Pre-Application Safety Assessment Report
related to the Advanced CANDU Reactor 700 MWe

U.S. Nuclear Reactor Commission
Office of Nuclear Reactor Regulation

October 2004

TABLE OF CONTENTS

INTRODUCTION AND SUMMARY viii

ABBREVIATIONS x

1. CLASS 1 PRESSURE BOUNDARY DESIGN (FT1) 1-1
    1.1 Review Scope 1-1
    1.2 Regulatory Review 1-2
    1.3 Potential Policy Issues 1-5
    1.4 Technical Issues 1-5
    1.5 Conclusion 1-9
    1.6 Pressure Tubes Materials Assessment Review Scope 1-9
    1.7 Regulatory Review 1-10
        1.7.1 Appendix A, “General Design Criteria,” to 10 CFR Part 50 1-10
        1.7.2 10 CFR 50.55a, “Codes and Standards” 1-12
        1.7.3 10 CFR 50.60, 10 CFR 50.61, and Appendices G and H to 10 CFR Part 50 1-15
    1.8 Potential Policy Issues 1-20
    1.9 Technical Issues 1-20
        1.9.1 AECL Research Programs 1-20
        1.9.2 Reduction in Creep Life from Environment Effects 1-20
        1.9.3 Delayed Hydride Cracking 1-21
        1.9.4 Safety Margins 1-23
        1.9.5 Inservice Inspection 1-24
        1.9.6 Pressure Tube Fracture Toughness 1-25
        1.9.7 Gas Side Corrosion and Hydrogen Ingress 1-25
        1.9.8 Garter Spring Contact 1-26
        1.9.9 Lead Plant Monitoring Programs 1-26
        1.9.10 Data Scatter 1-27
        1.9.11 Removal of Data from Correlations 1-27
        1.9.12 Blockage of Annulus Gas Flow by Organic Material 1-28
        1.9.13 Leakage of End Fitting Rolled Joints 1-28
        1.9.14 Galvanic Corrosion Between End Fitting and Pressure Tube 1-29
        1.9.15 Material for Garter Spring Spacers 1-29
        1.9.16 NRC Regulations and On-Power Fueling 1-30
    1.10 Materials Assessment Conclusions 1-30

2. DESIGN-BASIS ACCIDENTS AND ACCEPTANCE CRITERIA (FT2) 2-1
    2.1 Design-Basis Accidents—Review Scope 2-1
        2.1.1 Review Guidance 2-1
        2.1.2 Regulatory Review 2-1
    2.2 Potential Policy Issues 2-7
    2.3 Technical Issues 2-7
    2.4 Conclusions 2-8
    2.5 DBA Source Term Assessment 2-8
        2.5.1 Review Scope 2-8

2.5.2 Review Guidance 2-9
2.5.3 DBA Source Term Regulatory Assessment 2-9
2.5.4 Potential Policy Issues 2-10
2.5.5 Technical Issues 2-10
2.5.6 Conclusions 2-11

3. COMPUTER CODES AND VALIDATION ADEQUACY (FT3) 3-1
3.1 Review Scope 3-1
3.2 Regulatory Basis 3-2
3.3 Regulatory Issues 3-3
3.4 Potential Policy Issues 3-3
3.5 Evaluation of the RFSP-ISL, WIMS-ISL, and DRAGON-ISL Reactor Physics Computer Codes 3-3
3.5.1 Reactor Physics Code Descriptions and Review Scope 3-3
3.5.2 Technical Issues—Reactor Physics 3-7
3.5.3 CVR as an Example of Exercising the Code Package 3-18
3.5.4 Staff Audit Calculations—Reactor Physics 3-28
3.5.5 Reactor Physics Codes Assessment Summary 3-29
3.6 The CATHENA Thermal-Hydraulic Computer Code 3-33
3.6.1 Code Description and Theory Manual 3-33
3.6.2 Code Assessment—Experimental Data Comparisons 3-43
3.6.3 NRC Staff Audit Calculations—Thermal-Hydraulics 3-47
3.6.4 CATHENA Code Assessment Summary 3-50
3.7 NRC STAFF PIRT PANELS 3-51
3.7.1 Makeup of the PIRT Panels 3-51
3.7.2 Summary of Evaluations and Conclusions by the Nuclear Analysis PIRT 3-52
3.7.3 Summary of Evaluations and Conclusions by the Thermal-Hydraulic PIRT 3-54
3.8 Assessment of the Thermal-Hydraulic Experimental Database for the ACR-700 3-54
3.8.1 Introduction 3-54
3.8.2 Experimental Database Assessment Summary 3-66
3.9 Quality Assurance 3-67
3.10 Focus Topic 3 Conclusions 3-67

4. SEVERE ACCIDENT DEFINITION AND ADEQUACY OF SUPPORTING RESEARCH AND DEVELOPMENT (FT4) 4-1
4.1 Review Scope 4-1
4.2 Regulatory Issues 4-1
4.3 Potential Policy Issues 4-2
4.4 Technical Issues 4-2
4.5 Conclusions 4-4

5. CANADIAN DESIGN CODES AND QUALITY ASSURANCE STANDARDS (FT6) 5-1
5.1 Review Scope—Design Codes 5-1
5.2 Regulatory Issues—Design Codes 5-1
5.3 Potential Policy Issues—Design Codes 5-3

5.4 Technical Issues—Design Codes 5-4
5.5 Canadian Design Codes Assessment Conclusions 5-4
5.6 Review Scope—Quality Assurance Standards 5-5
5.7 Regulatory Issues—Quality Assurance Standards 5-5
5.7.1 Safety Classification—Safety Related 5-26
5.7.2 Safety Classification—Important to Safety 5-26
5.7.3 International Standards Organization Quality Programs 5-27
5.7.4 Conducting Activities Under Controlled Conditions 5-27
5.7.5 Nondestructive Examination Auditor Qualifications 5-28
5.7.6 Lead Auditor Qualifications 5-28
5.7.7 Audits Conducted by a Third Party 5-28
5.7.8 Documentation of Regulatory Requirements in Working Documents 5-29
5.7.9 Design Controls 5-29
5.7.10 Control of Design Documents Related to External Organizations . . 5-30
5.7.11 Qualification Testing 5-30
5.7.12 Design Approval 5-30
5.7.13 Documents in Use at Prescribed Activity 5-31
5.7.14 Document Review and Approval 5-31
5.7.15 Commercial-Grade Items 5-31
5.7.16 Contractor Performance Monitoring 5-32
5.7.17 Inspection Activities 5-33
5.7.18 Protective Environments 5-33
5.7.19 Notification of Nonconformances 5-33
5.7.20 Trained Auditors 5-34
5.7.21 NQA-1-1994, Part II 5-34
5.7.22 Legacy Issues 5-35
5.7.23 Procurement 5-35
5.7.24 Control of Subcontractors 5-35
5.8 Potential Policy Issues—Quality Assurance Standards 5-36
5.9 Technical Issues—Quality Assurance Standards 5-36
5.10 Quality Assurance Assessment Conclusions 5-36

6. DISTRIBUTED CONTROL SYSTEMS AND SAFETY CRITICAL SOFTWARE (FT7) . . . 6-1
6.1 Review Scope 6-1
6.1.1 ACR-700 Safety Systems 6-1
6.1.2 ACR-700 Safety Support Systems 6-1
6.2 Regulatory Basis 6-2
6.3 Potential Policy Issues 6-4
6.4 Technical Issues 6-4
6.5 I&C Assessment Conclusions 6-6
6.6 Human Factors Engineering Review Scope 6-6
6.7 Regulatory Issues 6-7
6.8 Potential Policy Issues 6-7
6.9 Technical Issues 6-7
6.10 Human Factors Assessment Conclusions 6-9

7. ON-POWER FUELING (FT8) 7-1
7.1 Review Scope 7-1

7.1.1 Introduction 7-1
7.2 System Descriptions 7-2
7.2.1 Description of On-Power Fueling Systems and Components 7-2
7.2.2 Description of Reactor Fueling Sequence 7-6
7.2.3 Description of Safety Features Associated with On-Power Fueling Systems 7-8
7.3 Regulatory/Policy 7-9
7.3.1 Regulatory Review 7-9
7.3.2 Potential Policy Issues 7-10
7.4 Technical Issues 7-10
7.4.1 Use of Canadian Codes and Standards 7-13
7.4.2 Inservice Testing Program 7-13
7.4.3 Classification of System Components 7-14
7.4.4 Review of NRC Regulations for Applicability 7-15
7.4.5 Evaluation of Operational Events and Failure Modes with Significant Consequences 7-18
7.4.6 Design Criteria Applicable to the Design of On-Power Fueling Systems 7-23
7.5 Conclusions 7-27

8. CONFIRMATION OF NEGATIVE VOID REACTIVITY (FT9) 8-1
8.1 Review Scope 8-1
8.1.1 Regulatory Basis 8-1
8.1.2 Overview of Coolant Void Reactivity in the ACR Nuclear Design 8-3
8.1.3 Review Activities 8-5
8.1.4 Information Reviewed 8-7
8.2 Regulatory Issues 8-9
8.3 Technical Issues 8-9
8.3.1 Summary of Evaluations and Conclusions by the Nuclear Analysis PIRT 8-9
8.3.2 Staff Insights from Independent Analysis of Coolant Void Reactivity 8-11
8.3.3 Summary of Staff Insights and Plans for Evaluating the Coolant Void Reactivity Validation Program 8-22
8.4 Potential Policy Issues 8-23
8.5 Conclusions 8-24

9. ACR PROBABILISTIC RISK ASSESSMENT METHODOLOGY (FT11) 9-1
9.1 Review Scope 9-1
9.1.1 Documents That Were Reviewed 9-1
9.1.2 Applicable Regulations, Regulatory Guidance, and Standards 9-2
9.2 Regulatory Issues 9-4
9.3 Potential Policy Issues 9-4
9.3.1 Background 9-4
9.3.2 Potential Policy Issue 9-6
9.4 Technical Issues 9-6
9.4.1 Overview of the ACR PRA Methodology 9-6
9.4.2 Methodological Issues 9-12
9.4.3 Aspects That Warrant Special Staff Attention 9-16

9.5 Conclusions 9-18

10. CANFLEX FUEL DESIGN (FT13) 10-1
10.1 Review Scope 10-1
10.2 Regulatory Issues 10-1
10.3 Potential Policy Issues 10-2
10.4 Technical Issues 10-2
10.5 Conclusions 10-8

11, REFERENCES 11-1

FIGURES

8-1 Fuel Channel Lattice Geometry in Conventional NU CANDU and ACR-700 Cores . . . 8-13
8-2 HELIOS-Computed Neutron Energy Spectra in Fuel Region for Cooled and Voided Core Lattices in Conventional CANDU (top) and ACR-700 (bottom) (0 GWd/t) 8-15
8-3 HELIOS CVR Results for Uniform or Homogeneous Coolant Density Reduction Versus Sector-Wide Heterogeneous Removal of Coolant within Channels (0 GWd/t) 8-18
8-4 HELIOS-Computed Four-Factor Spectral Components of ACR-700 Void Reactivity for Uniform Coolant Density Reduction Versus Sector-Wide Removal of Coolant within Channels 8-19
8-5 ACR-700 2x2 Core Lattice Modeled by HELIOS with Periodic Boundary Conditions . 8-20
8-6 2x2 Lattice Array with Mixed Fuel Burnup of 1.6 and 24.4 GWd/t 8-20

TABLES

3-1 Range of Parameters for Feedback Coefficients in Conventional CANDUs 3-7
3-2 Reported Changes in Reaction Rate Brought about by Coolant Voiding in a Typical CANDU Core 3-19
3-3 Reported WIMS-IST Overprediction of CVR in ZED-2 Experiments for Conventional CANDUs 3-20
3-4 Reported Effect of Input Choices in WIMS-IST 3-21
3-5 Reported Code-to-Code Comparison of Infinite Lattice CVR for Conventional CANDUs as a Function of Fuel Temperature and Burnup 3-22
3-6 Calculated Changes in Reaction Rate upon Voiding in a Conventional CANDU Lattice (Voided Minus Cooled in mk) 3-23
3-7 Comparison of Magnitude of the CVR from WIMS/RFSP-IST and MCNP Whole-Core Voiding in a Conventional CANDU Lattice 3-24
3-8 Lattice Parameters Used in ACR Design Study by AECL 3-25
3-9 Fuel Pin Composition by Lattice Cell Type 3-26
3-10 Reported Components of the Void Reactivity Lattice Cell 2 (2.7 Percent Enriched Uranium) 3-27
3-11 Comparison of Phenomena Contributing to CVR in ACR-700 and CANDU 3-28
3-12 Validation Priorities for CANDU Accident Analysis Thermal-Hydraulic Phenomena . 3-46
3-13 Validation Priorities for CANDU Accident Analysis Fuel and Fuel Channel Thermal-Mechanical Phenomena to Be Validated as Part of the CATHENA Validation Plan 3-47

3-14 T-H Processes Requiring Additional Data 3-56

5-1 Comparison of NQA-1-1994 to CSA-N286 5-7

7-1 NRC Design-Related Regulations Applicable to On-Power Fueling 7-16
7-2 Analysis of Failure Modes and Operating Experience for On-Power Fueling 7-19
7-3 Application of General Design Criteria to On-Power Fueling Systems 7-23

8-1 Effect on CVR from Core Design Choices in Conventional CANDUs and Reference ACR-700 8-14
8-2 Calculated CVR Components (mk) in Core Lattices of Conventional CANDU and Reference ACR-700 8-17
8-3 HELIOS Results for Checkerboard Voided Core Lattices with Uniform and Mixed 2x2 Burnup Arrays 8-21

INTRODUCTION AND SUMMARY

The ACR-700 is an advanced CANDU (CANada Deuterium Uranium) reactor (ACR) design that utilizes horizontal fuel channels passing through a heavy-water moderator tank. As with other CANDU designs, the ACR-700 will be refueled during power operation. Other features of the reactor system, coolant pumps, U-tube steam generators, and pressurizer are similar to U.S. pressurized-water reactor (PWR) designs.

On June 19, 2002, the Atomic Energy of Canada Limited Technologies, Inc. (AECL or the applicant) requested that the U.S. Nuclear Regulatory Commission (NRC) conduct a preapplication review of the ACR-700 for licensing in the United States. The company provided its proposed plan for NRC's preapplication review of the ACR-700 in a letter dated September 26, 2002. On December 18, 2002, the company submitted an amended plan which expanded the scope of the NRC's preapplication review to include the following focus topics (FTs) and their desired outcomes:

- FT1, "Class 1 Pressure Boundary Design"
- FT2, "Design-Basis Accidents and Acceptance Criteria"
- FT3, "Computer Codes and Validation Adequacy"
- FT4, "Severe Accident Definition and Adequacy of Supporting Research and Development"
- FT5, "Design Philosophy and Safety-Related Systems"
- FT6, "Canadian Design Codes and Standards"
- FT7, "Distributed Control Systems and Safety Critical Software"
- FT8, "On-Power Fueling"
- FT9, "Confirmation of Negative Void Reactivity"
- FT10, "Preparation of Standard Design Certification Docketing"
- FT11, "ACR Probabilistic Risk Assessment Methodology"
- FT12, "ACR Technology Base"

The NRC has conducted its ACR-700 preapplication review in two phases. Phase 1 consisted of a series of familiarization meetings designed to provide the staff with a general overview of the advanced CANDU reactor design. Phase 2 provided more specific and detailed information about the ACR design to facilitate the staff's review of the FTs identified in order to (1) develop inputs for the preapplication safety assessment report (PASAR), and (2) provide feedback to AECL in time to support its application for standard design certification.

In addition, given limited NRC resources, AECL requested that the NRC give priority to FTs 1, 3, 8, and 9 during Phase 2 of the preapplication review (see AECL Report No. 108-01200-440-001, "ACR R&D Status Report 2003).

This report contains the staff's assessment of each FT, except for FTs 5, 10, and 12. The remaining FTs cover the content of FTs 5, 10, and 12. The applicant added a new FT13, "ACR CANFLEX Fuel Design," to the scope of the preapplication review. In each section of this report, the staff discusses what was reviewed and what guidance was used to review it, to the extent that applicable guidance exists. The PASAR also identifies requirements for additional information, including any regulatory, policy, and technical issues that AECL must address during the design certification phase to allow the NRC to complete the ACR-700 design safety determination.

On the basis of its review of the materials submitted by AECL, including responses to requests for additional information, the staff concludes that the applicant will need to pursue a number of technical issues in more detail to reach satisfactory conclusions for design certification. The policy, regulatory, and technical issues involved are complex. The staff expects that the review and evaluation of the ACR-700 design will be more challenging, will involve expenditure of more resources, and may take longer to review than a typical light-water design. Notwithstanding, based on the information provided, the staff believes at this time that AECL will ultimately be able to satisfactorily address these policy, regulatory, and technical issues during the design certification review.

