May 14, 2004

Mr. Vince Langman
ACR Licensing Manager
Atomic Energy of Canada Limited (AECL) Technology, Inc.
481 North Frederick Avenue, Suite 405
Gaithersburg, Maryland 20877

SUBJECT: REQUESTS FOR ADDITIONAL INFORMATION - CATHENA CODE FOR ACR-700 APPLICATION

Dear Mr. Langman:

Atomic Energy of Canada Limited (AECL) submitted a formal request for a pre-application review of the Advanced CANDU Reactor (ACR-700) design on June 19, 2002.

The Nuclear Regulatory Commission (NRC) staff is reviewing technical information provided by AECL as part of the on-going pre-application review activities for the ACR-700 design. The NRC staff requests that AECL provide evaluation models for the various uses of CATHENA for the ACR-700 analysis in accordance with DG-1120 as there are many code options that will affect the analytical results. The requests for additional information (RAIs) are included in the enclosure. An advanced copy of the RAIs were sent to you via electronic mail on March 19, 2004. On May 3, 2004, AECL participated in a teleconference with the staff to clarify the content of the RAIs. Since the responses to these RAIs do not impact the preparation of Pre-Application Safety Assessments Report (PASAR), AECL agreed to provide the ACR-700 information requested in the RAIs prior to the design certification application submission.
If you have any questions or comments concerning this matter, you may contact the undersigned at (301) 415-4125 or jsk@nrc.gov.

Sincerely,

/RA/

James Kim, Project Manager
New Reactors Section
New, Research and Test Reactors Program
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Project No. 722

Enclosure: As stated

cc: See next page

May 14, 2004

Mr. Vince Langman
ACR Licensing Manager
Atomic Energy of Canada Limited (AECL) Technology, Inc.
481 North Frederick Avenue, Suite 405
Gaithersburg, Maryland 20877

SUBJECT: REQUESTS FOR ADDITIONAL INFORMATION - CATHENA CODE FOR ACR-700 APPLICATION

Dear Mr. Langman:

Atomic Energy of Canada Limited (AECL) submitted a formal request for a pre-application review of the Advanced CANDU Reactor (ACR-700) design on June 19, 2002.

The Nuclear Regulatory Commission (NRC) staff is reviewing technical information provided by AECL as part of the on-going pre-application review activities for the ACR-700 design. The NRC staff requests that AECL provide evaluation models for the various uses of CATHENA for the ACR-700 analysis in accordance with DG-1120 as there are many code options that will affect the analytical results. The requests for additional information (RAIs) are included in the enclosure. An advanced copy of the RAIs were sent to you via electronic mail on March 19, 2004. On May 3, 2004, AECL participated in a teleconference with the staff to clarify the content of the RAIs. Since the responses to these RAIs do not impact the preparation of Pre-Application Safety Assessments Report (PASAR), AECL agreed to provide the ACR-700 information requested in the RAIs prior to the design certification application submission.
If you have any questions or comments concerning this matter, you may contact the undersigned at (301) 415-4125 or jsk@nrc.gov.

Sincerely,

/RA/

James Kim, Project Manager
New Reactors Section
New, Research and Test Reactors Program
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Project No. 722

Enclosure: As stated

cc: See next page

ADAMS ACCESSION NUMBER: ML041040166

OFFICE	PM:RNRP	NRR:SRXB	RNRP:SC
NAME	JKim	WJensen	LDudes-jms for:
DATE	5/5/04	5/11/04	5/13/04

OFFICIAL RECORD COPY

Distribution for Requests For Additional Information dated May 14, 2004

Hard Copy
RNRP R/F
JKim
BSosa
LDudes
WJensen

E-Mail:
PUBLIC
AAttard
FAkstulewicz
LWard
PClifford
SMiranda
SSun
DBessette

REQUEST FOR ADDITIONAL INFORMATION - LETTER 6
ACR-700 Pre-Application Review - CATHENA Code for ACR-700 Application

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.

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 the emergency core cooling system (ECCS) analysis, five specific criteria described in 10 CFR 50.46 must be met for the loss-of-coolant accident (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 basis as well as the regulatory basis for acceptance.

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 the ACR-700. If the NUREG-0800 acceptance criteria will not be met, please provide the criteria that will be used and provide the technical basis 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 basis as well as the regulatory basis for acceptance of analyses for these events.

134. Appendix B to 10 CFR Part 50 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.

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

Requirements for Code Capability

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.

Methodology

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

For LOCA analyses, 10 CFR 50.46 provides the option of using one of two acceptable approaches. The first acceptable method is described in Appendix K to 10 CFR Part 50. 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 the ACR-700 and discuss when the uncertainty analyses and supporting material required by 10 CFR 50.46 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 10 CFR 50.46, please provide the technical as well as the regulatory basis for acceptance of this methodology.

