeder System", In Proceedings of the 4th International Topical Meeting on Nuclear Reactor Thermal-hydraulics, Karlsruhe (FRG), October 10-13, Vol. 1, 28-33.*
2. *R.S. Swartz, "RD-14M Facility Description and Characterization', Atomic Energy of Canada Ltd Report, COG-00-034-R1 (2003)*
3. *K.O. Spitz and S.Y. Shim, "Description for Header Flow Visualization Test Facility", Atomic Energy of Canada Ltd Report, COG-90-47 (1991)*
4. *J.L. Anderson and W.A. Owca, "Data Report for the TPFL Tee/Critical Flow Experiments", NUREG/CR-4164 Draft Report, June 1985.*

181b. Justify that the CATHENA code can adequately calculate break flow discharge during the blowdown and refill periods following a loss of coolant accident for connections to the header pipes and pressure tubes that are in the various orientations that will be used at ACR-700.

AECL Response
CATHENA break discharge modeling has been validated extensively. An overview of this effort is given in [1]. The validation matrix for the CATHENA break discharge model covers a wide range of conditions represented in a number of experiments as well as known analytical solutions [1]. For example, single-phase (gas or liquid) and two-phase blowdown tests (Edwards, Marviken, RD-12, RD-14 and RD-14M) under critical and sub-critical (Marviken bottom blowdown, RD-14) have been demonstrated to capture measured break flow rates within experimental scatter. Orientation effects have been studied (Marviken blowdown) and been shown to be predicted by CATHENA within the experimental measurement errors. Since the phase velocities near a break are large (inducing significant mixing particularly for large breaks) the impact of break orientation resulting from any phase distribution in the header is minimal.

AECL has recently assembled a team with the objective of improving our understanding and modeling of header and multiple channel behavior. As part of this objective, existing data will be analysed and its applicability to ACR-700 will be determined. This will be confirmed by validation of CATHENA against selected tests to determine applicability of the code version to ACR-700 analysis.

Reference:

1. *RC-2701-R1, "CATHENA MOD-3.5c/Rev.0 Systems Thermalhydraulic Validation Manual", edited by W. Won, X.M. Huang and G.M. Waddington, 2003 September.*

182. Four options are available in the code for calculating two-phase multipliers for valves and orifices. For the various valves and orifices modeled in CATHENA for the ACR-700, please indicate which model will be used and justify its use is appropriate for ACR-700 safety analysis.

AECL Response
A response to this RAI will be provided by April 2005.

183. Loss coefficient correlations are available for CANDU breakdown orifices used in two sizes of fuel channel feeder pipes (1 ½ inch and 2 inch). Please identify which of these two feeder pipe sizes is utilized in the ACR-700.

AECL Response
A response to this RAI will be provided by April 2005.

CATHENA Validation Plan RC-2240

184. Document RC-2204 "Validation Plan for CATHENA Mod-3.5c" presents in Table 1, 23 phenomena for which the CATHENA code will be validated.

184a. Please discuss the processes and the qualification of the personnel utilized in development of this table. Provide a comparison of this process with the PIRT process discussed in Draft Regulatory Guide DG-1120.

AECL Response
A response to this RAI will be provided by April 2005.

184b. Eighteen of the phenomena in Table 1 are shaded. Some are shaded darkly and some are shaded lightly indicating the priority of the phenomena for the various accident categories. Please discuss the significance of the degree of shading and how the degree of shading was determined for each accident category and for each phenomenon.

AECL Response
A response to this RAI will be provided by April 2005.

184c. For the various accident categories, phenomena are identified as primary or secondary phenomena. Please discuss the significance of this categorization and how it was determined.

AECL Response
A response to this RAI will be provided by April 2005.

CATHENA Validation Manual RC-2701

185. Report RC-2701 describes validation of CATHENA for 23 thermal/hydraulic phenomena relevant to CANDU accident analysis. For ACR-700 analysis CATHENA Mod-3.5d will be

utilized whereas the validation exercises were performed with Mods3.5b and 3.5c of CATHENA. For each of the 23 phenomena investigated in report RC-2701, please justify that the validation work performed on the earlier mods of CATHENA are valid for the version to be used for ACR-700 safety analysis. Compare thermal/hydraulic conditions measured in the tests with those expected in the ACR-700 under accident conditions.

AECL Response
A detailed response to this question, in the form of a report documenting AECL's plan for qualification of CATHENA MOD-3.5d for ACR-700 analyses, is being prepared, and will be provided in 2005 April. This report will examine each of the 21 Fuel Channel and System Thermal Hydraulics, and 9 Fuel and Fuel Channel Thermal-Mechanical Effects phenomena listed in the Technical Basis Document relevant to analyses using CATHENA. If the validation work performed for CATHENA MOD-3.5c or planned as part of the generic validation for the MOD-3.5d code version is considered adequate for ACR-700 application, justification will be provided. Plans will be included for the validation of any phenomena for which additional, ACR-specific validation is considered necessary.

Chapter 3.1 Break Flow Models

186. Section 3.1 describes validation of CATHENA for predicting break flow. Comparisons to data from 7 experiments are discussed. The test facilities were for various conditions of break flow. In most cases predicted to measured break flow was not actually compared but the degree of prediction was inferred indirectly from the pressure traces. Please provide the following information concerning the break flow validation.

186a. CATHENA provides several options for predicting break flow. For each of the 7 validation comparisons discussed in Section 3.1, identify the CATHENA break flow option that was used. Also state if the tested break flow option will be used for analysis of ACR-700 and identify the accident category and conditions for which the option will be utilized for ACR-700 analysis.

AECL Response
A response to this RAI will be provided by April 2005.

186b. Section 3.1.3 discusses an error in the ability of CATHENA to predict two-phase discharge rates under low pressure drop conditions. Please discuss the significance of this error for ACR-700 analysis. Has this error been corrected?

AECL Response
A response to this RAI will be provided by April 2005.

186c. Provide representative graphical comparisons of the break flows predicted by CATHENA to those of the experimental facilities. Justification should be provided that all break flow conditions significant to ACR-700 analysis are included.

AECL Response
A response to this RAI will be provided by April 2005.

Chapter 3.2 Coolant Voiding

187. Section 3.2 describes validation of the CATHENA code for prediction of coolant voiding following a postulated loss of coolant accident. The proper prediction of coolant voiding within the fuel channels is important for predicting the reactivity feedback for core power determination and for determination of fuel element heat transfer. Please provide the following information concerning this validation.

187a. In simulation of Marviken experiments, Christensen's power void experiments, RD-14 and RD-14M; noding of the heated section was found to significantly affect the results. Discuss the noding detail that was evaluated for these data comparisons and how these results were utilized in development of the CATHENA model for ACR-700.

AECL Response
For each experimental facility used for CATHENA validation, a reference nodalization has been developed. For each validation exercise, sensitivity studies are performed to demonstrate the spatial convergence of this nodalization.

For the validation of CATHENA for coolant voiding, the RD-14 fuel channel (which has the same length as ACR-700) was discretized into 12 axial segments (as is ACR-700). The sensitivity analysis showed little effect of axial nodalization on the predicted void fraction (RC-2332, Section 5.4). In the RD-14M validation exercise described in RC-2702-Rev.0 and RC-2332, the fuel channel was discretized into only 6 axial segments. Analyses showed that axial nodalization did affect the integrated channel void fraction (RC-2332, Figure 5.89). The recommended nodalization for RD-14M and ACR-700 fuel channels is now 12 axial segments. A more recent validation exercise, using 12-node discretizations of RD-14M fuel channels, shows that this is sufficient for coolant voiding (RC-2810, Section 5.3 and Figure 120).

The Marviken blowdown experiments were used to validate CATHENA for coolant voiding during fast depressurization transients. Although the finer nodalizations did improve the code's agreement with the measured void fraction in the discharge pipe (RC-2332, Figure 5.11), this test is not as applicable to ACR-700 as the RD-14M tests described in RC-2810. The Christensen power void tests were used to validate CATHENA for coolant voiding under nucleate boiling. The 1.27-m heated length was modeled with 10 axial segments, so each segment length is shorter than for the 12 segments in a 6-m ACR-700 fuel channel. However, the sensitivity analysis showed only *a small effect of the nodalization on the predicted void fraction, demonstrating the spatial convergence of the reference case (RC-2332, Figure 5.21).*

The CATHENA idealization of the ACR-700 follows the modeling approach used to represent RD-14M as closely as possible. A similar subdivision of piping is used, and the same user options are selected as far as is practicable. In particular, the ACR-700 fuel channel is modeled using 12 axial nodes, as is done for RD-14M. The same integration option for heat transfer from the fuel to the coolant is also used in the ACR-700 model.

The following reports are provided on the enclosed CD-ROM:
1. *RC-2332, "Validation of CATHENA MOD-3.5c for Coolant Voiding – Overview Report", by J.A.K. Reid, T.V. Sanderson and J.P. Mallory, 2000 April.*
2. *RC-2810, "Validation of CATHENA MOD-3.5c for Channel Coolant Voiding – RD-14M Large LOCA Experiments (B0104-B0109) and Power-Pulse LOCA Experiments (B0113, B0115, B0116 and B0117)", by D.F. Wang and T.V. Sanderson, 2002 June.*

187b. In comparisons to Christensen's power void experiments, it is stated that the CATHENA input option for splitting the heat flow between the steam and water phases within the heated channel significantly affected the results. Please describe this sensitivity study in more detail and relate the conclusions from this study to the basis for heat transfer splitting between the phases that will be utilized in the ACR-700 CATHENA model of the reactor fuel channels.

