AECL EACL

Assessment Document

GENERIC ACTION ITEMS - DISPOSITION AND/OR PLAN

ACR

108-00580-ASD-005

Revision 0

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LIST OF ACRONYMS

AECL	- Atomic Energy of Canada Limited
ACR™	- Advanced CANDU Reactor™*
AECB	- Atomic Energy Control Board
ASEP	- Accident Sequence Evaluation Program
ASME	- American Society of Mechanical Engineers
CANDU	- CANadian Deuterium Uranium
CCAFF	- Core Cooling in the Absence of Forced Flow
CHF	- Critical Heat Flux
CNSC	- Canadian Nuclear Safety Commission
COG	- CANDU Owners Group
CRL	- Chalk River Laboratories
CSA	- Canadian Standards Association
CT	- Calandria Tube
DBA	- Design Basis Accident
DDT	- Deflagration-to-Detonation Transition
ECC	- Emergency Core Cooling
EFADS	- Emergency Filtered Air Discharge System
FAC	- Flow Assisted Corrosion
FCV	- Filtered Containment Venting
FMS	- Feedback Monitoring System
GAI	- Generic Action Item
HFEPP	- Human Factors Engineering Program Plan
HIS	- Hydrogen Ignition System
HRA	- Human Reliability Analysis
HTS	- Heat Transport System
IAEA	- International Atomic Energy Agency
IEC	- International Electrotechnical Commission
ISA	- Instrument Society of America
ISO	- International Standards Organization
IST	- Industry Standard Tool(set)
LAC	- Local Air Coolers
LLOCA	- Large Loss Of Coolant Accident
LOCA	- Loss Of Coolant Accident
LOECC	- Loss Of Emergency Core Cooling

* ACR™ (Advanced CANDU Reactor™) is a trademark of Atomic Energy of Canada Limited (AECL).

LOM	- Loss Of Moderator
MFMI	- Molten Fuel Moderator Interaction
MTF	- Moderator Test Facility
NB	- New Brunswick
NFPA	- National Fire Protection Association
NGS	- Nuclear Generation Station
NIAC	- Nuclear Insurance Association of Canada
NOP	- Neutron Overpower Protection
OH	- Ontario Hydro
OPG	- Ontario Power Generation
OQAM	- Overall Quality Assurance Manual
P&IC	- Pressure & Inventory Control
PARs	- Passive Auto-Catalytic Hydrogen Recombiners
PD	- Pressure Drop
PHT	- Primary Heat Transport
PLEx	- Plant Life Extension
PLiM	- Plant Life Management
PPV	- POWDERPUFS-V
PSA	- Probabilistic Safety Assessment
PT	- Pressure Tube
QA	- Quality Assurance
QAP	- Quality Assurance Program
RB	- Reactor Building
ROP	- Regional Overpower Protection
RRS	- Reactor Regulating System
SCC	- Stress Corrosion Cracking
SDS1	- Shutdown System 1
SEU	- Slightly Enriched Uranium
SOE	- Safe Operating Envelope
THERP	- Technique for Human Error Rate Prediction
USNRC	- United States Nuclear Regulatory Commission
V&V	- Verification & Validation
VREA	- Void Reactivity Error Allowance
ZED-2	- Zero Energy Deuterium (Reactor)

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1. INTRODUCTION

1.1 CNSC Generic Action Items

Generic Action Items (GAIs) are a mechanism used by the regulator to formally identify and monitor the resolution of safety and licensing issues that are common to more than one licensee. They also provide a formal basis for communication of licensee progress in resolving identified safety issues. Since their inception in 1988, twenty-four Generic Action Items have been issued by the CNSC and its predecessor, the AECB. Many of them have been closed on the basis of commitments made and actions taken by the licensees.

Since 1998, CNSC staff position statements have been produced for most of the open Generic Action Items, to identify to the licensees the safety issues, and the specific actions required to resolve them to the satisfaction of the CNSC. Closure criteria stipulate certain analyses or evaluations that are expected to be performed, or identify design modifications or programmatic changes that are to be considered. The closure criteria listed in the position statements provide a list of the minimum requirements that would lead to satisfactory resolution of the issue. Position statements are subject to revision, as developments occur, as new information becomes available or in response to submissions by the licensee.

AECL has been actively involved in supporting the resolution of Generic Action Items at operating plants in Canada, through collaborative work sponsored by the CANDU Owners Group (COG), and by performing supporting research and development. Generic Action Item resolution status is an important element of regulatory feedback and is an integral part of AECL's formal design feedback process.

1.2 The Advanced CANDU Reactor (ACR)

The ACR is an evolutionary reactor designed for, among other things, enhanced safety features. The ACR design is based on the modular concept of horizontal fuel channels surrounded by a heavy water moderator, the same as with all CANDU reactors. The major innovation in ACR is the use of slightly enriched uranium fuel and light water as the coolant rather than natural uranium fuel and heavy water as the coolant as in the standard CANDU reactor design.

The safety enhancements made in ACR encompass safety margins, performance and reliability of safety related systems. Many of these safety improvements serve to address one or more of the GAIs and thus promote the licensability of the design.

The ACR incorporates design provisions to address Generic Action Items based on the specific approach determined to be necessary, within the context of overall industry plans, to reach satisfactory resolution on these issues with the CNSC.

This report provides a summary review of the GAIs from the ACR perspective, including a summary of the closure criteria, an assessment of the relevance to the ACR design, and the current status of actions taken in the ACR program to resolve the defined issues. The descriptions of the safety issues and the associated closure criteria (shown in italic fonts), are direct quotations from the first and second Canadian National Report for the Convention on Nuclear Safety dated 1998 and 2001 respectively, and CNSC (or AECB) Position Statements.

Many of the GAIs include a measure of design improvement in their resolution. If no direct design solution is feasible for a given GAI, AECL will ensure that the major contributors to risk and the major sources of uncertainty associated with the issue have been identified and addressed.

This report is organized as follows:

- Chapter 2 summarizes the scope and objectives of this assessment.
- Chapter 3 provides an executive summary of the ACR status for each of the Generic Action Items.
- Chapter 4 provides the detailed assessments for each GAI.
- Chapter 5 summarizes the overall conclusions of this assessment.

2. OBJECTIVES

The objective of this report is to review and evaluate the extent to which the ACR design is affected by the safety issues identified in the CNSC GAIs. This report provides a summary of the technical and programmatic issues to be resolved and the specific actions taken in the ACR design to resolve them. The relevance of the GAIs to the ACR design is assessed and the status of the ACR design with respect to the closure criteria established for each GAI is provided. Where relevant, future activities, such as analysis and R&D, needed to ensure that the issues raised in the GAIs are fully addressed are also identified. In cases where the issues raised in the GAIs are operational rather than design-based, this report identifies the actions that would need to be taken by the operating utilities, with the assistance of AECL.

This report covers the full set of GAIs that have been issued by the CNSC including those which have already been closed for one or more Canadian licensees.

3. EXECUTIVE SUMMARY OF GENERIC ACTION ITEMS

88G02, “Hydrogen Behaviour In CANDU Nuclear Generating Stations”

This action item addresses the depth of analysis concerning hydrogen mixing in containment and igniter capability. The CNSC believes more work is needed to reduce uncertainties related to: hydrogen concentration, standing flames, flame acceleration, flame propagation between compartments, and transition from deflagration to detonation.

ACR addresses this issue by employing hydrogen control measures that include Passive Auto-Catalytic Hydrogen Recombiners, and local air coolers. A comprehensive hydrogen assessment addressing the effects of fast deflagration and standing flames on major equipment in containment will be performed to demonstrate the adequacy of these design provisions.

89G03, “Ontario Hydro’s Pressure Tube Inspection Program”

The CNSC staff was concerned with hydride blisters related to PT/CT contact in some operating reactors several years ago. COG developed fitness for service guidelines and released them in 1991. The Canadian nuclear industry was judged to be responding adequately to this issue in late 1993 and the action item was closed in early 1994.

The fuel channel of the ACR will be designed and manufactured, based on successful experience gained, so as to minimize the potential for blister formation. AECL will support the purchasing utilities in the establishment of similar fitness for service guidelines appropriate to ACR operation. AECL has committed to apply relevant standards to the ACR, including the inspection program, CAN/CSA-N285.4.

89G05, “Use of Mercury Wetted Relays in Safety Related Systems”

The CNSC staff was concerned with the low reliability of non-tin doped mercury wetted relays.

This GAI was closed for all CANDUs as non-tin-doped mercury wetted relays are no longer used in safety related systems, therefore this GAI is not relevant to the ACR.

90G01, “Loss of Oil in Rosemount Pressure Transmitters”

Certain models of pressure transmitters manufactured by Rosemount had suffered a gradual deterioration in performance at some plants and the licensees were required to replace these particular models of Rosemount pressure transmitters.

This GAI was closed for all CANDU plants, as these transmitters are no longer used. This GAI is not relevant to the ACR.

90G02, “Core Cooling in the Absence of Forced Flow (CCAFF)”

CNSC staff indicated that they believed there were significant uncertainties associated with core cooling under partial inventory conditions should power to the primary coolant pumps fail. They advised that more analyses of the phenomena involved needed to be undertaken to understand the application of RD-14M test results to reactor scenarios.

Provisions have been made in the ACR design and safety analysis program to demonstrate that natural circulation is effective even if the heat transport system coolant inventory is somewhat depleted.

90G03, "Assurance of Continued Nuclear Safety/Management of Aging"

The CNSC staff is concerned with the effects of aging on plant safety. This GAI establishes expectations on the licensee for the establishment of a systematic and integrated program for managing the aging of safety significant systems, structures, and components throughout the plant design life.

ACR will be building upon AECL's experience from existing CANDU Plant Life Management (PLiM) and Plant Life Extension (PLEx) programs to ensure that the 60 year operational life target is achieved. ACR is also employing a reliability centred maintenance program to ensure all important plant components, large and small, are well maintained throughout the plant life.

91G01, "Post-Accident Filter Effectiveness"

This issue is directed primarily at units with vacuum building containment which must use filter systems to maintain releases within CNSC prescribed limits.

ACR will not employ a vacuum building containment concept. There is no requirement to include a containment venting system as part of the safety case for design basis events, and as such, there are no issues with respect to post-accident filter effectiveness in the ACR design.

91G02, "Operation with a Flux Tilt"

The CNSC staff expressed concern with slowly developing accidents leading to fuel overheating in some regions of the core. Licensees were asked to provide additional analysis to demonstrate the effectiveness of shutdown systems for slow loss of regulation accidents involving an initially tilted power distribution.

AECL is addressing the issues raised in this GAI in the design of the ACR, using the same approach and proven methodologies employed to successfully close this issue for CANDU 6 plants operating in Canada.

92G01, "Treatment of Human Factors in Ontario Hydro Reliability Analyses"

This issue centers on the methods used to make quantitative estimates of human reliability. Attention was originally drawn to uncertainty related to human action in the use of the filtered air discharge system at Pickering NGS.

The human reliability analysis methodology for ACR is based on internationally recognized procedures and previous experience. A Human Factors Engineering Program Plan was prepared to establish how human factors activities or tasks are to be carried out for the ACR.

94G01, "Best Effort Analysis of ECCS Effectiveness"

The CNSC staff identified a concern with the adequacy of validation for codes used to predict the performance of emergency core cooling.

The calculations of ECCS effectiveness for all LOCA cases for ACR will be performed using the CATHENA code. CATHENA has been extensively validated for current CANDU designs and the validation is being extended to cover the ACR design.

94G02, "Impact of Fuel Bundle Condition on Reactor Safety"

This GAI provides expectations for the establishment of a fuel and fuel channel program by the plant operator that integrates inspections, operating conditions, research results, design feedback and safety analysis.

The ACR design will take into account extensive experience from previous CANDU designs, and the efforts of others in the Canadian nuclear industry, as well as implementing a substantial fuel testing program for ACR CANFLEX fuel, to minimize the susceptibility to acoustics-based fuel bundle or pressure tube degradation. In addition, AECL will provide the plant owner with the full support necessary to implement an ongoing fuel and fuel channel inspection and maintenance program compliant with the requirements of this GAI.

95G01, "Molten Fuel/Moderator Interaction"

The CNSC staff identified the need for further research to confirm that analyses of single channel flow blockage, resulting in ejection of molten fuel into the moderator, are valid.

An industry program to provide additional experimental support for methods to assess the potential for in-core damage in existing CANDU designs is underway. Additional experimental work to address ACR-specific conditions will be performed if necessary. These experiments will permit AECL to validate generic and ACR-specific aspects of its molten fuel moderator interaction model and quantify important parameters. In their earlier review of CANDU 9 licensability, CNSC staff agreed that such experiments could generate sufficient information on important parameters, such as the heat conversion factor and the effect of rupture geometry, and could provide confidence that steam explosions in the calandria can be ruled out.

95G02, "Pressure Tube Failure with Consequential Loss of Moderator"

Based on the test results involving pressure tube and calandria tube integrity, CNSC staff identified the concern that pressure tube rupture leading to a guillotine pressure tube failure would result in ejection of the end fitting. The consequential loss of moderator inventory could be a problem in conjunction with scenarios involving assumed loss of emergency core cooling.

ACR is addressing this issue by using a thicker calandria tube thus the probability of the end fitting ejecting following a pressure tube rupture is extremely low. This will reduce the probability the entire accident sequence to allow it to be categorized as a severe core damage event rather than a design basis event.

95G03, "Compliance with Bundle and Channel Power Limits"

After reviewing the process that operating nuclear power plants use to demonstrate compliance with bundle and channel power limits, the CNSC staff identified the need for improvement in the methods used to demonstrate adequate compliance.

AECL will assist the plant operator with implementing appropriate monitoring and compliance procedures on an as-required basis. Compliance procedures, and ongoing programs to collect necessary operating data to confirm error allowances, will be developed for ACR consistent with and building upon the methodologies employed by the Canadian utilities to close this GAI.

95G04, "Positive Void Reactivity-Treatment in Large LOCA Analysis"

This GAI addresses the positive void reactivity effect in existing CANDU reactors. Unlike current natural uranium fuelled CANDU reactor designs, the ACR has a negative coefficient of reactivity associated with coolant voiding. This results in a considerably more benign response to events such as large break LOCAs which exhibit rapid voiding in the early stages of coolant blowdown following the break. The issue of positive void reactivity and its treatment in large LOCA analysis is therefore not directly relevant to the ACR.

Completion of the planned reactor physics R&D program for the ACR will demonstrate that safety analysis is being performed using acceptably accurate and validated reactor physics methods that provide appropriate treatment of void reactivity and experimentally-based allowances.

95G05, "Moderator Temperature Predictions"

The issue raised in this GAI is the demonstration, by analysis, that sufficient subcooling exists in the moderator at all times to prevent fuel channel failures following design basis events. The CNSC staff identified the need for additional code validation and demonstration.

The moderator circulation provisions for ACR will be examined. The 3-D Computational Fluid Dynamics computer code, MODTURC_CLAS will be further validated for ACR conditions. The 3-D MTF facility at Chalk River Laboratories will be modified to reflect the characteristics of moderator circulation in ACR and the code will be validated against these experimental results. ACR will use an improved moderator flow pattern relative to current CANDUs (as did CANDU 9) which gives increased margin to fuel channel failure.

96G01, "Fire Protection for CANDU Nuclear Power Plants"

In 1996, the Canadian Standards Association (CSA) issued CSA standard N293-95, Fire Protection for CANDU Nuclear Power Plants. CNSC staff considers that the nuclear industry should comply with relevant sections of this standard, and has requested the licensees to assess the adequacy of their fire protection programs against them.

The ACR Safety Design Guide on Fire Protection specifies requirements which will fully address the requirements of this GAI.

96G02, "Feeder Pipe Fitness for Service"

Inspections in several CANDU reactors revealed an unexpected reduction in the wall thickness of some outlet feeders. The rate of this degradation represents a departure from the original design predictions. The CNSC staff asked licensees to show that feeders are fit for service and to show sufficient understanding of the thinning phenomenon to prevent it from threatening the integrity of the feeders.

Current CANDUs use carbon steel feeders. ACR will use stainless steel that has a very low Flow Assisted Corrosion rate in the sensitive high velocity, high turbulence sections of the feeders. This will minimize the rate of feeder degradation. AECL will support the operating utility in developing fitness for service guidelines and a periodic inspection program appropriate to ACR operation.

98G01, "PHT Pump Operation under Two-Phase Flow Conditions"

The CNSC staff questioned the HTS piping fatigue analysis performed using a limiting forcing function (harmonic excitation) obtained from laboratory tests of full-scale HTS pumps. Further work was required to develop a mechanistic pump model from the available database and apply it to the HTS piping configuration.

The analysis methodologies employed by the Canadian licensees are considered to be well established and meet the requirements of the CNSC. The analysis performed for ACR to address these issues will be consistent with the overall approach taken by the Canadian licensees and ensure that the closure criteria for this GAI are met.

98G02, "Validation of Computer Programs Used in Safety Analysis of Power Reactors"

The CNSC is expecting that all computer codes used for safety analysis will be adequately validated prior to their application in accordance with the detailed requirements of Regulatory Guide G-149.

All computer programs to be used for ACR safety analysis have been validated for current CANDUs either by AECL or by AECL's Canadian industry partners as part of the collaborative Industry Standard Toolset program. Assessments of the qualification requirements of the computer programs for ACR have been performed and the relevant specific additional validation needs have been determined. AECL will ensure that the ACR validation and/or verification program will address the closure criteria identified for this GAI.

