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

Attention: Ms. A. Cubbage
Project Manager, ACR

Reference:
1. Letter W.D. Beckner to J. Polcyn, "Pre-application Safety Assessment Report for the Advanced CANDU Reactor (ACR-700)", October 27, 2004.

Re: AECL's Responses to PASAR Technical Comments on Reactor Physics

AECL's responses to technical issues on reactor physics issues raised in the Pre-Application Safety Assessment Report (Reference 1) are attached. We have extracted relevant statements from the PASAR and identified the section and page number for each statement. Each issue is followed by our response.

The attached PASAR responses are provided for information. We are not requesting comments on the PASAR responses. We will arrange a meeting with NRC physics specialists in the next two months to provide an update on progress in the reactor physics area.

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

/Attachments:
1. AECL's Responses to Reactor Physics Issues Identified in Section 3.5 and 8.3 of the NRC's Pre-Application Safety Assessment Report for the ACR-700

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ATTACHMENT 1
(Letter G. Archinoff to A. Cubbage, "AECL's Responses to PASAR Technical Comments on Reactor Physics", March 24, 2005)

AECL's Responses to Reactor Physics Issues Identified in Section 3.5 and 8.3 of the NRC's Pre-Application Safety Assessment Report for the ACR-700

This document provides AECL's responses to specific issues identified in Section 3.5 (Focus Topic 3 – Computer Codes and Validation Adequacy, Evaluation of the RFSP-IST, WIMS-IST, and DRAGON-IST Reactor Physics Codes) and Section 8.3 (Focus Topic 9 – Confirmation of Negative Void Reactivity, Technical Issues) of the NRC's Pre-Application Safety Assessment Report (PASAR) for the ACR-700 [1]. We have extracted specific issues from the text of the PASAR and reproduced them below. We provide our response in italic fonts following each of the PASAR issues.

1. **Section 3.5.1.3, page 3-6**

   Furthermore, RFSP incorporates a simpler cell code (POWDERPUFS-V), which is based on a "1.5 group representation." This code is only used for rough estimates and "tuned" to specific reactor and core designs. It is not expected that this simple cell code will be applied to ACR-700.

   AECL Response

   The lattice code POWDERPUFS-V is indeed a module of RFSP. However, POWDERPUFS-V will not be used for ACR design or safety analysis.

2. **Section 3.5.2.1.2, page 3-11**

   Finally, a burnable poison in the form of Dysprosium (which is different from its use as a fission product simulant) in the central element will have an as yet unknown spectral effect on the temperature coefficient. A comprehensive set of experiments should be carried out to validate the code set. This series should include fresh, partially burned, and end-of-life fuel compositions.

   AECL Response

   Experiments are planned to measure fine-structure reaction rates using demountable bundles representing fresh fuel and irradiated fuel. The irradiated fuel composition has been chosen to represent the latter half of its irradiation history and from a CVR perspective will be closely representative of end-of-life burnup. These tests are described in Section 3.3.4 of Reference 2.

3. **Section 3.5.2.1.3, page 3-11**

   AECL carried out critical experiments on the ZED-2 and the Zero Energy Experimental Pile (ZEEP) facilities. The temperature range being considered for the moderator was 30 °C (77 °F) to 100 °C (212 °F), although the highest temperature could not be reached. A shortcoming of the experiments was that the temperature applied to the entire lattice, including the fuel bundles. The contribution from heating the fuel is reported to be approximately 25 percent of the overall coefficient in conventional CANDUs. Finally, these measurements were carried out only for fresh fuel. Thus, no effects due to plutonium or fission products could be measured. The measured quantity is again the critical reactor height and the corresponding critical buckling (see Douglas, et al., November 2001).

AECL Response

The current procedure for moderator temperature experiments is to isolate the channel contents (i.e., fuel and coolant) from the moderator, which is confirmed by measuring the temperature in a representative sample of the channels using thermocouples positioned at the top of the fuel string. For elevated moderator temperatures we typically see differences of 10-15 degrees Celsius between the channel contents and moderator temperature. Moderator-temperature experiments have been performed using substitution lattices containing a few rods of plutonium-bearing MOX. Additional tests for ACR fuel will be performed and are described in Section 3.3.2 of Reference 2. In addition, the high-temperature MOX tests described in Section 3.3.3.3 of Reference 2 will include measurements of the moderator-temperature coefficient over a small range of moderator-temperature variation.