AECL Response
In CATHENA, the heat flux to the coolant is split into three components: heat transfer to the bulk liquid, heat transfer directly into vapor generation, and heat transfer to the bulk vapor phase. The wall-to-interface heat transfer correlations determine the split between the heat transferred to the bulk phases and heat transfer that directly results in vapor generation. There are a number of available options for the wall-to-interface heat transfer; the default setting is the Saha-Zuber onset-of-significant-void correlation, which is the option used in these validation exercises. Unlike most correlations, its uncertainty is not built in to the code, but in the validation exercises realistic estimates were used to examine its effect.

In the validation against the Christensen Power Void Experiments, the effect of the Saha-Zuber correlation was examined by applying an uncertainty of ±20%. The resultant changes in the calculated void fraction, in the subcooled region, approximately equal the experimental measurement uncertainty. The void fraction was not affected in the saturated boiling region.

In other validation exercises, for the nucleate boiling phenomenon (RC-2846-5), the sensitivity to the Saha-Zuber correlation was evaluated by applying an uncertainty of ±25.8% (±2σ, assuming σ=12.9% is the same as the wall-to-fluid heat transfer coefficient). For the validation of nucleate boiling using the subcooled boiling tests of

Bartolomei and Rouhani plus selected RD-14M small break blowdown tests, this assumed $\pm 2\sigma$ uncertainty in the wall-to-interface heat transfer resulted in no more than a ±3% difference in void fraction.

This sensitivity to the heat flux splitting correlation is not considered significant, and does not warrant any special treatment in CATHENA analyses of ACR-700 fuel channels. The effect of the uncertainty in the wall-to-interface heat transfer can be assessed through an uncertainty or sensitivity analysis as required.

The Saha-Zuber correlation is used in ACR-700 analyses in all components except the fuel channels, which applies a correlation (COG-01-209, 'WALL-INTERFACE-HEAT-TRANSFER' options 9, page 5-16) for the onset-of-significant void for a fuel bundle in a crept or uncrept channel. The effect of the uncertainty in this correlation on coolant voiding will be assessed through a confirmatory validation of CATHENA MOD-3.5d, to be completed in 2006.

The following reports are provided on the enclosed CD-ROM:
1. *RC-2332, "Validation of CATHENA MOD-3.5c for Coolant Voiding – Overview Report", by J.A.K. Reid, T.V. Sanderson and J.P. Mallory, 2000 April.*
2. *RC-2846-5, "Validation of CATHENA MOD-3.5c/Rev 0 for Nucleate Boiling — Overview Report", by G.M. Waddington and S.M. Froebe, 2003 August.*
3. *COG-01-209, "CATHENA MOD-3.5d/Rev 0 GENHTP Input Reference", edited by B.N. Hanna and T.G. Beuthe, 2003 July.*

187c. The Christensen's power void experiments which were for a vertical heated section appear to provide the only data for void formation within a heated pressure channel. Please provide a description of this facility including drawings and a description of the test procedure. Provide the complete set of the code to data comparisons, sensitivity studies performed and conclusions from these studies. Please justify that use of this data is an appropriate benchmark for the horizontal core channels of ACR-700.

AECL Response
The validation of CATHENA MOD-3.5c against the Christensen power void experiments is described in the coolant voiding validation overview report (RC-2332). This includes a description of the facility, comparisons of code predictions and experimental measurements, sensitivity studies and conclusions. These experiments were one of several sets of data used to validate CATHENA for void formation in a heated channel. The others include the McMaster, Bartolomei and Rouhani boiling experiments, plus several RD-14M tests. These are documented in RC-2810 and RC-2846-5. These validation exercises cover a range of channel geometries (tube, annulus and simulated fuel bundle) and test conditions. The experiments most representative of ACR-700 are the RD-14M tests, upon which the CATHENA idealization of ACR-700 is based. The CATHENA idealizations of RD-14M and ACR-700 are compared in 10810-03500-AR-005.

The following reports are provided on the enclosed CD-ROM:

1. *RC-2332, "Validation of CATHENA MOD-3.5c for Coolant Voiding – Overview Report", by J.A.K. Reid, T.V. Sanderson and J.P. Mallory, 2000 April.*
2. *RC-2810, "Validation of CATHENA MOD-3.5c for Channel Coolant Voiding – RD-14M Large LOCA Experiments (B0104-B0109) and Power-Pulse LOCA Experiments (B0113, B0115, B0116 and B0117)", by D.F. Wang and T.V. Sanderson, 2002 June.*
3. *RC-2846-5, "Validation of CATHENA MOD-3.5c/Rev 0 for Nucleate Boiling — Overview Report", by G.M. Waddington and S.M. Froebe, 2003 August.*

The following report has already been provided to the US-NRC:

4. *10810-03500-AR-005, "ACR-700 CATHENA Circuit Model", by H. Zhao, V. Yee, J. Lim, L. Bratu, 2004 August.*

187d. In comparisons to voiding data collected at locations outside the core channels from the RD-14 and RD-14M facilities, it was found that the test channels had to be forced into the CATHENA mixed flow regime to predict the data. Please discuss the implication of this finding for ACR-700 analysis. Please justify that assumptions made for flow mixing in the data comparisons are the same as those used for the ACR-700 and that the assumptions are appropriate for ACR-700 safety analysis.

AECL Response
A response to this RAI will be provided by April 2005.

187e. In comparisons to voiding data collected outside the core channels from the RD-14 and RD-14M facilities it was found that small errors in determining the flow split for fluid leaving the ends of the test section during a simulated loss of coolant accident could significantly affect the results. Please discuss the implications of this finding for ACR-700 analysis. What validation has been performed for the ability of CATHENA to predict core channel flow during a LOCA?

AECL Response
A response to this RAI will be provided by April 2005.

187f. Section 3.2.5 states that "none or the tests used in this validation provided coolant voiding rates within a CANDU representative channel subjected to a fast depressurization transient. However, experiments are currently underway in AECL's RD-14M facility to measure fast voiding within a CANDU-like channel using a neutron scatterometer device." Please provide the predictions of the CATHENA code for this data and compare the model used to that for analysis of ACR-700.

AECL Response
The validation of CATHENA MOD-3.5c against the RD-14M LOCA experiments, with in-channel void measurements using the neutron scatterometer, is now complete. These

experiments provide void fraction data for a horizontal heated section. The validation exercise is documented in RC-2810 and summarized in the updated validation manual RC-2701-Rev.1.

In general, only the default code model options are used in a validation exercise; any exceptions are documented and justified in the validation report. The same approach is used for the ACR-700 idealization, in both the RD-14M and ACR-700 simulations. The RD-14M and ACR-700 idealizations are similar; they are compared in 10810-03500-AR-005.

For the validation of CATHENA using the RD-14M tests with the neutron scatterometer, the code model options which affect coolant voiding in the channel are the nucleate boiling (wall-to-liquid) heat transfer coefficient and the wall-to-interface heat transfer correlation . Both RD-14M and ACR-700 simulations use a modifed version of Chen's correlation for wall-to-fluid heat transfer. For the wall-to-interface heat transfer, the RD-14M validation used the Saha-Zuber onset-of-significant void (OSV) correlation, while ACR-700 simulations use an OSV correlation recommended for CANDU fuel channels (see the response to question 187b). The validations exercises show that the sensitivity of coolant voiding to the uncertainties in these correlations is not significant.

The following reports are provided, in PDF format, on the enclosed CD-ROM:
1. *RC-2810, "Validation of CATHENA MOD-3.5c for Channel Coolant Voiding – RD-14M Large LOCA Experiments (B0104-B0109) and Power-Pulse LOCA Experiments (B0113, B0115, B0116 and B0117)", by D.F. Wang and T.V. Sanderson, 2002 June.*
2. *RC-2701-Rev.1, "CATHENA MOD-3.5c/Rev 0 System Thermalhydraulics Validation Manual", edited by W. Won, X.M. Huang and G.M. Waddington, 2003 September.*

The following report has already been provided to the US-NRC:

3. *10810-03500-AR-005, "ACR-700 CATHENA Circuit Model", by H. Zhao, V. Yee, J. Lim, L. Bratu, 2004 August.*

187g. Since small errors in predicting the initial voiding location and flow split from a depressurized channel can significantly affect the predicted results, size of the test section may have an effect on the result. Please discuss the effect of channel scale on the result of channel voiding and the advisability of performing separate effects experiments for a full scale channel for additional benchmarking of CATHENA.

AECL Response
A response to this RAI will be provided by April 2005.

188. In report 108US-03532-225-001 "CATHENA Simulation of RD- 14M Critical Break LOCA Experiment B9401," CATHENA was shown to significantly under predict the void fraction in the feeder tubes leading to and from the affected fuel channels after about 50 seconds into the test so that more cooling water was predicted to be flowing to and from the core than

was actually the case. See figures 10 and 11. Although the under prediction of voiding did not appear to greatly affect the cladding temperature comparisons, for ACR-700 analysis the effect of voiding might be of more significance for certain accident scenarios. Please identify the code deficiencies that caused this under prediction and discuss how they will be corrected.

AECL Response
A response to this RAI will be provided by April 2005.