ABBREVIATIONS

1/v	v is neutron velocity
ABWR	advanced boiling-water reactor
ACI	American Concrete Institute
ACR	advanced CANDU reactor
ACRS	Advisory Committee on Reactor Safeguards
AC	acceptance criteria
AECL	Atomic Energy of Canada Limited Technologies, Inc.
AISI	American Iron and Steel Institute
ALWR	advanced light-water reactor
ANS	American Nuclear Society
AOO	anticipated operational occurrence
ASEP	Accident Sequence Evaluation Program
ASME Code	American Society of Mechanical Engineers Boiler and Pressure Vessel Code
AST	alternative source term
B	boron
BNL	Brookhaven National Laboratory
BOL	beginning-of-life
BTP	Branch Technical Position
BWR	boiling-water reactor
CANDU	CANada Deuterium Uranium
CCF	common-cause failure
CCFL	counter-current flow limitation
CCFP	conditional containment failure probability
CDF	core damage frequency
CDS	core damage states
CE	containment event
CET	containment event tree
CFR	Code of Federal Regulations
CHF	critical heat flux
CNSC	Canadian Nuclear Safety Commission
COL	combined license
CRL	Chalk River Laboratories
CS	core support
CSA	Canadian Standards Association
CTR	calandria tube rupture
CVR	coolant void reactivity
CWIT	cold water injection test
DBA	design-basis accident
DBE	design-basis event
DG	draft regulatory guide
DHC	delayed hydride cracking
DNB	departure from nucleate boiling
DNBR	DNB ratio
Dy	dysprosium
ECC	emergency core cooling
ECCS	emergency core cooling system

ECI	emergency coolant injection
ECR	equivalent cladding reacted
EFF	end fitting failure
EFPY	effective full-power year
EOP	emergency operating procedure
FB	flow blockage
FCI	fuel-coolant interaction
FCS	fueling control system
FCTR	Fuel Channel Technology Report
FES	fuel element simulators
FH	fuel handling
FHDCS	fuel-handling distributed control system
FNU	fresh natural uranium
FoM	figures of merit
FPD	full power day
FR	Federal Register
FSAR	final safety analysis report
FT	focus topic
GDC	general design criterion/criteria
GENHTP	Generalized Heat Transfer Package
gpm	gallons per minute
Gwd/t	gigawatt days per ton
HFE	human factors engineering
HFEPP	Human Factors Engineering Program Plan
HFESR	Human Factors Engineering Summary Report
HRA	human reliability analysis
HSI	human-system interface
HTS	heat transport system
I	iodine
I&C	instrumentation and control
IAEA	International Atomic Energy Agency
i.a.w.	in accordance with
IEEE	Institute of Electrical and Electronics Engineers
IN	information notice
IPE	individual plant examination
ISI	inservice inspection
ISO	International Standards Organization
IST	Industry Standard Toolset
K	Kelvin
LASH	large-scale header test
LBB	leak-before-break
LCDA	limited core damage accident
LCDF	limited core damage frequency
LERF	large early release frequency
LOCA	loss-of-coolant accident
LOCA/LOECC	LOCA with coincident loss of emergency core cooling
LOF	loss-of-flow
LOFW	loss-of-feedwater

LOOP	loss of offsite power
LOR	loss-of-regulation
LRF	large release frequency
LTC	long-term cooling
LVRF	low void reactivity fuel
LWR	light-water reactor
MAAP	Modular Accident Analysis Program
MCNP	Monte Carlo N-Particle
MFMI	melt expulsion experiments
mk	milli-k
mk/EC	milli-k per degree celsius
mk/Wt%	milli-k per weight percent
MOX	mixed oxide
MSLB	main steamline break
MW	megawatt
MWe	megawatt electric
MWh/kg(U)	megawatt hours per kg of Uranium
NDT	nondestructive testing
NRC	U.S. Nuclear Regulatory Commission
NU	natural uranium
OM	Operations & Maintenance
PARCS	Purdue Advanced Reactor Core Simulator
PASAR	preapplication safety assessment report
PDS	plant damage state
PIE	postirradiation examination
PIRT	phenomena identification and ranking table
ppb	parts per billion
ppm	parts per million
PRA	probabilistic risk assessment
PSA	probabilistic safety assessment
PTs	pressure tubes
PTS	pressurized thermal shock
Pu	plutonium
PWR	pressurized-water reactor
QA	quality assurance
R&D	research and development
RAI	request for additional information
RCPB	reactor coolant pressure boundary
RCS	reactor coolant system
RES	Office of Nuclear Reactor Regulations (NRC)
RFSP	Reactor Fueling Simulation Program
RG	regulatory guide
RIH	reactor inlet header
RPV	reactor pressure vessel
S/U	sensitivity and uncertainty analysis
SAFDL	specified acceptable fuel design limit
SCC	stress-corrosion cracking
SCDA	severe core damage accident

SCDF	severe core damage frequency
SCFB	severe channel flow blockage
SDS1	Shutdown System 1
SDS2	Shutdown System 2
SER	safety evaluation report
SEU	slightly enriched uranium
SFB	stagnation feeder break
SMA	seismic margins analysis
SRM	staff requirements memorandum
SRP	Standard Review Plan
SS	stainless steel
SSC	structure, system, and component
SSE	safe-shutdown earthquake
T-H	thermal-hydraulic
TEDE	total effective dose equivalent
TMI	Three Mile Island
TRC	trip computer
U	uranium
U.S.	United States
UPM	unified partials method
V&V	validation and verification
w/o	without
wt%	weight percent
ZEEP	Zero Energy Experimental Pile

1. CLASS 1 PRESSURE BOUNDARY DESIGN (FT1)

1.1 Review Scope

By letter dated July 30, 2003, Atomic Energy of Canada Limited Technologies, Inc. (AECL or the applicant), requested feedback from the U.S. Nuclear Regulatory Commission (NRC or the staff) on several focus topics (FTs) as part of the advanced CANDU reactor—700 megawatt electric (MWe) output (ACR-700) preapplication review. Focus Topic 1 involves the Class 1 pressure boundary design. The Class 1 pressure boundary includes the reactor coolant system (RCS) pressure boundary of the ACR-700 design. AECL specifically requested that the staff accept the principle design features of the ACR-700 RCS pressure boundary (i.e., the use of zirconium-alloy (Zr-2.5wt%Nb) pressure tubes (PTs), rolled joints, closure plugs, modified American Iron and Steel Institute (AISI) type 403 martensitic stainless steel (SS) end fittings, and fueling machines as part of a Class 1 pressure boundary). The ACR USA Report 108US-31100-LS-001, Revision 0, "The Technology of CANDU Fuel Channels" (Fuel Channel Technology Report (FCTR)) describes these design features. This assessment addresses the mechanical design aspects of the AECL request. The materials assessment described in Section 1.7.2 of the ACR USA Report discusses additional material considerations associated with the zirconium-alloy PTs and the type 403 SS end fittings.

The staff reviewed the mechanical design aspects of the ACR-700 pressure boundary design against the current regulations to determine whether exceptions or alternatives to the existing NRC regulations are necessary. The staff considered the following regulations in its review:

- Title 10, Section 50.55a(c)(1), of the *Code of Federal Regulations* (10 CFR 50.55a(c)(1)) requires that components which are part of the reactor coolant pressure boundary (RCPB) meet the requirements for Class 1 components in Section III of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), except as provided in paragraphs (c)(2), (c)(3), and (c)(4). The staff reviewed the mechanical design of the ACR-700 PTs, closure plugs, end fittings, and fueling machines for conformance with this requirement.

- General Design Criterion (GDC) 2, "Design Bases for Protection Against Natural Phenomena," requires, in part, that structures, systems, and components (SSCs) important to safety be designed to withstand the effects of natural phenomena, such as earthquakes, without losing the capability to perform their safety functions. GDC 2 also requires that the design basis for these SSCs reflect appropriate combinations of normal and accident conditions with the effects of the natural phenomena. The staff reviewed the ACR-700 PT, end fitting, and closure plug mechanical design for conformance with this aspect of GDC 2.

- GDC 4, "Environmental and Dynamic Effects Design Bases," requires, in part, that SSCs important to safety be designed to accommodate the effects of and be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents (LOCAs). GDC 4 also requires that these SSCs be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures. GDC 4 further states that the dynamic effects associated with

postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping. The staff reviewed the ACR-700 PT, end fitting, and closure plug mechanical design for conformance with GDC 4.

- GDC 14, "Reactor Coolant Pressure Boundary," requires that the RCPB be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross leakage. The staff reviewed the PT rolled joint and closure plug mechanical design for conformance with this criteria.

- GDC 15, "Reactor Coolant System Design," requires, in part, that the RCS be designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including anticipated operational occurrences (AOOs). The staff reviewed the PT, end fitting, and closure plug mechanical design for conformance with this criteria.

Section 3.9.3 of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," (hereinafter referred to as the SRP) provides staff guidance regarding acceptable methods to satisfy the requirements of GDC 2 and 4 for component design load combinations and stress limits. The staff also reviewed the mechanical design of the Class 1 pressure boundary design against the guidance for load combinations and stress limits provided in SRP Section 3.9.3.

1.2 Regulatory Review

AECL provided a general description of the reactor design in the ACR USA Report. The CANDU reactor consists of a large cylindrical tank with end shields joined by tubular penetrations called calandria tubes. The PTs are contained within the calandria tubes and are separated from them by spacers in the annular gap. The combination of the calandria tubes and PTs are called fuel channels. The fuel bundles are contained within the PTs, and the coolant flows through the PTs. The PTs are connected to end fittings which contain openings for reactor coolant inlet and outlet flow. The PTs are sealed by closure plugs, except during fueling. The PTs, end fittings, and closure plugs form part of the RCPB.

The CANDU reactor differs significantly from current boiling- and pressurized reactor designs licensed under 10 CFR Part 50. However, the staff considers many of the regulations applicable to the RCPB design, in accordance with the provisions of 10 CFR 52.83. Therefore, the staff used the applicable provisions of 10 CFR Part 50 as the basis for this assessment. As required by 10 CFR 50.55a(c)(1), components that are part of the RCPB must meet the requirements for Class 1 components in Section III of the ASME Code, except as provided in paragraphs (c)(2), (c)(3), and (c)(4). The RCPB, as defined in 10 CFR 50.2, includes, in part, those pressure-containing components of boiling and pressurized water-cooled nuclear power reactors, such as pressure vessels, piping, pumps, and valves, which are part of the RCS. The staff reviewed the ACR-700 design against the 10 CFR 50.2 definition of the RCPB.

A unique feature of the CANDU reactor design is the on-power fueling. Fueling machines at each end of the reactor remove spent fuel bundles and insert new fuel bundles while the reactor

is operating. The outboard ends of each PT end fitting are sealed by removable channel closures that provide access for the fueling machines to remove irradiated fuel from the channel and insert new fuel into it, while the reactor is operating. As a consequence, the fueling machines become part of the RCPB during the fueling operation. The channel closures form the RCPB the remainder of the time.

The NRC regulations at 10 CFR 50.2 provide the following definition of the RCPB:

> Reactor coolant pressure boundary means all those pressure-containing components of boiling and pressurized water-cooled nuclear power reactors, such as pressure vessels, piping, pumps, and valves, which are:
>
> - Part of the reactor coolant system, or
> - Connected to the reactor coolant system, up to and including any and all of the following:
>
> (i) The outermost containment isolation valve in system piping which penetrates primary reactor containment,
>
> (ii) The second of two valves normally closed during normal reactor operation in system piping which does not penetrate primary reactor containment,
>
> (iii) The reactor coolant system safety and relief valves.

The regulations also specify, in 10 CFR 50.55a(c)(1), the following ASME Code classification for RCPB components:

> Components which are part of the reactor coolant pressure boundary must meet the requirements for Class 1 components in Section III of the ASME Boiler and Pressure Vessel Code, except as provided in paragraphs (c)(2), (c)(3), and (c)(4) of this section.

In lieu of meeting the Class 1 requirements of the ASME Code, 10 CFR 50.55a(c)(2) specifies the following requirements for proposed alternatives:

> Components which are connected to the reactor coolant system and are part of the reactor coolant pressure boundary as defined in § 50.2 need not meet the requirements of paragraph (c)(1) of this section, provided:
>
> (i) In the event of postulated failure of the component during normal reactor operation, the reactor can be shut down and cooled down in an orderly manner, assuming makeup is provided by the reactor coolant makeup system; or
>
> (ii) The component is or can be isolated from the reactor coolant system by two valves in series (both closed, both open, or one closed and the other open). Each open valve must be capable of automatic actuation and, assuming the other valve is open, its closure time must be such that, in

> the event of postulated failure of the component during normal reactor operation, each valve remains operable and the reactor can be shut down and cooled down in an orderly manner, assuming makeup is provided by the reactor coolant makeup system only.

AECL provided a description of the extent of the RCPB in a March 31, 2004, submission to the NRC, "Response to NRC's Requests for Additional Information (RAIs) #5 on Class 1 Pressure Boundary Design and Materials Review of Fuel Channels and On-Power Fueling." Figure 91-1 of the submission shows the extent of the Class 1 RCPB when the fueling machine is not attached. The RCPB includes reactor components, such as the PTs, end fittings, and closure plugs. The RCPB also includes those portions of the major components in contact with the reactor coolant, such as the steam generators, pumps, pressurizer, and connecting piping. Some of the piping attached to the main coolant loop show the Class 1 designation only out to the first isolation valve. According to AECL, some of the valve classifications currently shown for the ACR-700 may change to meet U.S. regulations, as stated above. Safety classification of ACR-700 RCPB components to satisfy the requirements of 10 CFR 50.55a(c)(2) will require further review during design certification.

Figure 91-2 of the AECL March 31, 2004, submission shows the extent of the Class 1 RCPB with the fueling machine attached to a PT. The closure plugs no longer form part of the RCPB. Instead, the fueling machine and its associated process connections form part of the RCPB. AECL indicated that small-diameter penetrations of the RCPB use threaded joints. The ASME Code Class 1 piping rules do not allow the use of threaded joints without an additional pressure boundary sealing mechanism. AECL also indicated that Swagelock fittings have been used for small-diameter tubing. The ASME Code Class 1 piping rules contain criteria applicable to tubing fittings. AECL should apply the ASME piping rules to the design of tubing and fittings that are part of the Class 1 pressure boundary. These issues will require additional review during design certification.

AECL further indicated that parts of the fueling machine pressure boundary, not designated as Class 1, are acceptable, according to the definition of the RCPB provided in 10 CFR 50.55a(c). In order to satisfy this requirement, AECL should either extend the Class 1 boundary to the second isolation valve for each line attached to the RCPB, or demonstrate that coolant loss from failure of the line will not exceed the normal makeup capacity. Application of 10 CFR 50.55a(c)(2) for small-diameter penetrations will require further review during design certification.

As indicated above, safety classification of the ACR-700 RCPB components is an issue that will require further review during the design certification. AECL must demonstrate that the RCPB classification satisfies the requirements of 10 CFR 50.55a(c)(2).

AECL states that the ACR-700 pressure boundary design satisfies the existing regulations, with the exception of the zirconium-alloy and AISI type 403 SS materials used for the PTs and end fittings. AECL plans to use the provisions of 10 CFR 50.55a(a)(3) to propose an alternative to using ASME Code-approved materials for these components. However, the Director of the Office of Nuclear Reactor Regulation must authorize the use of the provisions in 10 CFR 50.55a(a)(3). Section 1.7.2 of this report provides additional discussion of this topic.

The applicant's March 31, 2004, submission indicates that it will use the 2001 edition of the ASME Code to design the RCPB piping. The staff will determine applicable ASME Code editions and addenda at the time of design certification, in accordance with the requirements of 10 CFR 50.55a(c)(3).

GDC 4 requires that SSCs be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures. As stated previously, the CANDU reactor design differs significantly from current light-water reactor (LWR) designs. The CANDU reactor contains a large number of individual PTs which form part of the RCPB. The ACR-700 design includes consideration of a PT rupture. The end fittings are extensions of the PTs that project outside the calandria assembly. These end fittings are sealed by closure plugs. The end fittings and closure plugs are part of the RCPB. The applicant's March 31, 2004, submission indicates that the ACR-700 design did not consider the effects of PT end fitting failures (EFFs). It will likely be necessary to evaluate the compliance of PT EFFs and end fitting closure plug failures with GDC 4 requirements. This issue will require further review during design certification.

1.3 Potential Policy Issues

The staff did not identify any policy issues associated with the mechanical design of the ACR-700 PTs, end fittings, or closure plugs.

1.4 Technical Issues

A key design feature of the fuel channel is the use of a rolled joint between the PT and the end fitting. The rolled joint forms part of the RCPB and, consequently, must remain leaktight. Normally, a seal weld at the rolled joint would assure leaktightness. However, because the zirconium-alloy PT material cannot be welded satisfactorily to steels due to the formation of brittle intermetallic compounds, the design does not use a seal weld. Canadian Standards N285.0-95, "General Requirements for Pressure Retaining Systems and Components in CANDU Nuclear Power Plants," and N285.2-99, "Requirements for Class 1C, 2C, and 3C Pressure Retaining Components and Supports in CANDU Nuclear Power Plants," contain criteria for evaluating the PT design.

In the FCTR, AECL indicated that it evaluated the PT and end fitting design using the criteria contained in the ASME Code. AECL indicated that the only aspect of the design of the rolled joint that does not meet ASME Code criteria is the material selections for the PTs and end fittings. AECL has proposed alternative criteria provided in Canadian Standard CAN/Canadian Standards Association (CSA) N285.6, "Material Standards for Reactor Components for CANDU Nuclear Power Plants," in accordance with the requirements of 10 CFR 50.55a(3). The materials assessment of Section 1.6 of this PASAR discusses issues related to the material selection. The following paragraphs discuss the mechanical design of the joint.

The rolled joint configuration for the PT end fitting contains three circumferential grooves in the end fitting hub. The rolled joint is made by inserting one end of a PT into the hub at the inboard end of an end fitting after it has been heated, which, upon cooling, forms a zero-clearance fit. The PT is then rolled into this hub with a nominal 13.5 percent reduction in wall thickness, which causes the PT material to be extruded into the grooves of the end fitting hub. The cold working

of the PT into the hub produces compressive residual stresses. These compressive residual stresses provide a seal to prevent reactor coolant leakage through the joint.

GDC 14 requires that the RCPB be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. In order to satisfy GDC 14, AECL must demonstrate that the rolled joint remains leaktight throughout its service life. AECL proposes to use ASME Code Class 1 criteria to evaluate the mechanical design of the joint. ASME Code Class 1 requirements generally provide a design basis sufficient to satisfy GDC 14 requirements. However, the ASME Code does not contain specific rules for the fabrication or testing of roll-expanded joints in Class 1 vessels. Therefore, AECL specifies additional requirements for fabrication, testing, and inspection of the roll-expanded joint in CAN/CSA-N285.2, which requires that the joint be designed in accordance with the rules of Section III, paragraph NB-3200, of the ASME Code. It also requires that the prototype joints be subjected to pullout and performance tests to determine the structural integrity of the joints under simulated service conditions. The performance testing of the prototype joints must be performed in conditions that simulate actual service conditions in order to demonstrate that the joint can remain leaktight throughout its design life.

The FCTR cites the operating experience with the current CANDU reactor designs as a demonstration of the reliability of the rolled joint design. However, the report also indicates that the end fitting diameter in the ACR-700 has been reduced because of a smaller lattice pitch in its design. In addition, the ACR-700 PTs are thicker than those in previous CANDU reactors. These changes impact the joint fabrication process and could affect the leaktightness of the joint during its service life. AECL Report No. 108-01200-440-001, “ACR R&D Status Report 2003” (hereinafter referred to as the ACR status report), indicates that design of the ACR-700 rolled joint has not been finalized. The final rolled joint design requires prototype joint testing to satisfy the design requirements. The final rolled joint design, as well as its performance during prototype testing, will require review during design certification.