99G01, "Quality Assurance of Safety Analysis"

In recent years CNSC staff has become aware of an increasing number of occurrences of inadequate quality assurance in safety analyses performed by licensees. This GAI requires the licensee to carry out safety analyses in accordance with a well-established quality assurance program compliant with the CSA N286 series Standards.

Since AECL will carry out safety analyses of the plant as part of the planned design activities and in support of licensing by the plant owner, AECL will meet the requirements of this GAI. AECL has an established Quality Assurance Program for compliance with the CSA N286 series of Standards and there will be regular audits to demonstrate compliance.

99G02, "Replacement of Reactor Physics Codes used in Safety Analysis of CANDU Reactors"

Recent experimental data, as well as reviews of key older computer codes, have identified several shortcomings in the reactor physics area. Currently, the industry is in the process of retiring some older reactor physics computer codes. The CNSC staff has identified concerns with the absence of a structured and comprehensive program for replacement of various reactor physics codes by Canadian licensees.

This GAI is not directly relevant to the ACR. AECL is employing modern state-of-the-art CANDU physics analysis tools for the ACR design. Computer codes to be used in reactor physics analysis and nuclear design will be based on the CANDU Industry-Standard Toolset, modified and validated as required to address ACR physical design parameters. The tools will meet the CNSC requirements for compliance with CSA standard N286.7 and CNSC Regulatory Guide G-149.

00G01, "Channel Voiding During LOCA"

The CNSC staff has identified a concern that the computer codes used for prediction of overpower transients for CANDU reactors with a positive coolant void coefficient have not been adequately validated.

Unlike the current natural-uranium-fuelled CANDU reactors, the ACR reactor core design is characterized by a negative coolant void coefficient of reactivity. With a negative void reactivity, there is very little sensitivity of the results of safety analysis to voiding rate. Therefore, this GAI is not directly relevant to the ACR design. Nonetheless, the magnitude of channel voiding will be adequately characterized such that the thermalhydraulic and physics analysis tools used in the ACR safety analysis are adequately validated. AECL will use validated state-of-the-art computer programs for physics and thermal-hydraulics analysis. These tools (and any modifications undertaken to meet ACR requirements) will address the requirements identified in GAI 98G02 and in GAI 99G02 for compliance with CSA Standard N286.7 and CNSC Regulatory Guide G-149.

01G01, "Fuel Management and Surveillance Software Upgrade"

The CNSC staff has identified concerns with the fuel management and surveillance software and compliance methodologies used by some Canadian licensees. This GAI was established to track specific commitments made by Bruce Power and Ontario Power Generation to close GAI 95G03. Licensees are required to undertake a structured program for reactor core surveillance that should cover the fuel management software upgrade and validation, and the validation and qualification of the error compliance methodology.

This GAI does not apply directly to ACR. AECL will use modern physics and fuel management tools, WIMS-IST and RFSP-IST codes to perform core calculations in ACR, including core-tracking studies. These tools (and any modifications undertaken to address ACR requirements) will meet the requirements identified in GAIs 98G02 and 99G02 for compliance with the CSA Standard N286.7 and the CNSC Regulatory Guide G-149. AECL will provide support to the plant operator in implementing a structured program for reactor core surveillance that will cover the fuel management software implementation and validation, and the validation and qualification of the error compliance methodology.

4. GENERIC ACTION ITEMS – DETAILED REVIEW

This section provides a detailed review of each GAI, in chronological order.

4.1 88G02 "Hydrogen Behaviour in CANDU Nuclear Generating Stations"

4.1.1 Safety Issue (As Stated by CNSC)

"The hydrogen released due to high temperature interaction between zirconium alloys and steam during certain Design Basis Accidents (DBAs) may produce flammable gas mixtures in some regions of containment, in the short term. Flammable gas mixtures may also develop due to water/steam radiolysis and metal corrosion in various containment regions such as the sump regions and the calandria, over time. The mechanical and thermal loads generated by potential ignition of these gas mixtures may then threaten the integrity and functions of the containment envelope, critical internal structures and necessary safety-related system components and instrumentation.

To mitigate the combustion behaviour of the short term releases in the multi-unit NGSs, Ontario Power Generation (OPG) has installed Hydrogen Ignition Systems (HISs) to remove the hydrogen via benign burns initiated by a number of igniters. Assessments of the short term behaviour of hydrogen with the 3-D/1-D hybrid code GOTHIC were then carried out to demonstrate the effectiveness of the HISs. Based on a derived safe load criterion, OPG has concluded that all potential hydrogen burns in its NGSs are benign, even for impaired or ineffective HISs and Local Air Cooling units (LACs).

No short term mitigating measures have been installed in the single unit NGSs. It was postulated that dilution via mixing by natural circulation can mitigate the short term hydrogen threat. Scoping analyses with the 1-D code PRESCON2 to identify the potentially limiting cases, have been completed. Assessments with the 3-D/1-D hybrid GOTHIC code to determine the bounding gas mixtures and combustion modes in certain critical containment regions, have been carried out, as well.

No analysis of the threat posed by the hydrogen released over the long term in multi and single unit NGSs, have been carried out yet because the utilities were of the opinion that effective mitigation was provided by mixing via natural circulation or by venting via the Filtered Air Discharge System (EFADS).

As requested by CNSC staff, the utilities plan to demonstrate the effectiveness of current and planned mitigation measures including Passive Auto-Catalytic Hydrogen Recombiners (PARs) in the short and long terms." (Reference [1]).

4.1.2 Closure Criteria (As Stated by CNSC)

"Licensees are required to:

- *Consider the entire spectrum of design basis accidents including loss of coolant accident plus loss of emergency core cooling (LOCA + LOECC) and continue to use the methodology that maximizes the hydrogen and fission product source terms (limiting steam flows to the fuel channels), when evaluating the short-term and long-term hydrogen releases for each NGS.*

- *Carry out conservative, comprehensive and limiting assessments of mixing and transport for each NGS to:*
  - *determine the local hydrogen distributions in critical containment regions via 3-D assessments, crediting any hydrogen mitigating measures which are in place or planned; and*
  - *identify the envelope of hydrogen combustion and its consequences:*
  - *demonstrate that all local pockets of sensitive gas mixtures (combustible clouds) cannot lead upon ignition to Deflagration-to-Detonation Transitions (DDTs) and fast deflagrations with potentially unacceptable combustion loads in any region of containment;*
  - *demonstrate via well-supported calculations and/or experiments that gas mixtures of composition outside the DDT and fast deflagration envelope (i.e., in the slow deflagration domain), do not have, if ignited, consequences detrimental to the containment boundary, supporting internal structures and required safety-related equipment, taking into account turbulence and flame acceleration generated by obstacles, in particular;*
  - *demonstrate via well-supported calculations and/or experiments that potential standing flames do not threaten the survivability and/or function of vulnerable containment boundaries, critical internal structures and essential post-LOCA equipment;*
  - *ensure the survivability and/or functions of threatened essential post-LOCA equipment, vulnerable containment boundaries and critical internal structures, if any, via environmental qualification and/or augmented protection and mitigation, and*
  - *demonstrate, unless calandria refilling is assured, that, for LOCAs + LOECC with consequential moderator drain, the releases of hydrogen within the calandria, do not threaten the integrity of the calandria and that of necessary system components and monitoring instrumentation within it or next to it.*
- *Evaluate Passive Autocatalytic Recombiners for their potential to enhance the effectiveness of the short-term and long-term mitigating measures, and reduce the risk from potential hydrogen burns."* (Reference [1]).

4.1.3 Relevance of the GAI to the ACR

This GAI was issued for all licensees. The issues associated with hydrogen behaviour in the ACR will be similar to those in current CANDU reactors, and hydrogen mitigation provisions (to either dilute or remove hydrogen) will address both short and long term releases of hydrogen following design basis events and severe accidents.

4.1.4 Status for ACR

To control hydrogen in containment, ACR will follow the overall approach adopted for CANDU 9. The effects of any fast deflagration and standing flames on major equipment will be assessed and factored into the layout. The results of an extensive R&D program performed by the Canadian industry through COG to address major sources of uncertainty in these analyses will be accounted for.

The GOTHIC code will be used to study the thermalhydraulic behaviour of the ACR containment system under accident conditions. GOTHIC is a general purpose containment analysis code. The code has lumped parameter/1D/2D/3D modelling capability as well as hydrogen distribution/burning capability. It has been applied to active and passive CANDU containment behaviour under accident conditions, including containment pressurization, environmental qualification, and hydrogen mixing analyses. GOTHIC has been validated extensively, as an Industry Standard Tool (IST), for use in CANDU safety and licensing analysis, including hydrogen combustion calculations (refer to the assessment of GAI 98G02 for more details on the code validation programs). For the ACR containment, GOTHIC will be used to evaluate containment behaviour under both design basis and severe accident conditions.

The ACR Safety Design Guide on Environmental Qualification (Reference [2]) requires that components of the safety related systems which perform essential safety functions during events causing harsh environmental conditions either be protected from the harsh environment, or be qualified to withstand it.

In the current ACR design, Passive Auto-Catalytic Hydrogen Recombiners (PARs) will be installed based on appropriate analysis and taking into account the recommendations of an Industry Task Team formed by the CANDU Owners Group to establish design considerations and requirements for hydrogen mitigation systems in CANDU reactors. PARs qualification activities are being undertaken by AECL in conjunction with the Canadian nuclear industry activities.

Seismically qualified local Air Coolers will help disperse hydrogen released in the reactor vault and dome area of the containment as a result of events such as LOCA with loss of Emergency Core Cooling (LOECC), by mixing the containment atmosphere to prevent local regions of high hydrogen concentration.

The design basis for hydrogen is prevention of deflagration and the relevant event is large LOCA. The severe accident basis is prevention of the deflagration-detonation transition and this will be analysed for dual failures using a more realistic set of analysis assumptions.

Note that in past analyses of LOCA plus LOECC for CANDU reactors, the steam flow rate per channel was varied parametrically in the safety analysis, so as to maximize hydrogen production. The worst case corresponded to a ‘trace flow’ of about 10 g/s per channel. The industry has taken the position that this methodology is unduly pessimistic, and has proposed to the CNSC that more realistic LOCA + impaired ECCS analyses be performed for the dual failure event. LOCA + LOECC + loss of Moderator is much less likely in ACR than in previous CANDU designs because of the thicker, improved calandria tubes (see Section 4.13.4) allowing this event to be classified as a severe core damage event. With a high degree of confidence, it is expected that the actions taken in the ACR design to address the issue of hydrogen in containment will be sufficient to address the closure criteria for this GAI.

4.2 89G03 “Ontario Hydro’s Pressure Tube Inspection Program”

4.2.1 Safety Issue (As Stated by CNSC)

“In 1989, CNSC (then AECB) staff concluded that OPG’s (then OH) proposed in-service inspection program did not adequately cover all degradation mechanisms. The CANDU Owners

Group (COG) has since developed fitness for service guidelines that were acceptable to CNSC staff. OPG has committed to using these guidelines, and this was confirmed through ongoing licensing and surveillance activities. Surveillance programs have also served to verify the threshold for blister formation." (Reference [1]).

4.2.2 Closure Criteria

No closure criteria were defined. CNSC requested pressure tube degradation mechanisms to be well characterized and a proper periodic inspection program established accordingly.

4.2.3 Relevance of the GAI to the ACR

The GAI was originally addressed to the operating CANDU plants. The issues identified by this GAI will be addressed in ACR through the development of pressure tube fitness-for-service guidelines applicable to the new pressure tube design. In addition, design efforts to reduce the potential for hydride blister formation on pressure tubes will also assist in addressing the issues raised in this GAI.

The formation of hydride blisters on pressure tubes during plant operation is a potential precursor to pressure tube failure. The way in which hydride blisters can form on a pressure tube is the combined effect of:

a) achieving a concentration of hydrogen isotope in the pressure tube material, above a defined threshold for blister formation that is a function of the local temperature conditions, and
b) the pressure tube sagging into contact with its calandria tube. This creates a cold spot on the pressure tube at the contact point. Hydrogen can then migrate towards this cold point and blisters may develop at the point of contact.

Based on the considerable knowledge accumulated in this area over the last two decades, the ACR design incorporates a number of improvements at the design and manufacturing stage to minimize hydrogen levels, such as:

a) Careful control of the ingot melting (and re-melting, known as "quad" melt in the industry) process, with particular attention to vacuum conditions, which reduces the hydrogen concentration in the bulk of the ingot. In addition, as hydrogen tends to concentrate at the surfaces of both ingots and forgings, judicious machining to remove these hydrogen-rich layers ensures a very low hydrogen concentration in the finished tube. Early CANDU pressure tubes had hydrogen concentrations which varied from 5 ppm to 16 ppm, with a mean value of 10 ppm. Pressure tubes fabricated using current methods have a very narrow range of hydrogen concentration, below 5 ppm. The specification for the maximum hydrogen concentration allowed in pressure tubes has been changed from 25 ppm to 5 ppm, to reflect the improved practice.
b) In addition to the above measures to reduce hydrogen levels, progress has been made to ensure proper separation between the pressure tube and its calandria tube throughout the plant-operating lifetime. Early CANDU reactors had only two loose fitting garter spring type spacers between the pressure tube and the calandria tube. In the mid 1970's, the number was increased from two to four, to reduce the distance between spacers and to prevent contact between the two tubes. Later, during inspections occurring in 1983 and subsequently, the loose fitting spacers were found to have moved (during construction and commissioning,

mostly), and the design was again changed to a tight fitting spacer design. At this time, the spacer material was changed from zirconium alloy to Inconel, to ensure that the spacers themselves were immune to hydriding. The design of all subsequent CANDU fuel channels featured four tight fitting Inconel spacers. These spacers were expected to remain in their design positions over the entire life of the fuel channels.

Based on the proactive measures taken in design and manufacturing to minimize the likelihood of hydride blister formation, and the expectation that the utilities would implement an appropriate pressure tube inspection program supported by applicable fitness for service guidelines, CNSC (then AECB) staff considered that GAI 89G03 was satisfactorily resolved by AECL for CANDU 9. ACR design will make use of the approach and improvements made for the CANDU 9 design.

4.2.4 Status for ACR

A great deal of knowledge has been accumulated in pressure tube design and operation, and the ACR pressure tubes will benefit fully from the experience gained.

It continues to be a design requirement for ACR that the initial hydrogen content of the pressure tubes shall be less than 5 ppm.

Direct contact between the pressure tube and the calandria tube during the normal operating life of the fuel channels will be prevented in ACR by means of an adequate number of annulus spacers (four per channel). ACR will use tight-fitting Inconel annulus spacers similar to those proposed for CANDU 9.

To address the difference in pressure tube design and coolant conditions between ACR and CANDU 9, experiments will be conducted under temperature and coolant chemistry controlled conditions to measure hydrogen uptake rates on prototypical ACR material.

All of the above design/manufacturing features will minimize the likelihood of hydride blisters being formed. In combination with this, the utility is expected to conduct an appropriate pressure tube inspection program. They are also expected to have a program in place which can disposition inspection results, using pre-established rules for making decisions regarding continued station operation. AECL has committed to apply to ACR a number of relevant standards including the latest inspection standard CAN/CSA-N285.4 (Reference [3]) that specifies procedures and acceptance criteria for pressure tube inspection. AECL will review the COG fitness for service guidelines for applicability and support the operating utility, on an as required basis, in developing fitness for service guidelines appropriate to ACR operation based on utility experience and the latest revision of the COG guidelines.

In summary, the pressure tubes of the ACR will be designed and manufactured based on extensive experience gained in CANDU operation, so as to minimize the potential of blister formation once in service. AECL will support the operating utility, on an as required basis, in developing fitness for service guidelines appropriate to ACR operation. On this basis, the issues raised in this GAI are considered to have been resolved for the ACR.

4.3 89G05 "Use of Mercury Wetted Relays in Safety Related Systems"

4.3.1 Safety Issue (As Stated by CNSC)

"Failure rates of the 5 amp non-tin-doped mercury wetted relays were found to be unacceptably high. In 1989, CNSC (then AECB) staff had requested the licensees ensure that the reliability of the safety related systems using such relays still met their targets, or replace them. All licensees have replaced these relays." (Reference [4]).

4.3.2 Closure Criteria

No closure criteria were defined. CNSC requested licensees to either ensure that the reliability of the safety-related systems using such relays still met their targets, or to replace such relays.

4.3.3 Relevance of the GAI to the ACR

As for other recent CANDU designs, ACR will not use non-tin-doped mercury-wetted relays. Therefore, this GAI is not relevant to the ACR.

4.3.4 Status for ACR

The GAI is not applicable to the ACR.

4.4 90G01 "Loss of Oil in Rosemount Pressure Transmitters"

4.4.1 Safety Issue (As Stated by CNSC)

A CNSC (then AECB) letter (Reference [5]) addressed to Bruce Nuclear Generating Station stated:

"Certain models of Rosemount pressure transmitters, particularly Models 1153 Series B, 1153 Series D and 1154 have suffered a gradual deterioration in performance due to loss of oil from the sealed sensing module."

"We would like your assurance that these particular models of Rosemount transmitters are not used at your station in special safety or safety support systems service. (Note that some of these transmitters may have been supplied by other companies such as Bailey Controls (formerly Bailey Meter) and Fisher Controls)."

"It is understood that, currently, other versions of these transmitters, i.e. models 1151 and 1152 which are used at the Bruce site, have not exhibited this failure mode. However, we would also like your assurance that your technical staff are fully cognizant of this problem, are aware of the usual failure symptoms and are monitoring performance of these instruments."