4. **Section 3.5.2.1.3, page 3-12**

The above experimental estimates are based on fresh fuel. No data are available for the cases with either MOX fuel or fission-product-containing fuel. AECL will have to address these cases for the ACR-700 validation. In addition, the experimental data developed for the above validation will have to be repeated for ACR-700 conditions.

AECL Response

Please see the response to Item 3 regarding the use of simulated MOX fuel to account for burnup effects.

5. **Section 3.5.2.1.4, page 3-13**

The determination of moderator purity was carried out on the ZED-2 critical facility. For each different value of moderator composition, the critical height and corresponding buckling were measured. This resulted in a variation of buckling with weight percent of heavy water in the moderator. The WIMS-IST computed variation of buckling with moderator purity resulted in a prediction that was approximately 8 percent higher than that measured, with a standard deviation of ±2.5 percent. Reported comparisons to Monte Carlo calculations indicate that the MCNP predictions are slightly lower, but not enough to close the gap. This indicates that the scattering kernels in the nuclear data might need re-evaluation.

AECL Response

We will account for the discrepancies between calculation and measurement with bias and uncertainty terms in the calculation. Additional experiments are planned to reduce the uncertainty term. We also have an ongoing program to keep our nuclear data libraries up to date with the state of the art, and we look forward to any improvements in nuclear data that reduce the bias term in our calculations.

We are planning on doing additional moderator-purity experiments for ACR-type lattices that will be dedicated to determining the effect of moderator purity on lattice reactivity and CVR. When the current moderator charge has to be upgraded we will do two sets of experiments. Before the upgrade we will perform full-core flux-map experiments to determine the CVR with downgraded moderator. We will then upgrade the moderator and repeat the flux maps using exactly the same lattice. These tests are described in Section 3.3.6 of Reference 2.

6. **Section 3.5.2.1.4, page 3-13**

If the ACR-700 is controlled in the same manner as the current CANDU reactors, the above experiments will have to be repeated for the fuel composition, channel pitch, coolant, and expected poison variations characteristic of its design. The concerns regarding the moderator scattering kernels would still apply in this case.

AECL Response

Experiments are included in the revised work plan that will be dedicated to measuring the effect of moderator purity on lattice reactivity and CVR. Full-core flux maps will be performed to measure CVR in an ACR-type lattice with downgraded moderator. The moderator will be upgraded and the flux-map experiments will be repeated. These experiments will be analysed using MCNP to determine if there are any deficiencies in the heavy-water kernels. These tests are described in Section 3.3.6 of Reference 2.

7. **Section 3.5.2.1.5, page 3-14**

Predictions of multiplication factor variation with temperature, using the appropriate measured buckling, are all slightly below unity, except for the LVRF cases, but they are essentially constant. The flatness of the curve indicates that the feedback coefficient can be clearly defined. The predicted and measured slopes of the critical buckling versus temperature curve are the feedback coefficients in buckling units. It was reported that these two values agree, for all measurements, to within 10 percent with a confidence level of 95 percent. The error results in an over-prediction of the fuel coefficient. Reported results from code-to-code comparisons indicate that the resonance treatment in the newres option and the currently used nuclear data library for Pu-239 (and possibly for U-238) needs to be re-evaluated.

AECL Response

Improvements have been made to the WIMS version being developed for ACR application and an ACR-specific library has been generated. In particular, the "newres" resonance treatment methodology has been revamped and significantly improved. The approach used in testing the improvements to the WIMS developmental version has been through code-to-code comparison with the Monte Carlo code MCNP. A detailed description of the improved resonance treatment will be included in the WIMS Theory Manual, which will be updated as part of the standard software quality assurance process. The relevant experiments are described in Section 3.3.4.1 and 3.3.4.2 of Reference 2.