As stated previously, CAN/CSA-N285.2 requires that the PT end fitting rolled joints be designed in accordance with the requirements of Section III, paragraph NB-3200, of the ASME Code. Several technical issues related to the design of the PTs and end fittings must be resolved during design certification. These technical issues include the effect of the reactor coolant environment on the fatigue life of the PTs and end fittings, the creep behavior of the PTs, and the potential for delayed hydride cracking (DHC) of the PTs. Section 1.9.3 of this PASAR provides an additional discussion of issues related to creep and DHC of the PTs. The following paragraphs discuss the fatigue evaluation of the PTs and end fittings.

Paragraph NB-3200 of Section III of the ASME Code requires an evaluation of the PTs and end fittings for cyclic operation. Although the prototype testing of the rolled joint discussed above can provide additional assurance that the joint will remain leaktight during service conditions, it will not demonstrate that ASME Code design criteria have been satisfied for the PT or end fitting. Therefore, an ASME Code evaluation for cyclic operation is necessary. CAN/CSA-N285.2 requires development of irradiated fatigue curves for the evaluation of the PTs, since the existing ASME Code design fatigue curves do not apply to the irradiated PT material.

Research for LWR materials has shown that the environment can have a significant effect on the fatigue and stress-corrosion cracking (SCC) resistence of austenitic and ferritic steels. In pure water with oxygen levels in the parts per billion (ppb) range (similar to oxygen levels in PWRs), the fatigue life of austenitic SS components is reduced relative to the fatigue life in air. The fatigue life of austenitic SS components may be reduced by a lesser extent for oxygen levels in the parts per million (ppm) range (similar to boiling-water reactors (BWRs)). On the other hand, for oxygen levels in the ppm range, the susceptibility of austenitic SS components to SCC is increased. For ferritic steels, a significant reduction in fatigue life can occur when oxygen levels are in the ppm range, and a moderate reduction in fatigue life can occur at the ppb level. Ferritic steels are more resistant to SCC, but less resistant to erosion corrosion.

AECL indicated that a limited amount of information exists regarding the effects of environment on the fatigue behavior of the PT. Fatigue testing of zirconium-alloy PT material at 300 EC (572 EF) in both air and low-oxygen water established fatigue design curves for the material for both the axial (longitudinal) and transverse directions of the PT. The testing demonstrated a reduction in the fatigue life in low-oxygen water relative to that in air for the unirradiated material. However, fatigue tests of the PT material were not performed at the ACR outlet temperature conditions. Therefore, the impact of the higher ACR temperature on the design fatigue curves will require review during design certification. Limited testing of irradiated PT material in air was also carried out. The results show a reduction in fatigue life for irradiated material. AECL needs to develop fatigue curves that account for the conditions of the ACR PT environment, including oxygen level, temperature, and radiation effects. Development of design fatigue curves that apply to the ACR environmental conditions is a topic that will require further review during design certification.

AECL indicated that fatigue analysis of the end fitting and closure plug material is based upon fatigue curves provided in Appendix 1 to ASME Section III, Division 1. AECL also indicated that it applied the ASME rules for fatigue analysis to all Class 1 components including the PTs, end fittings, and closure plugs. The staff notes that AECL did not consider environmental effects in the fatigue design for the end fitting and closure plug material. The current ASME Code does not take into account the reduction in fatigue life caused by the environmental conditions of the reactor water. Austenitic SSs and some nickel-alloy components have been observed to experience a significant reduction in fatigue life at low levels of oxygen in pure water in experiments conducted under LWR conditions. The effect of the ACR-700 coolant environment on the fatigue life of the end fittings and closure plugs will require review during design certification.

The FCTR indicates that the PTs are evaluated for both beginning-of-life (BOL) and end-of-design-life conditions to account for PT dimensional changes that occur over the life of the fuel channel. These dimensional changes include the effects of thermal expansion and creep deformation, as well as irradiation-induced creep and growth deformation. The FCTR discusses the form of the equations used to evaluate creep deformation of the PT. According to AECL, neutron irradiation of the stressed material is the primary cause of the creep deformation. No staff guidance is applicable to the review of irradiation creep for pressure boundary design. The staff will need to perform a detailed review of the development and the application of the equations used to evaluate creep deformation of the PTs. This is a topic that will require in-depth technical review during the design certification.

The creep deformation causes the PT dimensional changes. These dimensional changes result in an increase in PT stresses resulting from the design loads. The FCTR indicates that the tensile strength of the PT material also increases with irradiation. The report further indicates that credit is taken for part of the strength increase caused by irradiation in order to maintain the intended design margins at the end-of-design-life conditions. The FCTR does not specify the amount of strengthening that will be credited in the evaluation. No staff guidance is applicable to the review of radiation strengthening of pressure boundary material. This topic will require further review during design certification.

Another key design feature of the fuel channel is the end closure design. The end closure relies on a mechanical seal to maintain leaktightness of the RCPB. During reactor operation, the outboard end of each end fitting is normally closed and sealed by a closure plug. The closure plug must be removed during reactor operation to facilitate on-power fueling. To allow the on-power refueling of a channel, the fueling machine is connected and sealed to an end fitting before the closure plug is removed. The closure plug relies on a flexible metallic ring that forms a seal against an edge in the end fitting. The seal ring is supported and retained by the body of the closure plug, which is held in position by jaws engaging a groove in the end fitting. CAN/CSA-N285.2-99 requires that components of the end closure that support the flexible seal ring be designed, manufactured, and tested to determine if they meet the requirements for ASME Class 1 pressure-retaining components. The standard also requires that the end closures be locked in place by closure safety locks, and that these closures be leak tested each time they are installed before removal of the fueling machine.

The ACR status report indicates that a new end closure seal design is being developed to improve the leak tightness of the closure. The report indicates that this new end closure seal design is based on the existing CANDU-6 design. The report also indicates that design verification testing of the new end closure design will be integrated with the testing of the fuel-handling system and end fitting design. The final end closure design, as well as its performance during design verification testing, will require review during design certification.

GDC 15 requires, in part, that the RCS be designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including AOOs. GDC 2 requires, in part, that SSCs important to safety be designed to withstand the effects of natural phenomena, such as earthquakes, without losing the capability to perform their safety functions. GDC 2 also requires that the design basis for these SSCs reflect appropriate combinations of normal and accident conditions with the effects of the natural phenomena. Section 3.9.3 of the SRP provides staff guidance on load combinations and stress limits that satisfy the GDC requirements.

AECL listed the load combinations and stress limits it used for the design of the RCPB in its March 31, 2004, submission to the NRC. AECL indicated that the load combinations for the design of Class 1 pressure boundary components and piping will be consistent with the staff guidance provided in SRP Section 3.9.3. Although the load combinations and stress limits proposed by AECL are consistent with the guidance provided in SRP Section 3.9.3, the stress limits for earthquake loads appear to conflict with the criteria referenced in the FCTR. The FCTR indicates that a more conservative stress limit is used for the seismic design of the fuel channel assembly. The stress limit used for load combinations that include earthquake loads will require clarification during design certification.

AECL also indicated that the ASME special stress limits in ASME Code, Section III, paragraph NB-3227.3, applicable to progressive distortion of nonintegral connections, will be satisfied for the PT/end fitting rolled joints. The rolled joint relies on the residual compressive stresses between the PT and end fitting produced by the rolling process to ensure a leaktight joint. The staff believes that this joint constitutes a nonintegral connection. A nonintegral connection is subject to loosening of the mating parts if the material yields under the applied loads. The ASME Code provision limits the primary plus secondary stress intensity to the material yield stress to prevent progressive loosening of the mating parts. The staff considers it important that these stress limits be satisfied for all load combinations to assure that the rolled joints remain leaktight under all postulated design conditions.

GDC 4 states that SSCs important to safety must be protected from the dynamic effects associated with postulated pipe ruptures or other equipment failures, unless the probability of the postulated piping rupture can be demonstrated, based on analyses reviewed and approved by the Commission, to be extremely low under conditions consistent with the design basis for the piping. The FCTR indicates that the PT design is based on leak-before-break (LBB) principles.

The application of LBB requires the use of “analyses” reviewed and approved by the Commission. The Commission has accepted, for this purpose, those analyses which have been performed in accordance with NUREG-1061, Volume 3, “Report of the U.S. Nuclear Regulatory Commission Piping Review Committee, Evaluation of Potential Pipe Breaks,” and draft SRP Section 3.6.3, “Leak-Before-Break Evaluation Procedures.”

Based on information in the applicant’s March 31, 2004, submission, it is the staff’s understanding that the ACR-700 design includes consideration of the dynamic effects of a PT rupture up to and including the complete severance of a PT. Therefore, AECL does not intend to request approval of LBB for the PT design. AECL specifically noted that the calandria tube which surrounds a PT will have an extremely low probability of failure in a PT rupture event. The staff will perform a more detailed review of the PT rupture evaluation during design certification to ensure that the ACR-700 PTs meet the requirements of GDC 4.

1.5 Conclusions

On the basis of the information provided by AECL during the preapplication review, and pending resolution of the regulatory and technical issues discussed above, the staff has not identified any issues related to the mechanical design of the PTs that would preclude certification of the ACR-700 design.

1.6 Pressure Tubes Materials Assessment Review Scope

For the preapplication evaluation of this topic, the NRC staff reviewed the FCTR, which was provided by AECL to support its preapplication of the ACR-700. Based on this review, the staff identified a number of regulatory and technical issues in the materials engineering area. The following information describes the scope of this review and the issues identified. Although the following discussion applies specifically to the fuel channels in the ACR-700, many of the issues potentially apply to the Class 1 pressure boundary and other SSCs in the ACR-700 design. In addition, other potential issues unrelated to the fuel channels, but identified during the course of

the staff's review, are also discussed. Although the following sections document the staff's review, AECL should note that additional issues will likely be identified during the course of the more detailed design certification review.

As required by 10 CFR 52.48, "Standards for Review of Applications," applications must meet the requirements of 10 CFR Part 50 and its appendices, provided those standards are technically relevant to the design proposed for the facility. From a materials engineering perspective, the staff identified the following regulations as pertinent to fuel channel design:

- Appendix A to 10 CFR Part 50, GDC 1, 4, 14, 15, 30, 31, and 32, "General Design Criteria for Nuclear Power Plants"
- 10 CFR 50.55a, "Codes and Standards"
- 10 CFR 50.60, "Acceptance Criteria for Fracture Prevention Measures for Light-Water Nuclear Power Reactors for Normal Operation"
- 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events"
- Appendix G, "Fracture Toughness Requirements," to 10 CFR Part 50
- Appendix H, "Reactor Vessel Material Surveillance Program Requirements," to 10 CFR Part 50.

Each of these regulations is discussed further below. To the extent applicable, the staff performed its review based upon the acceptance criteria in the SRP. In the sections that follow, the staff discusses the extent to which the ACR-700 design appears to comply with these regulations, and identifies those areas for which the staff requires additional information to complete the safety evaluation during design certification. In the subsequent materials engineering sections of this report, the staff identifies specific 10 CFR Part 50 requirements that may not be technically relevant to the ACR-700 design. As stated in 10 CFR 52.48, the NRC will review design certification applications against standards set forth in various requirements. For those standards that are technically relevant to the design, the staff will need to address, as part of the design certification review, how to apply the requirements of 10 CFR 52.48 to the ACR-700 design, and whether the applicant had appropriately documented in the design certification those requirements that are not technically relevant.

1.7 Regulatory Review

1.7.1 Appendix A, "General Design Criteria," to 10 CFR Part 50

With respect to the GDC contained in Appendix A to 10 CFR Part 50, 10 CFR 50.34(a)(3)(i) indicates, in part, that this appendix establishes the minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have previously been issued by the Commission. This appendix also provides guidance to applicants for construction permits in establishing principal design criteria for other types of nuclear power units. For the materials engineering sections of this

report, the staff considers the ACR-700 design to be similar to previously licensed water-cooled nuclear power plants. As such, during this review, the staff considered the GDC as the minimum requirements for the principal design criteria. However, the staff will need to generically determine (1) whether, for the purposes of 10 CFR 50.34(a)(3)(i), the ACR-700 is a water-cooled nuclear power plant, similar in design and location to plants for which construction permits have previously been issued by the Commission, and (2) whether the GDC establish the minimum requirements for the principal design criteria for the ACR-700. These determinations involve regulatory and technical questions that the design certification process will need to address.

GDC 1, 4, 14, 15, 30, 31, and 32 are the principal criteria that apply to the staff's materials engineering review of the ACR-700 fuel channel design. These GDC are:

GDC 1 - Quality standards and records
GDC 4 - Environmental and dynamic effects design bases
GDC 14 - Reactor coolant pressure boundary
GDC 15 - Reactor coolant system design
GDC 30 - Quality of reactor coolant pressure boundary
GCD 31 - Fracture prevention of reactor coolant pressure boundary
GDC 32 - Inspection of reactor coolant pressure boundary

In terms of the materials engineering aspects of the design of the ACR-700, the staff will typically verify compliance with these GDC by reviewing an application against the NRC's SRP. The SRP establishes the criteria the NRC staff uses in evaluating whether an applicant meets the Commission's regulations. The SRP is not a substitute for the regulations, and compliance is not a requirement. Nonetheless, as required by 10 CFR 50.34(h), applications for a light-water-cooled nuclear power plant must include an evaluation of the facility against the SRP. Pursuant to 10 CFR 50.34(h)(2), this evaluation must include an identification and description of all differences in design features, analytical techniques, and procedural measures proposed for a facility, as well as the corresponding features, techniques, and measures detailed in the SRP acceptance criteria. Where a difference exists, the evaluation must discuss how the proposed alternative provides an acceptable method of complying with those Commission rules or regulations, or portions thereof, that underlie the corresponding SRP acceptance criteria.

The materials engineering sections of the SRP are divided in several different ways. For example, there is a section on RCPB materials (Section 5.2.3), as well as separate sections on specific SSCs (e.g., steam generators are addressed in Section 5.4.2.1; reactor vessels are addressed in Sections 5.3.1, 5.3.2, and 5.3.3). Because the PTs are unique to the ACR-700 design, the SRP does not include a specific section pertaining to them. As a result, the staff reviewed the PT design using guidance consistent with that in SRP Section 5.2.3 and other SRP sections, although this guidance is not completely applicable to the ACR-700 PTs. The staff used the guidance contained in SRP Section 5.2.4 for the inspection of PTs. This guidance is also not directly applicable to the ACR-700 PTs. This lack of complete guidance for review, evaluation, and acceptance of the PT design raises a potential regulatory issue which the NRC staff will need to address before design certification. This issue, which may be applicable to the review of other components, represents a challenge to the staff's review which ensures that all SSCs important to safety are reviewed and evaluated for compliance with the regulations.

The SRP sections related to RCPB integrity and inspection, to a large extent, indicate that if the design (e.g., materials selection), fabrication, erection, and testing of the RCPB follow the ASME Code, then the applicable GDC are met. For the most part, AECL plans to follow the ASME Code for the design, fabrication, erection, and testing of the ACR-700; therefore, the NRC staff does not anticipate a significant impediment, in terms of the SSCs, to the certification of the ACR-700 design, with respect to compliance with the GDC. However, because of the unique nature of the ACR-700 design, several aspects of the ACR-700 are not addressed by either the SRP or the ASME Code. For the PTs, end fittings, closure plugs, and their support structures, AECL will rely on the design, fabrication, erection, testing, and periodic inspection requirements of the CAN/CSA-N Series standards. Section 1.7.2 of this report discusses the regulatory issues associated with AECL's reliance on the CAN/CSA-N Series standards. Review of the CAN/CSA-N Series standards is likely to lead to technical issues yet to be identified.

The SRP, in several instances, includes some additional guidance not contained within the ASME Code (e.g., ultimate strength limits). In addition, specific regulatory requirements establish some of the necessary conditions which must be met by light-water nuclear power reactor designs in order to comply with the GDC that address RCPB integrity. These criteria would include, but may not be limited to, GDC 14, 15, 30, 31, and 32. The staff discusses this topic in Section 1.7.3 of this report.

The staff expects that resolution of these issues will be challenging for both the NRC and AECL. Notwithstanding, the staff believes that it is feasible for AECL to provide the necessary information to resolve them. Resolution of these issues and technical and regulatory acceptance of the CAN/CSA-N Series standards or other standards, as necessary, would lead the staff to determine that the applicable GDC are satisfied.

1.7.2 10 CFR 50.55a, "Codes and Standards"

10 CFR 50.55(a) provides requirements for the use of codes and standards in nuclear power plants. In part, it provides requirements relating to the materials, design, fabrication, and testing of systems and components during construction, as well as requirements relating to inservice inspection (ISI) during the service life of a nuclear power facility. It requires the use of ASME Code, Section III, for construction, and ASME Code, Section XI, for ISI. The requirements of the ASME Code are subject to the limitations and modifications specified in 10 CFR 50.55a.

The acceptability of plants seeking design certification depends, in part, upon how the plant will be operated. The NRC has a body of requirements that apply to the manner in which plants are operated. To grant a design certification, the NRC must conclude that a plant design can be safely operated during the interval between ISIs. In the case of the ACR-700, ASME Code, Section XI, ISI requirements will be applied to Class 1, 2, and 3 components, other than the reactor assembly components and on-power fueling machine components. For these components, AECL intends to apply the inspection rules in Canadian standards. To reach a conclusion about the adequacy of reactor assembly components and on-power fueling machine components to perform their design function in service, the staff believes that as part of the design certification review, it is necessary to evaluate the ISI requirements that will be used by the applicant. The staff will consider CANDU operating experience, results of qualification

testing for ACR-700 design conditions, inspection techniques, sample size, frequency, and other factors to reach this conclusion.

A significant aspect of 10 CFR 50.55a is that it includes provisions for authorizing alternatives to the ASME Code requirements and granting relief from the requirements of the ASME Code that are impractical to meet. As stated in 10 CFR 50.55a(a)(3), proposed alternatives to the ASME Code may be used when authorized by the NRC, if (i) the proposed alternatives would provide an acceptable level of quality and safety, or (ii) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Pursuant to 10 CFR 50.55a(g)(6)(i), the Commission may grant relief and may impose alternate requirements when the ASME Code requirements are deemed impractical, giving due consideration to the burden upon the licensee if the ASME Code requirements were imposed on the facility.