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 design control document (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.

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.

Scaling Reports

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

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. AECL has provided CATHENA validation reports for Mod-3.5c of the code. We understand that additional code validations have and will be performed 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 the NRC to identify significant thermal/hydraulic phenomena for the 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 commiserate with their ranked importance.

Uncertainty Analysis Reports

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

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 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 break 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.

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 super-heated fluid conditions that might occur at ACR-700.

138. Experience in application of RELAP5 to the N-Reactor showed that when emergency core cooling (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..

139. Prediction of limiting conditions for countercurrent flow of steam and water is significant for the ACR-700 since following a LOCA, 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 counter-current 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.

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.

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 the ACR-700.

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 the ACR-700. Are there circumstances when a sudden depressurization is analyzed when the rigorous treatment of superheated water might affect the result?

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?

144. Section 4.8 describes the use of empirical spacial dependant velocity and void fraction coefficients in CATHENA over the cross section of a conduct. 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.

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:

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

  b. The discussions in Section 4.9 appear to relate to rectangular flow geometries. Justify that the model is adequate for determining the two-phase level within circular fuel channels containing ACR-CANFLEX fuel bundles.

  c. Verification 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 the comparison of CATHENA results with level swell data from large vertical tanks. Justify that the level swell model has been adequately verified for level swell within circular fuel channels containing CANFLEX fuel bundles.

  d. 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.

  e. 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 the ACR-700.

  f. What provision is made for accounting for level swell in vertical stacks of CATHENA nodes such the modeling of the ACR-700 steam generators. How is the layering of a two phase mixture and pure steam in vertical stacks containing multiple CATHENA nodes prevented?

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 phenomena? Compare the orientation of the feeder tube to the fuel channel of the test facility to those of ACR-700.

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.

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.

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?

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?

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 the 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?

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 the ACR-700 analysis discuss how the conductance is determined for inputting into the code.

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

   a. 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.

   b. 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 is not analytically valid and a statement that the effect of pressure tube ballooning on thermal/hydraulics or heat transfer is not included. Justify that it is appropriate to use these models for the ACR-700 safety analysis.

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 the ACR-700. Identify in each case whether the transient or accident is part of the design basis or beyond design basis.

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 is stated to be user input or the point reactor kinetics model. The point kinetics model is described in Section 7.1.5.6. For analysis of the ACR-700 please provide the following information:

   a. We understand that CATHENA has the ability of being 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 the ACR-700 will be analyzed using point-kinetics and which will be analyzed using the more detailed methodology.

   b. 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.

   c. 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 10 CFR Part 50, Appendix K be met concerning decay heat? If you propose to use the 1979 or the 1994 American

Nuclear Society (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.

d. 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 the ACR-700.

e. 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 10 CFR Part 50. Provide this information for the limiting design basis LOCA analyzed for the ACR-700.

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.

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.

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.

<u>Chapter 6. Numerical Methods</u>

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.

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 critical heat flux (CHF) earlier than that for the one-dimensional calculation. Please address the applicability of the channel average CHF approach to capture 3-D effects.

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:

   a. Comparisons with experimental data are referred to first for determining the X_L 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 the ACR-700 fuel channels.

   b. 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 T_{sat} and T_{max}. The determination of T_{max} is not clear. Please describe how T_{max} is determined. How has the T_{max} model been verified to be accurate?

   c. 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.

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 the 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.

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.

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

b. 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.

c. 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.

d. 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.

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.

a. 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 the ACR-700.

b. 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 the ACR-700 safety analysis of various postulated break sizes.

c. 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 verification using the facility specific model is valid for the ACR-700.

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.

a. 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.

b. 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.

166. The Groeneveld table lookup CHF (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 the ACR-700 safety analysis.

167. The Groeneveld table lookup CHF (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 8mm in diameter. Correction factors are provided pipes of larger diameter and for non-vertical orientation. Please describe the basis and verification of the correction factors and justify they produce results that are conservative for the ACR-700 safety analysis.

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.

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.

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.

171. Section 7.9 describes the “Fisher Valve Model.” Please discuss the use of Fisher valves in the 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 the ACR-700.

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 of 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.

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.

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

    a. 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 Standard Review Plan (SRP) 6.2.1.1.A.

    b. 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.

    c. Will the GTM be utilized to determine net positive suction head (NPSH) for safety-related equipment following an accident? If so, please demonstrate that the analysis meets the requirements of NRC Generic Letter (GL) 97-04.

    d. We understand that the GTM will be utilized to model the pressurizer for the 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.

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

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 the ACR-700. The staff would like to use point kinetics in audit calculations using RELAP5. Please provide the following data for the 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.