Another CNSC (then AECB) letter (Reference [6]) on this topic stated:

"Due to the possible safety concerns that affect all Canadian reactors this topic has been designated as a Generic Action Item, number 90G01."

4.4.2 Closure Criteria

No closure criteria were defined. CNSC requested that the problematic pressure transmitters be replaced.

4.4.3 Relevance of the GAI to the ACR

This GAI is not relevant to the ACR since the problematic models of Rosemount pressure transmitters will not be employed in the design.

4.4.4 Status for ACR

While the final decision on the selection of pressure transmitters has not been made yet for the ACR, the issues raised by the GAI are considered to have been adequately dispositioned due to the fact that, by 1989, Rosemount stopped production of the faulty transmitters.

4.5 90G02 "Core Cooling in the Absence of Forced Flow (CCAFF)"

4.5.1 Safety Issue (As Stated by CNSC)

"In some postulated accident scenarios, the primary heat transport pumps are assumed to fail. In this case, removal of residual heat relies on natural circulation of the coolant. Although natural circulation with the primary heat transport system full was shown to be effective, some partial inventory natural circulation experiments done at the Whiteshell RD-14M test facility have shown degraded cooling in some channels.

At issue are these unexpected results, which were observed in two-phase natural circulation (thermosyphoning) experiments in the RD-14M test facility. RD-14M is a scaled representation of the major components of a CANDU reactor primary and secondary heat transport systems. A fraction of these tests resulted in fuel element simulator heatup at unexpectedly high coolant inventory.

It should also be noted that this GAI addresses only two-phase thermosyphoning following faults from full power. Any related issues of faults at shutdown, e.g., loss of shutdown cooling, are to be tracked as station specific action items and are not discussed here." (Reference [1]).

4.5.2 Closure Criteria (As Stated by CNSC)

"In order to achieve closure, licensees are required to demonstrate, to the satisfaction of the CNSC, that experimental results do not invalidate current safety analyses that credit core cooling without forced flow. Alternatively, licensees may review the safety analyses with the new knowledge acquired from the experiments and, where necessary, implement adequate design modifications." (Reference [1]).

4.5.3 Relevance of the GAI to the ACR

Cooling without forced flow is an issue especially for CANDU plants that have two separate HTS loops. In the event of a LOCA and pump trip, natural circulation cooling is required in the intact loop. The ACR-700 design has a single HTS loop, therefore this is not an issue, however the ACR-1000 design may have two loops. This issue will be addressed for the ACR-1000 when that design is finalized. For now, this GAI will be addressed for the ACR-700 design only.

Extensive analyses have been undertaken by the Canadian utilities in response to this Generic Action Item. Early work focussed on demonstrations that the test results from RD-14M could not be directly related to reactor behaviour. AECL was involved in that analysis.

Fundamental mechanisms of single and two-phase natural circulation are generally well understood in simple geometries. Flow rates are governed by local density differences, elevation distances and flow resistances. The effectiveness of natural circulation in maintaining adequate channel cooling for a single-channel-per-pass geometry was extensively demonstrated in the RD-14 thermalhydraulic test facility. CATHENA code simulations showed that single- and two-phase natural circulation flows were well predicted in the single-channel-per-pass RD-14 facility (Reference [7]).

Natural circulation in parallel channel geometries is considerably more complex. Tests in the multiple-channel-per-pass facility, RD-14M, revealed the occurrence of bi-directional flow under two-phase conditions. Bi-directional flow occurs when the flow reverses in some, but not all channels in the same pass. In some tests, this phenomenon is believed to be responsible for the breakdown of adequate channel cooling at high primary side inventories (>85%).

The transition from unidirectional to bidirectional flow is caused by the creation of an adverse above header pressure gradient developed in the steam generators. Analysis has identified two categories of adverse above header pressure gradients: “steady” and dynamic. In “steady” cases, a time-independent adverse pressure drop develops presumably due to the preferential distribution of steam in the cold leg of the steam generator U-tubes. The magnitude of this adverse pressure drop increases as the primary inventory level is decreased, and eventually offsets the favourable pressure gradient present in the highest, individual channel feeders causing flow reversal. There has been some success in modelling these results (Reference [8]).

The flow instabilities in the dynamic cases have been well understood. The transition from shutdown cooling to natural circulation tests has been analysed. CATHENA simulations have shown that the major phenomena seen in these cases can be predicted (7 tests have been simulated) (Reference [8], [9]).

4.5.4 Status for ACR

This issue is resolved by demonstrating by analysis that channel stagnation and/or flow reversal will not occur during the relevant events. The specific resolution criteria are:

a) Demonstration that the HTS is subcooled after the HTS pumps have run down. Channel cooling during subcooled thermosyphoning is not a concern.

b) If a period of circuit void occurs after HTS pump rundown, channel cooling will not be a concern if the circuit void is low enough or the period of two-phase flow is brief enough.

ACR-700 is a single HTS loop reactor. The coolant flow in the ACR-700 Heat Transport System is configured in the “figure-of-eight” loop arrangement used in all CANDU reactors, with the heat transport pumps in series and the coolant making two passes through the core as it completes a full circuit. The equipment arrangement results in bi-directional coolant flow through the core. The headers and feeders are arranged such that the fuel channels served by each inlet header (50% of total) are uniformly distributed throughout the core. The headers, steam generators and pumps are all located above the reactor. This ensures natural circulation of the coolant on loss of power to the heat transport pumps.

Tests in RD-14 with ACR specific conditions (larger outlet header interconnect pipe and higher pressures) have been planned to validate CATHENA’s ability to predict thermalhydraulic conditions in ACR.

In summary, the following provisions of the ACR design and safety analysis program will adequately address the safety concerns that have been raised in this GAI for other CANDU designs:

a) The ACR-700 employs a single HTS loop, which eliminates the unfailed loop thermosyphoning concern. Channel stagnation may still occur for a limited period of time for specific loss of coolant accidents but channel cooling in these events is essentially an ECCS effectiveness issue covered by GAI 94G01.
b) Tests are planned in RD-14 to validate CATHENA against ACR conditions.
c) Plant transient analysis (i.e., loss of Class IV power and one pump seizure) will be performed to verify the adequacy of the design.
d) Future analysis of LOCA plus loss of Class IV power will also be performed to support the ACR ECCS design.

4.6 90G03 "Assurance of Continuing Nuclear Safety/Management of Aging"

4.6.1 Safety Issue (As Stated by CNSC)

"*Recognizing that the effects of ageing degradation on critical systems, structures and components can result in design safety margins being diminished and safety analyses being invalidated, the CNSC has embarked on the development of a regulatory position on requirements for the management of ageing.*" (Reference [1]).

"Degradation of components and systems should be anticipated where practical and potentially detrimental changes should be systematically identified and dealt with before defence-in-depth is challenged. The identification of all risk-significant systems, structures and components and their failure criteria using a systematic method is a necessary pre-requisite to a program for the management of age-related degradation." (Reference [11]).

4.6.2 Closure Criteria (As Stated by CNSC)

"*In 1990, the CNSC required that each of the licensees submit for its review a summary of the means by which it is assured of the continued safe operation of its nuclear power station as it ages. The notice required the licensees to address the following:*

- *the continued validity of steady state and dynamic analyses of the station, where key characteristics, such as heat transfer rates and flow-rates, have changed;*
- *the scope of the review of degradation mechanisms that could impact significantly on safety, and which might therefore require changes to surveillance and testing programs;*
- *the continued validity of reliability assessments of special safety systems, safety-support and safety-related systems in light of known or anticipated changes in component failure rates; and*
- *the adequacy of the planned maintenance program.*

In the early 1990's, the CNSC issued draft recommendations for a regulatory position on the requirements for the management of ageing. The program's fundamental requirement is to ensure that degradation of nuclear power plant systems, structures and components, due to ageing, is managed such that their contribution to the risk to public, worker safety, and to the *environment, from operation of the nuclear power plant, remains within the bounds claimed and accepted as the licensing basis for the facility.*

It was recommended that the program should be auditable and provide for the effective management of:

- *ageing degradation of any component that could increase the probability or consequences of process system failures;*
- *ageing degradation of any safety-support or other safety-related system that could render a special safety system less effective or less reliable; and*
- *ageing degradation that causes key system parameters such as flow rates, heat transfer rates and pressure drop to change to the extent that they impact on the limits assumed in the Safety Report."* (Reference [1]).

CNSC staff presented a draft position statement on GAI 90G03 to the Canadian industry in a meeting held in May 2000. A final position statement has not yet been issued.

In the draft position statement, the following expectations were outlined for this action item in connection with the establishment of a systematic and integrated program for managing the ageing of important safety components:

"Program Documentation:

Should include

- *the organization responsible for implementation of the program*
- *identification of roles and responsibilities of different groups directly involved with the program*
- *the means of assuring the adequacy and completeness of the program*

Work Required:

- *Assessment*
    - *risk-significant systems, structures and components should be assessed to identify and evaluate potential age-related degradation mechanisms and their rates*
    - *the consequences to safety of the failure caused by this age-related degradation should be identified.*
- *Monitoring*
    - *surveillance activities should be performed on all risk-significant systems, structures and components which have been identified as susceptible to age-related degradation*
    - *surveillance activities should be carried out at a frequency that is appropriate for detecting the presence and progress of potential age-related degradation*
- *Mitigation*
    - *activities of calibration, maintenance, or replacement should be performed at intervals which give assurance that any age-related degradation is mitigated before risk-significant systems, structures and components become incapable of meeting their defined specification.*

- *Recording*
    - *the results of the assessment, monitoring and mitigation of age-related degradation should be recorded and analysed.*
    - *the results of the assessment should ensure that the predictions of age-related degradation accurately reflect the actual performance of the risk-significant systems, structures and components*
    - *the analysis should enable the detection of any trends.*
- *Reporting*
    - *the results of the ageing management program should be reported at least annually*
    - *information which should be reported should include the description of program activities during the year, discovery of any age related degradation or failures, and any changes to the program scope or status."* (Reference [11])

4.6.3 Relevance of the GAI to the ACR

This GAI is directly relevant to the ACR and the approach to resolution will fully account for the experience gained over three decades of operation in previous CANDU designs.

The ACR is designed to achieve an operating life of 60 years. Key factors that will support achievement of the plant design life are:

- Design practices and operational requirements that will ensure that non-replaceable components operate in the design environment for 60 years.
- The plant design, supported by research, development and testing, will ensure that all components in the ACR plant will last for the operating life or can be replaced in a cost-effective manner.
- Utilization of life assurance and monitoring programs for critical components that will predict the behaviour of such components and establish requirements for rehabilitation, refurbishment and replacement.

This GAI establishes expectations for the establishment of a systematic and integrated program for managing the aging of important safety components throughout the plant design life and is therefore directly related to the achievement of the plant design life objective.

4.6.4 Status for ACR

ACR will be building upon the experience from existing CANDU Plant Life Management (PLiM) and Plant Life Extension (PLEx) programs to ensure the 60 year operational life is achieved. ACR has employed an operations and maintenance consultant, previously responsible for the plant ageing program at Darlington, to aid the ACR project to prepare a reliability centred maintenance (RCM) program. The RCM program is a maintenance program for all of the safety critical and economically important small equipment (i.e. valves). A detailed list of the important equipment is produced, maintenance activities are selected with their rationale, and any specialty maintenance rooms, tools etc. that are necessary for maintenance, replacement or testing of these components are included in the design. In addition, in the design phase, provisions are made to ensure good accessibility to equipment that will require maintenance, replacement and/or testing.

The fuel channel assemblies are designed to allow for replacement. This is planned to take place on a large-scale for reactor refurbishment at reactor mid-life, but could also be carried out on individual channels at anytime.

Major, non-replaceable equipment (i.e. turbines, steam generators) are considered as part of the PLiM program. Under PLiM, inspection and repair programs are prepared for these pieces of equipment.

In this way, the overall ageing program is ready to be implemented on the first day of operation.

AECL's design feedback procedures and operating instructions are established in compliance with the requirements of Canadian standard N286.2 (Reference [12]) which calls for measures to ensure that operational experience (including information relevant to plant ageing effects) is documented, assessed and incorporated into the design and operation of the plant and/or its QA programs as appropriate. It also calls for making this available to personnel in other phases of the plant's life cycle.

To meet these objectives, AECL has established a centralized Feedback Monitoring System (FMS) to address the feedback received from others, the evaluation status, and providing feedback to users. Governing procedural documents have been established, which describe the feedback processes and the corresponding interface mechanisms, including:

- Feedback to Design
- Plant Operations Feedback to Design
- Regulatory Feedback
- Engineering Experience from Built Projects
- Feedback of R&D into the CANDU Product
- Market Feedback

It is expected that continuing activities in this area for ACR will be sufficient to fully address the closure criteria for this GAI.

4.7 91G01 "Post-Accident Filter Effectiveness"

4.7.1 Safety Issue (As Stated by CNSC)

"Following a loss of coolant accident the emergency filtered air discharge system (EFADS) in multi-unit stations would be used to maintain containment pressure below the atmospheric level in the long term. The EFADS filters are relied upon to limit radioactive releases during the venting of containment. The single-unit stations do not depend on filtered air discharge systems to ensure the effectiveness of their containment systems. However, filtered venting at these stations may be considered as a long term post-accident management option. If during venting the filters do not perform as designed, this would result in a higher than expected public hazard. It is essential, therefore, that all licensees have programs to support the credited effectiveness of filters and to ensure prompt detection of any deficiency which could prevent the filters from performing as designed." (Reference [1]).

4.7.2 Closure Criteria (As Stated by CNSC)

"To achieve closure of this GAI, licensees are expected to have effective programs in place which will provide a continued assurance that under all credible post-accident conditions filter effectiveness will match or exceed that credited in safety analyses. These programs may, as necessary, consist of laboratory and in-situ tests, research and analytical efforts, adequate QA, testing, maintenance, and operational procedures and should:

- *ensure that there are appropriate tests for all filter elements;*
- *demonstrate that tests are performed under representative conditions;*
- *provide evidence confirming that filters are capable of performing as credited under harsh post-accident environmental conditions;*
- *provide an assurance that filter degradation can be reliably detected by existing tests;*
- *address consequences of hydrogen burns on filter performance;*
- *confirm that there are proper QA, maintenance and operation procedures in place;*
- *justify frequency of tests;*
- *provide an assurance that instrumentation and control equipment is such that it will allow operation of filters as designed under all representative conditions;*
- *demonstrate the adequate availability of filters; and*
- *identify areas, if any, where continued design, experimental and/or analytical efforts are required."* (Reference [1]).

4.7.3 Relevance of the GAI to the ACR

This GAI applies primarily to operating CANDU plants with vacuum buildings, where the performance of filtered air discharge systems is credited as part of the safety case. For the single-unit stations that do not depend on filtered air discharge systems to ensure the effectiveness of their containment systems, the licensees are also requested to assure that the potential use of installed filters as part of post accident management strategies do not result in a higher than expected public hazard. Since ACR will not employ a vacuum building as part of its containment design, this GAI is not relevant.

4.7.4 Status for ACR

The ACR containment design is based on the containment design of CANDU 9. A steel-lined pre-stressed concrete containment with a target leak rate of no more than 0.2 vol%/day is provided. Like the CANDU 9 design, this will also preclude any immediate need to perform filtered discharge to control leakage after an accident.

There is no intent to include a containment venting system as part of the safety case for design basis events, and as such, there will be no concern with respect to Post-Accident Filter Effectiveness. The accident analysis will show that doses to an individual at the exclusion zone boundary will be within prescribed limits.

4.8 91G02 "Operation with a Flux Tilt"

4.8.1 Safety Issue (As Stated by CNSC)

"The adequacy of Regional Overpower Protection (ROP) or Neutron Overpower Protection (NOP) trips for reactor operation with a flux tilt is demonstrated by analyses, which take into account different plant states for which continued operation is permitted.

ROP/NOP system design is based on information derived from simulations of certain reference and perturbed flux shapes in the reactor core. Trip setpoints are established from these simulations to prevent any channel reaching its critical power limit in case of a bulk loss of regulation. One key component in the analysis is the relationship between flux values at detector locations and channel powers for various flux shapes. The analyses assume that the ratios of changes in fluxes and channel powers due to perturbation, called simulation ratios, are invariant with respect to the reference flux shape. On this basis a limited number of combinations of credible perturbed flux shapes and an untilted reference flux shape is analyzed to derive trip setpoint values. Furthermore, these trip setpoint values provide coverage for different plant states. Differences in the reference flux shape used in the analyses and actual flux shapes are accounted for by regular detector calibration.

At issue is the adequacy of error allowances used to derive the trip setpoints values, since these can be sensitive to the accuracy of reference and perturbed flux shapes and associated simulation ratios. CNSC staff identified in 1991 an apparent deficiency in licensees' analyses and practices: while tilted-flux operations were not considered as initial conditions in ROP/NOP analyses, the reactors are permitted to operate with relatively large flux tilts. For initial flux shapes with relatively large tilts the invariance of simulation ratios needs to be demonstrated to support the adequacy of error allowances used to derive the trip setpoint values for all plant states for which continued operation is permitted. In order to demonstrate the adequacy of ROP/NOP trip setpoints, licensees were requested to:

- *determine the maximum tilt permitted by the current operating procedures for prolonged operation with a flux tilt, prior to any operator's action;*
- *generate a steady state flux distribution, corresponding to the maximum tilt permitted by the current operating procedures, and design-basis and abnormal perturbation flux shapes, corresponding to this steady state shape; and*
- *assess simulation ratios (ratios of changes in fluxes and channel powers due to perturbations) for the above flux shapes, and assessment of the ROP/NOP trip coverage by determining whether the ratios are invariant within any postulated error allowance."* (Reference [1]).