8. **Section 3.5.2.1.7, page 3-17**

Because the ACR-700 will represent a departure in fuel and code design from currently operating CANDU fuel, a very comprehensive set of experiments needs to be carried out to validate the methods and fundamental nuclear data. The increased burnup will generate a larger inventory of fission products, which will also affect the fuel temperature feedback coefficient. It might be necessary to reevaluate fundamental cross-section data for selected nuclides. In addition, an explicit scattering kernel for UO2 and PuO2 might have to be developed to accurately account for the potential up-scattering caused by the oxygen in fuel elements containing MOX fuel. Such data already exist for UO2 and may only need augmentation.

AECL Response

A new ACR-specific library has been developed for both WIMS and MCNP, to account for the effects mentioned in the comment. Our experiments tend to be integral measurements, such as the high-temperature MOX tests described in Section 3.3.3.3 of Reference 2, and there will be multiple analyses using WIMS, WIMS/RFSP, MCNP, etc. The response to item 14 addresses the specific comment on oxygen, and notes that we intend to add cross sections for O bound in UO_2 *to the WIMS and MCNP ACR data libraries.*

9. **Section 3.5.2.1.7, page 3-17**

In view of the significant departure from standard CANDU configurations, it is important to revisit all the assumptions made in creating the code/library package during the certification review. In particular, the following might be important:

- The WIMS-IST will have to be validated for the new cell arrangement, as outlined in Sections 3.2.1.1 and 3.3.2.1.8 of this report.
- The RFSP code will have to be validated for acceptable accuracy in those regions where diffusion theory is known to break down. Because the core is smaller, the reflector effects caused by spatial flux gradients may be more significant.
- Spectral changes around the periphery of the core may be more significant, particularly since this core is intended to have a negative value for the CVR. This effect might necessitate rethinking the number of groups needed for RFSP.

AECL Response

Experiments have been added to the revised ZED-2 experimental plan (Reference 2) to provide validation data that explicitly measure neutronic interaction between an ACR-type core and the reflector interface (refer to Section 3.3.4.1.3 of Reference 2). A multicell capability is being developed for WIMS-AECL to improve the modelling between a lattice cell and its surrounding environment (refer to Section 4.2.1.2 of Reference 3). Local regions in the core, where environment effects on lattice-cell properties are important, will be treated with multicell models. WIMS/RFSP validation will include comparisons against the full ZED-2 core configurations (refer to Section 5.7.2 of Reference 3). The accuracy of WIMS/RFSP full-core calculations is also being assessed through comparisons to MCNP. The adequacy of a 2-group representation has been assessed by comparisons to multigroup DONJON calculations.

10. Section 3.5.2.1.8, page 3-18

In the current CANDU reactor designs, the analysis is generally carried out at the RFSP step of the analysis. This implies that the few group cross sections have already been generated (using WIMS) and the appropriate cross sections (voided vs. cooled) are entered into RFSP. This is an approximation, and implies that the changing neutron energy spectrum in a voided channel does not affect the neutron energy spectrum in a neighboring channel that has not voided yet, or vice versa. Ignoring these possible spectral shifts might be an acceptable strategy in the case of the current CANDU reactors. They have a relatively large pitch (fuel channels far apart), and the coefficient is always positive (spectral shift would only affect the rate at which the reactivity increases). However, in the case of the ACR-700, the validity of this approximation will have to be re-examined.

AECL Response

A multicell capability is being developed for WIMS (refer to Section 4.2.1.2 of Reference 3). This will allow the generation of "color-set" cross sections that accounts for environment effects and differences in voiding rate in neighboring fuel channels. Proof-of-concept tests have been carried out and demonstrated that with this improvement in the WIMS/RFSP methodology, there is reasonable agreement between WIMS/RFSP and MCNP in modeling transient checkerboard-voiding core configurations.

11. Section 3.5.2.1.8, page 3-18

It may be necessary to generate few group cross sections for RFSP that are not only dependent on all the variables that are already included in the WIMS input (burnup, position in fuel channel, and fuel channel in core, temperature, and coolant density, etc.), but that are also dependent on the voiding pattern. Thus, it might be necessary to carry out a multi-channel WIMS-type analysis to determine the appropriate cross sections. This would imply that WIMS would have to handle a checkerboard input file recognizing different coolant densities in the neighboring fuel channels, assuming a simple checkerboard voiding pattern. More complicated voiding patterns (e.g., stratified partial voiding) may be identified based on ACR-700 system fluid dynamics calculations.