ASME publishes updated editions of Sections III and XI (and other sections) once every 3 years and publishes addenda once each year between issuance of the editions. The NRC periodically revises 10 CFR 50.55a to update the ASME Code editions and addenda incorporated by reference. These revisions to 10 CFR 50.55a may also reflect limitations or additional requirements, as noted above. A provision in 10 CFR 50.55a(g)(4) addresses the need for updating the applicable requirements of the ASME Code, Section XI. The regulations require that ISI of components and system pressure tests conducted during the first 10-year interval, as well as subsequent intervals, comply with the requirements in the latest edition and addenda of Section XI of the ASME Code, incorporated by reference in 10 CFR 50.55a(b), 12 months before the start of the 120-month interval, subject to the limitations and modifications listed therein.

In general, the rules of 10 CFR 50.55a are applicable to the Class 1 components of the ACR-700. The rules of 10 CFR 50.55a applicable to the ACR-700 include ASME Code, Section III, requirements for materials, design, fabrication, inspection, and testing, in addition to the ASME Code, Section XI, requirements for ISI. Except for the reactor assembly and the on-power fueling machines, the ACR-700 design is similar to conventional LWRs. The PTs and end fittings within the RCPB are made from materials not specified by the ASME Code. Because the ASME Code does not address ACR-700 PTs and end fittings, it does not specify materials for these applications. AECL could request that these materials be reviewed by ASME for adoption; however, AECL contends that this would be a lengthy process and has declined to pursue this option because it may result in an unacceptable delay. Further, these materials have been adopted in Canadian standards. Some of the materials specified in the ASME Code could potentially be suitable for use as PTs or end fittings. If the ASME Code does specify materials that are suitable, and AECL is proposing alternative materials, the staff will review the use of these non-ASME Code materials in the ACR-700 as alternatives to the ASME Code, Section III, requirements pursuant to 10 CFR 50.55a(a)(3)(i). If the ASME Code does not specify any materials suitable for use as PTs or end fittings, the staff will consider the ASME Code requirements for the materials selection for PTs and/or end fittings as not relevant to the ACR-700 design. As part of design certification, AECL should provide the basis for its determination of which of these avenues is correct for both the PTs and end fittings.

The staff is aware that 10 CFR 50.55a contains rules that are not relevant to the ACR-700 design. For example, because the ACR-700 design does not contain a reactor pressure vessel (RPV), as used in conventional LWRs, the reactor vessel ISI rules of the ASME Code are not

technically relevant to the ACR-700. As part of the design certification review, AECL will need to identify to the NRC whether there are any additional areas in which 10 CFR 50.55a contains rules that are not technically relevant to the ACR-700 design. Such areas may be documented as not technically relevant, in accordance with 10 CFR 52.48. However, if it is determined that these areas are not in compliance with the 10 CFR 50.55a requirements, the staff can authorize alternatives to these requirements. This regulatory issue will need to be addressed during the design certification review.

The ASME Code, Section III, contains rules for design by analysis that apply to PTs, end fittings, and the rolled joints; however, it does not contain specific rules for fabrication, inspection, and testing of these components. Similarly, ASME Code, Section XI, does not contain rules for ISI (including flaw evaluation and repair/replacement rules) for fuel channel components. For those areas in which the Section III and Section XI rules are not applicable or need to be supplemented, AECL relies upon the CAN/CSA-N285 Series standards. For example, AECL relies upon CAN/CSA-N285.6 for materials selection, properties, fabrication, and testing of the PT and end fitting materials. AECL relies upon CAN/CSA-N285.4, "Periodic Inspection of CANDU Nuclear Power Components," for minimum inspection requirements for pressure boundary components, such as fuel channel components, not addressed by ASME Code, Section XI. The staff requests that during the design certification review, AECL explain the Canadian standards and the basis for their acceptability. The staff's review of this information may lead to technical issues not yet identified. Resolution of these issues may require the NRC to develop limitations or additional requirements.

ASME Code rules pertaining to design and inspection incorporated by reference in 10 CFR 50.55a have the effect of NRC requirements. Accordingly, Canadian standards adopted to address those technical areas for which the ASME Code rules are either not applicable or have to be supplemented will need an appropriate level of regulatory control. For example, the Canadian standards with limitations or additions, as appropriate, could be incorporated by reference as requirements in the design certification appendix to 10 CFR Part 52 applicable to the ACR-700. This approach would provide regulatory control by making the Canadian standards, with limitations or additions as appropriate, NRC requirements. The staff refers to the approach of incorporating the Canadian standards by reference in the design certification appendix as option 1.

Past design certification reviews have not addressed requests for authorization of alternatives to ASME Code requirements, requests for granting relief from ASME Code requirements that are impractical to meet, and ASME Code ISI program updates because they only involved ASME Code, Section XI, ISI programs implemented by the combined license (COL) applicant. For the ACR-700, review of the Section XI ISI programs would similarly be part of a COL application. The staff would make a determination about their adequacy before issuing a COL, rather than as part of the design certification. However, the staff believes that the design certification review will need to address ISI programs based on rules outside ASME Code, Section XI.

The areas of authorization of alternatives, requests for relief, and ISI program updates for the ACR-700 present unique challenges that the staff may need to address during the design certification review. Specifically, the NRC has not addressed the issue of using and updating Canadian standards to regulate ISI of the reactor assembly and on-power fueling system. It may be appropriate to discuss how AECL proposes that a COL applicant would obtain

regulatory approval of alternatives to and relief from Canadian standards, and address revised ISI requirements in updated Canadian standards applied to periodic inspection of the ACR-700 reactor assembly and on-power fueling system. If the design certification appendix applicable to the ACR-700 codifies the Canadian standards, the staff may need to consider incorporating provisions in the design certification rule for authorizing alternatives to, and granting relief from, Canadian standards that are adopted as NRC requirements. The Canadian standards are updated and reissued periodically, although not as frequently as the ASME Code. Further, if the design certification appendix applicable to the ACR-700 codifies the Canadian standards, the staff may need to consider incorporating a mechanism for periodically updating those requirements of the Canadian standards that are incorporated by reference into the NRC regulations.

As an alternative to option 1, specific Canadian standards with limitations and additions as agreed upon by the NRC could be identified as Tier 2* material. This Tier 2* approach would provide regulatory control by requiring the licensee to obtain NRC approval to change/modify these standards. The Tier 2* approach would also permit the 10 CFR Part 52 licensee amendment process to be used for the approval of alternatives, granting of relief, or owner updating of the applicable edition of the standard. The staff refers to this alternative as option 2. Option 2 differs from option 1 in terms of how the Canadian requirements are updated. Under option 2, the licensee would periodically update CAN/CSA requirements, rather than such updates being specified in the rule.

As another alternative, Tier 2* material could include specific provisions from the CAN/CSA-N Series standards that are related to the design and inspection requirements for those areas in which ASME Code Sections III and XI rules are not applicable or need to be supplemented. The staff refers to this alternative as option 3. This option differs from option 2 in that the Tier 2* material would include the specifics of design and inspection based on the CAN/CSA-N Series standards, with limitations and additions agreed to by the NRC; in option 2, the Tier 2* material would include limitations and additions to referenced CAN/CSA-N Series standards.

Issues involving design and inspection requirements not covered by the ASME Code, and the appropriate regulatory control of design and inspection requirements, will need to be resolved during the design certification process.

1.7.3 10 CFR 50.60, 10 CFR 50.61, and Appendices G and H to 10 CFR Part 50

The NRC regulations which address RCPB fracture prevention during normal operating, hydrostatic/leak test, and specific off-normal (i.e., during a pressurized thermal shock (PTS) transient) conditions include 10 CFR 50.60, 10 CFR 50.61, and Appendices G and H to 10 CFR Part 50. This group of regulations establish some of the necessary conditions which light-water power reactor designs must meet in order to comply with the GDC which address RCPB integrity. Specifically, this would include, but may not be limited to, GDC 14, 15, 30, 31, and 32. Because 10 CFR 50.60, 10 CFR 50.61, and Appendices G and H to 10 CFR Part 50 work in unison, it is first necessary to discuss the scope of each of these sections or appendices to evaluate their applicability to the NRC staff's ACR-700 preapplication review.

Although it is the enabling regulation that invokes Appendices G and H to 10 CFR Part 50, 10 CFR 50.60 itself does not contain any specific technical requirements germane to the ACR-700 preapplication review.

Appendix G, "Fracture Toughness Requirements," to 10 CFR Part 50 specifies "fracture toughness requirements for ferritic materials of pressure-retaining components of the RCPB of light-water nuclear power reactors to provide adequate margins of safety during any condition of normal operation, including AOOs and system hydrostatic tests, to which the pressure boundary may be subjected over its service lifetime." For the majority of the ferritic RCPB, Appendix G to 10 CFR Part 50 refers to components meeting the fracture toughness requirements in Section III, Division 1, of the ASME Code. For light-water nuclear power reactor vessels, additional requirements are given and are based on meeting the conditions in Section XI of the ASME Code and the specific minimum temperature requirements detailed in Table 1 of Appendix G to 10 CFR Part 50. Finally, Appendix G to 10 CFR Part 50 invokes the requirements in Appendix H to 10 CFR Part 50, as they apply to monitoring changes in reactor vessel material fracture toughness resulting from neutron radiation damage.

Appendix H, "Reactor Vessel Material Surveillance Program Requirements," to 10 CFR Part 50 states that "[t]he purpose of the material surveillance program required by this appendix is to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region of light-water nuclear power reactors which result from exposure of these materials to neutron irradiation and the thermal environment." The information obtained from reactor vessel material surveillance program testing is used to support reactor vessel evaluations which demonstrate compliance with the requirements of Appendix G to 10 CFR Part 50 and 10 CFR 50.61.

10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," was enacted as a supplement to the requirements of Appendix G to 10 CFR Part 50. Whereas the requirements of Appendix G to 10 CFR Part 50 apply to "any condition of normal operation, including AOOs and system hydrostatic tests, to which the pressure boundary may be subjected over its service lifetime," 10 CFR 50.61 addresses the potential for reactor vessel failure caused by lower frequency PTS transients. As established in 10 CFR 50.61, specific material property screening criteria must be met in order to demonstrate that a facility's reactor vessel will be adequately protected from failure during a PTS event. Title 10, Section 50.61, of the Code of Federal Regulations also refers to Appendix H to 10 CFR Part 50 for reactor vessel material surveillance program results which may be applicable to the reactor vessel material property evaluations necessary to demonstrate compliance with 10 CFR 50.61.

The NRC staff's first step in the preapplication review of the ACR-700 design was a regulatory evaluation to determine the applicability of the requirements of 10 CFR 50.60, 10 CFR 50.61, and Appendices G and H to 10 CFR Part 50. The NRC staff noted that, as written, 10 CFR 50.60 and Appendices G and H to 10 CFR Part 50 state that they specifically apply to "light-water nuclear power reactor" designs. As written, 10 CFR 50.61 states that it is applicable to "pressurized water nuclear power reactor" designs. The "Definitions" section of 10 CFR Part 50, does not provide specific definitions of "light-water nuclear power reactor" or "pressurized water nuclear power reactor" relative to the applicability of 10 CFR Part 50 requirements. Hence a question exists as to whether the ACR-700 would meet either, or both, of these categorizations.

From a purely technical perspective, the ACR-700 constitutes a pressurized (i.e., the bulk fluid at primary system operating conditions is “pressurized” above thermodynamic saturated water conditions), light-water cooled, heavy-water moderated nuclear power reactor design. The extent to which the ACR-700 may, or may not, meet the definitions of “light-water nuclear power reactor” or “pressurized water nuclear power reactor” remains as yet unresolved. However, under the assumption that 10 CFR 50.60, 10 CFR 50.61, and Appendices G and H to 10 CFR Part 50 would legally apply to the ACR-700 design, the NRC staff proceeded with a review of the technical relevance of these regulations to the ACR-700 design.

The NRC staff next reviewed 10 CFR 50.60, 10 CFR 50.61, and Appendices G and H to 10 CFR Part 50 in light of the requirements given in 10 CFR 52.48, “Standard for Review of Applications,” which states that “[a]pplications filed under this subpart will be reviewed for compliance with the standards set out in 10 CFR Part 20, Part 50 and its appendices, and Parts 73 and 100…as those standards are technically relevant to the design for the proposed facility.” The staff focused its review of the technical relevance of 10 CFR 50.60, 10 CFR 50.61, and Appendices G and H to 10 CFR Part 50 on two specific aspects, (1) the limitation in scope of these regulations to ferritic RCPB materials, and (2) the intent of parts of these regulations to address the “nuclear power reactor vessel” component.

Appendices G and H to 10 CFR Part 50 (and by back reference, 10 CFR 50.60 which invokes Appendices G and H) clearly apply only to ferritic RCPB materials. Further, Appendix H to 10 CFR Part 50 only applies to a defined subset of all those materials. The definition of “ferritic material” in Appendix G to 10 CFR Part 50 states that a “[f]erritic material means all carbon and low alloy steels, high alloy steels including all stainless alloys of the 4xx series, and maraging and precipitation hardening steels with a predominantly body-centered cubic crystal structure.” This regulatory emphasis on ferritic RCPB materials is warranted because of the relatively sharp transition in material properties which occurs in iron-based, body-centered cubic (i.e., ferritic) materials as a function of temperature around the material’s “ductile-to-brittle transition” temperature. A ferritic RCPB material’s ductile-to-brittle transition temperature must be well understood and accounted for in the design and operation of a nuclear power reactor in order to ensure, as required by GDC 31, that the RCPB will behave in a nonbrittle manner to minimize the probability of rapidly propagating fracture

In addition, 10 CFR 50.61, Appendix H to 10 CFR Part 50, and Section IV.A.1 and IV.A.2 of Appendix G to 10 CFR Part 50 specifically apply to the monitoring and protection of ferritic nuclear power reactor vessel materials. The distinction between the reactor vessel and the rest of the ferritic RCPB in typical LWR designs currently in operation, or those that have received design certification for future construction and operation, in the United States is important for two reasons. First, the ferritic, beltline material of a reactor vessel is the only part of the RCPB in a typical LWR design to experience sufficient exposure to neutron radiation to possibly result in significant material property changes. This observation supports the application of the material surveillance program required by Appendix H to 10 CFR Part 50 only to this subset of RCPB materials. Second, the reactor vessel in a typical LWR design is a unique RCPB component. The reactor vessel is a nonredundant component, the gross failure of which could lead to a beyond-design-basis accident (DBA) (i.e., an RCPB rupture greater than the double-ended guillotine break of the largest RCPB piping, which is the limiting rupture assumed for emergency core cooling system (ECCS) design under 10 CFR 50.46). The unique nature of the reactor vessel in typical LWR designs must be recognized in order to appropriately

understand the need for specific regulatory requirements to ensure that it is adequately protected from failure.

Several relevant technical observations can readily be made with respect to the relationship between 10 CFR 50.60, 10 CFR 50.61, and Appendices G and H to 10 CFR Part 50 and the ACR-700 design. First, there is no reactor vessel, nor any analogous, single component within the ACR-700 design, the failure of which could potentially lead to a beyond-DBA, given that the design of the ACR-700 includes the ability to mitigate up to and including the double-ended guillotine break of the largest RCPB piping. Second, it is the staff's understanding that the only components within the ACR-700 pressure boundary which are both manufactured from a ferritic material and subject to potential material property changes resulting from irradiation are the modified 403 SS end fittings into which the PTs are rolled. The zirconium-alloy PTs themselves, which are the RCPB components subject to the most intense radiation fields, are by definition not ferritic materials, and thus not subject to the provisions of 10 CFR 50.60, 10 CFR 50.61, and Appendices G and H to 10 CFR Part 50. However, detailed technical requirements beyond the application of the ASME Code Section III requirements, as addressed in 10 CFR 50.61 and Appendices G and H to 10 CFR Part 50, also cannot be considered to be technically relevant to the design of the ACR-700 modified 403 SS end fittings, as these technical requirements were developed for massive, low-alloy steel LWR vessels.

Considering the discussion above concerning the application of the requirements of 10 CFR 50.60, 10 CFR 50.61, and Appendices G and H to 10 CFR Part 50 to ferritic RCPB materials, and specifically the reactor vessel in typical LWR designs used, or certified for use, in the United States, the NRC staff then evaluated how these regulations apply to the ACR-700 design review. The NRC staff concludes the following, contingent upon resolution of the issues identified above regarding the applicability of the regulations to "light-water nuclear power reactor" designs and "pressurized water nuclear power reactor" designs:

- To the extent that Appendix G to 10 CFR Part 50 requires that all ferritic materials (per the definition of ferritic materials given in Appendix G to 10 CFR Part 50) of a nuclear power reactor's RCPB meet the fracture toughness requirements of ASME Code Section III, Division 1, this requirement applies to all such materials incorporated into the ACR-700 design. The regulatory provisions of 10 CFR 50.60, inasmuch as they invoke these requirements in Appendix G to 10 CFR Part 50 would also apply to the ACR-700 design review.

- Per 10 CFR 52.48, the technical requirements specified in 10 CFR 50.61, Section IV.A.1 and IV.A.2 of Appendix G to 10 CFR Part 50, and Appendix H to 10 CFR Part 50 are not technically relevant to the ACR-700 design review. These specific regulations were promulgated to address the monitoring and evaluation of ferritic LWR pressure vessels, and the ACR-700 design does not appear to have any functionally and materially equivalent components.

Subsequent to the NRC staff's review of the specific applicability of 10 CFR 50.60, 10 CFR 50.61, and Appendices G and H to 10 CFR Part 50 to the ACR-700 design, the staff then considered the underlying intent of the requirements found to be technically irrelevant to the ACR-700 design. The staff considered whether a need exists to develop similar requirements for ACR-700 RCPB components to ensure that the ACR-700 design will meet all of the GDC and embodies a level of quality adequate to protect the public health and safety.

Based on the depth of its ACR-700 preapplication review process, the staff has concluded that at least three types of components within the ACR-700 design merit additional evaluation with respect to issues related to the requirements of 10 CFR 50.60, 10 CFR 50.61, and Appendices G and H to 10 CFR Part 50. These components are the ACR-700's zirconium-alloy PTs, Zircaloy-4 calandria tubes, and the modified 403 SS PT end fittings. The following paragraphs address the basis for the staff's conclusions for each of these component types.

With regard to the ACR-700's zirconium-alloy PTs and 403 SS PT end fittings, the NRC staff has noted that these components are, at least from a functional standpoint, the analog in the ACR-700 design closest to a typical LWR vessel. The components serve to support the fuel elements, maintain a coolant inventory, and channel coolant flow to carry heat away from the fuel elements; they also serve as a boundary to radioactive release in the event of fuel element failure. In addition, these components appear to be subjected to radiation fields far in excess of what would be experienced by a typical LWR vessel during its design life.