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 the ACR-700 will be determined and utilized in a conservative manner for safety analysis.

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.

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

    b. In the United States, 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.

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?

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?

181. CATHENA horizontal connector separation models are described in Section 7.17.2. These models provide for calculation of 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.

    a. 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.

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

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 that its use is appropriate for the ACR-700 safety analysis.

183. Loss coefficient correlations are available for CANDU breakdown orifices used in two sizes of fuel channel feeder pipes (1½ inch and 2-inch). Please identify which of these two feeder pipe sizes is utilized in the ACR-700.

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.

a. 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.

b. 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.

c. 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.

CATHENA Validation Manual RC-2701

185. Report RC-2701 describes validation of CATHENA for 23 thermal/hydraulic phenomena relevant to CANDU accident analysis. For the ACR-700 analysis CATHENA Mod-3.5d will be utilized whereas the validation exercises were performed with Mods 3.5b 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 the ACR-700 safety analysis. Compare thermal/hydraulic conditions measured in the tests with those expected in the ACR-700 under accident conditions.

<u>Chapter 3.1 Break Flow Models</u>

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.

a. 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.

b. 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 the ACR-700 analysis. Has this error been corrected?

c. 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 the ACR-700 analysis are included.

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.

a. 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 correlations and how these results were utilized in development of the CATHENA model for the ACR-700.

b. In correlation of 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 basis for the heat transfer splitting between the phases that will be utilized in the ACR-700 CATHENA model of the reactor fuel channels.

c. 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 the ACR-700.

d. 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.

e. 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?

f. Section 3.2.5 states that "none of 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 the ACR-700.

g. 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.

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.

<u>Chapter 3.3 Phase Separation</u>

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

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.

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 the ACR-700. Compare the specific speeds for the pumps used in the tests to those of the ACR-700.

192. Page 49 of RC-2701 describes significant discrepancies in simulating pump characteristics in the transition from single-phase to highly voided two-phase 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.

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.

a. For each test facility provide a comparison of the CATHENA model that was used to predict the test data and that which will be utilized to analyze the 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 the ACR-700 analysis.

b. 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 the ACR-700 CANFLEX fuel.

c. Convective heat transfer to steam tests at the CHAN facility were used to validate CATHENA for these conditions. Above 700°C 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°C needed to be utilized to validate the code. Please provide these data comparisons.

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.

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.

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

b. 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 the ACR-700 safety analysis.

c. Data from a simulated 37-element CANDU fuel bundle tests was correlated. Please provide correlations by CATHENA with data that models ACR-700 CANFLEX fuel. Provide uncertainty analyses so that the margin to CHF for the 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 SRP 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.

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

e. 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.

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.

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 correlation of 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 the 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.

198. Following a LOCA signal high pressure ECCS water will be injected into the inlet headers of the 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.

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 the 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.

a. 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.

b. 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 the ACR-700 analysis.

c. 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 LOCAs 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.

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

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. Correlation of 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 the ACR-700 safety analysis.

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 underpredict fuel cladding oxidization for both of these test series. Please justify the conservatism of models in CATHENA to be used for the ACR-700 safety analysis in light of these results.

Chapter 3.14 Reflux Condensation

202. Section 3.14.5 and Table 4 of RC-2701 identifies deficiencies in the verification 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.

Chapter 3.15 Counter Current Flow

203. Prediction of limiting conditions for countercurrent flow of steam and water is significant for the ACR-700 since following a LOCA, 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.

a. Countercurrent air/water tests were conducted at Dartmouth. When this test data was correlated by CATHENA, CATHENA overpredicted 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 the ACR-700.

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

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

d. 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 the ACR-700. Justify that this data is appropriately scaled for accident conditions at the ACR-700.

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.

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.

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 correlate the temperature vs time data for the simulated fuel pins.

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.

Chapter 3.20 Steam Condensation Induced Waterhammer

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 FY-2002/2003. If CATHENA is to be used to evaluate steam induced waterhammer for the ACR-700 safety analysis please provide this validation.

209. We understand that the water used in the emergency coolant injection (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.

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 FY-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 probabilistic risk assessment (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.

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 channel during a LOCA and provide validation that CATHENA can adequately describe phenomena involving the dissolved gas.

<u>Other Validation Issues</u>

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

   a. 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 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.

   b. Will analyses be performed for the 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.

   c. 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 (LPI) 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.

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 the 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.

214. The RD-14M facility contains 10 channels in five 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.

215. For stagnation header breaks, ACR-700 fuel sheath temperatures are predicted to increase early in the transient until 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.

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 the ACR-700.

217. If CATHENA is to be used to model anticipated plant transients such as are described in Chapter 15 of SRP 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.