4.8.2 Closure Criteria (As Stated by CNSC)

"To achieve closure, licensees were required to perform the following:

- *determine the adequacy of various error allowances and assumptions, such as the invariance of simulation ratios used in the analyses, to cover the actual maximum tilt permitted by the revised and improved operating procedures: there should be a specific error allowance for simulation ratio variation due to flux tilts in ROP/NOP error analysis, its magnitude should* *be sufficient to cover variations found in analyses that have been performed, as well as other cases which are not analyzed;*

- *determine the sensitivity of ROP/NOP analysis results to flux tilt definition; at issue is the fact that while the RRS flux tilt definition is based upon bulk regional averaged parameters, the ROP/NOP requirement is to prevent the critical power in any individual channel being reached; and*
- *determine the potential impact of factors not fully covered in current analyses on the effectiveness of the ROP systems; the main factors not completely covered are: the effect of transient xenon, boiling in a region of channels, and replacement of a failed detector."* (Reference [1]).

4.8.3 Relevance of the GAI to the ACR

The ACR design incorporates spatially distributed in-core detectors on each of Shutdown Systems 1 and 2 to provide the overpower trip function, consistent with the approach employed over the past two decades in previous CANDU designs. The tight neutronic coupling in the relatively small ACR core (smaller than the current CANDU 6 design) and the negative power coefficient provide exceptional stability with respect to flux tilts. The ACR reactor has no unstable spatial xenon oscillations. These ACR design features result in much lower likelihood of operation with flux tilts than in previous CANDU designs. Nonetheless, basic tilt control (side-side, top-bottom and end-end) is provided in the ACR core to guard against tilts caused by refuelling or other effects and the potential for flux tilts will nonetheless be considered in the design and analysis of regional overpower trip systems.

The determination of detector layout and the analysis of required trip setpoints for the Regional Overpower trip systems will be performed using methods and tools similar to those used in previous CANDU designs. As part of the design process, it will be shown that tilted flux shapes are adequately addressed.

4.8.4 Status for ACR

This GAI has been closed for all Canadian reactors. Licensees responded to the issues identified above by performing additional analyses and implementing operating rules and restrictions (e.g., flux tilt limits, tilt alarms and modified fuel management practices) to improve confidence in the trips in the existing ROP systems during operation with a tilted flux distribution. CNSC staff concluded that the operating rules implemented at various stations to limit permissible flux tilts have greatly enhanced the confidence in the adequacy of the ROP setpoints. Moreover, in AECL's analysis of the CANDU 6 ROP systems, a large number of configurations (tilted core with a subsequent design basis or abnormal perturbation) were analysed using AECL's standard ROP methodology. This provided significant confidence that there were no material inadequacies in coverage for tilted core states.

To demonstrate that the ROP trip system of the ACR is robust, the detector layout will be assessed for non-design basis perturbations including perturbations from a tilted flux distribution. Expert judgement is used to select perturbations from initially tilted distributions which represent the greatest challenge to the ROP system. This includes the effect of flux tilts on highly peaked shapes which may depend on only a few detector locations for coverage. There are a number of measures that can be implemented at the design stage to address issues related to tilted operation. These include modifying the detector locations in an assembly, modifying the number and location of assemblies, increasing the number of detectors per trip channel, increasing the number of flux shapes included in the design basis set, optimizing the ROP reference channel power distribution, changing handswitch selection rules and annunciating flux tilts. These design steps provide significant assurance that trip coverage for tilted operation will be acceptable.

AECL will be fully addressing the issues raised in this GAI using the same approach and proven methodologies employed to successfully close this issue for CANDU 6 plants operating in Canada.

4.9 92G01 "Treatment of Human Factors in Ontario Hydro Reliability Analyses"

4.9.1 Safety Issue (As Stated by CNSC)

"In 1991, OPG (then OH) submitted a reliability analysis in support of a proposed design change on the Pickering filtered air discharge system. The reliability study included an assessment of human reliability. The CNSC (then AECB) had concerns regarding the methodology used in the human reliability assessment, and has raised them as a generic action item based on the request of the licensee. CNSC has also initiated a research project to assess OPG's human reliability methodology called Technique for Human Error Rate Prediction (THERP). The results of this assessment confirmed the acceptability of this methodology. CNSC is also monitoring OPG human factor analyses on a case by case basis." (Reference [1]).

4.9.2 Closure Criteria

No closure criteria were defined. CNSC requested licensees to ensure that acceptable methodology was used in making quantitative estimates of human reliability.

4.9.3 Relevance of the GAI to the ACR

Human reliability assessment is an integral part of the ACR design process. Well-established human reliability modelling methodology will be employed in making quantitative estimates of human reliability, such that the issues raised in this GAI can be considered to have been adequately addressed.

An important aspect of any Probabilistic Safety Assessment (PSA) is the analysis of the human actions, commonly referred to as human reliability analysis (HRA). Given the high degree of hardware reliability and redundant design associated with nuclear power plant systems, human interaction with the systems are often significant contributors to system unavailability. The purpose of human reliability analysis is to identify potential human errors and to quantify the most significant of these. This covers the analysis of the human actions of potential concern identified during the PSA process.

The GAI focuses on methods used to make quantitative estimates of human reliability. Analysis methods were presented to and discussed with the CNSC. While further improvements in human reliability modelling were deemed desirable in the longer term, the generic issue was closed.

The procedures for incorporating human interactions into PSA studies, and the associated data requirements, are well documented and studied, e.g., for CANDU 9.

4.9.4 Status for ACR

The human reliability analysis methodology to be used for ACR is based on previous work in this area done in AECL and on industry accepted methods and guidelines.

For pre-accident and post-accident diagnosis, and in part for recovery human actions, HRA methodology is based on the experience accumulated during the PSA analyses for CANDU plants and for other facilities. Particular attention in the ACR PSA is given to modelling the post-accident execution errors in line with the internationally recognized Accident Sequence Evaluation Program (ASEP) HRA procedures (Reference [13]). Where necessary to meet CANDU-specific requirements and practice, these procedures are adequately modified. The overall impact of human reliability on the ACR will be derived from modelling of human error in the event trees which indicate ultimate plant response to an initiating event. AECL completed a Generic Probabilistic Safety Assessment (PSA) Methodology and Reference Analysis, which includes a description of methods for modelling human actions.

Human Factors considerations will be systematically incorporated into the design of the ACR. All aspects of plant design for which there is an interface with plant personnel will incorporate Human Factors considerations. Human Factors driven design features will be applied consistently plant-wide under all postulated plant regions and system states.

A Human Factors Engineering Program Plan (HFEPP) was prepared to establish how human factors activities or tasks are to be carried out for the ACR. The ACR HFEPP was developed with input from guidelines contained in a CANDU Owners Group guide and CNSC Human Factors Guides (Reference [14]). The ACR HFEPP is based on evolutionary design processes that seek to evolve and improve the design performance of the plant.

In summary, the issues raised in this GAI have been addressed in the ACR design. The HRA methodology for ACR is based on internationally recognized procedures and previous experience. A HFEPP was prepared to establish how human factors activities or tasks are to be carried out for the ACR.

4.10 94G01 “Best Effort Analysis of ECCS Effectiveness”

4.10.1 Safety Issue (As Stated by CNSC)

“The effectiveness of the ECCS has not been fully demonstrated for any CANDU nuclear power station to the satisfaction of the AECB. Computer codes may not be sophisticated enough to predict all relevant phenomena occurring during LOCA. Furthermore, the available experimental database was not adequate to demonstrate that the system would be fully effective. In the early 1980s, AECB asked the licensees to develop improved computer codes to predict, with sufficient confidence, the effectiveness of the ECCS during a LLOCA. In 1993, the licensees submitted their Best Effort Analysis which relied on improved computer codes. This GAI requested a demonstration of adequate validation of these methods and codes.

The licensees have initially concentrated their validation efforts on the first few seconds of the LLOCA. They have submitted a number of validation reports for phenomena occurring in the *first few seconds. The future plans aim at validating codes for the remainder of the event."* (Reference [4]).

4.10.2 Closure Criteria

No closure criteria were defined. To achieve closure, the addressees of this GAI were asked to adequately validate the computer codes used to predict ECCS effectiveness.

4.10.3 Relevance of the GAI to the ACR

The GAI was addressed to Ontario Power Generation (OPG) and Bruce Power only (both were then part of Ontario Hydro), as they had submitted 'limit consequence' analyses as part of the original LOCA safety case. Limit consequence analyses relied heavily on the moderator as a heat sink and minimized reliance on the ECC system. Subsequent 'best-effort' analyses submitted to demonstrate ECCS effectiveness relied on improved analysis tools and methods, but validation was incomplete. The CANDU 6 Canadian licensees were not asked to address this GAI.

The scenario of the ECCS behaviour following a Large LOCA is well understood. The rate of refilling core passes depends on a few factors, such as whether the HT pumps are operating or not, and the locations of core passes (upstream or downstream of the break).

In the analysis of CANDU 6 and CANDU 9 designs, AECL uses a more traditional deterministic safety analysis of LOCA with ECC. Best-estimate models (incorporated in codes) are combined with conservative assumptions on input data and plant state. A "limit consequence" analysis is not used.

Since the prediction of the effectiveness of the ECCS during a LLOCA is an important part of safety analyses for ACR, AECL will ensure the adequate validation of the methods and computer codes used for the prediction of the effectiveness of the ECCS during a LOCA. The validation of the computer codes will meet the requirements of GAI 98G02 with respect to the validation of computer programs used in safety analysis of power reactors.

4.10.4 Status for ACR

CANDU 9 addressed this GAI by demonstrating the ability of CATHENA, the computer code used to calculate the effectiveness of the ECC system at refilling the reactor channels, under LOCA conditions. The CNSC staff were satisfied with AECL's progress for CANDU 9 on this issue (Reference [10]). Since then, a considerable amount of additional validation work has been performed for CATHENA to meet the requirements of GAI 98G02.

To resolve this GAI, ACR will follow the approach taken for CANDU 9. The calculations of ECCS effectiveness for all LOCA cases considered for ACR will be performed using the CATHENA code. A comprehensive validation program was conducted for CATHENA as used in CANDU safety analysis, and the documentation was submitted to the CNSC (see GAI 98G02, Section 4.20). AECL will ensure that CATHENA will be adequately validated for the ACR design conditions.

4.11 94G02 "Impact of Fuel Bundle Condition on Reactor Safety"

4.11.1 Safety Issue (As Stated by CNSC)

"The condition of certain fuel bundles irradiated in CANDU reactors has been observed to differ from that predicted and accounted for in design, operation, and safety analysis documentation. The fuel bundles in question have shown signs of more-than-expected degradation such as end plate cracking, spacer pad wear, element bowing, sheath wear, bearing pad wear, sheath strain, disappearance of CANLUB layer, oxidation of defective fuel, and fission product release.

Fuel bundle degradation depends on the reactor design, fuel channel, fuel design, fuel manufacturer, and operating conditions. Since theoretical models have been unable to correlate these factors adequately to the fuel condition, fuel and pressure tube inspections are necessary. Owing to the number of factors upon which the degradation depends, the inspection program must be extended beyond inspection of defective fuel to observe these changes. Such expanded inspections, when done at Bruce A and B, showed that degradation has been occurring throughout the life of Bruce B.

Fuel bundle degradation is sometimes also accompanied by fretting and scratching of the pressure tube, these being superimposed onto other changes, such as creep.

The effect(s) of some of this bundle degradation on reactor safety is not known partially because of a lack of experiments and safety analysis methods to account even for changes to pressure tube geometry as the fuel channel ages. As such, the important fuel and fuel channel parameters to measure are not known.

Although some fuel inspections have been conducted and the results submitted to the CNSC, licensees do not have a formal process to ensure that the fuel and fuel channel conditions are identified and accounted for." (Reference [1]).

4.11.2 Closure Criteria (As Stated by CNSC)

"To achieve closure, licensees are required to perform the following:

- *Implement an action plan to eliminate excessive fuel and channel degradation in acoustically active channels (where applicable).*
- *Implement an effective formal and systematic process for integrating fuel design, fuel and channel inspection (in situ), fuel and channel laboratory examination, research, operating limits and safety analysis. This process must have the following features:*
  - *annual review (by the licensee) to demonstrate effective implementation and adequate corrective actions taken for deficiencies identified in the review;*
  - *sufficient resources for each participating group (design, inspection, examination, research, safety analysis, and operation) to ensure that the fuel condition is known and accounted for adequately;*
  - *clearly defined maximum allowable limits, under normal operation, on fuel condition in terms of sheath strain, element bowing, wear (spacer pad, bearing pad, end plate), pressure tube scratching and wear, burnup and residence time; design documentation and pressure tube fitness-for-service guidelines should be updated accordingly;*

- *a determination, for the full range of the operating envelope, the power boost sheath failure threshold for CANLUB fuel and the chemistry effects of CANLUB on centerline temperature and fission product release;*
- *assurance that the safety analysis accounts for the allowable fuel condition when combined with aging effects such as pressure tube creep, the effect of CANLUB in the fuel, and any chemistry effects on temperature and fission product release, including a calculation of the number of sheath failures resulting from a bounded loss of power control; and*
- *a surveillance program that demonstrates compliance with identified limits, e.g., detection of significant changes in fuel condition caused by changes in fuel fabrication and factors affecting acoustic resonance."* (Reference [1]).

4.11.3 Relevance of the GAI to the ACR

Previous CANDU designs have experienced fuel bundle deformation and pressure tube fretting. Some of these effects are due to pressure pulsations associated with Heat Transport System acoustics. This has pointed to the need to ensure that fuel bundle design has considered all possible in-service degradation mechanisms and that design, operations and safety analysis documentation account for these effects. It has also driven the requirement that a comprehensive in-service fuel bundle and fuel channel inspection program be implemented by the plant owners. The ACR design has taken into account the experience gained from previous CANDU designs and incorporates features that will minimize the potential for acoustics-initiated fuel bundle or pressure-tube degradation during routine operation. The fuel, fuel channel and heat transport system for the ACR are modified designs that build upon extensive experience derived from over three decades of CANDU plant operation. Acoustic analysis will be performed, supported by testing, to demonstrate acceptability of the modified design.

4.11.4 Status for ACR

This GAI has been closed (or closure is expected shortly) for all of the Canadian Utilities based on implementation of a comprehensive in-service fuel and fuel channel program that integrates inspections (including post-irradiation hot cell examinations), operating conditions, research results, design feedback and safety analysis.

AECL supports regular, periodic post-irradiation examination of fuel bundles from each reactor to detect evidence of degradation, and of fuel channels to detect flaws. AECL will support the ACR plant owner in establishing an effective fuel and fuel channel program that provides for ongoing collection of operating, R&D, manufacturing and inspection information relevant to the ACR. AECL will support evaluation and integration of this information to identify trends (i.e., those which may be precursors to safety or performance issues) and the impacts of these trends such that corrective action may be taken by the plant owner as required.

The overall approach being adopted by AECL for the ACR is to produce a design that will take into account extensive experience from previous CANDU designs so the susceptibility to acoustics-based fuel bundle or pressure tube degradation is minimised. The acoustic characteristics of the heat transport system will be addressed to ensure that potential problems such as excessive fuel bundle fretting and end-plate cracking are eliminated. An extensive testing program in a test rig of a full ACR channel is planned for ACR CANFLEX fuel including:

- Endurance tests for pressure tube and bundle fretting with ACR pressures, temperatures and flows accurately reproduced;
- Refuelling impact tests;
- Cross flow tests;
- Pressure drop tests; and
- Sliding wear tests.

In addition, there are plans to study the sheath corrosion at the higher temperatures expected in ACR and to simulate the power boost seen during refuelling with irradiation tests. Acoustic resonance analysis will be performed via adaptation of well-established methods.

To demonstrate that fuel bundle performance in the ACR is acceptable, AECL will ensure that its R&D, engineering testing and qualification programs provide appropriate experimental evidence to support the assessment, identification and management of fuel bundle degradation and demonstrate fuel bundle integrity under ACR operating conditions. Operating limits will be developed as required to establish a clearly defined Safe Operating Envelope for the plant operator. Inspection criteria (for both the commissioning and in-service phases of plant operation) will be developed to allow examination and monitoring of the relevant degradation mechanisms.

This GAI is addressed in the design and operation of ACR as described above. AECL will provide the plant owner with the full support necessary to implement an ongoing fuel and fuel channel inspection and maintenance program compliant with the requirements of this GAI.

4.12 95G01 "Molten Fuel/Moderator Interaction"

4.12.1 Safety Issue (As Stated by CNSC)

"A single channel event, such as a severe flow blockage or a stagnation break of an inlet feeder, could reduce flow through the channel without detection by the reactor regulating or shutdown systems. If the flow blockage is severe (>90%), then the coolant in the channel will boil-off, and melting of cladding, spacers, and fuel would occur. The accident proceeds rapidly because the reactor remains at full power. This situation leads to an accident scenario where the pressure tube (and subsequently the calandria tube) may fail at high pressure (10 MPa), ejecting the melt into the moderator.