AECL Response

Please see the response to Item 10. More complicated voiding patterns such as the top half of the channel voided can be properly modeled in the new WIMS version. However, stratified void does not occur in LLOCA transients, nor does it occur during any design-basis accident while the reactor is still at power, i.e., prior to scram.

12. Section 3.5.3.3, page 3-28

This comparison (see table 3-11: Comparison of Phenomena Contributing to CVR in ACR-700 and CANDU) indicates that a completely different situation exists between the CVR components for the ACR-700 and the existing CANDU reactors. The above comparison indicates that it might be sufficient to accurately predict the U-238 contribution to the CVR in the current CANDU reactor designs to make an estimate of the CVR. This will not be the case in the proposed ACR-700. It will be important to predict the global neutron leakage from the core, contributions from the fissile isotopes (which are large and of opposite sign), and contributions from U-238 and Dysprosium. Thus, a new WIMS-IST/DRAGON-IST/RFSP-IST set of code inputs and models will need to be created and validated against appropriately designed experiments.

AECL Response

Assessment of physics codes and models for ACR applications, and code development and verification work plans are in place and some of the key development tasks are well advanced. The physics toolset qualification program is based on a comprehensive experimental program in ZED-2 to provide ACR specific data for code validation purposes. The experimental program is discussed in Reference 2, and the code qualification plan is discussed in Reference 3.

13. Section 3.5.5, page 3-29

Based on the information provided by AECL during the preapplication review, and pending resolution of the issues discussed above, the staff believes WIMS-IST, DRAGON-IST, and RFSP-IST, do not have any fundamental errors or shortcomings that would preclude certification of the ACR-700 design....Thus, the currently used codes (WIMS-IST, DRAGON-IST, and RFSP-IST) will have to be modified and revalidated for the new conditions.

AECL Response

We note and appreciate the conclusion that the physics codes do not have "any fundamental errors or shortcomings". Our plans for enhancing the codes and for performing experiments in support of their validation are described in References 3 and 2, respectively.

14. Section 3.5.5, page 3-30

WIMS - Because the safety performance of the ACR-700 depends significantly on correct treatment in the resonance range, the effect of binding oxygen in the fuel crystal lattice should be evaluated.

AECL Response

Preliminary calculations taking into account bound oxygen in UO_2 indicate a 0.2 to 0.3 mk effect in coolant void reactivity. It is planned to add cross sections for O bound in UO_2 to the WIMS and MCNP ACR data libraries.

15. Section 3.5.5, page 3-30

WIMS - the ACR-700 is a completely new concept, which will demand that a new, fast-running resonance treatment be developed and validated, in a similar way in which the newres treatment was validated for the current CANDU design. This validation effort will call for a significant set of experiments (see below), and a large number of Monte Carlo calculations to evaluated gaps where no experimental data exist.

AECL Response

A new resonance treatment has been added to the version of WIMS to be used for ACR applications. The validation of this version is included in the analysis of the planned ZED-2 experiments. Preliminary results show good agreement with MCNP.

16. **Section 3.5.5, page 3-30**

    WIMS - The revised version of WIMS-IST will need to treat checkerboard voiding patterns with variable burnup in adjacent fuel channels. This version will also have to be validated against appropriately designed critical experiments.

    AECL Response

    Per the response to similar earlier comments, a multicell option is being implemented in WIMS and experiments have been added to the revised experimental plan to validate this option.

17. **Section 3.5.5, page 3-30**

    WIMS - The burnup chains used in WIMS will have to be modified to allow for the proper treatment of burnable poison located in selected fuel pins. In addition, the inclusion of appropriate fission product treatment consistent with the higher burnup and associated higher fission product concentrations will need to be included in the WIMS code.

    AECL Response

    Burnable dysprosium has been added to the WIMS library. Comparisons between burnup calculations performed using WIMS and Helios show good agreement. The intention is to replace the existing dysprosium data in the WIMS library with data derived from the most recent ENDF/B-VI evaluation. Based on the Helios comparison the expected effect on CVR is about 0.09 mk.