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 the ACR-700, please provide the following information if the phenomena are to be assessed using CATHENA.

a. 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.

b. Comparisons of CATHENA models with experimental data for flow and heat transfer through ballooned fuel channels is 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.

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 model this condition. Please clarify if molten fuel heat transfer will be evaluated for the 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.

CATHENA Thermal-Mechanical Validation Manual

220. Report RC-2702 describes validation of CATHENA for eight thermal-mechanical phenomena relevant to CANDU accident analysis. For the 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 the ACR-700 safety analysis.

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 verification is adequate for the pressure tube/calandria tube geometry of the ACR-700. Section 3.3.5 states that for the verifications, 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 verified for transient conditions.

222. Section 3.4 describes CATHENA validation for predicting calandria tube-to-moderator heat transfer. CHF and post-dry-out model verification is stated to be completed in FY-2002 to 2003. If these models are to be utilized for the 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.

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.

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.

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.

263. Will fuel element sagging occur for any of the accidents to be evaluated for the ACR-700 DCD? If so, please describe how the degree of sagging will 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.

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.

ACR-700

cc:
Mr. Charles Brinkman
Westinghouse Electric Co.
Washington Operations
12300 Twinbrook Parkway, Suite 330
Rockville, MD 20852

Mr. Thomas P. Miller
U.S. Department of Energy
NE-20, Rm. A286
Headquarters - Germantown
19901 Germantown Road
Germantown, MD 20874-1290

Mr. David Lochbaum
Nuclear Safety Engineer
Union of Concerned Scientists
1707 H Street, NW, Suite 600
Washington, DC 20006-3919

Mr. Paul Gunter
Nuclear Information & Resource Service
1424 16th Street, NW, Suite 404
Washington, DC 20036

Mr. James Riccio
Greenpeace
702 H Street, NW, Suite 300
Washington, DC 20001

Mr. Ron Simard
Nuclear Energy Institute
Suite 400
1776 I Street, NW
Washington, DC 20006-3708

Ms. Patricia Campbell
Winston & Strawn
1400 L Street, NW
Washington, DC 20005

Mr. Paul Leventhal
Nuclear Control Institute
1000 Connecticut Avenue, NW
Suite 410
Washington, DC 20036

Mr. Jack W. Roe
SCIENTECH, INC.
910 Clopper Road
Gaithersburg, MD 20878

Mr. David Ritter
Research Associate on Nuclear Energy
Public Citizens Critical Mass Energy
and Environmental Program
215 Pennsylvania Avenue, SE
Washington, DC 20003

Mr. James F. Mallay, Director
Regulatory Affairs
FRAMATOME, ANP
3315 Old Forest Road
Lynchburg, VA. 24501

Mr. Tom Clements
6703 Gude Avenue
Takoma Park, MD 20912

Mr. Vince Langman
Licensing Manager
Atomic Energy of Canada Limited
2251 Speakman Drive
Mississauga, Ontario
Canada L5K 1B2

Mr. Victor G. Snell
Director of Safety and Licensing
Atomic Energy of Canada Limited
2251 Speakman Drive
Mississauga, Ontario
Canada L5K 1B2

Mr. Glenn R. George
PA Consulting Group
130 Potter Street
Haddonfield, NJ 08033

J. Alan Beard
GE Nuclear Energy
13113 Chestnut Oak Drive
Darnestown, MD 20878-3554

Mr. Ian M. Grant
Canadian Nuclear Safety Commission
280 Slater Street, Station B
P.O. Box 1046
Ottawa, Ontario
K1P 5S9

Mr. Gary Wright, Manager
Office of Nuclear Facility Safety
Illinois Department of Nuclear Safety
1035 Outer Park Drive
Springfield, IL 62704

Dr. Gail H. Marcus
U.S. Department of Energy
Room 5A-143
1000 Independence Ave., SW
Washington, DC 20585

Mr. Ronald P. Vijuk
Manager of Passive Plant Engineering
AP1000 Project
Westinghouse Electric Company
P. O. Box 355
Pittsburgh, PA 15230-0355

Dr. Greg Rzentkowski
Canadian Nuclear Safety Commission
P.O. Box 1046, Station 'B'
280 Slater Street,
Ottawa, ON, K1P 5S9
Canada

Mr. Ed Wallace, General Manager
Projects
PBMR Pty LTD
PO Box 9396
Centurion 0046
Republic of South Africa

Mr. John Polcyn, President
AECL Technologies Inc.
481 North Frederick Avenue
Suite 405
Gaithersburg, MD 20877

Mr. Russell Bell
Nuclear Energy Institute
Suite 400
1776 I Street, NW
Washington, DC 20006-3708