Therefore, based upon the requirements specified in GDC 14, 15, 30, 31, and 32, the staff has identified a need to develop appropriate review guidance and requirements related to maintaining the integrity of these components in light of their service environment. For the zirconium-alloy PTs, this review guidance should be integrated with guidance regarding the evaluation of other known potential degradation mechanisms (e.g., DHC) which may not require exposure to significant radiation fields to become active. For the 403 SS end fittings, the effect of the thermal environment should be considered, along with the existing radiation fields, when evaluating potential material property changes. The staff is considering the definition of a material surveillance program similar to that described in Appendix H to 10 CFR Part 50 but which may include the periodic testing of zirconium-alloy PTs and 403 SS PT end fittings that have been removed from service in lieu of surveillance specimens. In addition, the staff is also considering operating limits for these components, similar in intent to those specified in Sections IV.A.1 and IV.A.2 of Appendix G to 10 CFR Part 50, as well as protection from other design-basis events (DBEs), like those addressed in 10 CFR 50.61 for typical LWR vessels. The emphasis of the staff's review guidance, with respect to the protection of the PTs and PT end fittings from DBEs beyond normal operation and AOOs, should be to achieve adequate assurance that the probability of consequential multiple PT/end fitting ruptures, to the extent that a beyond-design-basis condition is achieved, is extremely low.

The Zircaloy-4 calandria tubes are unique to the ACR-700 design and are functionally different from typical LWR components. First, as was discussed previously with regard to GDC 4, the staff understands that the calandria tubes will be credited within the ACR-700 design as a component which mitigates the dynamic effects of a postulated design-basis rupture of one of the zirconium-alloy PTs. To this extent, maintaining adequate assurance with regard to the structural integrity and material properties of the calandria tubes for their design lifetime appears to be necessary to permit certification of the overall ACR-700 design. Therefore, the NRC staff is considering the need to define review guidance requiring the development of a material surveillance program to monitor property changes in the calandria tube material resulting from radiation exposure. It is anticipated that, like the material surveillance program discussed above for the zirconium-alloy PTs, a material surveillance program for the Zircaloy-4 calandria tubes would have an intent similar to a material surveillance program for typical LWR vessels, as described in Appendix H to 10 CFR Part 50.

In the course of the ACR-700 preapplication review, AECL has provided the NRC with a significant amount of information regarding work to date on the evaluation of the properties of the materials chosen for the ACR-700 PTs, calandria tubes, and PT end fittings, as well as the provisions which have been made within the ACR-700 design for their monitoring and inspection. Based on this information, the NRC staff has reasonable assurance at this time that adequate material property monitoring and ISI programs can be developed to ensure that the ACR-700's zirconium-alloy PTs, Zircaloy-4 calandria tubes, and type 403 SS PT end fittings will meet the provisions of GDC 14, 15, 30, 31, and 32. The staff will, however, perform a more detailed review of this topic during the application review phase to ensure that the ACR-700 design is adequate.

1.8 Potential Policy Issues

Under "Regulatory Issues" in Section 1.7 of this PASAR, the staff has identified a number of issues. Depending upon the approach taken, certain issues may necessitate Commission guidance and, therefore, be considered policy issues. However, at this time the staff has not identified any specific policy issues requiring Commission guidance.

1.9 Technical Issues

1.9.1 AECL Research Programs

AECL is using qualification testing to support design and periodic inspection requirements of fuel channel components. AECL qualification testing is intended to demonstrate that the fuel channel design and inspection programs meet the various regulations, including Appendix A to 10 CFR Part 50. AECL and the staff agree that a meeting is needed to review the research programs which support extending qualification of the component materials from existing CANDU operating conditions to ACR-700 operating conditions for each of the degradation mechanisms. AECL and the staff will also discuss the status and basis of the research programs and the available results. The staff anticipates that this meeting will take place before AECL submits the design certification.

1.9.2 Reduction in Creep Life from Environment Effects

The FCTR describes extensive creep testing, including thermal creep, irradiation creep, and irradiation growth. This testing relates to maintaining the integrity of the RCPB and, hence, to ensuring that various regulations, including Appendix A to 10 CFR Part 50, are met. In this regard, the staff is concerned with the lack of available test data that compares creep behavior of the component materials in typical heavy water (including impurities, such as ppb to ppm levels of oxygen) to that in typical light water (including oxygen and other impurities) and in air. The coolant environment (including impurities) could reduce the creep life of components, as has been observed for fatigue life (see the discussion on fatigue in Section 1.4 of this report).

In its March 31, 2004, submission, AECL responded to question 114 with a description of creep testing of capsules irradiated in sodium-potassium-alloy (NaK) that has been performed or is planned. The objective of the additional testing program is to confirm the expected behavior for the regions of the ACR-700 PTs that operate at higher temperatures.

This discussion further notes that examination of removed PTs from operating CANDU reactors reveals no indication of adverse effects, other than surface oxidation, that would reduce the creep life of the PT in a water environment. Although the oxide layer from these tubes exhibited cracks, the cracks do not extend to the metal, and they do not induce cracking in the metal. However, the staff is aware that at temperatures in the creep regime, the environment can affect creep rate, even though the oxide remains intact.

During the design certification, AECL will need to provide the technical bases to demonstrate that analyses and testing have adequately considered the environmental effects on creep design noted above.

1.9.3 Delayed Hydride Cracking

1.9.3.1 Applicability of Experience with Hydride Formation from Heavy Water

During the preapplication review, the staff raised questions regarding the lack of available test data that compares DHC, corrosion, and SCC behavior of the component materials in typical heavy water (including impurities, such as ppb to ppm levels of oxygen) to that in typical light water (including oxygen and other impurities) and in air, where appropriate. These questions relate to maintaining the integrity of the RCPB and, hence, to ensuring that various regulations, including Appendix A to 10 CFR Part 50, are met.

AECL response focused on DHC; AECL believes that DHC is the only potential mechanism of crack initiation and crack growth in PTs. AECL has not experienced any known environmental cracking mechanisms under normal operating conditions in PTs.

AECL concludes that DHC requires sufficient hydrogen concentration to precipitate hydrides under the temperatures being considered, and sufficient stress intensity to produce crack initiation, growth, and potential fractures. DHC can be characterized through material surveillance, experimental, and theoretical work and, unlike SCC, DHC is more predictable and can be avoided. Although some data exist with respect to corrosion and DHC, AECL makes the assumption, with respect to the materials' behavior, that light and heavy water are very similar from a chemical viewpoint. Differences between light water and heavy water in terms of corrosion and ingress of hydrogen isotope for zirconium-alloy materials are generally within uncertainties of the experimental determinations. It could be expected that differences in behavior between hydrogen and deuterium could appear in factors dependent upon diffusion rates, and that any such differences would be of the order of the ratio of the square roots of the isotopic masses (i.e., $\sqrt{2}$ (1.41)). DHC testing, historically, has been carried out by using hydrogen instead of deuterium and, hence, the data available already meet ACR-700 requirements in terms of using the relevant isotope of hydrogen. AECL relies on data from two sources for DHC testing. The first source is unirradiated laboratory specimens that contain hydrogen from the manufacturing process and additional, intentionally added hydrogen for DHC testing. The other source of data is from the PTs removed from service. The removed tubes contain hydrogen from the tube manufacturing process and deuterium from service in heavy-water-cooled CANDUs. The data from these two sources indicate no difference in DHC testing within the experimental scatter. Therefore, the staff considers that AECL has at this time satisfactorily addressed this issue, which should be included in the design certification application for completeness.

1.9.3.2 Hydriding and Deuteriding in Zr-2.5Nb Pressure Tube Material

In a letter dated April 14, 2004, AECL provided information demonstrating the similarities and differences between hydriding and deuteriding in the Zr-2.5Nb PT material. The ACR-700 reactor coolant is light water, while much of the materials testing AECL is using to ensure integrity was performed in heavy water. This materials testing relates to maintaining the integrity of the RCPB and, hence, to ensuring that various regulations, including Appendix A to 10 CFR Part 50, are met. Part B of this topic discusses potential differences in solubilities of hydrogen versus deuterium and provides a more specific discussion of uptake (ingress) of hydrogen versus deuterium.

The isotopes hydrogen and deuterium behave in a similar manner in the PT material. The terminal solid solubilities are basically indistinguishable, provided that the solubilities are expressed on an atomic fraction basis (i.e., taking into account that the masses of the two species differ by a factor of 2). This has been demonstrated in a number of studies. In Reference 121.1 to its April 14, 2004, letter, AECL describes a comprehensive study of the solubility behavior of deuterium in PT material. The results are shown to be consistent with other experiments measuring the solubility of the hydrogen isotope in zirconium alloys.

Much of the corrosion and hydrogen ingress experimental work carried out over many years has used light water as the corrosion medium. Recently, in support of the ACR-700, additional experiments have been undertaken to determine if any measurable differences do exist in the corrosion and hydrogen isotope uptake behavior between heavy water and light water. The following paragraph summarizes the interim results of this ongoing study.

Two sets of Zr-2.5Nb coupons and one set of Zircaloy-4 coupons are being tested for hydrogen and deuterium uptake in H_2O and D_2O, respectively. The data show that after up to 600 days of exposure, within experimental errors, the hydrogen isotope uptake (expressed as atomic fraction) is similar in H_2O and D_2O. The tests will continue for a few more years to obtain longer term data and increase the confidence in the measured ingress rates. Based on the information provided, the staff considers that AECL has at this time satisfactorily addressed this issue, which should be included in the design certification application for completeness.

1.9.3.3 Hydrides at Inlet and Outlet in Pressure Tube

DHC requires both hydride precipitation and a tensile stress above a certain threshold. Preventing DHC relates directly to maintaining the integrity of the RCPB and, hence, to ensuring that various regulations, including Appendix A to 10 CFR Part 50, are met. From the information provided by AECL, hydrides will be present in the ACR-700 at the inlet and outlet of the PT after approximately 14 effective full-power years (EFPYs) of operation. Figure 9-2 in this report depicts the presence of hydrides and Figure 9-14 in the FCTR is a schematic representation. AECL will need to identify those areas where the hydrogen concentration does exceed the solubility limit, and will need to demonstrate that DHC will not occur in those locations. AECL will need to demonstrate during the design certification review that normal operating stresses (e.g., channel vibration, startup, and shutdown), transient stresses, and DBA stresses will not exceed the DHC threshold stress. This is discussed in more detail below.

At the inlet end of the PT, a deuterium concentration of approximately 40 ppm is required for the total hydrogen isotope concentration to exceed the solubility limit at the operating

temperature. This concentration is exceeded only over a short distance from the end of the tube, and generally is mostly or entirely within the compressive stress zone of the rolled joint. Similarly, at the outlet end, the solubility limit is generally not exceeded at operating temperature. The intent of the fuel channel design for the ACR-700 is to maintain the concentrations of hydrogen below the solubility limit at operating conditions for the life of the PTs and for all parts of the PT between the compressive regions of the rolled joints. This would imply that at operating temperatures, independent of the stress state, DHC could not occur.

For an unflawed tube, the FCTR indicates no operating transients will result in a stress in the PT in excess of the stress required to initiate DHC at a smooth surface. Because all operating transients have not yet been finalized for the ACR-700, specific stresses cannot be provided. Nevertheless, AECL is confident in this conclusion because operating and transient stresses, such as those associated with startup and shutdown, are generally low. The CAN/CSA-N285.2 Standard requires that the maximum tensile stresses under design level A (operating) and level B (upset), plus the maximum initial residual tensile stress, will not exceed 67 percent of the tensile stress required to initiate DHC in laboratory tests of unnotched specimens (i.e., 450 MPa based upon tests of irradiated specimens). The staff is unclear about the limits under level C and D conditions.

AECL uses a blunt notch analysis to evaluate anomalies, such as gouges from fuel bundle fretting. During the design certification review, the staff will need to evaluate the AECL blunt notch analysis.

The staff is concerned that the hydrogen concentration in the PT which exceeds the hydrogen solubility will move inboard of the compressive region after long reactor operating times, leading to the potential for shearing of the PT by DHC just inboard of the end fittings. For example, during a seismic event with the fueling machine attached, the staff can envision peak bending moments and consequent tensile stresses in the PT near the inboard end of the end fitting. During the design certification phase, the staff will evaluate the AECL model and CAN/CSA-N285.2 analysis to determine whether meeting the CAN/CSA Code requirements satisfies these staff concerns regarding DHC of the PTs.

1.9.4 Safety Margins

The increase in the maximum operating temperature of the ACR-700 to 325 EC (617 EF) will increase the kinetics of many of the materials degradation processes. This increase in temperature may also introduce additional modes of degradation. In a letter dated March 19, 2004, the staff asked AECL whether data exist on different degradation processes and modes at the higher ACR-700 temperatures and in the appropriate coolant environments. The staff also asked whether adequate safety margins existed at these higher temperatures. AECL responded in a letter dated March 31, 2004.

Testing has been carried out at temperatures relevant to ACR-700 outlet temperatures on a number of factors that must be taken into account in assessing potential degradation. Such tests include corrosion and hydrogen isotope uptake testing in a heavy-water loop in the Halden high-flux reactor in Norway. Heavy water has been used because it improves the ability to measure relatively small changes in concentration attributable to the corrosion. Testing has been carried out at 325 EC (617 EF), and a range of parametric tests have been carried out to

aid in improving the corrosion and hydrogen ingress models. Corrosion tests of the ACR-700 prototype PT material under irradiation will begin this year.

Testing of ACR-700 PT rolled joints in a loop facility upgraded to ACR-700 conditions will also be carried out. These tests will provide information on rolled joint corrosion and hydrogen isotope uptake performance in long-term tests (a minimum of 3 years) to validate the predictive model for rolled joint hydrogen ingress for ACR-700.

Deformation testing will be carried out at ACR-700 temperatures to confirm the steady-state creep rates. This information will be used in the design to preclude contact between the PT and the surrounding calandria tube during the PT lifetime.

Tests to measure characteristics of DHC in irradiated PT material from current reactors at ACR-700 operating temperatures have begun. These tests will be supplemented by tests of prototype ACR-700 material in both the unirradiated and irradiated material conditions.

The staff understands that information exists or is being assembled on the different degradation processes and modes at the higher temperatures and in the appropriate coolant environments. However, the staff considers this general area of safety margins to be an open issue. This issue relates to maintaining the integrity of the RCPB and, hence, to ensuring that various regulations, including Appendix A to 10 CFR Part 50, are met. During the design certification, the staff will need to understand how AECL determines margin to failure to account for uncertainties. For example, Appendix 5 to Section II of the ASME Code requires creep rate and creep rupture strength data at 50 EC (90 EF) intervals to 50 EC (90 EF) above the maximum intended use temperature. AECL has no plans to conduct creep at design temperature plus 50 EC (90 EF). AECL will need to present the technical basis to demonstrate that there is adequate safety margin to account for uncertainties in creep analysis, such as heat-to-heat variability, data scatter, and model uncertainties.

1.9.5 Inservice Inspection

Chapter 11 of the FCTR provides a discussion of the techniques used in CANDU reactors and planned in the ACR-700 for inspection and monitoring fuel channels. During the preapplication review, the staff requested a reconciliation summary between CAN/CSA-N285.4 and the ISI rules of ASME Code, Section XI. AECL provided its response by letter dated March 31, 2004. The FCTR and this response provide information on the inspections performed, the reasons for conducting these inspections, the frequency of inspections, and a comparison with the principal elements embodied in the requirements of ASME Code, Section XI, and CAN/CSA-N285.4. The information provided by AECL addresses the staff's question at this time. However, during the design certification phase, the staff expects to review ISI in more detail before reaching an overall conclusion. For example, the staff will consider the results of qualification testing at the ACR-700 fuel channel design conditions, with respect to the potential for new degradation mechanisms. The staff also expects to review the ISI of all reactor assembly components, the bases for sample sizes and sample expansion, frequency of inspection, performance demonstration requirements, and flaw acceptance criteria.

During the preapplication review, the staff requested that AECL identify fuel channel components that are not accessible for ISI. By letter dated March 31, 2004, AECL indicated the

both the calandria and lattice tubes are accessible for postfabrication inspection, but that they become inaccessible for inspection after the PTs are installed. These components are supporting structures for the pressure boundary components and do not form part of the RCPB. Based on the design, materials, environment, and the expected stress levels, AECL is not aware of any inservice degradation mechanisms that would affect these components and necessitate periodic inspection. Moreover, the lattice tubes and calandria tubes are monitored continuously during service by the annulus gas system. The information provided by AECL addresses the staff's question at this time. However, during the design certification phase, the staff may ask additional questions in this area regarding the potential for different design conditions for the ACR-700 to change the performance of these components. The staff may also ask additional questions concerning the capabilities of the monitoring systems.

1.9.6 Pressure Tube Fracture Toughness

The minimum stress intensity factor for DHC initiation is called $K_{1H.}$ By letter dated April 14, 2004, AECL provided a technical basis and data to support the relationship discussed in Section 12.2.1.3 of the FCTR that identifies the lower bound of K_{1H} as 4.5 MPa(m)^0.5. The lower bound K_{1H} value relates to maintaining the integrity of the RCPB and, hence, to ensuring that various regulations, including Appendix A to 10 CFR Part 50, are met. The information submitted was proprietary. The staff reviewed this information, which describes the test techniques used by AECL, as well as other contributors to CANDU research, to measure K_{1H} in zirconium-alloy PT material in both unirradiated and irradiated conditions over a period of many years. AECL maintains a database of K_{1H} measurements on irradiated material which have been taken together as a single distribution and provide the mean, standard deviation, and 95 percent prediction interval. Based on the information provided, the staff considers that AECL has satisfactorily addressed this issue, which should be included in the design certification application for completeness.

1.9.7 Gas Side Corrosion and Hydrogen Ingress

The FCTR discusses the gas side corrosion and hydrogen ingress in terms of maintaining an oxidizing, dry environment. In a letter dated March 31, 2004, AECL indicated that creep, fatigue, and SCC testing have not been conducted in this environment of carbon dioxide, hydrogen, oxygen, water, and other possible impurities to evaluate the effect on PT, end fitting, and calandria tube life under operating temperatures and stresses. However, AECL has no indication that the environment in the annulus gas has had any adverse effect on the integrity of any of these components.

Hydrogen ingress is of interest because the presence of hydrogen in the zirconium-alloy PT can lead to DHC under specific circumstances. By scavenging the hydrogen in the annulus gas, the presence of oxygen acts to maintain the hydrogen level at a very low concentration. Even in the absence of an oxygen addition, the CO_2 annulus gas would remain oxidizing so that a protective oxide layer on the zirconium alloy would minimize the ingress of hydrogen into the zirconium-alloy PT.