18. **Section 3.5.5, page 3-30 to 3-31**

    WIMS - Currently two neutron energy groups are sufficient to carry out the core simulation calculations for conventional CANDUs. Cross sections are prepared by WIMS-IST and used as input to RFSP-IST (see below). Because of the smaller core dimensions, spectral shifts at the core/reflector interface, importance of the resonance energy range, and importance of leakage effects (see below), a different few group structure might be necessary (more groups with different group boundaries). This would necessitate a change in the WIMS system.

    AECL Response

    Please see the response to Item 12. Also, the fine-structure experiments planned to validate the physics code suite are relevant to this comment (refer to Section 3.3.4 of Reference 2), particularly measurements to be taken at the edge of the lattice (refer to Section 3.3.4.1.3 of Reference 2).

19. Section 3.5.5, page 3-31

WIMS - The kinetic parameters for a core composed of relatively high burnup fuel, light-water coolant, and a tighter pitch will need re-evaluation using the modified WIMS.

AECL Response

Agreed. The kinetics parameters will be evaluated as part of the design and safety analysis.

20. Section 3.5.5, page 3-31

WIMS - The thermo-physical properties used in WIMS to describe the fuel will need changes to recognize the fact that at the end of life, the burnup is essentially three times that of the current CANDU fuel at the end of life.

AECL Response

Fuel isotopic composition as a function of burnup is computed by WIMS depletion calculations. NRU irradiations and PIE data have been used to validate such calculations. The same validation will be performed for ACR fuel at extended burnups. This is discussed in Section 5.5 of Reference 3.

With regard to fuel thermo-physical properties, fuel temperature is the parameter of interest that is determined in part by the thermo-physical properties. Fuel temperature is an input parameter in the WIMS calculation. It is correlated to bundle power (computed by RFSP) based on ELESTRES predictions (ELESTRES is our fuel performance code). ELESTRES validation considers the higher burnup conditions for ACR fuel.

From the reactor physics perspective, we need to address the question of cross-section dependence on fuel temperature and burnup. With the ACR design, the ranges of fuel temperature and fuel burnup are not outside of the PWR experience base, where the same nuclear database is applied.

21. Section 3.5.5, page 3-31

DRAGON – The discussion regarding nuclear data outlined above (WIMS, page 3-30, first bullet) also applies to DRAGON.

AECL Response

DRAGON is used primarily for device incremental-cross-section calculations, and this comment does not apply to DRAGON device incremental-cross-section calculations. Essentially DRAGON is used to do 3D supercell calculations with and without a device present. The material cross sections are provided as input to the 3D model. These material cross sections, however, are derived using either DRAGON or using WIMS-AECL. The standard methodology is to use WIMS-AECL in the so-called "side-step" method. Thus, the question of basic nuclear data applies to WIMS-AECL, and the DRAGON supercell calculations make use of WIMS-based cross sections. Also the differences between the calculations of the cases with and without devices subtract off many of the uncertainties common to both cases.

22. Section 3.5.5, page 3-31

DRAGON - The DRAGON code has a sophisticated method of determining escape probabilities. Therefore, the code could be used to determine the validity of approximations that might be useful in WIMS.

AECL Response

Our standard lattice code is WIMS-AECL. DRAGON is not used as the standard lattice code. Our strategy for WIMS validation is through comparisons to experiments and inter-code comparisons with MCNP.

23. Section 3.5.5, page 3-31

RFSP - The larger spectral shifts expected at the core/reflector interface during a coolant voiding event will compel AECL to examine the current few group structure very carefully. A revised few group structure may need to be developed (see above under WIMS).

AECL Response

We have performed an assessment on the accuracy of 2-group full-core calculations relative to N-group calculations. The key parameters examined included maximum channel and bundle powers and CVR for full-core void and checkerboard void. We will continue to assess the biases and uncertainties associated with a 2-group representation in the full-core calculations. The biases and uncertainties will be quantified and documented in the Validation Manual. In the longer term, we plan to develop multigroup nodal solution method in RFSP, as discussed in Section 4.2.2.3 of Reference 3.

24. Section 3.5.5, page 3-31

RFSP - The fuel management scheme built into RFSP is quite flexible, but its application has been limited to relatively low burnup operation. The extended burnup will call for validation that the fuel duration and burnup are correctly determined.