The safety reports for currently operating CANDU plants account for the damage that the reactor sustains from spontaneous rupture of a fuel channel. Of concern is whether molten fuel/moderator interaction in a loss of flow event can cause additional damage that prevents the shutdown of the reactor, trigger rupture of other fuel channels, or fail the calandria vessel." (Reference [1]).

4.12.2 Closure Criteria (As Stated by CNSC)

"This generic action item will be closed when, following the tests:

- *the dominant mode of interaction can be determined, using the resulting pressure history and particle size distribution; or*
- *in the event that the experimental results are inconclusive, the safety margin or potential damage resulting from the interaction, regardless of its mode, is determined, using the measured pressure transient (signature) following the pressure tube rupture as the primary tool for the evaluation."* (Reference [1]).

4.12.3 Relevance of the GAI to the ACR

The sequence of events during a postulated flow blockage accident in an ACR does not change from that of existing CANDU designs. However, the details of events will change. The extent of damage to in-core structures depends on the amount of molten material in the channel at the time of pressure tube failure and it is possible that there will be more molten material produced in an ACR than a CANDU 6 reactor. Thicker pressure tubes, the use of CANFLEX type fuel, and a flatter axial channel power profile in the ACR may result in a change in event timing prior to channel failure. In addition, the lattice pitch and fuel channel design may have an impact on the extent of in-core damage. Note that for ACR, the potential reduction in shutdown depth for SDS1 due to in-core shutoff rod guide tube damage should not an issue since the discharge of light water reactor coolant into the heavy water moderator will keep the reactor sub-critical following in-core rupture events.

4.12.4 Status for ACR

Historically, analyses submitted by the licensees indicated that the mode of fuel-moderator interaction would be forced interaction as opposed to free interaction. The forced interaction model postulates that the melt is ejected from the channel at sufficiently high velocity to be fragmented and rapidly quenched within milliseconds of the ejection. The fragmentation and dispersion were expected to occur due to hydrodynamic processes. Fragmentation and dispersion were also expected to occur due to mechanical impact on adjacent in-core structures. The free interaction model postulates that the ejected melt first accumulates outside the channel as a coarse mixture of melt and water. This coarse mixture is then triggered by a shock to disperse into fine fragments which quench rapidly, causing a more severe pressure pulse than that generated by forced interaction.

An industry task group examined the modelling of energetic fuel-coolant interactions and the available data that supports the models. This group concluded that both the theory and the available CANDU-specific test data supported the conclusion that forced fuel-coolant interaction will always occur following a CANDU single channel event. However, CNSC staff have taken the position that free interaction cannot be ruled out when relatively large amounts of molten material are ejected into the moderator.

The industry concluded that the most effective way to address CNSC staff concerns was to undertake experimental work on molten material/moderator interaction. The Molten Fuel Moderator Interaction (MFMI) experimental program is being undertaken jointly by AECL and the Canadian utilities and is planned to be completed in 2005. These experiments will permit AECL to validate generic aspects of its MFMI model and quantify important parameters. In their earlier review of CANDU 9 licenseability (Reference [10]), CNSC staff agreed that such experiments could generate sufficient information on important parameters such as the heat conversion factor and the effect of rupture geometry and could provide confidence that steam explosions can be ruled out. In the case of the CANDU 9 licenseability review, CNSC staff considered that AECL's participation in this experimental program was an acceptable approach to resolving the issue raised by this GAI.

As part of the process of defining an acceptable test program, the Canadian nuclear industry and the CNSC jointly convened an Independent Expert Advisory panel of three internationally-renowned fuel-coolant-interaction experts to review the proposed test program. The expert panel endorsed the final experimental program as well as the success criteria. The CNSC staff agreed that if the tests are performed as proposed and the results meet the success criteria, GAI 95G01 can be closed.

The molten fuel moderator behaviour for ACR will, to a large degree, be addressed within the planned industry activity and experimental envelope. This conclusion will be confirmed following completion of preliminary safety analyses of single channel events for ACR and the validation program for the TUBRUPT-IST code, which is used for in-core damage analyses (refer to the assessment of GAI 98G02, Section 4.20, for additional details of the planned validation activities for this code). ACR differs from current CANDUs in terms of the quantity of molten material (ACR will likely have more) and the internal pressures (ACR will be higher). The current round of tests planned include sufficient molten material to cover off that expected for ACR, but not the higher internal pressures expected. There are ACR specific tests planned in the long term (approximately 3 years) to address the higher pressures. It is expected that the higher pressures will help promote the forced interaction mode.

4.13 95G02 "Pressure Tube Failure with Consequential Loss of Moderator"

4.13.1 Safety Issue (As Stated by CNSC)

"At issue are the consequences of rupture of a pressure tube in an operating CANDU reactor, and the interpretation of fuel channel burst tests conducted by the industry.

Rupture of a pressure tube in an operating reactor could result in one or more of the following:

- *a loss-of-coolant accident (LOCA) inside or outside the reactor core;*
- *a breach of the moderator boundary leading to a loss of moderator heavy water (LOM); and*
- *damage to reactor systems, structures and components, including adjacent fuel channels, reactivity control mechanisms, the calandria, and ejection of fuel bundles into the calandria and/or the reactor vault.*

Analyses of such events are presented in the Safety Reports for each plant. However, the Safety Reports neglect scenarios involving a LOCA and a large loss of moderator (LOCA/LOM). Despite the fact that LOCA/LOM events are neglected, the industry approach to analysis of failures with ECC unavailable has been to credit the moderator as the ultimate heat sink for the reactor.

Events that combine a LOCA with a loss of moderator would potentially invalidate the conclusions of the safety reports. OPG has stated that "the ejection of an end fitting, in conjunction with the calandria tube failure, would seriously threaten the capability of the moderator as a heat sink if ECC is not available, because the moderator loss rate would far *exceed the available makeup rate". The absence of an effective heat sink could lead to a severe accident in which the fuel attains high temperatures and releases large quantities of radioactivity into containment. Furthermore, the results of fuel channel burst tests conducted by the industry suggest that pressure tube rupture events leading to a large loss of moderator are more probable than previously assumed. Out of about a dozen burst tests, several pressure tubes have suffered a guillotine break.*

It should also be noted that similar consequences may also occur in the case of a feeder stagnation break, especially if the initial break size is increased as a result of the channel rupture. Although most of the arguments presented in this statement address tests and observations done mainly by OPG, CNSC staff considers them applicable to all currently-licensed CANDU reactors." (Reference [1]).

4.13.2 Closure Criteria (As Stated by CNSC)

"To achieve closure, licensees are expected to:

- *demonstrate that the hydrogen mitigation measures are such that the integrity of the calandria and the containment are assured;*
- *submit proposals for a course of action that would result in the reduction in the risk associated with such an event; and*
- *submit the following, in the event that cost-benefit arguments are used in support of the proposals mentioned above:*
  - *a description of the cost-benefit assessment process;*
  - *the cost-benefit tools and associated documentation;*
  - *the consequence assessment methodology;*
  - *the consequence assessment results;*
  - *an examination of the various options (e.g., design, procedural,...) for event mitigation;*
  - *studies on pressure and calandria tube failures and end fitting ejection; and*
  - *the final cost-benefit analysis report."* (Reference [1]).

4.13.3 Relevance of the GAI to the ACR

The accident scenario of interest involves the spontaneous rupture of a pressure tube (PT) followed by failure of the calandria tube (either immediately due to initial pressurization/impact or later due to excessive strain arising from high-stress diametral creep), followed by guillotine failure of the PT, followed by ejection of one or both of the fuel channel end fittings. The ejection of the end fitting(s) coupled with the rupture of the calandria tube would allow moderator to drain from the calandria. The concern for current CANDUs is if this event sequence were coincident with the failure of the Emergency Core Cooling (ECC) system, then there would be inadequate cooling for the fuel channels. However, ACR has moderator makeup from the reserve water tank so even in the event of loss of ECC, moderator cooling would be maintained for some time. A detailed evaluation of this issue was recently undertaken by a Canadian nuclear industry task team as part of the program to resolve this GAI for existing CANDU reactors.

Two basic approaches are under consideration by existing CANDU utilities to address the LOCA/LOECC/LOM issue: 1) reduce the probability of some or all of the events (by design modifications and/or operating procedure changes) such that the entire sequence can be categorized as a severe core damage accident with low probability rather than a design basis accident, or 2) provide a system to replace the moderator that could be lost from the calandria in the scenario, thus reliably maintaining a heat sink for the affected channels.

4.13.4 Status for ACR

The ACR design of the fuel channel will incorporate features to significantly reduce the probability of CT failure through design changes so the entire sequence can be categorized as a severe core damage accident with low probability rather than a design basis accident.

In existing plants, in the event of a spontaneous pressure tube (PT) failure, the discharge of coolant through the PT crack will impact on the calandria tube (CT) causing a large symmetric pressure loading. Lateral loads on both the CT and PT in opposite directions from these interactions will cause some CT strain. Should the CT fail, there will be large lateral loads on the PT due to the discharge flow into the calandria. These loads may lead to circumferential cracking and guillotine PT failure resulting in ejection of the end fitting.

If the CT does not fail, the likelihood of a PT guillotine failure is significantly reduced.

The ACR fuel channel design includes a CT which is significantly thicker (2.5 mm vs. 1.4 mm) than the CANDU 6 CT. The manufacturing process for the CT will be selected to deliver properties such that the probability of CT failure in the event of spontaneous PT failure will be low. Preliminary analysis indicates that this can be achieved with a substantial margin. The ACR development program will include a test program to confirm analysis predictions.

Even if the ACR CT were to fail, the reserve water tank can provide makeup to the moderator, maintaining moderator cooling in this scenario.

In summary, the ACR design will reduce the probability of some or all of the events such that the entire sequence of LOCA+LOM+Loss of ECC can be categorized as a severe core damage accident with low probability rather than a design basis accident. Severe core damage accident management provisions implemented for the ACR will provide assurance that the beyond design basis event can be adequately managed. Therefore, it is considered that the issues identified by this GAI will be adequately addressed in the ACR design.

4.14 95G03 “Compliance with Bundle and Channel Power Limits”

4.14.1 Safety Issue (As Stated by CNSC)

“The limiting values for bundle and channel powers are specified in the Operating Licence for each station.

Licensees ensure compliance with these limits by following operating procedures, which are based on analyses. However, current validation of the channel and bundle power analyses method is such that the errors associated with their calculations are not well-defined. If larger allowances for uncertainties were needed, channel or bundle power may become more limiting than bulk power and derating may, therefore, be required.

At issue are several areas both in the compliance analyses and procedures where, in the view of the CNSC staff, improvements are needed to ensure adequate compliance with bundle and channel power limits.” (Reference [1]).

4.14.2 Closure Criteria (As Stated by CNSC)

“To achieve closure of the generic aspects of this issue, licensees are required to complete their related code validation program and perform additional analyses to address the identified issues related to methodology, models and computer codes. Specifically, the following should be performed:

- *determining the adequacy of error allowances and level of confidence, to cover the various sources of uncertainty in codes predictions, plant data measurements and methodology; allowances should, at 98% confidence level, account for:*
    - *error in total reactor power normalization;*
    - *error in code methodology and modelling;*
    - *error in measurements (instrumented channels and flux detector/mapping); and*
    - *xenon transients initiated by fuelling.*
- *determining the adequacy of the validation of computer codes used, for core-tracking and compliance with licence power limits, consistent with validation plans for other safety and licensing codes; and*
- *evaluating the acceptability of compliance procedures, addressing in particular:*
    - *consistency;*
    - *actions when limits exceeded; and*
    - *assurance of compliance during periods of time at which core-tracking runs are made.”* (Reference [1]).

4.14.3 Relevance of the GAI to the ACR

This issue is primarily an operational one, related to the validation of the codes used to predict bundle and channel powers and the methods employed to demonstrate compliance with specified limits. Channel and bundle power limits are used in safety analyses as initial operating conditions and safety analysis results are valid only if these limits are not exceeded. Compliance with these limits is important to safe operation. The WIMS and RFSP tools are being employed in the ACR design and analysis and a bundle/channel power compliance procedure incorporating adequate allowances for uncertainties in the monitoring procedure will be developed.

Development of these compliance procedures will explicitly account for the differences between the ACR core design and the design of existing CANDU reactors. For example:

- The ACR core is more compact than existing CANDU reactors. Compliance monitoring in ACR will be based in part on a flux-mapping system based on in-core detector readings. Changes in the design of monitoring provisions from CANDU 6 to ACR will be accounted for in characterising and quantifying inherent errors in the surveillance methods.
- Error models and error terms established for the CANDU 6 design to address the closure criteria of this GAI will be reviewed for application in ACR.

- The computer codes used in ACR compliance analysis will be appropriately validated.
- ACR uses CANFLEX fuel, which provides higher safety margins because of fuel element ratings that are over 10% lower than those in current CANDU 37-element bundle designs. CANFLEX fuel also exhibits higher critical heat flux, and therefore provides higher margin to fuel sheath dryout during transients and accidents. These benefits will be somewhat offset by the higher channel powers in ACR than in existing CANDU reactors. However it is expected that the higher margins will mean that the issue of compliance with bundle and channel power limits for operation at full power will be of less significance in the ACR design.

4.14.4 Status for ACR

This GAI has been closed for the Canadian licensees on the basis of commitments made to employ validated codes for core tracking and compliance with channel and bundle power limits, and to make improvements to bundle and channel power uncertainty analysis. Residual issues are being tracked under GAI 99G02 (Reactor Physics Code Replacement). As indicated in the assessment for GAI 99G02, the WIMS and RFSP tools being used in the ACR design and analysis, will be adequately validated. As indicated in the assessment of GAI 98G02, validation programs for both WIMS and RFSP have been established to provide an adequate basis for their use in ACR analysis.

Compliance procedures, and ongoing programs to collect necessary operating data to confirm error allowances, will be developed for ACR consistent with and building upon the methodologies employed by the Canadian Utilities to close this GAI. The procedures will be adjusted, if required, to account for the specific bundle and channel power envelopes derived for ACR as an outcome of the detailed accident analysis, to the extent that these limits differ in form from those employed in operating CANDU plants. The procedures will also take into account potential changes to the ACR design to enhance surveillance capability (on-line and/or off-line), advances in computer technology, and advances in reactor physics modelling and calculational techniques that may lead to overall improvements in the compliance processes.

AECL will assist the plant operator with implementing appropriate monitoring and compliance procedures on an as-required basis.

4.15 95G04 “Positive Void Reactivity - Treatment in Large LOCA Analysis”

4.15.1 Safety Issue (As Stated by CNSC)

“CANDU reactors have a positive void coefficient of reactivity. In the postulated event of a large LOCA there is an increase in core power, due to positive void reactivity feedback. CANDU reactors have specific engineered features designed to limit the voiding rate in the core and mitigate the power pulse. Two automatic independent shutdown systems have been designed to quickly insert negative reactivity to offset the positive void reactivity. This engineered design solution is based upon an inherent feature of CANDU heavy water-natural uranium lattice: a long prompt neutron life time. The timing and rundown characteristics of each shutdown system are expected to limit the magnitude and duration of the power pulse, and to ensure that the energy deposition in the fuel will not jeopardize the fuel and fuel channel integrity.

The safety analyses in support of the acceptability of the safety systems' performance to ensure meeting the fuel and fuel channel integrity acceptance criteria are based, to a large extent, on numerical simulations of the power pulse. It is therefore important that safety analyses account for the positive void coefficient of reactivity in a conservative manner. This requires the assessment of the accuracy in determination of this coefficient. For fresh fuel at cold conditions a significant amount of data is available from experiments performed in ZED-2 reactor at Chalk River Laboratory. However, the current validation of the theoretical models and computer codes used by the CANDU industry are such that errors associated with void reactivity calculations are not well defined due to a lack of specific experimental data at in-reactor operating conditions and fuel burnups. Although an allowance for uncertainty is included in the safety analysis, the adequacy of this allowance for power reactor conditions is not fully demonstrated, due to the lack of specific experimental data.

Initially, CNSC staff was concerned with the adequacy of the allowance and raised GAI 95G04 requesting the licensees to increase the uncertainty allowance and to provide more information on relevant research. The licensees initiated an industry-wide experimental program carried out in the ZED-2 reactor at Chalk River Laboratories.

Subsequent developments, including new experimental results from the experimental program indicated that the interim value of void reactivity error allowance (VREA), which is applied to predictions of the design and licensing code POWDERPUFS-V (PPV), is not adequate. Furthermore, it has been recognized that PPV significantly under-predicts the void reactivity effect for conditions specific to power reactors and typical average fuel burnup.

There are several areas where, in the view of CNSC staff, specific actions are needed to ensure a high confidence level of results of large LOCA analyses. These areas are:

- *the accuracy and validation of current reactor physics licensing methods and computer codes used for power pulse analyses;*
- *the suitability of the experimental program to support the validation of reactor physics codes and data for conditions specific to power reactors and anticipated accident conditions; and*
- *the acceptability of results of power pulse calculations performed with more accurate and validated methods, and adequate allowances, in support of safety system performance."* (Reference [1]).

4.15.2 Closure Criteria (As Stated by CNSC)

"To achieve closure, licensees are required to complete a suitable experimental program and related analyses based upon more accurate methods and adequate allowances, and undertake adequate interim measures. The following specific closure criteria should be met:

- *Perform a systematic and comprehensive review and assessment of the various uncertainties and sources of error involved in the large LOCA methodology regarding void reactivity;*
- *Provide further supporting evidence for the predictions of the void reactivity coefficient and details of the experimental program and analytical benchmarks. Ensure that the experimental program and analytical benchmarks address the following issues:*
  - *the effect of operating conditions, such as burnup, coolant purity, moderator poison, moderator purity, fuel temperature, and pressure tube creep;*

- *the effect of uncertainties in nuclear data; and*
- *the effect of uncertainties related to limitations of diffusion approximation, core voiding pattern during a LOCA accident, localized absorbers representation, and fuel burnup distribution.*

- *Revise the large LOCA analyses by using more accurate and validated reactor physics methods regarding void reactivity and experimentally-based allowances."* (Reference [1]).