Chapter 3.3 Phase Separation

189. Please provide report RC-2340, "Validation of CATHENA MOD-3.5c for Phase Separation-Overview Report."

AECL Response
Report "Validation of CATHENA MOD-3.5C for Phase Separation – Overview Report", RC-2340, was submitted to the NRC on April 13, 2004.

190. Page 35 of report RC-2701 states that "a sensitivity analysis showed that increasing the number of nodes steepened the predicted wave profile that is theoretically shown to be a vertical front for the bore and a parabolic profile for the depression wave." This study relates to the prediction of phase separation within the fuel channels. Discuss how this sensitivity study was implemented in determining the noding detail for the fuel channels of the ACR-700 CATHENA model.

AECL Response
A response to this RAI will be provided by April 2005.

3.5 Heat Transport Pump Characteristics

191. Section 3.5 describes validation of the CATHENA pump model. Please justify that this data is applicable to the reactor coolant pumps to be installed for ACR-700. Compare the specific speeds for the pumps used in the tests to those of ACR-700.

AECL Response
A response to this RAI will be provided by April 2005.

192. Page 49 of RC-2701 describes significant discrepancies in simulating pump characteristics in the transition from single-phase to highly voided two-phases flow and states that changes in the pump models are required. Please describe these changes and provide comparisons to appropriate experimental data to show that the pump model in CATHENA is now adequate.

AECL Response
No changes to the pump model have been included in CATHENA. A more fundamental pump model is not required for ACR-700 analyses.

In the validation performed [1 & 2], the calculated pump behavior was within the uncertainties for both single-phase and fully-degraded two-phase conditions. The validation specifically highlighted that the uncertainty is largest in the transition region and that this was consistent with large uncertainty in the pump head degradation ($M(\alpha)$, ±20%). The validation also highlighted the larger uncertainty for lower saturation temperatures that is in large part a consequence of larger flow measurement uncertainties in two-phase flows.

References:
1. *W. Won, X.M. Huang and G.M. Waddington, "CATHENA MOD-3.5c/Rev 0 System Thermalhydraulics Validation Manual", Atomic Energy of Canada Limited Report, RC-2701 Revision 1, September 2003.*
2. *D.M. Kawa, J.P. Mallory and D.J. Wallace, "Validation of CATHENA MOD-3.5c For Heat Transport Pump Characteristics – Overview Report", Atomic Energy of Canada Limited Report, RC-2443, 2000.*

Chapter 3.7 Convective Heat Transfer

193. Section 3.7 describes validation of CATHENA for convective heat transfer. Comparisons of code predictions to test data from several test facilities are described.

193a. For each test facility provide a comparison of the CATHENA model that was used to predict the test data to that which will be utilized to analyze ACR-700, in particular compare the noding detail for the test section and the equivalent component for the ACR. Compare the heat transfer option selected to predict the test data with that which will be used for ACR-700 analysis.

AECL Response
A response to this RAI will be provided by April 2005.

193b. Tests were performed to evaluate convective heat transfer at the CWIT facility and at RD-14 for 37-element CANDU fuel. What additional validation will be performed to validate CATHENA for convective heat transfer for ACR-700 CANFLEX fuel.

AECL Response
The validation of CATHENA MOD-3.5c for convective heat transfer included exercises with experiments covering a range of channel geometries: tubes (Harwell steam tests), 37-element bundles (RD-14 and CWIT) and 7-element bundles (RD-14M and CHAN). All this validation is considered to be applicable to the ACR-700, because the modified Chen and Heinemann heat transfer correlations (used to model liquid and vapor

convective heat transfer, respectively, in both the validation exercises and the ACR-700) are not bundle geometry dependent. An additional confirmatory validation exercise is planned. This new validation exercise will use data obtained in the RD-17 facility, at ACR-like pressures and temperatures, and is scheduled to be completed in 2005 October. The RD-17 test section consists of a single, 13-mm diameter heater element in a pressurized flow tube. The flow tube inside diameter is chosen using a scaling analysis based on CANDU 37-element and RD-14M 7-element bundles. Because the convective heat transfer correlations are not bundle geometry dependent, convective heat transfer tests using a full 43-element ACR-700 bundle are not considered necessary.

193c. Convective heat transfer to steam tests at the CHAN facility were used to validate CATHENA for these conditions. Above 700OC thermal radiation and zirconium-steam oxidation effects interfered with the use of this data for code validation. It was concluded that more suitable data from 700 to 1500^{O}C needed to be utilized to validate the code. Please provide these data comparisons.

AECL Response
The response to this RAI will be provided by April 2005.

Chapter 3.8 Nucleate Boiling

194. Section 3.8 discusses the need for nucleate boiling data to validate CATHENA. Please provide this validation for ACR-700 CANFLEX fuel. Identify the CATHENA heat transfer correlations that are being validated.

AECL Response
The validation of CATHENA MOD-3.5c for nucleate boiling heat transfer has been completed. It is documented in the nucleate boiling overview report (RC-2846-5) and is summarized in the revised validation manual (RC-2701-R1, Chapter 3.8). This validation included exercises with experiments covering a range of channel geometries: tubes (McMaster, Bartolomei), rectangular duct (Christensen), annulus (Rouhani), and 7-element bundles (RD-14M). All this validation is believed to be applicable to the ACR-700, because the modified Chen heat transfer correlation (used to model nucleate boiling heat transfer, in both the validation exercises and the ACR-700) is not bundle geometry dependent. An additional confirmatory validation exercise is planned. This new validation will use data from the RD-17 facility at ACR-like pressures and temperatures, and is scheduled to be completed in 2005 October. The RD-17 test section consists of a single heater element in a pressurized flow tube. Because the nucleate boiling heat transfer correlation is not bundle geometry dependent, tests using a full 43-element ACR-700 bundle are not considered necessary.

The following reports are provided on the enclosed CD-ROM:
1. *RC-2701-R1, "CATHENA MOD-3.5c/Rev.0 Systems Thermalhydraulic Validation Manual", edited by W. Won, X.M. Huang and G.M. Waddington, 2003 September.*

2. *RC-2846-5, "Validation of CATHENA MOD-3.5c/Rev.0 for Nucleate Boiling — Overview Report", by G.M. Waddington and S.M. Froebe, 2003 August.*

Chapter 3.9 CHF and Post Dryout Heat Transfer

195. Section 3.9 described CATHENA validation for CHF and post dryout heat transfer. Data comparisons are discussed for simulated fuel bundles and calandria tube heat transfer.

195a. For each test facility provide a comparison of the CATHENA model including noding and CHF correlation used to predict the test data with the ACR-700 analysis model.

AECL Response
In general, for both CATHENA validation and ACR-700 analyses the default or recommended correlations and options are used. If a non-default correlation or option is applied, then it is justified and documented (a sensitivity analysis may be provided). Spatial convergence should be demonstrated in the validation simulations. This is confirmed by a nodalization sensitivity analysis, which shows that increasing the number of nodes beyond what is used in the validation has no significant impact on the results. The results of the nodalization sensitivity studies are taken into consideration in the development of the ACR-700 nodalization.

Comparison of the CATHENA models for each facility and for ACR-700 analysis is as follows. Note that the default correlations for CHF and Post-Dryout (PDO) heat transfer in CATHENA MOD-3.5c are the Groeneveld-Leung look-up tables for CHF, the Bjornard & Griffith correlation for transition boiling, and the Groeneveld-Delorme correlation for film boiling. Please refer to RC-2518-5 for a detailed description of the validation exercises.

1. *37-element CHF tests: use 12 thermal hydraulic nodes for 6-m long horizontal heated channel; 37-element FES bundle was modeled using 4 pin groups. The default heat transfer correlations were applied in simulation.*
2. *Bennett post CHF tests: use 40 thermal hydraulic nodes for 219-in (1st series) and 144-in (2nd series) long vertical test section; the default heat transfer correlations were applied.*
3. *Pool boiling separate effect tests: use 1 thermal hydraulic node because the problems are zero-dimensional. The default heat transfer correlations were applied.*
4. *Horizontal tube rewetting/refilling tests: use 12 thermal hydraulic nodes for 3-m long heated horizontal zircaloy tube. The default heat transfer correlations were applied.*
5. *RD-14M blowdown tests: use 6 thermal hydraulic nodes for 6-m long heated horizontal channel; 7-element FES bundle was modeled using 3 pin groups. Overall model consists of 496 nodes, 511 links, and 159 heat transfer models for test B9802, and 287 nodes, 294 links, and 115 heat transfer models for test B0002. The default heat transfer correlations were applied. A recommendation from the validation exercises using RD-14M data (see question 187a) is to use 12 thermal hydraulic*

nodes to represent the heated channel in any future simulations of RD-14M experiments.

6. *ACR-700: use the same fuel channel nodalization as that recommended based on the validation. The CHF and PDO models differ from those used for the MOD-3.5c validation. For CHF, a new look-up table specific to CANFLEX bundles is used. The PDO look-up table has also been updated to improve its accuracy at ACR-typical conditions. To address deficiencies noted in the MOD-3.5c validation for PDO heat transfer, new models for developing PDO heat transfer were developed; ACR-700 simulations use a model specific to CANFLEX fuel bundles (COG-01-209, option 'DEV-PDO-3', page 5-25). A new CATHENA MOD-3.5d correlation (the CHF look-up table for 43-pin CANFLEX fuel bundle) will be applied in simulations of ACR-700. The validation of CATHENA using these new models is currently underway and completion is expected by 2005 April. More information is provided in the responses to questions 195.d and 195.e.*

The following report is provided on the enclosed CD-ROM:

1. *COG-01-209, "CATHENA MOD-3.5d/Rev 0 GENHTP Input Reference", edited by B.N. Hanna and T.G. Beuthe, 2003 July.*

195b. In many of the large scale tests involving multiple assembly bundles, CATHENA was found to over predict CHF in comparison to the test data. This indicates that the CHF correlations in CATHENA are not conservative for safety analysis. Please discuss how conservative predictions of CHF will be obtained for ACR-700 safety analysis.