The low level of hydrogen entering the PT from the annulus gas side is taken into account through inspection and surveillance, but would be indistinguishable from the hydrogen entering the tube from the water side as a result of corrosion. In the absence of contact with the

calandria tube, the outer diameter of the PT is at a higher temperature than the inner diameter because of gamma heating. Hydrogen in zirconium alloy diffuses to the lower temperature at the inner diameter where it would be monitored through an inspection program. Stresses in the PT are generally quite low, except near the rolled joint where they are limited to a fraction of the stress required to initiate DHC at a smooth surface. DHC is discussed in more detail above.

The gas-side corrosion and hydrogen ingress topics relate to maintaining the integrity of the RCPB and, hence, to ensuring that various regulations, including Appendix A to 10 CFR Part 50, are met. The staff considers the information provided by AECL at this stage adequate to address its question on gas-side corrosion and hydrogen ingress. The staff expects to revisit this issue on the gas-side environmental effects on fatigue, creep, and SCC during the design certification review.

1.9.8 Garter Spring Contact

Hydrides would form in the outer surface of a PT if they were to come into contact with calandria tubes. This would occur because of hydrogen diffusion to the lower temperature at the contact point and the dependence of the hydrogen solubility limit on temperature. Hydrides are undesirable because of the potential for DHC. AECL indicated that no measurements of the temperature at the contact point of the PT with the garter spring spacer have been taken. The temperature is calculated by analysis. For the ACR-700, the design of the spacer has not been completed and the thermal analysis has, therefore, not been done. The design will minimize the temperature reduction in the PT at the points of contact with the calandria tube to ensure that hydrides do not precipitate in the PT during normal operating conditions at the garter spring contacts.

The issue of potential DHC at garter spring contact locations relates to maintaining the integrity of the RCPB and, hence, to ensuring that various regulations, including Appendix A to 10 CFR Part 50 are met. During the design certification, the staff will evaluate the AECL analysis of the PT temperature at locations where garter spring spacers contact the PT to ensure that DHC at these locations in not a concern.

1.9.9 Lead Plant Monitoring Programs

The Canadian codes and standards require the monitoring of key PT physical properties, including fracture toughness and DHC velocity. Performance of these material surveillance tests require the removal of the PT and are only applied to the lead unit with a particular type of PT alloy (see Section 11.2.4.5 of the FCTR).

The NRC is aware of operating experience at U.S. LWRs where the lead plant (in terms of operating time) has not always been the first to experience a degradation mechanism. In addition, the NRC typically does not require a lead plant to perform more inspections, testing, or analyses than another plant.

Given these two issues, in a March 19, 2004, NRC letter to AECL, the staff requested that AECL address this issue for PTs and other situations in which the concept of a lead plant is used for determining monitoring programs. In response to the staff's question, AECL indicated in its March 31, 2004, letter that utilities in Canada have generally shared the costs of doing

such surveillance inspections through cooperative agreements. In addition, AECL indicated that improved manufacturing methods have resulted in more uniform properties and this is expected to translate into more uniform performance after irradiation. The ISI program at each plant monitors the aspects of PT integrity performance associated with water chemistry effects because these effects could potentially vary somewhat from one plant to another.

The staff considers this issue open for the following reasons:

- AECL provided (or referenced) no data to indicate that minor variations in material properties (within the specification) will not result in more significant material property changes.

- AECL provided (or referenced) no data which address the variability that could occur in the manufacturing process (e.g., tubes made to the same standard may have a different behavior because of manufacturing issues and/or because of the various combinations of the processing variables).

- AECL did not identify other SSCs that are potentially inspected/monitored under a lead plant concept.

1.9.10 Data Scatter

Many of the plots of corrosion rates (or oxide thickness) in the FCTR show a large amount of scatter. In addition, the integrity of an SSC may be affected by a number of degradation effects. Depending on how the scatter in these data is addressed and subsequently combined in the design and operation (inspection) of the various SSCs, the integrity of these components could be affected. The integrity of SSCs is addressed in various NRC regulations, including Appendix A to 10 CFR Part 50. In a letter dated March 31, 2004, AECL responded to a staff question about how it addressed and combined, from a conceptual standpoint, the scatter in the correlations. AECL indicated that predictions of performance are generally based upon bounding estimates (95 percent prediction limits at 50 percent confidence). In addition, AECL indicated that the effects of wall thinning of PTs from irradiation deformation and from corrosion are combined. The staff is satisfied with the response on this issue from a conceptual standpoint. However, the staff may have further questions on this topic during design certification as part of its detailed review on a component-specific basis, because the importance of the component and the scatter in the data may result in the need to take a more conservative approach.

1.9.11 Removal of Data from Correlations

AECL provided data supporting the various analyses it performed. Many of these analyses dealt with assessing the integrity of SSCs. Various NRC regulations, including Appendix A to 10 CFR Part 50, address the integrity of SSCs. In response to NRC questions regarding whether any data were removed from the correlations, AECL indicated in a letter dated March 31, 2004, that data would generally be removed only if it could be determined, with some confidence, that the data were not valid. AECL, however, does not have a documented company policy on the specific issue of removing data from correlations. The staff considers this response sufficient to address its question at this time; however, the staff expects to ask

further questions on this issue when it performs a detailed review of each SSC during design certification.

1.9.12 Blockage of Annulus Gas Flow by Organic Material

The annulus gas system is used to detect leaks and to maintain the protective oxide on the outer surface of the PT. The FCTR states that initial use of carbon dioxide in the annulus resulted in blocking of the gas flow. This was the result of organic material that formed in the cooler areas of the system and subsequently deposited in the "pig tails" (see Section 9.2.3 of the FCTR). To prevent formation and deposition of this material, oxygen was added to the annulus gas system. To ensure that the annulus gas was not adversely affecting the PT, consistent with the requirements of GDC 4, the staff asked AECL if the addition of oxygen was successful in preventing the formation of the organic material, how the blockage was detected, whether a blockage could go undetected during operation, and the implications of undetected blockage.

AECL responded in a March 31, 2004, letter that no blockages have occurred since oxygen additions were implemented. Before the oxygen additions, a small number of plants experienced blockages, but none of these have experienced blockages since adding oxygen. Blockage was detected by monitoring flow rotameters for each annulus gas string. AECL also stated that a blockage could potentially go undetected for an unspecified period because there is not a uniform practice for monitoring the rotameters at utilities operating CANDU reactors. According to AECL, blockage would result in a longer response time to system leakage, and the blockage would be cleared after the leakage caused a high enough pressure rise in the blocked annulus. The staff considers this issue open because AECL did not address the following: (1) the effects of operating without the annulus gas system for an unspecified period of time (i.e., the potential for the pressure boundary to degrade during a period when the gas flow was blocked); (2) the corrective actions to be taken following identification of a blocked pig tail (e.g., evaluating the impact on the garter springs); and (3) the potential need for a flow monitoring program.

1.9.13 Leakage of End Fitting Rolled Joints

The FCTR states that "the PT to end fitting rolled joints in a CANDU reactor have never come apart nor allowed excessive leakage of the reactor's coolant" (see Section 3.2.2 of the FCTR). These joints are part of the RCPB, and therefore must satisfy GDC 14, which requires, in part, that the RCPB have an extremely low probability of abnormal leakage. In response to an NRC question regarding the amount, detection, and consequences of leakage associated with these joints, AECL indicated in a letter dated March 31, 2004, that only one instance of a rolled joint leak has occurred and the leak monitoring system is capable of detecting very small leaks and determining the approximate location of the leak. In this one instance of rolled joint leakage, the leak rate was very low and there was no detectable effect of this leakage on any component. Given that AECL is still developing the rolled joint and performing qualification testing, the staff considers this response sufficient to address its question at this time. During the design certification review, the staff expects to ask for information on the qualification test results, particularly with respect to the behavior of the joint during accident conditions.

1.9.14 Galvanic Corrosion Between End Fitting and Pressure Tube

Galvanic corrosion between the end fitting and PTs is the primary source of hydrogen ingress to the PT (see Section 9.3.2.1 of the FCTR). If the dissimilar metal contact increases the rate of hydrogen generation (cathodic reaction), it may also increase the rate of the corresponding anodic reaction and cause localized, rapid penetration of the zirconium. If the hydrogen ingress is enhanced on the PT material, it may also be enhanced on the martensitic SS end fitting material, which is susceptible to hydrogen embrittlement under some conditions. The rolled joint is part of the RCPB and is, therefore, addressed in various NRC regulations, including Appendix A to 10 CFR Part 50.

In a letter dated March 19, 2004, the staff requested that AECL identify the corrosion reactions that form the galvanic cell, the location of this corrosion, the effect on degradation of the PT and end fitting, and the method for monitoring hydrogen ingress. AECL responded that corrosion has not occurred in a rapid and localized manner, the concentration of aggressive anions (e.g., chloride) is too low to create increasingly corrosive environments in crevices (e.g., rolling grooves), and only slight corrosion has been observed on the faces of the rolled joints removed from service after about 100,000 hours of operation. In its March 2004 meeting with the NRC staff, AECL confirmed that this galvanic corrosion takes place at operating temperature, where the high diffusivity of hydrogen in steel precludes embrittlement of the end fitting. According to the FCTR, hydrogen ingress and corrosion are monitored through periodic removal of single tubes from service and in-situ microsampling (“scrapes”). Existing surveillance and research programs determine the amount of corrosion at the higher operating temperature and pressure of the ACR-700. The staff considers that AECL has, at this time, satisfactorily addressed this issue, which should be included in the design certification application for completeness. However, it is likely that the staff will request additional information when more data from existing surveillance and research programs become available.

1.9.15 Material for Garter Spring Spacers

The garter spring spacers prevent contact between the pressure and calandria tubes. Contact between the pressure and calandria tubes can lead to DHC. These spacers are made from nickel-based Alloy X-750. As the FCTR notes, this alloy is susceptible to SCC in some aqueous environments (see Section 17.5 of the FCTR). Various NRC regulations, including Appendix A to 10 CFR Part 50, address the integrity of SSCs. Because failure of the garter spring could affect the integrity of the pressure boundary, the staff asked questions in a letter to AECL dated March 19, 2004, regarding the potential for the Alloy X-750 to degrade, given the background level of moisture in the gas annulus space between the tubes. AECL responded in a letter dated March 31, 2004 that there is no possibility of liquid water formation in the annulus, given the upper limit on the dew point in the annulus; therefore, AECL considered SCC to be impossible. In the event the annulus was flooded, additional ISI requirements would be considered in order to confirm that the flooding had not resulted in any garter spring degradation. The staff considers that AECL has at this time satisfactorily addressed this issue, which should be included in the design certification application for completeness.

1.9.16 NRC Regulations and On-Power Fueling

Many of the same regulations that the PTs will be required to meet also apply to the design, fabrication, erection, testing, and inspection of the on-power fueling system. To this end, in a letter dated March 19, 2004, the staff made several broad observations pertaining to the use of Canadian codes and standards, ISI, operating experience, limiting conditions for operation, and classification of system components. In response to the staff's questions, AECL indicated in a March 31, 2004, letter that it would provide additional details at a later date. Given the general nature of the questions, the staff considers that AECL has satisfactorily addressed the staff's issues because the company has addressed many of the broader issues in the context of the PT review.

1.10 Materials Assessment Conclusions

In the preceding sections, the staff identified the regulations that are pertinent to the design of fuel channels from a materials engineering perspective. Based on an assessment of the extent to which the ACR-700 complies with these regulations, the staff discussed regulatory and technical issues that should be addressed during the design certification. For example, the staff will evaluate the acceptability of CAN/CSA-N standards used for areas in which ASME Code, Section III and Section XI, rules are not applicable or need to be supplemented. Certain regulatory issues exist with respect to providing appropriate regulatory controls over the use of CAN/CSA-N standards. In addition, the staff has identified the possible need for a materials surveillance program similar to that addressed in Appendix H to 10 CFR Part 50, operating limits for PTs and end fittings similar in intent to those specified in Sections IV.A.1 and IV.A.2 of Appendix G to 10 CFR Part 50, and protection from DBEs similar to those addressed in 10 CFR 50.61 for typical LWR vessels.

As noted above, the staff identified a number of technical issues which it will pursue in more detail during design certification. The regulatory and technical issues involved are complex. The staff expects that the review and evaluation of the ACR-700 design will be challenging and will involve the expenditure of more resources and will take longer to complete than a typical light-water design. Notwithstanding, based on the information provided, the staff is confident at this time that AECL can satisfactorily address these regulatory and technical issues. In the preapplication review, the staff has not identified any material issues related to the fuel channel design that would preclude certification of the ACR-700 design.

2. DESIGN-BASIS ACCIDENTS AND ACCEPTANCE CRITERIA (FT2)

2.1 Design-Basis Accidents—Review Scope

The preapplication review proposal classifies the evaluation of DBAs and acceptance criteria (ACs) for the ACR-700 design as FT2.

2.1.1 Review Guidance

In the current licensing process, the NRC requires each applicant to perform analyses of various design-basis transients and accidents for its proposed LWR design using NRC-approved methods. The results of the analyses must demonstrate the design's compliance with the requirements of GDC 10, "Reactor Design," for fuel design limits, GDC 15, "Reactor Coolant System Design," for the RCPB pressure limits during non-LOCA conditions, and 10 CFR 50.46 for the core coolability requirements during LOCA conditions. Chapter 15, "Accident Analysis," of the SRP provides guidance for implementing the requirements of GDC 10, 15, and 10 CFR 50.46, as they relate to design-basis transients and accidents, as well as the associated ACs.

The staff based its review of the DBAs and the associated ACs for the ACR-700 preapplication on the guidance specified in Chapter 15 of the SRP for non-LOCA analyses and the 10 CFR 50.46 requirements for LOCA analyses.

2.1.2 Regulatory Review

The staff's review identified areas where AECL's submission deviated from the current NRC review regulations and guidance and detailed the technical, regulatory, and policy issues to be resolved.

2.1.2.1 ACR-700 Event Categorization

For the ACR-700 design, AECL defines the initiating events according to their anticipated frequency of occurrence, and categorizes Class 1 events with a yearly frequency of greater than 10^{-2}, Class 2 events with a yearly frequency ranging from 10^{-2} to 10^{-3}, Class 3 events with a yearly frequency ranging from 10^{-3} to 10^{-4}, Class 4 events with a yearly frequency ranging from 10^{-4} to 10^{-5}, and Class 5 events with a yearly frequency ranging from 10^{-5} to 10^{-6} (see AECL Report, "ACR-700 Limited Core Damage Accidents," presented to the NRC on April 5, 2004, and the letter from V. Langman (AECL) to the NRC, "Response to NRC's Request for Additional Information (RAIs) on Event Categorization," dated June 7, 2004). AECL further groups the event classes, according to their calculated results, into DBEs (including Class 1 through Class 3 events) and limited core damage accidents (LCDAs) (including Class 4 through Class 5 events). AECL indicated that for the DBEs, the analysis will use conservative assumptions to demonstrate that the amount of fuel cladding failures is limited (see Langman letter dated June 7, 2004) and that no PT failures occur (see AECL Report 108-03600-AB-003, Revision 0, "Analysis Basis: Safety Basis for ACR," July 22, 2003; Section 2 of AECL Report 10810-03510-AB-001, Revision 0, "Initial Conditions and Standard Assumptions Safety Analysis Report, August 1, 2003; AECL Report 108-03500-AB-004, Revision 1, "Fuel and Fuel Channel Safety Analysis Methodology," September 2003; and "ACR-700 Limited Core Damage

Accidents”). For the LCDAs, the analysis will use best-estimate assumptions (see “ACR-700 Limited Core Damage Accidents) to show that no fuel melting or propagation occurs beyond the initially affected channels (see Langman letter dated June 7, 2004). The NRC must approve all methodologies.

2.1.2.2 SRP Event Categorization

The SRP defines a DBE as an initiating event caused by a failure of a single process in combination with a single failure of another active component or system. It divides DBEs into AOOs and postulated accidents. The requirements of Appendix A to 10 CFR Part 50 define AOOs as conditions of normal operation and those transients that are expected to occur one or more times during the life of a plant; therefore, AOOs encompass Class 1 events for the ACR-700 design. Postulated accidents specified in SRP Chapter 15 encompass Class 2 and Class 3 events.

In the SRP, each event class contains a number of specific events. A comparison of the events considered for the ACR-700 design with those specified in the SRP shows that the events considered for analyses are significantly different (see Table 2 in AECL Report 108-03600-AB-003, Revision 0; Tables 2-1 and 2-2 in Section 2 of AECL Report 10810-035-AB-001, Revision 0; and Tables A2 through A4 in the Langman letter dated June 7, 2004). For example, many SRP events (e.g., feedwater flow increase, excessive steam demand, and turbine trip) are not considered for the ACR-700 design. Further, AECL does not provide justification for the SRP events that it did not consider for the ACR-700 analysis. In addition, because of the uniqueness of its design, many events (e.g., the off-stagnation feed break, the partial single channel flow blockage, and the PT/calandria tube failure) that are not included in the SRP, are considered as DBEs for the ACR-700 analysis. The staff also finds that the different sets of initiating events listed in AECL Report 108-03600-AB-003, Revision 0, Section 2 of AECL Report 10810-03510-AB-001, Revision 0, and the Langman letter dated June 7, 2004 are not always consistent with each other.

In order to complete its review, the AECL should provide a complete list of the specific events in each event class for the ACR-700 design. AECL should define the initiating events in each event class to be consistent with those specified in Chapter 15 of the SRP and provide justification for the SRP events that will not be analyzed for the ACR-700 design. For those events that are not included in the SRP, AECL should provide for each event adequate justification, including appropriate plant data, relevant and acceptable analytical results to determine the event classification.