4.15.3 Relevance of the GAI to the ACR

Unlike natural uranium fuelled CANDU reactors, the ACR has a negative coefficient of reactivity associated with coolant voiding. This results in a considerably more benign response to events such as large break LOCAs which exhibit rapid voiding in the early stages of coolant blowdown following the break. In particular, the initial overpower transient typical of natural-uranium fuelled CANDUs will not occur for a large break LOCA in the ACR. As a result, the anticipated safety system performance margins (i.e., with respect to meeting the fuel and fuel channel integrity acceptance criteria) are larger than those experienced in existing CANDU designs. This GAI is therefore not directly relevant to the ACR design. Nonetheless, safety analyses will be performed using acceptably accurate and validated reactor physics methods that provide appropriate treatment of void reactivity and experimentally based allowances. Additional information on the status of validation is provided in the assessment of GAI 98G02.

4.15.4 Status for ACR

This GAI remains open for existing (natural uranium fuelled) CANDU reactors. An extensive series of tests at the ZED-2 facility have been completed and analysed for natural uranium cores. These tests have provided data to validate the WIMS-IST code and its associated nuclear data library for use in calculating void reactivity in CANDU LOCA analyses. A comprehensive program of code-to-code comparisons has been completed and an error (with systematic and random components) has been defined. The Canadian nuclear industry has now submitted documentation to the CNSC covering all of the closure criteria.

AECL's approach for the ACR is to adopt WIMS as the standard lattice cell code (in place of the PPV code used in previous analysis of existing CANDU designs). Also, a WIMS-based RFSP model will be employed for ACR safety analysis, with appropriate code bias and uncertainty applied for physics parameters based on validation against measurement data and benchmarking against the MCNP code. WIMS and RFSP will be modified and validated incrementally, as required, to address ACR specific requirements such as the use of light water in the heat transport system, new fuel and fuel channel design, new reactivity device designs, etc.

A test program for the ZED-2 facility is being developed based on the assessed requirements of the code validation program for the ACR. Extensive experience gained in the performance of the previous ZED-2 tests, and the associated WIMS validation work for natural uranium fuelled CANDUs, will be employed in the design and assessment of a suitable testing program for the ACR. Completion of the planned reactor physics R&D program for the ACR, will demonstrate that safety analysis is being performed using acceptably accurate and validated reactor physics methods that provide appropriate treatment of void reactivity and experimentally-based allowances.

4.16 95G05 "Moderator Temperature Predictions"

4.16.1 Safety Issue (as stated by CNSC)

"In certain loss-of-coolant accidents (LOCA) events, the integrity of some fuel channels depends on the capability of the moderator to become the ultimate heat sink. As fuel channels heat up, pressure tubes expand and some may contact their respective calandria tubes. Fuel channels where contact has occurred will likely fail if the outside of the calandria tubes is dried out. One of the important parameters that determines calandria tube dry-out is the degree of subcooling available in the moderator. An accurate prediction of moderator temperature distribution is therefore required to demonstrate fuel channel integrity under accident conditions.

Since moderator flow is three-dimensional (3-D) in nature, licensees have developed a 3-D computer code to calculate the distributions of flow and temperature in the moderator for various reactor operating conditions. This code, MODTURC-CLAS, has been validated against data from a 2-D test facility and other separate effects experiments. The code has been validated also against in-reactor moderator temperature measurements taken for Pickering Unit 5 and Bruce NGS Unit 3. The data were very few, and the code did not predict the slow temperature oscillations that were measured in the reactor. This validation has left many questions unanswered, especially since the magnitude of the oscillations is much larger than the subcooling margins that exist for most CANDU stations.

CNSC staff has therefore concluded that there is a need for producing code validation data from a 3-D test facility that represents the geometry and operating conditions of the CANDU reactors. This generic action item was then raised, and subsequently a Moderator Test Facility was constructed and commissioned at AECL-CRL in 1999 to support the CANDU-9 development." (Reference [1]).

4.16.2 Closure Criteria (as stated by CNSC)

"To bring this generic action item to closure, the licensees are required to carry out:

- *3-D integrated test program under simulated reactor conditions; test results are to be analyzed and the underlying phenomena identified; and*
- *code validation against the integral test data; pre-predictions should be carried out as part of code validation; validation results should be used to determine code uncertainties for reactor applications."* (Reference [1]).

4.16.3 Relevance of the GAI to the ACR

As in the case of existing CANDU designs, the issue is the demonstration, by qualified analysis, that sufficient subcooling exists in the moderator at all times to prevent fuel channel failures.

4.16.4 Status for ACR

The Canadian nuclear industry has validated the MODTURC_CLAS-IST code as part of the Industry Standard Toolset. An integral scaled 3-D Moderator Test Facility (MTF) was built to investigate moderator behaviour, having the key characteristics of a full-scale CANDU nuclear reactor calandria vessel. Predicted three-dimensional fluid flow and temperature distributions within the MTF vessel are in good agreement with the experimental data, which represent a range of moderator operating conditions. Good quantitative agreement between code predictions and measured values of three-dimensional water temperature distribution in the MTF vessel has been obtained for both steady state and transient simulations. The predicted and measured flow and temperature distribution patterns in the MTF vessel also confirmed the stability of the CANDU 9 moderator system design, which forms the basis for the ACR design. For CANDU 9, CNSC staff accepted the approach adopted by AECL to validate the MODTURC_CLAS code as a means of addressing this GAI (Reference [10]).

The design of the system will include analyses using the MODTURC_CLAS code. To extend the validation of MODTURC_CLAS to the ACR core design, the MTF facility at Chalk River Laboratories will be modified to simulate moderator circulation design and scaled core parameters of the ACR.

The calandria inlet and outlet nozzle configuration in ACR and its effect on moderator circulation will be assessed using the computer code MODTURC_CLAS. The "dead spots" issue (regions of minimal flow circulation) caused by flow patterns in previous CANDU designs was resolved in the CANDU 9 design. ACR will follow the CANDU 9 design approach to ensure adequacy of moderator circulation within the calandria.

It is expected that the planned validation program extending the validation of MODTURC_CLAS to ACR will close all issues related to this GAI for the ACR design.

4.17 96G01 "Fire Protection for CANDU Nuclear Power Plants"

4.17.1 Safety Issue (as stated by CNSC)

"The AECB expects CANDU stations to be operated with minimal risk from fire, as fire can be a major risk contributor to the overall station safety. In 1996, the Canadian Standards Association (CSA) issued CSA standard N293-95, Fire Protection for CANDU Nuclear Power Plants. AECB staff considers that the nuclear industry should meet relevant sections of this standard, and has requested the licensees to assess the adequacy of their fire protection program against them." (Reference [4]).

4.17.2 Closure Criteria

No specific closure criteria have been defined.

4.17.3 Relevance of the GAI to the ACR

Fire protection systems and equipment at most existing CANDU plants were designed to comply with the requirements of the National Fire Code of Canada, the National Building Code of Canada, selected standards of the National Fire Protection Association (NFPA), and the Nuclear Insurance Association of Canada (NIAC). After CSA issued CAN/CSA standard N293-95 (Reference [15]), the CNSC considered that the guidelines presented in Appendix D of CSA N293 (Reviewing Fire Protection for Existing CANDU Plants) should be used to assess and upgrade fire protection in the existing CANDU plants. This GAI was addressed to the licensees of existing CANDU plants. In order to address the requirements of this GAI, the relevant sections of CSA N293 (Reference [15]) will be addressed by the ACR design.

4.17.4 Status for ACR

The comprehensive fire protection requirements for the ACR are documented in a Safety Design Guide. The guide identifies the safety criteria for fire protection for the purpose of shutdown, containment of release of fission products, monitoring and control as well as to ensure that support services remain available. The guide specifies the requirements for a program to initiate, co-ordinate and document the design activities associated with fire protection and includes design requirements for fire protection, prevention, detection, suppression and mitigation. The requirements in the Safety Design Guide are based on CSA Standard CAN/CSA-N293 (Reference [15]) and on recommendations, reflecting international experience, from IAEA Safety Guide 50-SG-D2 (Reference [16]).

In the ACR plant design, all relevant fire protection principles will be taken into account, to fully comply with the Canadian fire protection standard for nuclear plants, CSA N293 (Reference [15]).

The consequences of a fire are limited by the separation by fire barriers of the required safety related systems and the separation by fire barriers of selected redundant components. The use of combustible materials is minimized in the area of safety related systems.

Fire detection and automatic fire suppression is provided in all plant areas except those continuously occupied and in the reactor building. A seismically qualified fire water supply is provided for the reactor building and areas containing essential safety related equipment credited in the seismic event. A comprehensive Fire Hazards Assessment and Probabilistic Safety Assessment for internal fires will be completed to demonstrate the adequacy of the overall plant design against fires and to provide the plant operating staff with the documentation required for the fire protection program.

Implementation of the requirements specified in the ACR Safety Design Guide on Fire Protection will address the requirements of this GAI.

4.18 96G02 "Feeder Pipe Fitness for Service"

4.18.1 Safety Issue (as stated by CNSC)

"Inspections in several CANDU reactors revealed an unexpected reduction in the wall thickness of some outlet feeders. The rate of this degradation represents a departure from the original design predictions. When these findings were observed, feeder wall thicknesses were still adequate and were predicted to remain adequate for several years. However, the expected lifetime of some of the feeders could be limited by the current rate of degradation. Licensees were asked to show that feeders are fit for service. They were also asked to show sufficient understanding of the thinning phenomenon to prevent it from threatening the integrity of the feeders.

Licensees submitted a report to the CNSC including results of their investigation of the most likely cause for feeder thinning. They have also committed to a periodic feeder inspection program." (Reference [1]).

4.18.2 Closure Criteria

No formal Closure Criteria have been defined.

4.18.3 Relevance of the GAI to the ACR

This GAI was addressed to the operating CANDU plants. However, for future plants, the CNSC has identified the following expectations (Reference [17]):

a) A proper understanding of the degradation mechanism;
b) Elimination of the mechanism, if possible;
c) If not, its effect should be minimized as much as possible with design provisions;
d) If the feeders cannot be designed for the life of the plant, the maintenance program shall include replacement of feeders;
e) Inspections should take into consideration the identified degradation mechanism. The design phase should have identified proper locations and frequency for the planned inspections in order to confirm the expected rate of degradation, and should have established a proper rejection criterion. Also, inspections should include some activities aimed at capturing unexpected degradation.

Meeting these expectations in the ACR design will ensure that the issues associated with this GAI have been fully addressed. As part of the ACR design process, efforts will be made to minimize the rate of feeder degradation over the entire planned operating lifetime. ACR design will use stainless steel feeders that have a very low Flow Assisted Corrosion rate in the sensitive high velocity, high turbulence sections of the feeders, which will minimize the rate of feeder degradation.

4.18.4 Status for ACR

Canadian licensees have responded to the GAI and investigated the feeder thinning issue. It has been concluded, and CNSC has accepted, that the major cause of outlet feeder wall thinning at the operating CANDU plants is Flow Assisted Corrosion (FAC).

The rate of feeder degradation at ACR will be minimized by careful selection of feeder materials. At the ACR operating conditions, FAC is believed to be a concern for carbon steel feeder piping, which has been used in previous CANDU designs. To minimize these effects, alternative feeder materials (i.e., stainless steel) will be used in the ACR design.

The ACR project will make extensive use of knowledge gained from current CANDU plants. However, due to the compact core design of the ACR plant, small feeder pipes with 2.5 inch OD and 2 inch ID have to be used at the reactor face. This results in a high flow velocity in these segments of the feeders. To ensure feeder pipe “fitness for service” for the entire design plant life, alternative materials with better corrosion/erosion resistance will be required for a portion of the feeder run within the reactor face.

Based on design assessments that have been performed to date for the ACR design, some conclusions and recommendations have been made regarding the selection of pressure retaining materials for the feeders.

The tasks aiming at identification and verification of feeder materials have been scheduled. These will involve corrosion tests of feeder materials in a high temperature and pressure loop.

All of the above considerations will minimize the rate of feeder degradation in the ACR. In combination with this, the utility is expected to implement an appropriate periodic feeder inspection program. AECL has committed to applying to the ACR design many relevant standards including CAN/CSA-N285.4-94 (Reference [3]) that specifies procedures and acceptance criteria for feeder inspection. AECL will support the operating utility, on an as required basis, in developing fitness for service guidelines and a periodic inspection program appropriate to ACR operation.

In summary, in order to minimize the rate of feeder degradation over the planned operating lifetime, the ACR will use stainless steel with a very low FAC rate in the sensitive high velocity, high turbulence sections of the feeders and validate the performance through tests.

4.19 98G01 "PHT Pump Operation Under Two-Phase Flow Conditions"

4.19.1 Safety Issue (As Stated by CNSC)

"The operation of Primary Heat Transport (PHT) pumps under low suction pressure and significant void can be detrimental to the integrity of the PHT system piping due to the generation of large-amplitude pressure pulsations and excessive pumpset vibration. In the past, the PHT piping fatigue analysis was done using a limiting forcing function (harmonic excitation) obtained from laboratory tests of full-scale PHT pumps. Given the underlying assumptions, especially the amplitude and frequency of excitation, this approach was very sensitive to interpretation of the test data and their application to the PHT system. Consequently, the assessment of the piping fatigue life may not have been conservative.

Further work was therefore required to develop a mechanistic pump model from the available data base and apply it to the PHT system piping configuration. Compared to the use of some arbitrarily assumed limiting forcing function, this additional work was expected to give a more realistic representation of the behaviour of the PHT pump and piping under various accident conditions." (Reference [1]).

4.19.2 Closure Criteria (As Stated by CNSC)

"To achieve closure of this action item, for stations other than Darlington, licensees are required to perform the following:

- *Re-assess the PHT pump behaviour under two-phase flow conditions with emphasis on the unsteady feedback from the PHT system piping (consider assessment of the latest full-scale tests involving the Darlington PHT pump); and*
- *update the fatigue analysis of the PHT system piping."* (Reference [1]).

4.19.3 Relevance of the GAI to the ACR

Operation of the primary heat transport system pumps under two-phase flow conditions can occur under certain accident scenarios. In order to avoid damage to the pumps (e.g., seals) and piping due to extended operation under sustained low pressure, low temperature cavitation conditions, all four HT pumps are tripped automatically following a LOCA when the Heat Transport System pressure falls to below a specified minimum pressure for an extended period of time. This ACR design feature is similar to provisions which exist in a number of operating CANDU plants. The actual performance requirements for the ACR heat transport pumps (and the automatic pump trip function) under these conditions will depend in part on the detailed design of the ECCS.

Based on the CNSC position statement (Reference [18]) developed for existing CANDU designs, the work required to ensure that this GAI is adequately addressed for ACR is as follows:

- mechanistic models of the behaviour of the ACR Heat Transport System pumps under two phase flow conditions will be developed to permit Heat Transport piping fatigue life analysis to be undertaken. Issues to be addressed include:
  - Ensuring an adequate characterization of pump instabilities under two phase conditions (oscillation frequency and amplitude, pump characteristics, onset of instability, instability criteria, sensitivity to key parameters, etc.);
  - Formulation of appropriate scaling rules: amplitude and frequency of pump head oscillations as a function of system process conditions (temperature, void) and geometry (length, suction side orientation, formation of void cavities).
- Application of the mechanistic models to Heat Transport System piping to calculate the expected minimum fatigue life. The fatigue analysis will demonstrate that, at the end of station life, the Heat Transport System piping can withstand two phase flow conditions for all credible accident sequences for a period sufficient to credit either operator action or an automatic pump trip. The analysis will account for uncertainties related to the prediction of pump head oscillations (fatigue life is a strongly non-linear function of oscillation amplitude) and to the prediction of accident conditions. The analysis will involve:
  - Identification and characterization of various accident scenarios where pump-induced pulsations arising from two phase conditions are likely to be significant;
  - Formulation of an appropriate forcing function to predict heat transport system piping vibration (travelling versus standing waves, possible amplification or attenuation of pressure pulsations). This will include mechanically-transmitted vibration from the pumpset;
  - Prediction of fatigue life of the weakest part of the piping for the accident sequences under consideration.

4.19.4 Status for ACR

This Generic Action Item has been closed or is expected to be closed shortly for all of the Canadian licensees. The analysis methodologies employed by the Canadian licensees are considered to be well established and meet the requirements of the CNSC. The analysis performed for ACR to address these issues will be consistent with the overall approach taken by the Canadian licensees and will ensure that the closure criteria for this GAI will be met.

Work to be performed for ACR will include:

- Completion of the accident analysis to ensure that all relevant accident scenarios are considered in determining the HT piping loading conditions, and in establishing initiating setpoints and conditioning provisions for the automatic HT pump trip function.
- Testing of the HT pumps (if determined to be needed to supplement existing available data) followed by implementation of an appropriate validated model of HT pump behaviour addressing the issues identified above.

- Performance of the fatigue analysis to demonstrate that at the end of station life (based on the 60 year plant lifetime target), the Heat Transport System piping can withstand two phase flow conditions for all credible accident sequences for a period sufficient to credit either operator action or an automatic pump trip.