AECL Response

The fuel depletion (and hence fuel composition) as a function of burnup is evaluated with the WIMS lattice code. RFSP models the history of irradiation as the bundle is shifted along the channel, accounting for the local conditions at different positions. Issues on extended burnup in ACR are more related to WIMS' capability and accuracy in tracking fuel depletion and in providing homogenized cross sections as a function of burnup. Validation of WIMS burnup calculations will be based on NRU irradiation of ACR fuel bundles and PIE data.

25. Section 3.5.5, page 3-31

RFSP - If the homogenization scheme used in WIMS to determine the few group cross sections is changed in any manner, the change will have to be consistent with the algorithms being used in RFSP.

AECL Response

At the present time, there is no plan to change the homogenization scheme used in WIMS. If, in the future, a more sophisticated homogenization scheme is introduced, such as region-dependent cross sections within a lattice cell, then the RFSP model and/or algorithms will need to be modified to make use of the cross sections from WIMS.

26. Section 3.5.5, page 3-32

The experiments should reflect the prototypic enrichment at BOL. Ideally, a set of experiments should also be carried out to reflect the end-of-life composition. Suitable MOX fuel should be used to simulate the end-of-life conditions, including the effect of depleting the Dysprosium-containing central fuel pin.

AECL Response

Experiments are planned to measure fine-structure reaction rates using demountable bundles that are representative of fresh fuel and irradiated fuel. The irradiated-fuel composition has been chosen to represent the latter half of the irradiation history and from a CVR perspective will be closely representative of end-of-life burnup. The specific experiments that will be performed with simulated MOX fuel are discussed in various sections of Reference 2.

27. Section 3.5.5, page 3-32

To gain sufficient data to estimate uncertainty, measurement experiments may have to be repeated several times (see Douglas, June 2001). An experimental program designed to validate the computer code suite used to analyze the ACR-700 should account for the increased number of experiments needed to determine the uncertainty associated with a measured value.

AECL Response

We routinely compare uncertainties derived from the reproducibility of repeated measurements with uncertainties determined from analysis of a single experiment and have found good consistency between the two approaches for establishing experimental uncertainty.

As stated in Sections 1 and 2 of Reference 2 a much wider range of experiments and analysis of those experiments are planned, aimed largely at providing validation data for the entire physics toolset (WIMS/DRAGON/RFSP and MCNP). This will provide integral validation of the toolset and of MCNP. The accuracy of the experimental data used for determining reactivity coefficients (i.e. moderator temperature, moderator purity, coolant density, etc.) is defined by the accuracy of the measurement of the change in moderator critical height. We measure absolute critical height to an accuracy of +/- 0.02 cm and changes in critical height to an accuracy of +/- 0.01 cm.

Other systematic errors (e.g., moderator purity accuracy, moderator temperature accuracy) can be cancelled out by design of the experiment. Where an experiment uses a short series of critical height measurements to establish a coefficient value, residual errors over the series may be reduced to less than ±0.02 mk. The uncertainty is adequately determined by scatter of the data points within the series, in a single experiment.

28. **Section 8.3.1, page 8-10**

The phenomenon with the largest discrepancy between importance and knowledge level relates to spatial homogenization of cross sections for use in core simulations.

AECL Response

This comment is similar to comments made in Section 3 of the PASAR, and is addressed in the responses to those comments. See, for example, item 10.

29. **Section 8.3.1, page 8-10**

Leakage is significant between ACR channel nodes, which is especially important when the void is in a checkerboard pattern, and from peripheral channels to the adjacent reflector regions. Since the checkerboard situation exacerbates the homogenization problem, which was identified as particularly important, ZED-2 may also be very useful if it can represent such a configuration.

AECL Response

A large number of checkerboard-void experiments have been added to the revised experimental plan. Refer to Sections 3.3.1 and 3.3.3 of Reference 2.

30. **Section 8.3.1, page 8-10**

In general, it would be beneficial for the NRC to follow the ZED-2 work so that results can be used to validate the NRC's calculational methods and to resolve the leakage issues brought up in the PIRT.

AECL Response

Reference 2 describes the experimental program. We will provide the experimental results reports as they become available.