AECL indicates that the stagnation feeder break (SFB) and the severe channel flow blockage (SCFB) would result in a fuel channel rupture (see Langman letter dated June 7, 2004). AECL claims that the likelihood of occurrence of those two events is very low, and thus, classifies them as LCDAs (see Langman letter dated June 7, 2004). AECL will analyze them using “design-centered” (best-estimate) assumptions (see “ACR-700 Limited Core Damage Accidents”) without assuming a loss of offsite power (LOOP) (see Langman letter dated June 7, 2004). The staff does not agree with AECL’s approach for analyzing the events. The staff determined that those two LCDAs, the SFB of Class 4 events, and the SCFB of Class 5 events classified in “ACR-700 Limited Core Damage Accidents,” should be classified as DBEs (equivalent to postulated accidents specified in the SRP) because they are initiating events caused by failures of a single process, thus meeting the definition used in the current licensing

practice for DBEs. In addition, the SFB is a small-break LOCA and is within the purview of 10 CFR 50.46, which requires that LOCA analyses consider different break sizes and locations. Therefore, the staff has determined that AECL should analyze both events with approaches consistent with those applied to the analyses of DBEs using methods, computer codes, and conservative assumptions acceptable to the NRC. Further, a LOOP, in addition to the limiting single active failure, should be assumed in the analyses of the events.

AECL only includes LOOP assumptions in analyses of Class 1 through 3 events involving failures of the RCPB (see AECL Report 108-03600-AB-003, Revision 0; letter from V. Langman (AECL) to the NRC, "Regulatory Treatment of Limited Core Damage Accidents for the ACR-700," May 10, 2004; and Langman letter dated June 7, 2004). However, it does not include a LOOP in its analyses of other DBEs, such as Class 1 through 3 events caused by faults other than RCPB failures, an SFB, and an SCFB. The AECL approach may not comply with the requirements of GDC 17, "Electric Power Systems," which states the following:

> An onsite electric power system and offsite electric power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

Accordingly, the NRC expects that a LOOP will not be considered as a single-failure event, but should be assumed in the analysis for each of the AOOs and postulated accidents without changing the event classes. AECL should comply with the GDC 17 requirements in its classification of the DBEs (caused by RCPB failures or faults other than RCPB failures) in each event class for the ACR-700 design.

Regarding the assumption of the time of a LOOP, AECL indicated that it will assume in the analysis a consequential LOOP resulting from a turbine trip or a reactor trip (see Langman letter dated June 7, 2004). AECL assumption of a consequential LOOP in the analyses of DBEs is not consistent with the SRP guidance, which specifies that for the steamline break and the feedwater line break, the worst time of a LOOP should be considered.

Accordingly, AECL should assume the time of a LOOP consistent with the SRP guidance.

2.1.2.3 <u>ACR-700 Acceptance Criteria for Analyses of Design-Basis Accidents</u>

The ACs for analyses of DBEs are established in accordance with the basic principle that more frequent occurrences should have the potential to yield lower radiological consequences than those situations that are less likely to occur. Specifically, AECL Report 108-03600-AB-003, Revision 0, "Analysis Basis: Safety Basis for ACR," July 22,2003; Section 2 of AECL Report 10810-03510-AB-001, Revision 0, "Initial Conditions and Standard Assumptions Safety Analysis Basis," August 1, 2003; AECL Report 108-03500-AB-004, Revision 1, "Fuel and Fuel Channel Safety Analysis Methodology"; and "ACR-700 Limited Core Damage Accidents," list, for limiting radiological consequences, the following ACs for fuel limits, fuel channel limits, and RCPB limits:

Class 1 Events:

- No fuel failures and no PT failures occur in order to ensure fuel and fuel channel integrity.

- The RCPB pressure is less than the Level B limit (110 percent of the design pressures) for cases with Shutdown System 1 (SDS1) to trip, and is less than the Level C limit (120 percent of the design pressures) for cases with Shutdown System 2 (SDS2) to trip.

Class 2 Events:

- No fuel failures in the unaffected fuel channel and no PT failures in the unaffected fuel channel occur.

- The RCPB pressure is less than the Level C limit (120 percent of the design pressures) for cases with SDS1 to trip, and is less than the Level D limit for cases with SDS2 to trip.

Class 3 Events:

- Limited fuel failures occur, but no PT failures occur (except channels that are affected by the initiating event).

- The RCPB pressure is less than the Level C limit (120 percent of the design pressures) for cases with SDS1 to trip, and is less than the Level D limit for cases with SDS2 to trip.

Class 4/5 Events:

- No fuel melting or propagation occurs beyond the initially affected channels (see “ACR-700 Limited Core Damage Accidents”)

AECL Report 108-03600-AB-003, Revision 0, and Section 2 of AECL Report 10810-03510-AB-001, Revision 0, provide the limiting conditions to meet the ACs for the fuel channel integrity and the fuel integrity. For fuel integrity, the following conditions must be met:

- No fuel cladding dryout or flow stratification in the channel occurs.

- If dryout or flow stratification does occur, the fuel cladding temperatures must be less than 800 EC (1472 EF).

- If the fuel cladding temperatures are greater than 800 EC (1472 EF), then the following conditions must be met:

  (1) no fuel centerline melting

  (2) no excessive diametral strain—uniform cladding strain less than 5 percent for cladding temperatures less than 1000 EC (1832 EF)

  (3) no significant cracks in the surface oxide—uniform cladding strain less than 2 percent for cladding temperatures greater than 1000 EC (1832 EF)

(4) no oxygen embrittlement—oxygen concentration less than 0.7 weight percent over half the cladding thickness

(5) no cladding failure because of beryllium-braze penetration at bearing pad or spacer pad location

(6) no fuel cladding melting

For fuel channel integrity, the following conditions must be met:

- No fuel melting occurs.
- The PT temperature is less than 600 EC (1112 EF).
- If the PT temperature is greater than 600 EC (1112 EF) and the PT strains to contact its calandria tube, the calandria tube shall remain intact. This condition is satisfied if sustained film boiling does not occur on the calandria tube outside surface.
- No centerline melting occurs.

2.1.2.4 SRP Acceptance Criteria for the LOCA and Non-LOCA Analysis

The SRP specifies the ACs for each initiating event in accordance with the regulatory requirements, such as GDC 10 and 15 and 10 CFR 50.46. Specifically, Chapter 15 of the SRP specifies the following ACs for LOCAs and non-LOCAs:

Non-LOCAs with a LOOP include AOOs, AOOs with a single failure and postulated accidents with a single failure. The ACs are as follows:

AOOs

- Departures from nucleate boiling (DNB) and fuel centerline melting are not permitted in order to ensure fuel cladding integrity.
- The pressures in the reactor coolant and main steam systems should be maintained below 110 percent of the design values.

AOOs with any single failure of active component or system

- DNB occurrence on a limited number of the fuel rods is allowed. All fuel rods are assumed to fail if they experience DNB. An estimate of the number of potential fuel failures shall be provided for radiological dose calculations. There shall be no loss of function of any fission product barrier other than the fuel cladding.
- pressures in the reactor coolant and main steam systems below 110 percent of the design values

Postulated accidents with any single failure of active component or system

- The minimum DNB ratio (DNBR) can be used to assess the potential for core damage.

- Any fuel damage calculated to occur must be sufficiently limited to allow the core to remain in place and intact with no loss of core cooling capability.

- The pressures in the reactor coolant and main steam systems should be maintained below 110 percent of the design values (except for the double-ended guillotine feedwater line break; its pressures are limited to 120 percent of the design pressures).

LOCAs with Any Single Failure of Active Component or System and with a LOOP

- 10 CFR 50.46 requires that the LOCA analyses consider different postulated break sizes and locations, as well as the effects of the limiting single failure and a LOOP. The results of the analyses must satisfy the following acceptance criteria:

  – The peak cladding temperature is less than 1204 EC (2200 EF).

  – The maximum local oxidation of the cladding is less than 17 percent of the total cladding thickness.

  – The hydrogen generated from the chemical reaction of the cladding with water or steam is less then 1 percent of the total metal in the cladding cylinders surrounding the fuel.

  – No loss of core cooling capability occurs.

  – Long-term cooling (LTC) capability is maintained.

A comparison of the ACs and associated limiting conditions discussed in Sections 2.1.2.3 and 2.1.2.4 above shows that the fuel integrity protection principles are similar; specifically, for the most likely events (AOOs or Class 1 events), the fuel integrity is maintained, while for the less likely events (postulated accident or Class 2 and Class 3 events), a limited amount of fuel damage is allowed, but core coolability is maintained. However, significant differences exist in the specific ACs discussed in the SRP and those specified for the ACR-700 design. Specific differences include the following:

- The SRP provides two different sets of ACs for LOCAs and non-LOCA events, while AECL proposes one set of ACs applicable to both LOCAs and non-LOCA events.

- The SRP uses the DNBR limit to evaluate the fuel integrity and specifies that all fuel rods are assumed to fail if they experience DNB, while the ACR-700 uses the specific limits of the cladding temperatures, fuel centerline melting, cladding strains, oxygen embrittlement, and fuel cladding melting to determine fuel failures for various cladding and fuel temperatures, and flow conditions in the fuel channels.

- The SRP specifies that for LOCA analyses, the five ACs in 10 CFR 50.46 must be met. AECL indicates that the LOCA analyses to be submitted for the ACR-700 design certification will demonstrate compliance with 10 CFR 50.46 ACs (see Langman letter dated June 7, 2004). However, it does not explicitly address compliance with each of

the 10 CFR 50.46 ACs. AECL should submit the specific ACs and address its compliance with the 10 CFR 50.46 ACs.

- The ACs and associated limiting conditions for the fuel channel integrity are unique to the ACR-700 design and are not applicable to currently licensed LWRs. Therefore, they are not available in the SRP.

- The SRP specifies that the RCPB pressures are limited to 110 percent of the design pressures, with the one exception which limits the design pressure for the double-ended guillotine feedwater line break to 120 percent. AECL allows 110 percent of the design pressures (Level B limit), 120 percent of the design pressures (Level C limit) of the design pressures, and the Level D limit for various events at different plant conditions.

2.2 Potential Policy Issues

The staff determined that the SFB and the SCFB are DBEs because they are initiating events caused by failures of a single process. This meets the definition of DBE in the SRP. In categorizing DBEs, AECL stated no fuel melting or propagation beyond the initially affected fuel channels to occur for the LCDAs, including the SFB (a small-break LOCA) and the SCFB (a small-break LOCA resulting from the channel flow blockage), but it assumes cladding melting and fuel melting in the affected channels is expected could occur. The proposed ACs for these two DBEs deviate from the 10 CFR 50.46 requirements which do not allow cladding oxygen embrittlement, fuel cladding, or fuel melting to occur as a result of a LOCA. AECL will likely have to obtain an exemption from the 10 CFR 50.46 fuel performance ACs for these two DBEs. Pending acceptance of the approach by the staff, this is a policy issue and requires Commission approval.

2.3 Technical Issues

AECL should provide a complete list of the specific events in each event class applicable to the ACR-700 design for the staff to review. The initiating events in each event class should be consistent with those specified in Chapter 15 of the SRP, and AECL should provide justification for the SRP events that it did not analyze for the ACR-700 design. For those events that are not included in the SRP, AECL should provide for each event adequate justification, including appropriate data and relevant analytical results, to determine the event classification for a specific event. In its classification of initiating events, AECL should include a LOOP, in addition to single active failure, for each initiating event, without changing the event classes.

AECL should provide technical information to justify the six specific limits that are proposed to meet the ACs for fuel integrity in terms of fuel centerline melting, cladding strain, cladding cracks, beryllium-braze penetration, and fuel cladding melting. The information should include test data, calculational methods, and computer codes used to support the specific limits for fuel integrity.

AECL should submit its specific ACs and address compliance with the five ACs specified in 10 CFR 50.46 for LOCA analyses.

AECL should provide technical information to justify the conditions (discussed in Section 2.1.2.3 of this report) it proposed to meet the ACs for fuel channel integrity, in terms of PT temperature and stain, and modes of heat transfer on the calandria tube outside surface.

AECL should provide technical information to justify the proposed RCPB pressure limits of Levels B, C, and D (discussed Section 2.1.2.3 of this report) for various events at different plant conditions.

AECL should provide relevant analyses and applicable test data to justify the ACs that specify no fuel melting or propagation occurrence beyond the initially affected channels for the SFB and the SCFB.

2.4 Conclusions

Based on its review of the FT2 information, the staff determined that the SFB and SCFB, classified by AECL as LCDAs to be analyzed using best-estimate assumptions, should be analyzed as DBEs using methods, computer codes, and conservative assumptions for input parameters that are acceptable to the NRC. Regarding considerations of a LOOP in the event classification, the staff determined that AECL should include a LOOP, in addition to a single-failure event, in its classification of the AOOs and postulated accidents in each event class, without changing the event classes. In addition, the staff identified significant differences in the event categorization and the associated ACs for the ACR-700 design as compared to the SRP guidance and the 10 CFR 50.46 requirements. In order to resolve the identified differences, AECL must satisfactorily resolve the technical, regulatory, and policy issues stated above.

2.5 DBA Source Term Assessment

2.5.1 Review Scope

The staff performed a preapplication review of the FT2 information for its DBA source term and the radiological dose ACs proposed by AECL. The staff identified key policy and technical issues associated with the DBA source term.

AECL has provided information on the ACR-700 design, expected reactor transient behavior during hypothetical reactor accidents, AECL current experimental database, and proposed future AECL reactor accident research programs applicable to the ACR-700.

Specifically, the staff reviewed the following documents provided by AECL:

- ACR 108-126810-LS-001, “Phenomenology for Limited and Severe Core Damage Accidents in an ACR,” September 2003

- ACR 108-126810-LS-002, “ACR Limited and Severe Core Damage Accidents: Supporting R&D,” November 2003

- AECL letter to NRC, “Proposed Accident Source Terms for ACR-700,” dated May 28,2004

- AECL letter to NRC, “Regulatory Treatment of Limited Core Damage Accidents for ACR-700,” dated May 10, 2004

- AECL presentation, “Topical Meeting on Severe Accidents,” held at Chalk River Laboratory on July 20 - 22, 2004

2.5.2 Review Guidance

The regulatory requirement and guidance that the staff considered in its review of the ACR-700 preapplication DBAs and the ACs for the ACR-700 design include the following:

- 10 CFR 50.34, as it sets forth fission product release assumptions and radiological consequence dose acceptance criteria

- Regulatory Guide (RG) 1.183, “Alternative Radiological Source Terms for Evaluating Design-Basis Accidents at Nuclear Power Reactors,” dated July 2000, as it describes alternative source term (AST) and provides guidance on implementation of the AST

- NUREG-0800, Section 15.0.1, “Radiological Consequence Analyses Using Alternative Source Terms”

2.5.3 DBA Source Term Regulatory Assessment

The ACR-700 design is unique as compared to conventional LWRs. Some of the most notable characteristics of the ACR-700 design are modular horizontal fuel channels, on-power fueling, slightly enriched uranium (SEU) fuel with light-water cooling, and the separate low-pressure, low-temperature, and heavy-water moderation with relatively low neutron absorption. These unique design characteristics pose a challenge to the staff’s current ability to analyze postulated ACR-700 reactor accident scenarios.

In its preapplication submissions, AECL proposed a scenario-specific mechanistic reactor accident source term for the ACR-700 reactor design in lieu of the AST for LWRs described in RG 1.183. The proposed source term is based on a large-break LOCA with coincident loss of emergency core cooling (LOCA/LOECC).

The ECC consists of a direct high-pressure injection of makeup water into the RCS from the accumulator tanks containing water at relatively high pressure (emergency coolant injection (ECI) system) and pumped injection of makeup water into the RCS from the reactor building sump (long-term cooling system). In addition, the ACR-700 design includes an emergency water supply in the reserve water tank at a high elevation in the reactor building. The connections to the RCS and to the reactor building sump are automatically opened on the ECC initiation signal and water flow to the RCS and to the reactor building sump is by gravity.

In this accident sequence, AECL postulated that a large-diameter pipe in the primary system (one of the two inlet headers) is ruptured, discharging its light-water coolant into containment. The reactor trips, and although a signal is generated for injection of ECC flows as described above, multiple failures are postulated that prevent the addition of cooling water to the RCS.

The RCS water inventory continues to decrease and the heat transport pumps become less and less effective until they trip automatically.

AECL further postulated that as flow decreases, fuel and PT temperatures increase. Fuel heat up, clad oxidation, clad failure, and consequent fission product release occur in most channels. Eventually, the PT becomes hot enough to deform under the weight of the fuel in the channel. Any subsequent heat up of the PT may result in sagging, allowing it to contact a cold calandria tube, which causes its temperature to drop significantly. Heat is transferred from the calandria tube to the surrounding heavy-water moderator. AECL assumed that the moderator cooling system will remove heat from the moderator. AECL, therefore, postulated that no fuel melting will occur. AECL estimated that a LOCA/LOECC accident provides the highest fission product source term into containment—on the order of 10 percent of the total core fission product inventory.

2.5.4 Potential Policy Issues

The DBA source terms proposed by AECL are major deviations from the 10 CFR 50.34 requirement which states, in part, the following:

> the fission product release assumed for this evaluation should be based upon a major accident, hypothesized for purpose of site analysis or postulated from considerations of possible accidental events. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release into the containment of appreciable quantities of fission products.

Currently operating reactors were licensed (1) using a source term “hypothesized for purpose of site analysis,” and hence not mechanistic, and (2) assuming “substantial meltdown of the core.” Since the AECL approach deviates from the requirements of 10 CFR 50.34, this issue will require Commission consultation.

2.5.5 Technical Issues

AECL has not provided the scenario-specific mechanistic source term characteristics, including fission product composition and magnitude, chemical and physical form of the fission products, and the timing of the release of these fission products. AECL stated that these source term characteristics will be provided with the ACR-700 standard reactor design certification application at a later date.

The staff plans to modify the MELCOR severe accident code as part of the NRC’s Office of Nuclear Regulatory Research’s (RES) severe accident research program. This code will be used to (1) independently evaluate if core melt is a credible occurrence in the LOCA/LOECC accident sequence, and (2) independently confirm the source term characteristics that will be provided by AECL.

To assist the staff in identifying the physical processes most important in the analysis of ACR-700 reactor accidents, the recent staff efforts to develop the ACR-700 phenomena identification and ranking table (PIRT) process identified, among others, fission product source terms during fuel interaction with pressure and calandria tubes as a key technical issue that

must be addressed. Fission product release prediction during periods of fuel interaction with pressure and calandria tubes is complicated by a lack of data. From a source term perspective, the staff believes that it would be useful to consider the onset of extensive core debris interactions with PTs and calandria tubes as delineating the boundary between DBAs and severe accidents.