AECL Response

The CHF mechanism at ACR-700 flow conditions corresponds to liquid film dryout, where the clad-temperature rise beyond CHF is gradual and controllable. The bundle CHF table provides the best-estimate predictions of the initial CHF occurrence, which represents only a single dry spot in the entire 6-m long bundle string. This is considered conservative as the heat-transfer rate remains high (almost the entire bundle string remains under nucleate boiling) and the clad temperature remains low and controllable for Loss of Regulation and Loss of Flow types of accident scenarios. Slight differences in CHF between predictions and measurements could be encountered due to the prediction uncertainty.

In the validation exercise using the 37-element CHF tests (it is assumed that the question is referring to these tests), the heated length used in the validation was incorrect by approximately 3%, which would have reduced the overprediction by that amount. The CHF table in CATHENA MOD-3.5d is improved relative to the MOD-3.5c code version, so that the table predicts the data within the scatter (some high and some low). The methodology for predicting CHF for ACR-700 is outlined in response to RAI #166.

195c. Data from a simulated 37-element CANDU fuel bundle tests was predicted. Please provide predictions by CATHENA for data that models ACR-700 CANFLEX fuel. Provide uncertainty analyses so that the margin to CHF for ACR-700 fuel can be determined with a high degree of confidence. Discuss how the accuracy and confidence level for the prediction of CHF meets the guidance of Standard Review Plans 4.2 and 4.4. For the ACR-CANFLEX data please discuss how fuel channel flow distribution was included for radial creep which would increase the flow area between the top of the fuel bundle and the top of the fuel channel.

AECL Response
A response to this RAI will be provided by April 2005.

195d. Provide comparisons with post CHF data that appropriately models ACR-700 CANFLEX fuel. Include post CHF film boiling data as well as post dryout data. Evaluate the uncertainty in this data.

AECL Response
The current post-dryout heat-transfer correlations were derived with full-scale CANFLEX-bundle data obtained at conditions close to the ACR-700 operation (test pressure up to 11 MPa as compared to ACR-700 pressure of 12.5 MPa and test mass-flow rate up to 25 kg/s as compared to ACR-700 mass flow rate of 26 kg/s). These correlations are expressed in terms of a ratio of heat-transfer coefficient, and are applicable to the ACR-700 bundle (the differences in flow conditions between experiments and ACR-700 operation are minor). The post-dryout correlation for minimum post-dryout heat-transfer coefficient bounds all experimental values from the full-scale bundle tests. It follows asymptotically towards the fully developed post-dryout heat-transfer coefficient with increasing wall-superheat values. The fully developed post-dryout heat-transfer coefficient is predicted using the film-boiling look-up table. An assessment against Freon data of the CANFLEX bundle showed that the film-boiling look-up table predicts the experimental values of fully developed post-dryout heat-transfer coefficient with an average error of about –1.5% (standard deviation of 10%). Confirmatory data will be obtained with the ACR-700 bundle string at relevant conditions of interest.

195e. Table 4 of RC-2701 list four deficiencies in the CATHENA code for CHF and post dryout heat transfer. These are 1) Film boiling heat transfer rates are underestimated for flowing conditions. 2) Film boiling heat transfer rates are overestimated for stagnant, subcooled conditions and 3) Inconsistent results were identified for uncertainty analysis of transition boiling. Please describe how these deficiencies have been corrected.

AECL Response
Following the validation of the MOD-3.5c code version [1], the MOD-3.5d code version includes a number of modeling additions directly related to the calculation of heat transfer post-dryout (PDO) conditions. The modeling additions for the MOD-3.5d code version that are directly applicable to ACR-700 analyses are as follows:

1. *Specifically for the ACR-CANFLEX fuel geometry a best-estimate PDO heat transfer calculation methodology has been implemented in the MOD-3.5d code version. In this methodology, a developing flow PDO heat transfer calculation [2] has been implemented for application to conditions for bundles between CHF and fully developed film boiling. Heat transfer for fully developed PDO conditions is independent of the fuel bundle geometry. For fully developed PDO conditions, the film boiling heat transfer coefficient calculated is obtained from a PDO look-up table method based on pressure, flow, quality and wall superheat. Confirmatory validation for this best-estimate PDO heat transfer methodology is being performed and will be reported by April 2005.*
2. *The overestimate of film-boiling heat transfer rates for stagnant conditions subcooled was not considered a significant deficiency in CATHENA with respect either ACR or CANDU applications. As reported in Reference 1 (Section 3.9.4), the overestimate reported in PDO heat transfer for stagnant conditions is apparent for large subcooling . Since there is only a small amount of liquid contained within a fuel bundle, the liquid within a fuel bundle for these conditions will be at or near saturation. The validation results in Reference 1 (Section 3.9.4) regarding stagnant conditions are more applicable to severe accident analysis where, for example, CHF and PDO heat transfer conditions on the external surface of the calandria tube is being considered.*
3. *The inconsistency noted, during the validation reported in Reference 1(Section 3.9.4), in the uncertainty analysis for transition boiling resulted from a naming error within the uncertainty input recognized by the MOD-3.5c/Rev 0 code version which allowed the user no access to the uncertainty for the selected correlation. This error has been corrected in the MOD-3.5d code version.*

As a result of the changes to the PDO heat transfer methodology, the validation performed for CHF and PDO heat transfer with the MOD-3.5c/Rev 0 code version is being revisited and extended for the MOD-3.5d code version.
The detailed documentation for the PDO heat transfer methodology charges are included in the CATHENA MOD-3.5d Theoretical Manual in preparation.

References:
1. *W. Won, X.M. Huang and G.M. Waddington, "CATHENA MOD-3.5c/Rev 0 System Thermalhydraulics Validation Manual", Atomic Energy of Canada Limited Report, RC-2701 Revision 1, September 2003.*
2. *N. Hammouda, "Best-Estimate PDO Correlation for CANFLEX Mk-IV Bundle", Atomic Energy of Canada Limited Report, CANFLEX-170, FFC-FCT-477, March 2003.*

Chapter 3.10. Condensation Heat Transfer

196. Validation of condensation heat transfer models in CATHENA is discussed in Section 3.10 of report RC-2701. The comparisons with data indicate that noding detail is important for

predicting void fraction within the fuel element channels, headers and feeder tubes. The text describes how modifications were made to the CATHENA noding detail, heat transfer areas and coefficients to better match the test data. Please address each modification that was made to better match the test data and discuss how this experience is utilized in modeling of the ACR-700.

AECL Response
It should be clarified that the purpose of sensitivity and/or uncertainty analyses in the validation exercises is to quantify the uncertainty of the CATHENA code in predicting the key parameters for the phenomenon, not to "better match the test data".

The general guidelines used in preparing CATHENA models in the validations and in the ACR-700 simulations are listed in the response for question 195a. Comparison of the CATHENA models for each facility and for ACR-700 analysis is as follows. Note that the default correlation for condensation heat transfer in CATHENA simulations is the maximum of a conduction term and the Heinemann correlation for turbulent steam convective heat transfer.

- *McMaster subcooled boiling tests: use 40 thermal hydraulic nodes for 1.146-m (section A) and 1.09-m (section B) long vertical annular test section; the default heat transfer correlation was applied in the simulations.*
- *CWIT feeder refill tests: use 12 thermal hydraulic nodes for 6-m long heated horizontal channel; 37-element FES bundle was modeled using 10 pin groups. Overall model consists of 206 nodes, 207 links, and 1422 heat transfer surfaces. The default heat transfer correlation was applied.*
- *RD-14 blowdown tests: use 12 thermal hydraulic nodes for 6-m long heated horizontal channel; 37-element FES bundle was modeled using 10 pin groups and 24 thermal hydraulic nodes for the primary tubes of each steam generator. Overall model consists of 274 nodes, 279 links, and 647 heat transfer surfaces. The default heat transfer correlation was applied. The effect of the 'STM-GEN-COND' option was also evaluated. This option was applied to the primary side of the steam generators, which increases the condensation heat transfer to a more appropriate level for these components.*
- *RD-14M blowdown tests: use 6 thermal hydraulic nodes for 6-m long heated horizontal channel; 7-element FES bundle was modeled using 3 pin groups and 24 thermal hydraulic nodes for the primary tubes of each steam generator. The overall model consists of 529 nodes, 544 links, and 177 heat transfer models. The default heat transfer correlation was applied. The effect of the 'STM-GEN-COND' option, applied to the steam generators' primary side, was also evaluated. A recommendation from the validation exercises using RD-14M data (see question 187a) is to use 12 thermal hydraulic nodes to represent the heated channel in any future simulations of RD-14M experiments.*
- *ACR-700: use the same fuel channel nodalization as in the validation using the RD-14/14M facilities; i.e., 12 thermal hydraulic nodes for each 6-m fuel channel. The ACR-700 model uses 37 thermal hydraulic nodes for the primary tubes of each steam*

generator, compared with 24 nodes in the RD-14M model. This reflects the fact that the ACR-700 steam generator tubes are longer than the RD-14M tubes. The default heat transfer correlation will be applied in ACR-700 simulations. The 'STM-GEN-COND' option is applied to the primary coolant side of the steam generators.