31. **Section 8.3.1, page 8-11**

Benchmark calculations using methods more rigorous than those expected to be the norm are also suggested. The examples identified as being most urgent to resolve relate to calculating the bundle power near the core-reflector interface and the fuel element power everywhere, the latter requiring dehomogenization (flux reconstruction).

AECL Response

The code assessment program and code validation program described in Reference 3 will address the concerns identified in this comment. Specifically, inter-code comparisons with MCNP will ascertain the errors in the bundle powers near the core/reflector interface; and the peak fuel element power due to a global flux gradient across the bundle.

32. **Section 8.3.2.4, page 8-21**

Any modeling biases arising from modeling approximations of effects not present in the experimental benchmarks used for validation will be quantified by analysis of detailed versus approximated models and ultimately added, as appropriate, to the biases derived from such experiments. Modeling approximations to be analyzed would potentially include, among others, smearing the rim effect, modeling few versus many fuel pin composition regions, ring-averaging the burnup within fuel bundles, simplified treatment of localized three-dimensional lattice flux variations (e.g., from transverse reactivity devices, bundle end effects, core-periphery flux gradients), coarse versus detailed representations of radial temperature profiles in the fuel, or using single-channel versus multiple-channel lattice physics models for checkerboard voiding and/or fuel depletion. Other potentially significant issues to be analyzed include the sensitivity of CVR to (1) D_2O temperature and the pedigree of the S(..,..) data for bound thermal scattering in D_2O, (2) variable core power history, fuel temperatures, thermal fluid conditions, and fueling schemes, (3) the variable presence of control absorber rods in the core, (4) the presence of soluble boron or gadolinium in the D_2O moderator under certain operating conditions, and (5) zirconium cross-section modeling in the fuel cladding, PTs, and calandria tubes.

AECL Response

Each of the points raised in the comment is addressed individually.

(i) *Smearing the rim effect – This is addressed in Section 4.1.9 of Reference 3.*

(ii) *Modeling few versus many fuel-pin composition regions - Each fuel pin is sub-divided into 2 annuli radially and 2 sectors azimuthally. The fuel composition in each of the fuel sub-regions is separately tracked. This 2x2 subdivision is recommended based on considerations of accuracy in the results and computational speed. We will verify the applicability of this approximation through sensitivity studies.*

(iii) *Ring-averaging the burnup within fuel bundles - Unless there is a strong global flux gradient across the bundle, the assumption that the fuel-pin power and burnup in the elements of the same pitch circle are the same is a reasonable approximation. We will confirm that such gradients are not significant, and if they are we will quantify their effect.*

(iv) *Simplified treatment of localized three-dimensional lattice flux variations (e.g., from transverse reactivity devices, bundle end effects, core-periphery flux gradients) – These issues are being addressed by various aspects of the code qualification and experimental program. For example, there will be measurements at the edge of the lattice (Section 3.3.4.1.3 of Reference 2), end-flux-peaking measurements (Section 3.3.4.2.2 of Reference 2) and measurements with absorbers in the lattice (Section 3.3.4.1.2 of Reference 2). These will be used to validate the modeling of these phenomena and assign appropriate uncertainties.*

(v) *Coarse versus detailed representations of radial temperature profiles in the fuel – The radial temperature profile within a fuel pin is modelled with ELESTRES, and the pin-average value used in the WIMS model. Sensitivity studies are performed to assess the effect of this modeling approximation.*

(vi) *Using single-channel versus multiple-channel lattice-physics models for checkerboard voiding and/or fuel depletion – The reactor physics calculations are carried out for each individual fuel bundle in every fuel channel in the core. The thermal hydraulic boundary conditions, such as coolant density and coolant*

temperature, are inputs taken from thermal hydraulics computer codes. We have assessed the sensitivity of the checkerboard voiding effect to whether a single average channel is assumed from a thermal hydraulics perspective, or multiple channels are considered (Reference 4). The impact on checkerboard voiding and the resultant reactor power transient is very small.