4.20 98G02 "Validation of Computer Programs Used in Safety Analysis of Power Reactors"

4.20.1 Safety Issue (As Stated by CNSC)

"Safety analysis is used to establish certain safety-related information about the design and behavior of the reactor and the safety systems under various conditions including normal operation and certain postulated events such as loss-of-coolant accidents. Such information is provided in the Safety Report, and its updates, and is a primary definer of the station's licensing basis and the bounds of the safe operating envelope. Analysis is also used to demonstrate that the station is being operated within the conditions of the operating license.

The credibility of this safety-related information depends to a great extent on the degree of conservatism incorporated into the safety analysis and on the qualification of the individual safety analysis activities and tools such as computer programs, analysis methods, and input information. Increasing recognition is being given by both licensees and the CNSC to the importance of safety analysis qualification and one important element of this qualification is computer program validation.

In the past, CNSC staff undertook a number of assessments of licensees' computer programs and safety analysis methods and has identified a number of issues with respect to computer program validation. Issues identified include a lack of a managed process in performing validation of computer programs, insufficient documentation of computer program validation, inadequate applicability of validation due to the limited range of conditions in the validation experiments in comparison with the reactor analysis, inadequate assessment of the impact of dimensional scaling, and important phenomena for which adequate validation data does not exist. CNSC staff has concluded that these deficiencies are affecting the overall confidence in the safety analysis results.

The industry has developed a generic framework for computer program validation and CNSC staff has been relying upon this generic approach to improve the computer program validation. However, to date, the results of this generic approach have been limited, and few specific commitments have been made by the licensees. CNSC staff has concluded that licensees need to establish specific validation programs to improve computer program validation and to provide the necessary confidence in the safety analysis.

Currently, there are a number of open generic action items for which validation of computer programs is an outstanding issue. These GAIs include:

- *GAI 88G02 Hydrogen Behaviour in CANDU Nuclear Generating Stations*
- *GAI 90G02 Core Cooling in the Absence of Forced Flow*
- *GAI 95G04 Positive Void Reactivity - Treatment in Large LOCA Analysis*
- *GAI 95G05 Moderator Temperature Predictions"* (Reference [1]).

4.20.2 Closure Criteria (As Stated by CNSC)

"To achieve closure, licensees are required to:

- *undertake a code validation program;*
- *provide bi-annual summary reports describing the overall progress and the major milestones achieved; and*
- *submit the information identified below; the extent of information provided shall be sufficient to demonstrate that the expectations in this position statement have been met:*
  - *A list of the computer programs and corresponding applications to which the licensee considers this GAI to be applicable.*
  - *Description of the key elements of the validation process. (For the industry's proposed validation process, the purpose of each of the documents: technical basis document, validation matrices, and validation plans should be defined, their overall content described and the inter-relationships between the documents defined.)*
  - *A report for each computer program which summarizes the range of conditions over which the program is intended to be used and compares these with the ranges of conditions in the experimental database. The report will identify "gaps" in the validation experimental database and either the plans for their closure or a justification for leaving them open.*

In addition, the following information is required as related to each event or group of events. The way in which the information is organized into individual reports, and the extent of information needed for a particular computer program and application, are left to the discretion of the licensee:

- *Identify the inter-relationships between the type of safety analysis, the important safety limits to be met, the technical disciplines and the overall goals of the validation.*
- *In each of the technical disciplines, identify all phenomena for which validation is potentially required, and justification for any ranking used.*
- *Identify experimental facilities which can be used to validate the programs for the required phenomenon; indicate any gross physical distortions that may make the facility and the way in which it behaves atypical of the reactor: such distortions include geometric differences (including scaling), differences in material properties, differences in fluid physical properties.*
- *Identify specific experimental tests that will be used in the validation exercises; indicate the ranges of relevant conditions in comparison with those typical of the equivalent reactor case.*
- *Produce validation plans for each of the programs: the plans should include sufficient information (together with other documents) to demonstrate that once the validation is complete the programs will be adequately validated; the validation plans should include the rationale for the extent of validation.*
- *Produce validation reports for each of the programs; the reports should contain sufficient information to demonstrate the claimed accuracy of the program for the given application."* (Reference [1]).

4.20.3 Relevance of the GAI to the ACR

This GAI has recently been closed based on the validation of mainline CANDU computer programs used in safety analysis that has already been completed by AECL and its industry partners. All requirements in this GAI apply to the safety analysis computer programs used for ACR. The detailed schedule for completing any incremental validation tasks required for ACR will be determined based on the ACR project schedule.

The Canadian industry established a program to meet all of the GAI conditions for the critical safety analysis codes which were designated to be part of the Industry Standard Toolset. These codes form the basis for the analysis of the ACR reactor. For a limited number of codes that were not included in the IST, AECL has independently met the conditions of the GAI to validate these codes.

The development, validation and application of computer programs at AECL are performed in compliance with the "AECL Quality Assurance Manual for Analytical, Scientific and Design Computer Codes" (Reference [19]). The current version of this manual has addressed CNSC staff comments and has been made consistent with CSA Standard N286.7 (Reference [20]) and the CNSC Regulatory Guide, G-149 (Reference [21]). All validation and verification activities of computer programs used in the safety analysis of ACR will be carried out, documented and controlled in conformity with the requirements of this manual.

4.20.4 Status for ACR

The ACR design and safety analysis will meet the GAI requirements through the use of validated codes. ACR will use validated codes that form the set of the Industry Standard Toolsets to the greatest extent practical, along with commercially available safety analysis codes, and AECL proprietary codes.

Features in the ACR design that are different from the current CANDU design may require additional validation or modification of the codes used. A preliminary assessment of the key safety codes has been carried out to identify such requirements and development programs have been initiated to address high priority issues. In particular, R&D will be conducted to fully validate the physics analysis codes for the ACR fuel and lattice configuration, and experiments have been initiated to extend the validation of the thermalhydraulics analysis codes.

As part of the overall ACR development program, qualification plans will be developed and executed to qualify all of the codes used in the ACR safety analysis.

The validation status of the mainline computer programs to be used in the ACR safety analysis is presented in Table 1. Reference [22] contains more details of the ACR-specific validation work being done for safety analysis codes.

4.21 99G01 "Quality Assurance of Safety Analysis"

4.21.1 Safety Issue (As Stated by CNSC)

"The CNSC expects power reactor licensees to conduct their operations in accordance with a Quality Assurance Program (QAP). The QAP includes QA requirements for various safety-related activities. Safety analysis is one activity to which the QAP shall apply.

Safety analysis is used to establish certain safety-related information about the behaviour of the reactor and the safety-related systems (and in particular the special safety systems) under normal conditions and certain postulated events such as loss-of-coolant accidents. Such information is provided in the Safety Report, and its updates. The Safety Report is a key document that defines the station's licensing basis and the bounds of the safe operating envelope. The acceptability of this safety-related information depends to a great extent on the degree of conservatism incorporated into the safety analysis; it also depends on the credibility of the safety analysis tools and activities (such as computer codes, analysis methods, and input information).

It is important that licensees perform safety analyses in a systematic manner so that high confidence can be attributed to the definitions of the licensing basis and safe operating envelope for each of the licensees' stations.

In recent years CNSC staff has became aware of an increasing number of occurrences of inadequate quality assurance in safety analyses performed by licensees. These deficiencies have been identified through audits and assessments by both licensees and the CNSC. Examples are:

- *inadequate control over the use of computer codes resulting in the inappropriate use of these codes beyond their originally intended application;*
- *basic errors in fundamental conservation equations incorporated into codes;*
- *inadequate validation of computer codes;*
- *codes for which the theoretical bases have not undergone an appropriate level of peer review or independent technical scrutiny;*
- *inadequate independent peer review of safety analysis methods;*
- *inconsistencies between safety analyses input and plant data;*
- *significant errors in safety report updates;*
- *inadequate code documentation (controlled or otherwise);*
- *inadequate planning of safety analysis in support of outages, resulting in analyses being carried out in an unplanned and hasty manner, and*
- *inconsistent reporting of safety analysis errors to the CNSC.*

CNSC staff has concluded that the inadequate quality assurance with respect to safety analysis is resulting in a significant reduction in the overall confidence in the safety analysis results." (Reference [1]).

4.21.2 Closure Criteria (As Stated by CNSC)

"*To achieve closure, licensees are required to:*

- *have a QAP that includes requirements for safety analysis activities, and which meets applicable QA standards;*
- *undertake a QAP assessment in accordance with N286.0 to determine the effectiveness of the licensees' current QAPs specifically with respect to safety analysis activities; this assessment shall include a formal program review and audits; the program review shall determine the extent to which the QAP meets applicable QA standards and CNSC staff expectations;*

- *submit to the CNSC a report of the assessment identifying program deficiencies and a plan and schedule for their correction;*
- *implement the corrections to the QAP;*
- *submit to the CNSC sufficiently detailed information to demonstrate that the QAP meets the requirements of applicable QA standards and this position statement; and*
- *provide six-monthly progress reports on resolution of this GAI."* (Reference [1]).

4.21.3 Relevance of the GAI to the ACR

The GAI was addressed to Canadian licensees. The licensees were required to carry out safety analyses in accordance with a well-established QAP. AECL is not a power reactor operating organization. However, AECL carries out safety analyses of the plant as part of the design activities and in support of licensing activities carried out by power reactor operating organizations. Therefore, AECL is committed to addressing the requirements of the GAI. The ACR project has established a Quality Assurance Program including requirements for the conduct of safety analysis activities.

4.21.4 Status for ACR

The ACR project will follow the AECL's company-wide Quality Assurance Manuals and Procedures that have been subjected to careful scrutiny and been revised in light of Canadian regulatory documents and national standards. AECL's Overall Quality Assurance Manual (OQAM) (Reference [23]) is the top-tier QA manual in a series of manuals (e.g., company-wide design QA manual (Reference [24]) and other sub-tier QA manuals), procedures, and operating instructions that document an integrated approach, applicable to all life-cycle phases of CANDU nuclear power plants and other nuclear facilities. A table cross-referencing the requirements of CAN/ CSA N286.0 (Reference [25]) to the AECL OQAM and supporting Procedures has been developed, which shows that AECL's QAP fully complies with Canadian National QA Standards. Projects and facilities draw on the company-wide QA manuals to define project-specific and facility-specific QA programs that also provide assured compliance with CAN/CSA-N286.0 and other CSA Standards such as N286.1, N286.2 and N286.7. The effective implementation of the QAP is demonstrated by a program of QA reviews, audits, and assessments.

Safety analysis is an integral part of AECL's design QA program. AECL's design QAP includes requirements for safety analysis activities. AECL revised its company-wide Design QA Manual (Reference [24]) to meet the requirements of the revised CSA N286.2 Standard. AECL also revised its Quality Assurance Manual for Analytical, Scientific and Design Computer Codes (Reference [19]) in 2001. The current version of this manual has incorporated the CNSC's relevant comments and has been made consistent with CSA Standard N286.7 (Reference [26]) and the CNSC Regulatory Guide, G-149 (Reference [21]). In early 2001, AECL issued a specific procedure covering Computer Program Validation. Computer code validation issues are being addressed under the GAI 98G02, "Validation of computer programs used in safety analysis of power reactors". In general, AECL will ensure that all computer codes to be used for ACR safety analysis will be adequately validated prior to their applications.

The AECL company-wide QAP is supplemented by project specific QA manuals and procedures. A Quality Assurance Manual (Reference [27]) for ACR has been issued.

It is expected that the ACR QAM, together with the supporting procedures/documentation and the successful completion of future actions, will meet all applicable QA standards and regulatory requirements, and will address the closure criteria of this GAI.

4.22 99G02 "Replacement of Reactor Physics Computer Codes Used in Safety Analysis of CANDU Reactors"

4.22.1 Safety Issue (As Stated by CNSC)

"Licensees use reactor physics methods and computer codes to provide important safety-related information in various areas of interest, such as nuclear design, operation, and compliance with the safe operating envelope. There are high expectations concerning the accuracy and validation of these methods and codes, and their predictions for normal and accident conditions, because of their role in the fundamental support of the robustness of the design and in the confirmation of safe operation; consequently, they should reflect the current state of knowledge in the area of reactor physics.

Recent experimental data, as well as reviews of key computer codes, have identified several shortcomings in the reactor physics area. The most important are: inaccurate predictions of key parameters for accident conditions, lack of proper validation data for important phenomena and range of conditions, and a significant gap between the state of knowledge reflected in the licensees' computer codes and the current state of knowledge in this area. These shortcomings have had a negative effect on the overall confidence in predictions of the reactor physics analyses, especially, for those analyses of design-basis accidents where safety margins are small. The use of a given reactor physics computer code would impact the safety margins with respect to the derived acceptance criteria within the accuracy, and associated sensitivity, of this computer code to predicted reactor physics parameters.

Currently, the industry is planning to retire some old reactor physics computer codes, such as POWDERPUFS-V (PPV) and SMOKIN, and is carrying out a validation program of more accurate ones such as WIMS-AECL, DRAGON, and WIMS-AECL based RFSP. This validation program, however, does not address issues which are specific to the code replacement process itself.

At issue, is the apparent absence of a distinct and separate program on replacement of various reactor physics codes, which is needed to address appropriately several areas requiring specific actions. These areas are:

- *the approach to replacement of specific reactor physics computer codes, that would include firm schedules and dates of replacement;*
- *the impact of reactor physics computer codes replacement on safety margins; and*
- *the impact on safety report update programs, and coverage for any interim period, where a computer code has been replaced, during which period the full impact on all safety analyses may not have been fully addressed.*

This position statement therefore defines CNSC staff expectations with respect to replacement of current reactor physics computer codes. It also identifies the links between this generic action item and activities being performed under other GAIs." (Reference [1]).

4.22.2 Closure Criteria (As Stated by CNSC)

"To achieve closure, licensees are required to undertake a structured program of replacement of reactor physics computer codes. The program should include the following:

- *implementation of a structured approach of specific code replacement, including firm schedules and dates of replacement;*
- *replacement of all the relevant reactor physics computer codes used in safety analysis and operation, including PPV, MULTICELL, PPV-based RFSP, and SMOKIN; PPV should also be retired from use for fuel management and core tracking simulations;*
- *ensuring that validation of new codes is in accordance with requirements in GAI 98G02, Regulatory Guide G-149, and CSA Standard N286.7;*
- *assessment of the impact of code replacement on current safety margins and identify the limiting design-basis accidents whose safety margins might be significantly affected;*
- *assessment of the impact on safety report updates' programs and identify ongoing and future activities that might be affected by the reactor physics codes' replacement;*
- *definition of adequate interim allowances for use with key reactor physics parameters provided by PPV for safety analysis, such as void reactivity, delayed neutron fraction, fuel temperature reactivity, prompt neutron lifetime, until replacement of PPV; and*
- *adequate coverage for any interim period, where a computer code has been replaced, during which period the full impact on all safety analyses may not have been fully addressed."* (Reference [1]).

4.22.3 Relevance of the GAI to the ACR

This GAI is not directly relevant to the ACR. This is because from the outset of the ACR design process, AECL is employing state-of-the-art CANDU physics analysis tools (i.e., the WIMS, RFSP-IST and DRAGON code system and associated nuclear data libraries) for undertaking analyses in the areas of nuclear design, operation and compliance with the safe operating envelope. In addition, the ACR design does not have a positive coolant void reactivity coefficient, and physics codes' uncertainty is less important with respect to its impact on calculated safety margins for events such as Large LOCAs. Therefore, activities identified in the closure criteria related to code replacement, and transitional actions such as the use of interim allowances for reactor physics parameters and safety margin impact assessments, do not apply. The physics analysis tools will be modified as required and qualified (validated) for use in the ACR design (refer to the assessment of GAI 98G02 for more details on the validation programs). The tools will meet the requirements in this GAI and in GAI 98G02 for compliance with CSA standard N286.7 and CNSC regulatory guide G-149.

4.22.4 Status for ACR

A key component of the safety technology application for the ACR is the suite of validated computer codes that are required for safety analyses. Codes to be used in reactor physics analysis and nuclear design will be based on the CANDU industry-standard toolset, modified as required by ACR physical design parameters. ACR specific requirements such as the use of light water in the heat transport system, new fuel and fuel channel design, new reactivity device designs, etc., will be addressed explicitly in developing and validating ACR specific versions of the reactor physics codes.

The approach being taken for modifying and validating the ACR-specific physics analysis codes is as follows:

- a code assessment has been performed to determine the modifications required and to develop a preliminary assessment of the validation requirements for the modified codes;
- modified versions of the codes have been produced for use in preliminary safety analyses being performed in support of the concept design assessment;
- preliminary validation requirements have been used as an input to R&D planning, including the need for major test facilities;
- detailed validation plans are being prepared within the context and format of the existing code validation program, as an extension to that program. The assessment of GAI 98G02 provides additional detail on the code validation programs.

The execution of this code development program, validation and R&D support activities will result in the use of validated and verified tools in ACR reactor physics analysis, in accordance with CNSC requirements stated in this Generic Action Item.

4.23 00G01 “Channel Voiding During a Large LOCA”

4.23.1 Safety Issue (As Stated by CNSC)

“Reactivity effects of channel voiding in CANDU 6 reactors are positive. When a break occurs in the primary heat transport system, the system depressurizes. Some coolant in fuel channels associated with the broken pass turns into steam. This coolant voiding in the fuel channels results in a reactor power pulse due to the positive void reactivity feedback. The increase in reactor power must be arrested and limited by the two fast-acting shutdown systems. Safety assessments have been performed for Large LOCAs to show shutdown system effectiveness.