197. Condensation of steam within the steam generator tubes is an important phenomenon during recovery from small break LOCAs since the reactor is "crash cooled" by secondary system depressurization to facilitate ECCS performance. Section 3.10.3 describes how in comparisons to small break simulation data from RD-14M, an optional "STM-GEN-CONC" model was included in the CATHENA simulation. The STM-GEN-CONC model is not described in the CATHENA theory manual (COG-00-008). Will this model be utilized for ACR-700 analysis? If so, please describe the model and discuss how it is conservative for safety analysis. Justify that the condensation model utilized for ACR-700 analyses is conservative.

AECL Response
A response to this RAI will be provided by April 2005.

198. Following a LOCA signal high pressure ECCS water will be injected into the inlet headers of ACR-700. Condensation heat transfer in the headers will be important for determining the local pressures which will influence ECC flow into the feeder tubes and into the core channels. The headers of neither the CWIT facility nor the RC-14 facility are scaled to the ACR. Please address this apparent deficiency in the code validation and discuss how code validation will be accomplished for this phenomenon.

AECL Response
A response to this RAI will be provided by April 2005.

Chapter 3.12 Quench/Rewet Characteristics

199. Section 3.12 of RC-2701 describes validation of the quench/rewet models in CATHENA using separate effects data from the full scale CWIT facility with 37-element heater sections and from the integral RD-12, 14 and 14M facilities. Please provide the following additional information concerning these data comparisons and their applicability to ACR-700. We understand that parallel channel tests have been run at the CWIT facility. Has CATHENA been benchmarked against these tests. If so, please provide the validation report. If not, please discuss the merits of such benchmarking.

199a. Please provide the following reports. 1) RC-2466 describing the CWIT channel fill tests, 2) RC-1584-8 describing test at RD-12, 3) RC-1584-10 describing tests at RD-14M and 4) RC-2464 which is the quench/rewet overview report.

AECL Response
The following reports were submitted to the NRC on April 13, 2004:

- *"Validation of CATHENA MOD-3.5C for Quenc/Rewet Characteristics - CWIT Channel/Feeder Refill Tests", RC-2466*
- *"Validation of CATHENA MOD-3.5B/Rev 0 for ECC Effectiveness Analysis - RD-12 Large Break Blowdown Experiment B8223", RC-1584-8*
- *"Validation of CATHENA MOD-3.5B/Rev 0 for ECC Effectiveness Analysis - RD-14M Large Break Blowdown Tests with Pump Rundown", RC-1584-10*
- *"Validation of CATHENA MOD-3.5C for Quench/Rewet Characteristics – Overview Report", RC-2464*

199b. The prediction of quench/rewet by CATHENA is stated to be a function of the fuel channel noding. Please compare the axial, radial and circumferential noding used with CATHENA to predict test results with that which will be used for ACR-700 analysis.

AECL Response
A response to this RAI will be provided by April 2005.

199c. Quench/rewet phenomena are of considerable safety significance for the ACR-700. It is important that the conditions predicted for the ACR are encompassed by those of the tests. Please provide comparisons including the pressures, temperatures, channel power and flow rates from both fuel channel ends for a range of postulated loss of coolant accidents between those conditions predicted for CATHENA for the ACR and the conditions covered by tests at each facility. These comparisons should be for the time in the accident when coolant is beginning to reenter the channel until coolant channel voiding no longer occurs. In particular postulated break sizes in the inlet header producing flow stagnation should be included as well as the small break of a feeder tube producing flow stagnation in a single channel. The CATHENA analyses should assume operation of the ECCS with the limiting single failure.

AECL Response
A response to this RAI will be provided by April 2005.

199d. The full scale quench/rewet tests at the CWIT facility were for simulated 37 element CANDU fuel. Will similar tests be run for ACR-CANFLEX fuel? If not please describe the validation basis for the CATHENA code for quench/rewet analysis for this fuel.

AECL Response
No additional CWIT tests are planned using an ACR-CANFLEX simulated bundle. The validation performed to date for quench/rewet has included (see RC-2701 Rev 1, Section 3.12):

- *horizontal tube refilling test (no fuel);*
- *CWIT single-break tests using a 37-element, uniformly-heated test section with off-line end fitting simulators;*
- *CWIT double-break tests using a 37-element, axial-cosine heated test section and in-line, CANDU 6 end fittings;*
- *RD-12 test with two-7-element test sections per pass;*

- *RD-14 37-element uniformly heated test sections;*
- *RD-14M 7-element uniformly heated test sections.*

The validation showed that CATHENA consistently predicts longer channel refill times than those measured in the tests, regardless of the facility scale or geometry. The differences (predicted time minus measured time) range from–4 s to + 38 s.

The hydraulic diameter (basically the ratio of flow volume to surface area to be rewetted) could be used to characterize the geometric differences between the various heated channels. For the horizontal tube refilling tests, 37-element tests (CWIT or RD-14) and RD-14M tests, the hydraulic diameters of the test sections are 25.4 mm, 7.4 mm and 5.5 mm respectively. The hydraulic diameter of the ACR-700 fuel channel containing 43-element CANFLEX fuel is about 7.5 mm.

Thus we believe that the present database and validation are adequate, and no additional tests using a 43-element bundle in CWIT are considered necessary.

Chapter 3.13 Zirc/Water Thermal-Chemical Reaction

200. Section 3.13 of RC-2701 describes validation of CATHENA for prediction of the effects of zirconium/water reaction at elevated temperatures. Comparisons to data from several test facilities is discussed. CATHENA has several options for prediction of zirconium/water reaction. Please identify the model that was used for each comparison and compare these to the models that will be used for ACR-700 safety analysis.

AECL Response
The response to this RAI will be provided by April 2005.

201. Section 3.13.3 discusses CATHENA comparisons to zirconium-water reaction data from the Whiteshell Laboratory and from the CHAN facility. CATHENA was found to under predict fuel cladding oxidization for both of these test series. Please justify the conservatism of models in CATHENA to be used for ACR-700 safety analysis in light of these results.

AECL Response
A response to this RAI will be provided by April 2005.

Chapter 3.14 Reflux Condensation

202. Section 3.14.5 and Table 4 of RC-2701 identifies deficiencies in the validation of CATHENA for reflux condensation such as would exist within the steam generator tubes during a postulated small break LOCA event. The need to assess the code against more reactor typical primary side pressures and tube diameters is identified. Please address these deficiencies and discuss how they will be corrected.

AECL Response
A response to this RAI will be provided by April 2005.

Chapter 3.15 Counter Current Flow

203. Prediction of limiting conditions for countercurrent flow of steam and water is significant for ACR-700 since following a loss of coolant accident ECC water that is injected into the inlet headers must flow against the rising steam within the feeder pipes to reach the fuel channels. Validation of CATHENA for counter current flow is described in Section 3.15 of RC-2701. Please provide the following information concerning this validation.

203a. Countercurrent air/water tests were conducted at Dartmouth. When this test data was predicted by CATHENA, CATHENA over predicted the flooding limit so that water was predicted to be injected through the test section when the data showed that it would be ejected. Prediction of early liquid injection through the feeder tubes is not conservative for safety analysis. Discuss how CATHENA will be made to calculate conservative feeder pipe flooding for ACR-700.

AECL Response
The response to this RAI will be provided by April 2005.

203b. Please provide report RC-1584-3 describing CATHENA validation using Dartmouth countercurrent flow data.

AECL Response
Report "Validation of CATHENA MOD-3.5b/Rev 0 for ECC Effectiveness Analysis – Countercurrent Flow/Flooding in Vertical Pipes", RC-1584-3, was submitted to the NRC on April 13, 2004. Report "Validation of CATHENA MOD-3.5c/Rev 0 for Countercurrent Flow – Overview Report", RC-2475, was submitted to the NRC on May 5, 2004.

203c. Please provide report RC-1584-4 describing CATHENA validation using data from the WNRE elbow flooding tests.

AECL Response
Report "Validation of CATHENA MOD-3.5b/Rev 0 for ECC Effectiveness Analysis – Countercurrent Flow/Flooding in 90 Degree Elbows", RC-1584-4, was submitted to the NRC on April 13, 2004. Report "Validation of CATHENA MOD-3.5c/Rev 0 for Countercurrent Flow – Overview Report", RC-2475, was submitted to the NRC on May 5, 2004.

203d. Validation of CATHENA for countercurrent flow has been performed to date with only low pressure data. Please provide validation for these models at the pressures that will be expected during post LOCA recovery at ACR-700. Justify that this data is appropriately scaled fro accident conditions at ACR-700.

AECL Response
A response to this RAI will be provided by April 2005.

Chapter 3.16 Flow Oscillations

204. Section 3.16 states that validation of CATHENA to model density wave oscillations is scheduled for FY 2002/2003. Please provide the results of this validation.

AECL Response
The validation of CATHENA MOD-3.5c/Rev.0 for flow oscillations has been completed and is summarized in Revision 1 of the validation manual (RC-2701-R1, Chapter 3.16). More details can be found in the flow oscillations validation overview report (RC-2849-5).