Issues related to CVR:

1) *Sensitivity to moderator temperature is covered in Section 4.1.4 of Reference 3. Tests related to the effect of moderator temperature on CVR are discussed in Section 3.3.2.1 of Reference 2. Any other effects of the scattering data for D_2O in the WIMS library will become part of the bias and uncertainty established during validation, and will be allowed for in calculations.*
2) *Variable core power history, fuel temperatures, thermal fluid conditions, and fueling schemes - CVR sensitivities to various operating conditions are covered under Section 4.1.4 of Reference 3.*
3) *The variable presence of control absorber rods in the core - CVR sensitivity to control absorber insertion positions is covered under Section 4.1.4 of Reference 3.*
4) *The presence of soluble boron or gadolinium in the D_2O moderator under certain operating conditions - CVR sensitivity to the presence of moderator poison is covered under section 4.1.4 of Reference 3. Tests related to the effect of moderator temperature on CVR are discussed in Section 3.3.2.2 of Reference 2.*
5) *Zirconium cross-section modeling in the fuel cladding, PTs, and calandria tubes - Zr cross-section data (including self-shielding) in the WIMS library have been well established.*

33. Section 8.3.2.4, page 8-21

If ACR-700 void reactivity proves to be nearly zero or positive, then the applicant will need to address how to terminate the fission chain reaction in response to large LOCAs. The more constricted core geometry of the ACR-700 in relation to existing CANDUs has necessitated changes to the CANDU shutdown systems for the ACR-700. The impact on the reliability of scram insertion must be determined. The staff's continuing review activities related to void reactivity will therefore consider this fact as a basis for highlighting the need to ensure adequate shutdown system reliability in accidents.

AECL Response

With the latest ACR-700 fuel design, the void reactivity following a large LOCA is expected to be small and negative. The inherent reactivity feedback is expected to reduce reactor power in the initial seconds following a large LOCA, but a timely reactor scram will still be required to ensure that the relevant acceptance criteria are met. The safety analysis will demonstrate the effectiveness of scram in ensuring that the acceptance criteria are met.

34. **Section 8.3.3, page 8-23**

The staff intends to pursue with AECL the potential sources of applicable critical benchmark data from past measurements in the Deuterium Critical Assembly in Japan and from other heavy-water critical experiment facilities in the United States, the United Kingdom, Italy, and elsewhere.

AECL Response

At a meeting on Dec. 2, 2004, NRC staff requested that these reports be provided. AECL has provided the requested reports under separate cover (Reference 5).

35. **Section 8.5, page 8-25**

In making the transition to the anticipated design certification review phase, the staff will continue and expand its review efforts with AECL on

(1) analyzing void reactivity effects in AECL's nuclear design of the ACR-700 submitted for design certification,

(2) assessing the adequacy of AECL's planned set of void reactivity experimental benchmarks from ZED-2 and other facilities,

(3) identifying and addressing needs for additional benchmark experiments and other measured data,

(4) using the emerging set of benchmark results to quantify bias and uncertainty in nuclear code calculations of void reactivity and other safety-related nuclear characteristics of the ACR-700 design,

(5) developing specific NRC review guidance, and

(6) further developing and using the staff's independent nuclear analysis tools and capabilities as needed for simulating ACR-700 operating states, transients, and accidents.

AECL Response

To facilitate NRC staff's review, AECL has provided a description of the planned experimental program (Reference 2) and a description of the code enhancement program (Reference 3) to NRC staff.

References

1. Letter W. Beckner to J. Polcyn, "Pre-Application Safety Assessment Report (PASAR) for the Advanced CANDU Reactor (ACR-700)", Oct. 27, 2004.
2. M.B. Zeller, "Planned Physics Experiments in ZED-2 in Support of ACR", 108-123110-LS-001, Rev. 0, March 2005[1].
3. H. Chow, "Qualification Plan for Reactor Physics Toolset for Use in ACR Design and Safety Analysis", 108-119190-LS-001, Rev. 0, March 2005[1].

1 References 2, 3 and 4 are AECL PROTECTED – Proprietary reports and are submitted to the NRC under separate cover.

4. D. Jenkins, "Effect of Zone Controllers and Multiple Thermalhydraulic Channels on ACR Void Reactivity and LOCA", 108-115530-ASD-002, Rev. 0, March 2005[1].

5. G. Archinoff to A. Cubbage, "Non-ZED-2 Experiments Considered for ACR Validation", AECL file 108US-ACNU05-0016L, March 24, 2005.