Two significant parameters that determine the magnitude of the power pulse are:

- *Rate and extent of coolant voiding; and*
- *Reactivity changes due to voiding.*

These parameters are predicted by computer models. However, they have not been adequately validated as there are no direct void fraction measurements applicable to CANDU fuel channel conditions. This GAI is intended to deal with the channel voiding issue, since the void reactivity issue is being dealt separately under another GAI.” (Reference [1]).

4.23.2 Closure Criteria (As Stated by CNSC)

"To bring this generic action item to closure, the licensees are expected to carry out:

- *channel void measurements during large LOCAs relevant to reactor conditions; the effects of heat transfer rate and scaling on channel voiding should also be quantified.*
- *validation exercises with the relevant safety analysis computer codes against the channel void data. There should be sufficient information to demonstrate the claimed accuracy of the code for the given application, and*
- *an impact assessment of the safety margins in the safety report."* (Reference [1]).

4.23.3 Relevance of the GAI to the ACR

Unlike the current natural-uranium-fuelled CANDU reactors, the ACR reactor core design is characterized by a negative coolant void reactivity coefficient. As discussed in the assessment of GAI 95G04, this results in a considerably more benign response to events such as large break LOCAs which exhibit rapid voiding in the early stages of coolant blowdown following the break. In particular, the initial overpower transient typical of natural-uranium fuelled CANDUs will not occur for a large break LOCA in the ACR. As a result, the anticipated safety system performance margins (i.e., with respect to meeting the fuel and fuel channel integrity acceptance criteria) will be larger than those experienced in existing CANDU designs. Nonetheless, the magnitude and rate of coolant void generation in the fuel channels and their associated uncertainties will be adequately characterized such that the thermalhydraulic analysis tools used in the ACR safety analysis are adequately validated, consistent with the requirements of GAI 98G02.

4.23.4 Status for ACR

The Canadian nuclear industry is collaborating, through the CANDU Owners Group, on the completion of an experimental program at the RD-14M thermalhydraulic facility to measure channel voiding under Large LOCA conditions representative of those expected in current CANDU designs. The key deliverables committed by the industry team to close this GAI include:

- A report summarizing the performance of a state-of-the-art neutron scatterometer developed by AECL;
- Development of a plan to address the effects of scaling on code validation exercises using void generation experimental results obtained in the RD-14M facility;
- Reports on the analysed data from the RD-14M LOCA tests with transient channel void measurement;
- Validation reports documenting validation exercises performed with the relevant safety analysis computer codes (i.e., CATHENA in the case of AECL) against the channel void data.

The validation exercises performed with the CATHENA thermalhydraulics code will form an important part of the validation basis for this analysis tool even though channel voiding is a less significant factor in the safety case for ACR than it is for the existing natural-uranium-fuelled heavy-water-cooled CANDU design.

Based on the work completed to date, and the work now underway, the issues raised in this GAI are being addressed for the ACR design.

4.24 01G01 "Fuel Management and Surveillance Software Upgrade"

4.24.1 Safety Issue (As Stated by CNSC)

"The compliance with reactor physics safety limits that defines the Safe Operating Envelope (SOE), such as channel and bundle power limits, are essentially based on analyses performed with a fuel management code. Recent more rigorous scrutiny of the accuracy of methods, acceptance criteria, assumptions and results of safety analyses of various design-basis accidents led to significant restrictions to operating parameters, including channel and bundle powers, and the introduction of additional physics parameters for compliance purposes, such as fuel string relocation reactivity, minimum margin to axial constraint, etc. As such, the significance of compliance with safety-related reactor physics limits has increased. This has enhanced the need for an improved analytical model validated over a broader range of applications and conditions, and better defined compliance allowances and more consistent procedures.

Two main areas of improvement related to the current fuel management code have been explicitly identified:

i) code methodology, modeling and data, and

ii) validation

There are also various issues related to methodology for deriving the compliance error allowances at 98% confidence level and station specific compliance procedures and practices.

At issue are several areas both in the compliance analyses and procedures where, in the view of the CNSC staff, improvements are needed to ensure adequate compliance with OP&P limits related to reactor physics parameters and reactor core status at various stations."
(Reference [28])

4.24.2 Closure Criteria

This GAI has been issued only to Ontario Power Generation and Bruce Power, to track specific actions identified as a follow-up to closure of GAI 95G03 "Compliance with Channel and Bundle Power Limits". GAI 95G03 had been closed for these two licensees on the basis that they had committed to an extensive program to upgrade their fuel management code SORO up to the current level of the industry standard state-of-the-art tool RFSP-IST, which is being used by AECL in the design of ACR. Since this program represented a significant departure from previous plans by OPG and Bruce Power to close GAI 95G03, it was considered appropriate to raise a new GAI to address remaining unresolved issues and to identify closure criteria specific to the SORO code upgrade program and the associated compliance methodology and monitoring program.

The closure criteria defined by CNSC staff for this GAI are as follows:

"*To achieve closure, licensees are required to undertake a structured program for reactor core surveillance that should cover the fuel management software upgrade and validation, and* *validation and qualification of the error compliance methodology. The program should include the following:*

a) *software upgrade to a level at least similar to the IST reactor physics code RFSP-IST. The software upgrade activities should meet the requirements in GAI's 98G02 and 99G02 for compliance with CSA Standard N286.7 and CNSC Regulatory Guide G-149;*

b) *validation, verification and qualification for production use. The related activities should meet the requirements in the licensee's governing QA process for creation, maintenance, verification and validation of safety analysis software and associated reference data, to ensure compliance with CSA Standard N286.7 and CNSC Regulatory Guide G-149;*

c) *verification, validation and qualification of error compliance methodology and associated assessment data base for the full range of computed parameters and applications;*

d) *estimation of compliance uncertainties for computed parameters. The allowances should, at the 98% confidence level, account for;*

   - *error in total reactor power normalization;*
   - *error in code methodology and modelling;*
   - *error in measurements;*
   - *xenon transients initiated by fuelling;*
   - *implementation of a station-specific monitoring program for periodic confirmation of accuracy of fuel management code predictions and compliance uncertainties for computed parameters to ensure reactor operation is within SOE. The program should include periodic measurements and analyses with the fuel management code for the plant actual conditions. The program should address issues such as changes in reactivity device worth (for example: cobalt adjuster burnout), changes to the NOP/ROP reference power shape, evaluation of ageing effects on reactor physics calculations (impact of core and fuel channel geometry distortion on reference power shape and bundle and pin maximum powers), effects of xenon transients initiated by fuelling."* (Reference [28]).

4.24.3 Relevance of the GAI to the ACR

This GAI does not apply directly to ACR. This is because AECL is employing the WIMS-IST and RFSP-IST codes (rather than the corresponding OPG tools PPV and SORO) to perform core calculations in ACR, including core-tracking studies. As indicated in the assessment for GAI 98G02, these tools (and any modifications undertaken to address ACR requirements) will meet the requirements identified in GAI 98G02 and 99G02 for compliance with CSA Standard N286.7 and CNSC Regulatory Guide G-149. New Brunswick Power, the operator of the CANDU 6 unit at Point Lepreau, achieved closure of the related GAI 95G03 "Bundle and Channel Power Compliance" in May 2001 by committing to the implementation of the WIMS/RFSP code suite for fuel management and safety analysis.

Items c) and d) of the closure criteria are applicable to the ACR since they represent general requirements placed on the bundle and channel power compliance methodology employed by the plant operator. These criteria are equivalent to those identified for closure of GAI 95G03, and their disposition for the ACR is discussed in the assessment for that GAI.

4.24.4 Status for ACR

This GAI can be considered to be adequately addressed in ACR from a design perspective, for the reasons outlined in Section 4.24.3. Once the plant is in operation, the plant operator will be required to implement, as part of their ongoing fuel management program, a number of measures related to ongoing validation of fuel management simulations and compliance with safety limits (i.e., consistent with items c) and d) of the closure criteria for this GAI, and the equivalent items in GAI 95G03). AECL will provide support to the plant operator in implementing a structured program for reactor core surveillance that will cover the fuel management software implementation and validation, and the validation and qualification of the error compliance methodology. These methods are consistent with those developed in Canada by plant operators with AECL participation and support.

Based on current and planned activities, the closure criteria for GAI 01G01 are considered to have been fully addressed in the ACR program.

5. CONCLUSIONS

The extent to which the ACR design addresses the licensing issues identified in the CNSC GAIs has been assessed. The approach employed in the ACR to address the issues identified in the CNSC GAIs has been summarized with references to additional supporting information. The overall conclusion from this assessment is that the issues defined in the GAIs are considered to have been adequately addressed by design provisions or via current and planned ACR program activities. In cases where resolution is dependent upon implementation of specific on-going programs by the plant owner, AECL will provide the necessary support to develop and implement these programs.

6. REFERENCES

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Table 1
Mainline Computer Programs to be Used in ACR Safety Analysis and Their Validation Status (Preliminary)

Computer Programs	IST*	Use	Validation Status
Group 1, Major verification and validation work required:			
RFSP	Yes	This is a three-dimensional two energy group neutron diffusion code, with both static and kinetics capability.	The two-energy-group method implemented in RFSP-IST has recently been validated by the industry for CANDU reactors and could be used for the ACR design. However, consideration is being given to increasing the number of energy groups for use in analyses with SEU fuel. Some major changes have been made to the code for the purpose of both current and future CANDU safety analysis. These changes have been approved and verified. The validation of this code for ACR conditions is in progress. It is expected that this validation program will close all issues related to ACR compliance with 98G02 requirements.
WIMS-AECL, 2-5d	Yes	This is a multi-group neutron transport code for reactor lattice calculations and reactivity coefficients.	WIMS has been adequately validated for the present CANDUs. No potential changes have been identified for ACR purposes. Because of the use of light water coolant in ACR, guidelines used for standard CANDUs for preparing WIMS input models will be reviewed for ACR. There is a program in place to validate WIMS for ACR. It is expected that this validation program will close all issues related to ACR compliance with the requirements of GAI 98G02.
MODTURC_CLAS	Yes	Moderator thermalhydraulics. MODTURC_CLAS is a 3-D IST computer program that calculates the moderator flow and temperature distribution in the calandria.	MODTURC_CLAS has been adequately validated by the Canadian nuclear power industry for current CANDUs (refer to GAI 95G05). A preliminary assessment program identified some changes in the code that need to be made to address ACR specific design parameters. There is a program underway to validate the computer code for ACR. The MTF facility will be modified to simulate moderator circulation in the ACR geometry and the code will be compared to these experimental results. A specific validation plan is under preparation.

* Industry Standard Toolset

Computer Programs	**IST***	**Use**	**Validation Status**
			It is expected that this validation program will close all issues related to ACR compliance with the requirements of GAI 98G02.
TUBRUPT	Yes†	Calandria loading analysis. TUBRUPT was selected for validation as the single code to assess the consequences of an in-core fuel channel rupture.	A Code Validation Project was carried out for TUBRUPT-IST by AECL and OPG. The Validation Exercise Reports, Validation Manual and Tool Qualification Report have been submitted to the CNSC. AECL and IST partners have recently received comments on TUBRUPT-IST validation from CNSC and will be addressing these comments in a timely manner. Assessment will be made on the applicability of the code to ACR geometry. If changes in the code are required, an incremental validation program will be prepared to ensure code qualification to ACR design.
CATHENA	No**	This is a 1-D two-fluid thermalhydraulics network code used for steady state and transient analysis.	The validation work that has been completed to date is specific to code version CATHENA MOD-3.5c/Rev 0. An assessment of CATHENA for ACR was completed in 2001 together with a review of validation needs. A more formal review of the applicability of CATHENA to ACR safety analyses, including a validation plan of the code, is being prepared. CATHENA MOD-3.5d is designed to specifically address ACR needs. It is expected that this verification and validation program for CATHENA MOD-3.5d will close all issues related to ACR compliance with the requirements of GAI 98G02.
NUCIRC	No	Heat transport system analysis.	Major changes have been introduced in the NUCIRC-MOD2.001 code (leading to the MOD2.002 version). The PD and CHF models for CANFLEX fuel implemented in NUCIRC-MOD2.002 for ACR applications satisfy present ACR design requirements. Verification of the implementation of the models is being performed and comparison with independent calculations. Validation of the model calculations will be performed against planned tests. As per the Validation Plan for NUCIRC-MOD2.000 and MOD2.001, a number of validation exercises were performed. A validation plan for the

† Although the IST version is available an assessment will be made as to whether the IST or pre-IST version is more appropriate for use in ACR.

** Three-party validation has been done (AECL, Hydro-Quebec, NB Power).

Computer Programs	IST*	Use	Validation Status
			NUCIRC-MOD2.002 code with regard to the ACR applications is being prepared. Once appropriate data for CANFLEX fuel are generated (in a test program at the Stern laboratory), the validation of the ACR options will be performed. It is expected that this verification and validation program will close all issues related to ACR compliance with the requirements of GAI 98G02.
ELESTRES	Yes	Fuel qualification.	The IST version of this code has been qualified for the burnups seen in natural uranium CANDU reactors. The high burnup seen in the ACR core has resulted in the production of a new version of ELESTRES for use on the project. It is expected that the incremental validation and verification program to be conducted on the modified version of the code will close all issues related to ACR compliance with the requirements of GAI 98G02.
Group 2, Minor verification work required:			
SMART	Yes	Radionuclides and aerosols calculation.	SMART has been validated and applied in the past for CANDU 6 & CANDU 9 safety analysis. No major modifications have been deemed necessary for ACR safety analysis. However, some minor changes may be needed, which will require verification. The verification plans for this code are being prepared. It is expected that this verification program will close all issues related to ACR compliance with the requirements of GAI 98G02.
ELOCA	Yes	Transient fuel temperature.	ELOCA-IST 2.1 has been adequately validated by Canadian nuclear power industry for current CANDUs. The code has not been specifically validated for the ACR conditions. No changes have yet been made to the code. However, the models for specific heat capacity and thermal expansion of UO_2 are being upgraded to account for the higher burnups planned for ACR fuel. The changes are deemed minor and will require verification. It is expected that this verification program will close all issues related to ACR compliance with the requirements of GAI 98G02.

Computer Programs	IST*	Use	Validation Status
Group 3, It is not clear yet if changes will require verification or validation:			
DRAGON	Yes	This is a 3-D neutron transport lattice code for super cell calculation.	As a part of the IST, this computer code has been validated by the Canadian nuclear power industry for application in existing CANDU designs. A potential decision on validation of the program for ACR depends on the results from ZED-2 experiments on ACR lattices. It is expected that the validation and verification performed for this code will close all issues related to ACR compliance with the requirements of GAI 98G02.
SOPHAEROS-IST	Yes	Calculation of fission product transport and retention in the Heat Transport System.	SOPHAEROS-IST 2.0 has been assessed for use in ACR safety analysis. In general, the code is suitable for calculations related to ACR however some coding changes related to materials and geometry have been deemed necessary. SOPHAEROS-IST 2.0 validation for application to current CANDU designs is in progress. A validation plan was prepared for application of SOPHAEROS 2.0 to current CANDU safety analysis. However, an incremental validation plan for SOPHAEROS-IST 2.0 is required to ensure ACR conditions are adequately addressed. It is expected that the validation and verification performed for this code will close all issues related to ACR compliance with the requirements of GAI 98G02.
SOURCE	Yes	SOURCE models fission product release from the uranium oxide within fuel elements to the surrounding environment.	Initial validation of SOURCE-IST 2.0 beta-6 for use in safety analysis of the current CANDU reactors was conducted in 2001. The main conclusion from the validation (including some verification work done in parallel) was that the SOURCE code is not currently suitable for use in CANDU safety analysis. Code modifications are required to correct some errors, verification will be completed and the code has to be re-validated. The applicability of the SOURCE IST 2.0 code to ACR safety analysis was assessed. Based on this assessment, no major code changes beyond the generic changes already under way will be required for application to ACR safety analysis. A validation plan has been prepared for application of SOURCE IST 2.0 to current CANDU designs. However, additional experiments to obtain additional data, and additional validation exercises have been recommended for the application to the

Computer Programs	IST*	Use	Validation Status
			ACR design. It is expected that the validation and verification performed for this code will close all issues related to ACR compliance with the requirements of GAI 98G02.
MAAP-CANDU	Yes	MAAP-CANDU is a code to analyse the response of a CANDU plant to severe core damage accidents.	MAAP CANDU is currently being validated for CANDU 6 applications. It is expected that some code changes will be required for ACR. Code changes have not been identified since the design is still under development. If code changes are implemented that are specific to ACR design, additional incremental validation might be required to address all requirements of this GAI.
Group 4, No verification or validation work required:			
GOTHIC	Yes	General purpose containment analysis code. The code has lumped parameter/1D/2D/3D modelling capability as well as hydrogen distribution/burning capability.	The GOTHIC code is being validated by the industry for application in CANDU 6 and CANDU 9 designs. The generic validation of GOTHIC-IST is expected to cover the ACR specific containment design, including the possible use of passive systems. An assessment of GOTHIC-IST for ACR safety analysis has been performed. The assessment did not propose any changes to the current version. Hence, no further validation or verification is needed.
ADDAM	Yes	Dose and dispersion calculation.	The IST validation of this code is in progress. Once validated, the code is expected to be qualified for ACR application, assuming that the exclusion area boundary will not be less than 500 meters.