The validation for flow oscillations was conducted using two RD-14 flow stability tests, two RD-14M flow stability tests, and two RD-14M natural circulation tests. For all these tests, CATHENA predicted flow oscillations where oscillations occurred in the experiments. The predicted and measured oscillation amplitudes and frequencies were compared. The tests cover a range of conditions (e.g., pressure, channel powers, flow rates) that are directly applicable to the conditions under which flow oscillations may occur in postulated transients in ACR-700; therefore the validation for this phenomenon is directly applicable to ACR-700.

The following reports are provided, in PDF format, on the enclosed CD-ROM:
1. *RC-2701-R1, "CATHENA MOD-3.5c/Rev.0 Systems Thermalhydraulic Validation Manual", edited by W. Won, X.M. Huang and G.M. Waddington, 2003 September.*
2. *RC-2849-5, "Validation of CATHENA MOD-3.5c/Rev.0 for Flow Oscillations — Overview Report", by D.F. Wang and W. Won, 2002 September.*

Chapter 3.17 Natural Circulation

205. Section 3.17 states that validation of CATHENA to model natural circulation phenomena is scheduled for FY 2001/2002. Please provide the results of this validation including validation against RD-14M data from the series of tests for natural circulation when the test assembly was partially drained.

AECL Response
The validation of CATHENA MOD-3.5c/Rev.0 for natural circulation has been completed and is summarized in Revision 1 of the validation manual (RC-2701-R1, Chapter 3.17). More details can be found in the natural circulation validation overview report (RC-2777-5).

The validation for natural circulation was conducted using six CWIT standing start tests, three RD-14 natural circulation tests, three RD-14M natural circulation tests, and three RD-14M transition to natural circulation tests. Overall, CATHENA predicted more than half of the single-phase liquid and two-phase natural circulation loop and channel flow rates to within 20% of the experiments. However, the validation did show that the natural circulation flow rate was sensitive to the pump-stopped pressure loss coefficient under two-phase conditions. The tests cover a range of conditions (e.g., pressure, channel powers, liquid subcooling) that are directly applicable to the conditions under which natural circulation may occur in postulated transients in ACR.

The following reports are provided, in PDF format, on the enclosed CD-ROM:

1. *RC-2701-R1, "CATHENA MOD-3.5c/Rev.0 Systems Thermalhydraulic Validation Manual", edited by W. Won, X.M. Huang and G.M. Waddington, 2003 September.*
2. *RC-2777-5, "Validation of CATHENA MOD-3.5c/Rev.0 for Natural Circulation — Overview Report", by X.M. Huang, 2003 March.*

Chapter 3.18 Fuel Channel Deformation

206. Section 3.18.2.3 describes tests used to validate the code for fuel channel deformation and circumferential fuel channel temperature distribution. In these tests simulated CANDU 37-element or 28-element fuel bundles were allowed to boil down so that the pressure tube would heat and deform. In these tests the pressure tube ballooned so as to make contact with the calandria tube so that fuel channel heat could be removed at the outer surface of the calandria tube. One area of interest for these tests is the ability of CATHENA to predict the temperatures within the simulated fuel pins for these tests. Please provide copies of the report describing the ability of CATHENA to predict the temperature vs. time data for the simulated fuel pins.

AECL Response
A response to this RAI will be provided by April 2005.

207. The fuel channel walls of the ACR-700 are to be thicker than those of the test apparatus described in Section 3.18.2.3 and the gap between the pressure tube and the calandria tube is to be larger. In addition, the ACR design uses tight fitting garter springs in the gap between the fuel channels and the calandria tube so that pressure tube sag will not result in contact. Please discuss how CATHENA will be validated to predict fuel channel deformation, possible contact with the calandria tube, post contact heat transfer and post contact fuel element temperatures for the ACR-700 fuel channel design with 43-element ACR-CANFLEX fuel.

AECL Response
The phenomenon of fuel channel deformation is the same for the ACR-700 as it is for existing CANDUs, for which the experiments used in the validation were designed. The only difference is that the ACR-700 will have pressure tubes with thicker walls. The thicker wall will result in stresses within the tube, under normal operating conditions,

that are within the range of experience for the existing pressure tube design. In the fuel channel deformation model implemented in CATHENA, the pressure tube wall thickness is provided as a user input parameter. Thus the existing model is capable of handling thicker walled pressure tubes and straining/rupture at ACR-700 conditions. Therefore, the existing validation for fuel channel deformation is applicable to ACR-700 analysis. Similarly, the existing validation for pressure tube to calandria tube contact heat transfer is also applicable, because the phenomenon is no different in the ACR-700. In the ACR-700, due to the larger annulus gap between the pressure tube and calandria tube (compared with existing CANDUs), pressure tube sag is the only credible mode of pressure tube to calandria tube contact. In the CATHENA solid-to-solid contact model, this mode of heat transfer is modelled using a user-supplied contact area and contact thermal conductance. The garter springs are not expected to be an issue (see the answer to question 222).

Additional confirmatory validation for fuel channel deformation is planned and will be performed when ACR-specific experimental data becomes available. These experiments and the CATHENA validation are in the R&D scope for ACR. The ACR-700 pressure tube deformation tests are currently scheduled for 2006. The confirmatory validation of CATHENA will follow these experiments.

<u>Chapter 3.20 Steam Condensation Induced Waterhammer</u>

208. Section 3.20 states that CATHENA has not yet been validated to predict steam condensation induced waterhammer but that this work is scheduled for 2002/2003. If CATHENA is to be used to evaluate steam induced waterhammer for ACR-700 safety analysis, please provide this validation.

AECL Response
A response to this RAI will be provided by April 2005.

209. We understand that the water used in the ECI accumulators will be degassed. The potential for waterhammer for degassed water is considerably greater than that for water that is saturated with dissolved gases and the magnitude of any waterhammer that occurs is considerably larger. During recovery from a LOCA, cold degassed water from the ECI accumulators will refill hot steam filled piping of the reactor system. Please provide analyses of the resulting waterhammers that will occur and discuss how further damage to the reactor system will be prevented.

AECL Response
A response to this RAI will be provided by April 2005.

Chapter 3.21 Non-Condensable Gas Effects

210. Section 3.21 states that CATHENA has not yet been validated to predict the effect of the presence of non-condensable gas on safety analysis predictions but that this work is scheduled for 2002/2003. Dissolved gases in the reactor coolant as well as hydrogen gas from potential zirconium-water reaction are listed as non-condensable gas sources. Another source of non-condensable gas is the nitrogen that is used to pressurize the accumulator tanks. We understand that during LOCAs the accumulator tanks will be automatically isolated on low level so that the nitrogen gas will not be released into the reactor system. If valve failures are considered in the analyses either for the design basis or for the PRA, the effect of this nitrogen on core cooling and natural circulation will have to be considered. If CATHENA is to be used in these evaluations for ACR-700, please provide the appropriate code validation.

AECL Response
The response to this RAI will be provided by April 2005.

211. We understand that water used in the ECI accumulators will be degassed since dissolved air in the injected water might affect core cooling in the horizontally oriented core channels during a LOCA. Since there is a nitrogen cover gas above the ECI accumulator water please discuss how nitrogen solution in the water will be prevented. Please discuss the consequences of release of the dissolved nitrogen within the core channels during a LOCA and provide validation that CATHENA can adequately describe phenomena involving the dissolved gas.

AECL Response
A response to this RAI will be provided by April 2005.

Other Validation Issues

212. The NRC staff has run the critical inlet header break for ACR-700 using the CATHENA executable and input that were provided by AECL. The staff has the following questions concerning this analysis.

212a. Following opening of the break, the sheath temperature of the fuel elements in the average fuel channel adjacent to the break reaches a peak temperature of 1061°C at 7.2 seconds and then decreases. We understand that the first engineered safety feature to act to provide core cooling is the opening of the outlet header cross connect line. We understand that CATHENA has not yet been validated to predict the affect of opening of this line on core cooling. Please describe how this validation will be accomplished and on what schedule.

AECL Response
A response to this RAI will be provided by April 2005.

212b. Will analyses be performed for ACR-700 either for the design basis or for the PRA in which it is assumed that the outlet header cross connect line fails to open? If so, please provide code validation for the conditions which are calculated to occur in these analyses.

AECL Response
A response to this RAI will be provided by April 2005.

212c. In the core channels adjacent to the break complete voiding occurs immediately. Then the channels are refilled by the ECI flow. When ECI flow is exhausted at approximately 260 seconds, low pressure injection begins immediately and comes to full flow at 325 seconds. The affected core channels remain filled until LPI reaches full flow. Then these channels void. They void and refill intermittently until 813 seconds. Please describe the phenomena that are occurring at this time. How has CATHENA been validated to model these phenomena? Describe comparisons to any available test data.

AECL Response
A response to this RAI will be provided by April 2005.

213. For the RD-14M tests of the critical header break compare the orientation of the channel having the highest sheath temperature in the tests to that predicted to have the highest sheath temperature for ACR-700. Consider the location of the fuel channels in the core as well as the orientation of the feeder pipes as they connect to the headers and to the core channel.

AECL Response
A response to this RAI will be provided by April 2005.

214. The RD-14M facility contains 10 channels in 5 levels. For representative break sizes please provide comparisons of CATHENA predictions to the test data for quenching time and location. Also, provide comparisons for the peak sheath temperature and location of the peak sheath temperature for each channel.

AECL Response
A response to this RAI will be provided by April 2005.

215. For stagnation header breaks ACR-700 fuel sheath temperatures are predicted to increase early in the transient eventually the LOCA interconnect line opens to provide a source of coolant flow. Analyses by CATHENA for the period before the interconnect line opens, predict small flows in the affected channels which are driven by small pressure differences across the channels. These small flow rates are predicted to mitigate the rise in sheath temperature during the stagnation period. Please demonstrate that the CATHENA code has been adequately validated by comparison to experimental channel flow data during this stagnation period to predict these small flows or discuss how the CATHENA calculations will be supplemented by suitably conservative bounding calculations.

AECL Response
A response to this RAI will be provided by April 2005.

216. Occurrence of flow stagnation in the individual core channels following an inlet header break will depend on the resistance of each individual channel and its connected feeder tubes including the effect from the alignment patten of the fuel bundles in each core channel. Discuss how the variation in resistance will be accounted for in evaluation of header stagnation breaks for ACR-700.

AECL Response
A response to this RAI will be provided by April 2005.

217. If CATHENA is to be used to model anticipated plant transients such as are described in Chapter 15 of the Standard Review Plan NUREG-0800, the code should be validated against transient data from operating plants to the extent possible. Please provide code comparisons to representative plant transients including those causing a decrease in secondary system heat removal, increases in secondary system heat removal, loss of coolant flow and changes in core reactivity.

AECL Response
A response to this RAI will be provided by April 2005.

CATHENA Thermal-Mechanical Validation Plan

218. Section 3.1.2 of RC-2151 discusses phenomena that are not modeled by CATHENA but states that CATHENA is capable of describing certain of the phenomena. Since these phenomena may be addressed in the design certification document for ACR-700, please provide the following information if the phenomena are to be assessed using CATHENA.

218a. Comparisons of CATHENA models with experimental data for fuel bundle behavior following disassembly and rearrangement at the bottom of a pressure tube is discussed. Will CATHENA be used to evaluate severe accidents of this type? If so please provide descriptions of the models to be used including the theoretical equations, user input instructions, and the validation document.

AECL Response
CATHENA will not be used to model fuel channels after fuel bundle disassembly and rearrangement at the bottom of the pressure tube have occurred. However, CATHENA will be used to simulate the initial part of severe accident scenarios in order to provide initial conditions for a continuation of the scenario analysis using an integrated code such as MAAP.

218b. Comparisons of CATHENA models with experimental data for flow and heat transfer through ballooned fuel channels are discussed. Will CATHENA be used to evaluate ACR-700 conditions in which the fuel channel might be ballooned? If so please provide descriptions of the models to be used including the theoretical equations, user input instructions and validation document. Please include considerations for pressure drop, and heat transfer for the various element locations within the fuel bundle. Please include considerations for two-phase flow as well as single phase flow.

AECL Response
A response to this RAI will be provided by April 2005.

219. Section 3.6.3 describes validation of CATHENA for local melt heat transfer to the pressure tube (phenomenon FC15). Table 2 of report RC-2702 also lists phenomenon FC15 as one that is to be validated as part of the CATHENA Fuel-Channel Validation Plan. Section 3 of RC-2702 states that phenomenon FC15 should not have been included in the validation plan since there are no models within CATHENA to evaluate this condition. Please clarify if molten fuel heat transfer will be evaluated for ACR-700 using CATHENA. If so please provide descriptions of the models to be used including the theoretical equations, user input instructions and validation document.

AECL Response
Local melt heat transfer in ACR-700 may occur in one channel during a severe flow blockage event or a stagnation feeder break event. The consequences of local melt contact with the pressure tube will not be analyzed by CATHENA, but will be assessed based on relevant experimental information. The experimental information will be used to bound the time of pressure tube failure after melt contact occurs.

CATHENA Thermal-Mechanical Validation Manual

220. Report RC-2702 describes validation of CATHENA for 8 thermal-mechanical phenomena relevant to CANDU accident analysis. For ACR-700 analysis CATHENA Mod-3.5d will be utilized whereas the validation exercises were performed with Mods3.5b and 3.5c of CATHENA. For each of the phenomena investigated in report RC-2151, please justify that the validation work performed on the earlier mods of CATHENA are valid for the version to be used for ACR-700 safety analysis.

AECL Response
A detailed response to this question, in the form of a report documenting AECL's plan for qualification of CATHENA MOD-3.5d for ACR-700 analyses, is being prepared, and will be provided in 2005 April. This report will examine each of the 21 Fuel Channel and System Thermal Hydraulics, and 9 Fuel and Fuel Channel Thermal-Mechanical Effects phenomena listed in the Technical Basis Document relevant to analyses using CATHENA. If the validation work performed for CATHENA MOD-3.5c or planned as part of the generic validation for the MOD-3.5d code version is considered adequate for ACR-700 application, justification will be provided. Plans will be included for the

validation of any phenomena for which additional, ACR-specific validation is considered necessary.

221. Section 3.3.5 of report RC-2702 discusses pressure tube to calandria tube heat transfer in including the thermal conductance for contact between a pressure tube and the surrounding calandria tube for the condition of a sagged pressure tube. Please justify that the validation is adequate for the pressure tube/calandria tube geometry of ACR-700. Section 3.3.5 states that for the validations the contact conductance was held constant. The report recommends that validation of this model be accomplished using transient data since the contact conductance is expected to vary during an accident. Please discuss how the models in CATHENA will be validated for transient conditions.

AECL Response
As described in report RC-2702, the validation of CATHENA for this phenomenon was divided into pre-contact (i.e., conduction, convection and thermal radiation) and post-contact (i.e., direct metal-to-metal contact) heat transfer modes.

A combination of tests (two-surface-enclosure thermal-radiation numeric test, CANRAD fuel-channel separate effect tests, and CHAN thermal-chemical component tests) was used to validate the pre-contact heat transfer modes. This validation is applicable and adequate for ACR-700 analysis; because the conductive, convective and radiative heat transfer across the annulus gap between the pressure tube and calandria tube are not affected by any design differences between the ACR-700 and existing CANDUs.

Post-contact heat transfer or direct metal-to-metal contact heat transfer is modelled in CATHENA using the solid-solid contact model, either alone or in conjunction with the fuel-channel deformation model. The solid-solid contact model requires the user to supply at least the contact conductance upon which the metal-to-metal heat transfer rates are calculated. Validation of this aspect of the phenomenon was accomplished using the Contact Heat Transfer Numerical test and CANRAD-4 experimental test data. Because the annulus gap is larger, and the pressure tube is stronger, in the ACR-700 than in existing CANDUs; pressure tube sag is the only credible mode of pressure-tube-to-calandria-tube contact. In the CATHENA solid-solid contact model, this mode of heat transfer is modelled using user-supplied contact area and contact thermal conductance. Because the user is responsible for supplying appropriate values (whether constant or transient) for these input parameters, additional validation of the code for ACR-700 conditions is not required for this phenomenon. The values for a transient contact conductance may be determined from the analysis of experimental data (from tests such as ACR-700 pressure tube deformation tests currently scheduled for 2006), and these values become part of the input file for ACR-700 analysis.

222. Section 3.4 describes CATHENA validation for predicting calandria tube-to-moderator heat transfer. CHF and post dryout model validation is stated to be completed in FY 2002 to 2003. If these models are to be utilized for ACR-700 safety analysis, documentation of this

validation should be provided to the NRC staff. In Section 3.4 it is further stated that before code validation of these phenomena can proceed, the contact conductance between the pressure tube and the calandria tube must be known. The garter springs that separate ACR pressure tubes from the calandria tubes will affect the area of contact and will perhaps prevent contact in the vicinity of the garter springs. Please consider these ACR features in your validation of these phenomena.

AECL Response
The validation of CATHENA MOD-3.5c/Rev.0 for CHF and post-dryout heat transfer has been completed and is documented in Section 3.9 of the updated thermal hydraulics validation manual (RC-2701-R1). The validation exercises which are applicable to calandria-tube-to-moderator heat transfer are the McMaster horizontal cylinder CHF tests, the Whiteshell calandria tube CHF tests and the Sakurai horizontal cylinder film boiling tests.

The contact conductance is required to predict the heat transfer between a strained pressure tube in contact with the calandria tube. In the ACR-700, due to the thicker and stronger pressure tube and the larger annulus gap between the pressure tube and calandria tube (compared with existing CANDUs), pressure tube sag is the only credible mode of pressure tube to calandria tube contact. For this case, tests with the CANDU 6 fuel channel geometry have shown that the garter-spring affected zone is small in comparison to the length of the fuel channel and therefore a reduction to contact conductance near the garter spring will have little effect. The pressure tube cannot maintain a hot spot when adequate cooling is provided on either side of the garter spring (assuming adequate moderator subcooling exists to maintain nucleate boiling on the outside surface of the calandria tube). Confirmatory tests for an ACR-700 fuel channel are scheduled for 2006.

References:
1. *RC-2701-R1, "CATHENA MOD-3.5c/Rev.0 Systems Thermalhydraulic Validation Manual", edited by W. Won, X.M. Huang and G.M. Waddington, 2003 September.*

223. Section 3.6 describes CATHENA validation for calandria tube deformation and failure. Two sets of data are described: one utilizing molten zircalloy-4 in contact with the calandria tube and the other involving heating the simulated fuel channel tube until it came in contact with the calandria tube. Please justify that these tests appropriately describe the ACR configuration with a thicker fuel channel and garter springs separating the calandria tube from the fuel channel. Section 3.6.5 indicates that the pressure range for the tests may not be adequate to cover reactor conditions. Please justify that both the pressure and temperature ranges of the validation tests are adequate for the conditions predicted in ACR-700 safety analysis.

AECL Response
A response to this RAI will be provided by April 2005.

Additional Requests

261. CATHENA simulation of RD-14M experiment B9401 is described in report 108US-03532-225-001. The staff notes that for code simulation of the test, the inlet and exit headers are described using 4 fluid nodes. The reactor inlet and outlet headers of ACR-700 are modeled in the current CATHENA input description as single nodes. The headers are 11 meters long and have connections all along the lengths so that use of a single node model may not be valid. Will the headers be modeled differently in the CATHENA input description used for DCD analysis? Please provide validation for the header model to be used for the ACR-700 DCD by comparison with experimental data from a facility that is properly scaled for ACR-700.

AECL Response
A response to this RAI will be provided by April 2005.

262. Section 5.3.3.3 of the CATHENA theory manual COG-00-008 states that changes in pressure tube geometry (ballooning) is not included in thermal/hydraulic calculation (i.e., flow area or hydraulic diameter) or heat transfer calculations. The thermal radiation view factor matrix changes that would result from ballooning are also not included in the calculations. Are these effects important to analyses to be performed for ACR-700? If so, please discuss how these effects will be evaluated for ACR-700 safety analysis and how the models used in these calculations will be validated.

AECL Response
A response to this RAI will be provided by April 2005.

263. Will fuel element sagging occur for any of the accidents to be evaluated for the ACR-700 DCD? If so, please describe how will the degree of sagging be evaluated. If sagging is calculated to occur please discuss how the perturbations on channel flow and heat transfer will be evaluated in the safety analyses since these effects are not modeled in CATHENA.

AECL Response
A response to this RAI will be provided by April 2005.

264. For ACR-700 it has been postulated that following a large LOCA caused by a header break that voids will form in alternate channels at the core face closest to the break in a checkerboard fashion. Furthermore for an inlet header break the fuel bundles affected first will be the freshest bundles which were loaded last. The checker board effect may produce a different reactivity feedback than if the voiding were uniform across the core. Please discuss the importance in accurate prediction of local channel voiding on reactor power for the period before reactor trip. If this effect is determined to be significant, then provide validation of CATHENA for local void prediction.

AECL Response
A response to this RAI will be provided by April 2005.

AECL
TECHNOLOGIES INC.

Attachment 2
List of Enclosures and References in AECL's Responses to NRC's Request for Additional Information on CATHENA

(Letter G. Archinoff to B. Sosa, "Response to the NRC's Requests for Additional Information (RAIs) on the CATHENA Computer Code", December 24, 2004)

Table 1: List of Enclosures

Enclosure	*File Name*	*RAI #*
COG-01-200 – CATHENA Input Manual MOD3.5d Rev0	*COG-01-200 - Cathena Input Manual MOD35d Rev0.pdf**	*135*
COG-01-209 – GENHTP Input Manual MOD3.5d Rev0	*COG-01-209 - GENHTP Input Manual MOD35d Rev0.pdf**	*135, 187b, 195a*
P. Revelis, S. Pereira, N.U. Aydemir, B.N. Hanna, "A Model for Level Swell in Horizontal Pipes", Atomic Energy of Canada Limited Report, COG-92-410	*COG-92-410.pdf*	*145a*
M.E. Lavack, "Level Swell Implementation in CATHENA MOD-3.5b/Rev 0", Atomic Energy of Canada Limited Report, RC-1640 (1996)	*RC-1640.pdf*	*145a*
T.V. Sanderson and D.J. Wallace, "Validation of CATHENA MOD-3.5c for Coolant Voiding - RD-14M Critical Break Test B9902", Atomic Energy of Canada Ltd Report, RC-2341 (2000).	*RC-2341.pdf*	*149*
"MATRIX-1.05 A Stand-Alone Preprocessor Utility for CATHENA Users", J.B. Hedley, Atomic Energy of Canada Limited Report, COG-99-232.	*COG-99-232.pdf*	*157*
Steam-bubble-application-Appendix.pdf	*Steam-bubble-application-Appendix.pdf*	*161*
RC-2810, "Validation of CATHENA MOD-3.5c for Channel Coolant Voiding – RD-14M Large LOCA Experiments (B0104-B0109) and Power-Pulse LOCA Experiments (B0113, B0115, B0116 and B0117)", by D.F. Wang and T.V. Sanderson, 2002 June.	*RC-2810.pdf*	*187a, 187c, 187f*
W. Won, X.M. Huang and G.M. Waddington, "CATHENA MOD-3.5c/Rev 0 Systems Thermalhydraulic Validation Manual", RC-2701 Revision 1, September 2003.	*RC-2701_(R1).pdf*	*187f, 194, 204, 205*

* *Please note that the properties of the available version of this document does not allow full compliance with the requirements for submission of electronic documentation.*

Enclosure	*File Name*	*RAI #*
J.A. Keith Reid, T.V. Sanderson and J.P. Mallory, "Validation of CATHENA MOD-3.5c for Coolant Voiding - Overview Report", Atomic Energy of Canada Ltd Report, RC-2332 (2000).	*RC-2332.pdf*	*187a, 187b, 187c*
RC-2846-5, "Validation of CATHENA MOD-3.5c/Rev 0 for Nucleate Boiling — Overview Report", by G.M. Waddington and S.M. Froebe, 2003 August.	*RC-2846-5_(R0).pdf*	*187b, 187c, 194*
RC-2849-5, "Validation of CATHENA MOD-3.5c/Rev.0 for Flow Oscillations — Overview Report", by D.F. Wang and W. Won, 2002 September.	*RC-2849-5_(R0).pdf*	*204*
RC-2777-5, "Validation of CATHENA MOD-3.5c/Rev.0 for Natural Circulation — Overview Report", by X.M. Huang, 2003 March.	*RC-2777-5_(R0).pdf*	*205*

Table 2: List of References

RAI #	*Reference*
140	*1. C.W. Snoek and L.K.H. Leung, "A Model for Predicting Diabatic Pressure Drops in Multi-Element Fuel Channels", Nuclear Engineering and Design, 110 (1989) 299-312*
141	*1. Kowalski, J.E., 1987, "Wall and interfacial shear stress in stratified flow in a horizontal pipe", AIChE Journal Vol. 33, (2), 274–281.* *2. Ransom, V.H. et al., 1982, "RELAP5/MOD1 code manual Volume 1: system models and numerical methods", NUREG/CR-1826, EGG-2070.* *3. Rohatgi, U.S., J. Jo, and L. Neymotin, 1982, "Constitutive relations in TRAC-PD2", NUREG/CR-3073.*
142	*1. Edwards, A.R. and O'Brian, T.P., "Studies of Phenomena Connected With the Depressurization of Water Reactors", UKAEA Report, Journal of the British Nuclear Society, Volume 9 (1970)* *2. X.M. Huang and J.P. Mallory, "Validation of CATHENA MOD-3.5c for Break Discharge Characteristics - Overview Report", Atomic Energy of Canada Ltd, RC-2339 (2000).*
143	*1. R. Kouyoumdjian and J.P. Mallory, 2001, "Validation of CATHENA MOD-3.5c/Rev 0 for Countercurrent Flow – Overview Report", Atomic Energy of Canada Ltd Report, RC-2475*
145f	*1. "CATHENA MOD-3.5c/Rev 0 Systems Thermalhydraulic Validation Manual", RC-2701*
147	*1. "CATHENA MOD-3.5c/Rev 0 Systems Thermalhydraulic Validation Manual", RC-2701* *2. "CATHENA MOD-3.5c/Rev 0 Theoretical Manual", COG-00-008*
149	*1. T.V. Sanderson and D.J. Wallace, "Validation of CATHENA MOD-3.5c for Coolant Voiding - RD-14M Critical Break Test B9902", Atomic Energy of Canada Ltd Report, RC-2341 (2000).* *2. J.E. Kowalski and X.M. Huang, "Validation of CATHENA MOD-3.5c for Quench/Rewet Characteristics - CWIT Channel/Feeder Refill Tests", Atomic Energy of Canada Ltd Report, RC-2466 (2001).*

RAI #	*Reference*
165b	*1. Mac Gregor, R.K. and Emery, A.P., "Free Convection Through vertical Plane Layers: Moderate and high Prandtl number Fluids", J. Heat Transfer, Vol. 91, p.391 (1969)* *2. Hollands, K.G.T., Raithby, G.D. and Konicek, L., "Correlation Equations for Free Convection Heat Transfer in Horizontal Layers of Air and Water", Int. J. Heat and Mass Transfer, Vol. 18, p.879 (1975)* *3. "CATHENA MOD-3.5c/Rev 0 Theoretical Manual", COG-00-008*
166	*1. Leung, L.H.K., S.K. Yang, Y.J. Guo and W.W.R. Inch, "A Look-up Table of Critical Heat Flux for the CANFLEX Mk-IV in Crept and Uncrept Channels", Atomic Energy of Canada Limited Report, CANFLEX-161, FFC-FCT-383, 2001.* *2. "CATHENA MOD-3.5c/Rev 0 Theoretical Manual", COG-00-008*
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