2.5.6 Conclusions

Based on the gaps between the current knowledge base and the information required to address pertinent regulatory issues for the ACR-700, AECL should address the following ACR-700 DBA source term modeling and assessment issues in order for the NRC staff to reach satisfactory conclusions for design certification:

- Cross section data for radionuclide inventory in the ACR-700 core
- Scenario-specific accident source term including fission product composition and magnitude, chemical and physical form of fission products, and the timing of release of these fission products
- Iodine chemistry in containment, particularly the formation of gaseous iodides
- Fission product release during fuel interaction with pressure and calandria tubes
- Fission product and aerosol behavior in primary heat transport system and containment including aerosol removal characteristics in ACR-700 specific geometry

The staff plans to perform the following activities in order to develop independent capabilities for safety analysis of the ACR-700 design.

- The severe accident code, MELCOR, will be modified as part of the RES ACR-700 severe accident research program to model the unique ACR-700 configuration.
- The staff will use the modified MELCOR to:

  (a) verify the LOCA/LOECC accident sequence postulated by AECL
  (b) predict fission product release source terms
  (c) evaluate if core melt is a credible occurrence
  (d) address iodine chemistry in the containment, in particular, formation of gaseous organic iodides and the gas phase decomposition of gaseous iodides.

On the basis of the information provided by AECL during the preapplication review, and pending resolution of the policy and technical issues discussed above, the staff did not identify any issues related to the DBA source term that would preclude certification of the ACR-700 design.

3. COMPUTER CODES AND VALIDATION ADEQUACY (FT3)

3.1 Review Scope

The ACR-700 standard design is an evolutionary extension of the CANDU reactors that have been operating in Canada and in other countries for many years. The basic CANDU reactor core design differs from the designs of reactors currently operating in the United States. The remainder of the ACR-700 nuclear steam supply system is similar to the designs of PWRs operating in the U.S. The ACR-700 design includes two U-tube-type steam generators, a pressurizer, and four reactor coolant pumps. The fuel bundles are contained and cooled within horizontal PTs inside of calandria tube that is surrounded by a heavy-water moderator tank (calandria). Each PT comprises a fuel channel. Unlike operating CANDU reactors, the reactor coolant that flows within the fuel channels will be ordinary water instead of heavy water.

The PT walls are a part of the RCPB so that the moderator tank is only slightly above atmospheric pressure. There are 292 horizontal fuel channels in the latest ACR-700 design, each containing 12 fuel bundles. Each fuel bundle is 4 inches in diameter and 19.5 inches long; each contains 43 individual fuel pins, grouped in circular arrays. A design feature of the ACR-700, as with operating CANDU reactors, is that the reactor can be refueled while at power. Refueling machines, which are located on either side of the horizontal reactor core, add and remove fuel bundles in preset pattens.

In a meeting with the NRC staff on September 26, 2002, AECL proposed a plan for preapplication review of the ACR-700 (see “A Plan for the Pre-Application Review of the Advanced CANDU Reactor (ACR)” dated September 26, 2002) which was supplemented by letter (see Langman (AECL) to Sosa (NRC), “Phase 2 of ACR Pre-Application Review,” dated July 30, 2003). This proposal called for an early review of certain computer codes that the NRC will review in the AECL safety analysis of the ACR-700; the proposal also called for the NRC staff to provide a PASAR describing the status and conclusions from that review. The review proposal, accepted by the staff, categorizes the review of the computer codes and the adequacy of their validation as Focus Topic 3 (FT3).

To address FT3, the NRC staff developed a preliminary review plan to initiate the early review requested by AECL. The early review process included the following:

- the principal thermal-hydraulic (T-H) and reactor physics computer codes
- development of an independent PIRT for major safety analysis functions to be performed by the computer codes
- assessment of the experimental database for the computer codes
- development of independent NRC staff confirmatory analysis capabilities

This section contains the staff’s preliminary evaluation of FT3. It provides the NRC staff comments on the adequacy of the computer codes and summarizes the status of the staff’s review. This report evaluates four computer codes, RFSP-IST, WIMS-IST, and DRAGON-IST,

used to evaluate the reactor core physics, and the CATHENA code, used for the T-H analysis of the reactor system during postulated transient and accident conditions.

3.2 Regulatory Basis

Pursuant to 10 CFR 52.47(a)(1)(i), an application for design certification must contain the technical information required of applicants for construction permits and operating licenses in 10 CFR Part 50, and which is technically relevant to the design and not site-specific. Such information includes that which is required by 10 CFR 50.34, specifically including analysis and evaluation of emergency core cooling system (ECCS) cooling performance following postulated loss-of-coolant accidents in accordance with the requirements to 10 CFR 50.46. In addition, 52.47(a)(1)(ii) requires that the application demonstrate compliance with any technically relevant portions of the Three Mile Island requirements (for analyzing certain transients, among other things) set forth in 10 CFR 50.34(f), except for 50.34(f)(1)(xii), (f)(2)(ix), and (f)(3)(v). Chapter 15 of the SRP describes these accidents and transients. The NRC staff will review how the computer codes of FT3 are used for the ACR-700 safety analysis to ensure that the requirements of the Commission's regulations are met.

For the analysis of LOCAs, 10 CFR 50.46 provides two acceptable approaches. Appendix K to 10 CFR Part 50 describes one acceptable approach. Appendix K provides a set of specific requirements that evaluation models should meet. Appendix K was written to apply to PWRs and BWRs operating in the United States, all of which have cores containing vertically mounted fuel assemblies. Some of the Appendix K requirements may not apply to CANDU reactors, which have fuel assemblies within horizontal fuel channels.

The alternate acceptable approach described in 10 CFR 50.46 is to demonstrate that the analytical model realistically describes the behavior of the reactor system during a LOCA. Using comparisons to applicable experimental data, uncertainties must be identified and accounted for in the calculated result to give a high level of probability that the acceptance criteria for LOCAs listed in 10 CFR 50.46 are not exceeded. At this time, AECL has not documented which of these approaches it will take for the analyses of LOCAs. Therefore, the NRC staff will defer this part of the evaluation to a later time.

The staff has prepared suggested means by which the general requirements for computer codes used in reactor safety analysis can be met. An acceptable approach for preparing applications for approval is described in Draft Regulatory Guide (DG) 1120, "Transient and Accident Analysis Methods," dated December 2002. In its review, the staff compared the AECL submission to the approach suggested by the DG. The NRC staff reviewers of safety analysis computer codes have utilized the guidance provided in draft SRP Section 15.0.2, "Review of Analytical Computer Codes." The DG and SRP divide the review into six areas:

(1) documentation
(2) evaluation model
(3) accident scenario identification process
(4) code assessment
(5) uncertainty analysis
(6) quality assurance (QA) plan

The review approach described in the DG and draft SRP Section 15.0.2 provides for an ordered procedure by which the analytical purposes of computer codes are established and their applications are thoroughly described. A significant feature of the process is the development of PIRT. The PIRT allows for a comprehensive validation process which allows attention to be focused on the more significant physical processes and phenomena which the code will be called upon to evaluate. To aid in the preapplication review, the NRC staff assembled its own PIRT panels. Section 3.5 of this report discusses the findings of these panels relevant to the FT3 computer codes. The staff has also requested AECL to submit the material and reports discussed in DG-1120, including its own PIRT, for each application of the FT3 computer codes.

3.3 Regulatory Issues

The staff did not identify any regulatory issues, such as rules, rulemaking, or exemptions, that will need to be resolved before completing its evaluation of the computer codes being reviewed under FT3 at this time.

3.4 Potential Policy Issues

In completing its review of the computer codes discussed in FT3, the staff did not identify any policy issues that will need upper management or Commission guidance for resolution at this time.

Although the staff did not identify any regulatory or policy issues associated with the review of the codes themselves, based on discussions with AECL, the staff understands that for certain potential break sizes to be analyzed for the ACR-700, the results may not fall within the ACs for LOCAs outlined in 10 CFR 50.46. The staff understands that AECL plans to file an exemption request to this regulation based on the low probability that a break will occur within this size range. Until the analytical results for the ACR-700 have been submitted, the staff will not be able to assess the need for an exemption to this regulation or other regulations for which the results predicted for, or the acceptability of the ACR-700 may not fall within the requirements of NRC regulations.

3.5 Evaluation of the RFSP-ISL, WIMS-ISL, and DRAGON-ISL Reactor Physics Computer Codes

3.5.1 Reactor Physics Code Descriptions and Review Scope

This section describes three analysis codes, WIMS-ISP, DRAGON-ISP, and RFSP-ISP, together with a discussion of their application in the analysis of the CANDU core behavior. The analytic techniques have all been developed and validated for the current CANDU design (see Douglas, et al., “System Validation Manual for WIMS-IST/DRAGON-IST/RFSP-IST Reactor Physics Code Suite,” dated November 2001). Thus, the major focus of this review is on the applicability of the currently used code package to the ACR-700 design.

The primary differences between the currently designed CANDU reactors and the ACR-700. from a reactor physics standpoint, are as follows (see Altiparmakov, “Calculation of Coolant Void Reactivity Components in Lattices of NG CANDU Type,” FFC-RRP-398, January 2002):

- Lattice spacing between the fuel bundles is reduced from 28.75 centimeters (cm) to 22 cm, in an effort to reduce the heavy-water inventory and to ensure a negative value for the coolant void reactivity (CVR).

- Light water rather than heavy water is used as a coolant to reduce the heavy-water inventory and to simplify the heat removal system.

- The gap between the PT and calandria tube is increased in order to decrease the moderator to fuel ratio, which is intended to help maintain a negative value for the CVR.

- SEU will be used to enhance the burnup and improve fissile material utilization.

- The fuel bundles consist of 43 elements, rather than a smaller number (37 or 28) in the current designs.

- The amount of fissile material per fuel bundle is reduced to allow for the higher fission gas inventory expected as a result of the higher burnup.

- Burnable poison (dysprosium (Dy)) is added to the central fuel element to reduce the reactivity swing between fresh and fully burned fuel and to reduce the value of the CVR.

- Enrichment zone loading (SEU to natural uranium (NU)) across the fuel bundle may be introduced. The enrichment variation may help reduce the CVR.

A primary reason cited by the applicant for making the changes is to reduce the CVR to negative values. The applicant's stated expectation (see ACR-700 Technical Description, Revision 0) is that the CVR will be negative in all operating conditions (i.e., burnup, voiding pattern, and fuel management scheme).

Many of the nuclear analysis documents reviewed for FT3 describe the codes and their validation as applied to the analysis of conventional CANDU reactors fueled with natural uranium. The following subsections summarize the reviewed information, including that pertaining to conventional CANDU neutronics, and provide preliminary staff comments on technical issues that could necessitate modifications to the codes and modeling techniques and extensions to the validation data base for ACR-700 analysis. During the preapplication review, the applicant described its ongoing efforts to identify and complete necessary changes and improvements in these areas.

3.5.1.1 WIMS

WIMS is a two-dimensional, multigroup neutron transport code routinely used in the analysis of CANDU reactors. The code is coupled to dedicated libraries of microscopic nuclear data (see Altiparmikov, "WIMS-AECL Theory Manual," COG-00-07, dated September 2001). It encompasses the physics aspects of the reactor lattice cell calculations, from few group cross section preparation to cell homogenization and burnup calculations based on detailed space-energy neutron flux distributions. In practice, standard input models are used in order to decrease the computation time. The combination of WIMS and a standard input model is known as WIMS-IST, which is used for the validation calculations referred to below. WIMS-IST has many input options, since there is more than one way of carrying out a cell calculation.

Because CANDU reactors have more than 4000 fuel bundles, each with unique properties to be analyzed on the cell level, the WIMS-IST calculations need to be both acceptably accurate and fast. The following are basic features of the WIMS standard input model:

- The full neutron energy range is collapsed from 89 groups to 33 groups when carrying out the transport theory solution. The 33-group structure is chosen with conventional CANDU reactors in mind.

- The more accurate two-dimensional spatial mesh is replaced by a hybrid model, which uses a combination of a two-dimensional and a one-dimensional representation. The selection of the mesh spacing is a compromise between speed and accuracy.

- The cell outer boundary condition is not treated as a two-dimensional "white" boundary (scattered isotropically), but as a one-dimensional cylinder with a white boundary.

- The resonance treatment is based on an approximate cluster treatment, which is faster than the two-dimensional collision probability method. The approximate treatment is known as the *newres* cluster treatment.

- Azimuthal variation in the fuel properties is not allowed in the standard model, although it is possible with WIMS.

In addition to the above choices, the standard input model also recommends the following:

- the Benoist treatment for determining the diffusion coefficients

- the B-1 approximation for incorporating the buckling into the cell-averaged leakage estimate

3.5.1.2 DRAGON

DRAGON-IST solves the three-dimensional, steady-state neutron transport equation using the collision probability method (see Marleau and Hebert, "A User Guide for DRAGON: Version DRAGON," Release 3.04, IGE.174, dated April 2000). It has flexibility regarding the choice of spatial regions and the number of energy groups. It has the capability to treat resonance self-shielding and to carry out burnup calculations. DRAGON includes several boundary condition options. Finally, DRAGON could serve as an infinite lattice code similar to WIMS.

The primary reason for using DRAGON is its three-dimensional capability which can determine input for the Reactor Fueling Simulation Program (RFSP) in complex geometric situations encountered in the vicinity of control elements, which are introduced at right angles to the fuel channels. A standard input file and a frozen version of the code have been identified for design calculations, thus creating a DRAGON-IST.

3.5.1.3 Reactor Fueling Simulation Program

The RFSP is a major design code used in the analysis of CANDU reactor performance (see Rouben, "RFSP Program Description," Revision 1, AECL Report TTR-370, dated April 1995). It

can calculate both static and time-dependent neutron flux and power distributions in the core. The static distributions can be determined in two different ways:

(1) solution of the three-dimensional finite-difference neutron diffusion equations assuming two energy groups

(2) a flux synthesis method using an expansion of the flux in a series of precalculated modes to fit in-core vanadium detector responses

To carry out the diffusion theory calculation, RFSP uses the two group cross section data generated by WIMS-IST and DRAGON-IST for the various lattice cells. In addition, RFSP includes a simple cell model that can be used to vary the state points (e.g., fuel temperature, coolant density, etc.) without having to carry out a complete WIMS calculation. Furthermore, RFSP incorporates a simpler cell code (POWDERPUFS-V), which is based on a "1.5 group representation." This code is only used for rough estimates and "tuned" to specific reactor and core designs. It is not expected that this simple cell code will be applied to ACR-700.

RFSP can perform fuel management calculations and can simulate a reactor operating history at specific intervals, taking burnup steps and channel refueling into account. Using the diffusion theory option, the program can calculate time-averaged flux, power, and irradiation distributions and channel refueling frequencies. In addition, the following quasi-static capabilities are included:

- determination of harmonic flux modes used in flux mapping
- estimation of fluxes at specific in-core locations by three-dimensional parabolic interpolation
- calculation of the reactivity increase expected when channels are refueled
- transients due to xenon-135 (Xe-135) and iodine-135 (I-135)

Furthermore, RFSP can follow fast transients, such as shutdown system performance in terminating large LOCAs. This is an important transient for the currently designed CANDU reactors; because the CVR is positive, there will be an increase in power before the shutdown system responds. It may also be important for the ACR-700 (see Section 8 of this report). A standard version of this code and its input package has been created and is known as RFSP-IST.

3.5.1.4 Feedback Coefficients To Be Reviewed

The feedback coefficients to be reviewed are functions of a variety of parameters, with a range as defined in Table 3-1. The parameters correspond to the currently designed CANDU reactors. The definitions cover both operating and accident conditions.

Table 3-1 Range of Parameters for Feedback Coefficients in Conventional CANDUs

Feedback coefficient and Parameter description	Normal range	Accident condition following reactor trip
Coolant density (gm/cc)	0.8–1.1	0.0–1.1
Coolant temperature (EC)	260–310	40–315
Moderator density (gm/cc)	1.06–1.1	0.1–1.1
Moderator temperature (EC)	30–85	30–100
Moderator poison concentration (ppm)	B: 0–11 Gd: 0–3	B: 0–11 Gd: 2600
Moderator purity	99.75 percent–99.96 percent	97.43 percent –99.96 percent
Fuel temperature (K)	850–1300	850–1800
Coolant purity	97 percent–100 percent	97 percent–100 percent

Normal conditions for the ACR-700 will be different from those presented above, since it operates at a higher pressure (approximately 2200 psi), thus causing its outlet temperatures and densities to be different. The values in Table 3-1 are used in the report authored by Douglas, et al. that forms the basis for this study of code validation and applicability to the ACR-700.

3.5.1.5 Guidance for Review

The NRC staff review has followed the recommendations of DG-1120, as discussed in Section 3.2 of this report. The applicant did not present all areas outlined in DG-1120 for staff review at this time.

In the current review, accident scenarios are of reduced importance, since the feedback coefficients are determined by arbitrarily perturbing a specific state point about a nominal operating point. Documentation on QA will not be reviewed explicitly, because AECL uses the Industry Standard Toolset (IST) system. IST is based on accepted, validated, and frozen codes and input options, which ensures that an internal QA system is in place.

The following sections outline the technical issues to be analyzed using the above code package. In addition, the ability of the codes to predict the values of parameters for the currently operating reactors will be discussed. Finally, one parameter, the CVR, will be selected and the implications of its value and underlying phenomenology for the ACR-700, as compared to the current CANDU reactors, will be discussed. Related discussions are also provided in Section 8 of this report.

3.5.2 Technical Issues—Reactor Physics

This section outlines the technical issues related to reactor physics and discusses the application of the code package to their determination. The CVR is used as an example of an important feedback mechanism that is different from the currently designed CANDU reactors, and that has regulatory and safety significance. The major technical issues to be addressed are the feedback mechanisms and power shape determination, which are discussed below.

3.5.2.1 Determination of Major Feedback Mechanisms and Power Distribution

This subsection discusses the feedback mechanisms outlined above in Table 3-1. In particular, for each mechanism, this subsection outlines the physics of the phenomena, discusses validating experiments and the comparison of the IST versions of the codes' predictions, and summarizes implications for the ACR-700.

3.5.2.1.1 Coolant Density Change

A variety of phenomena can induce a coolant density change, such as changes associated with temperature or pressure; however, the most significant phenomena would be in the limit of zero density, implying complete removal of the coolant. This latter case is also known as the CVR worth and has significance in the LOCA scenario.

Changes in the neutron energy spectrum occur when the coolant density changes, leading to several competing processes of opposite sign, the sum of which results in the CVR worth. The most important reactions in this case are those associated with the fissile and fertile materials (uranium-235 (U-235), plutonium-239 (Pu-239), and U-238, respectively). The moderator and coolant determine the overall neutron spectrum, which is characterized by their respective temperatures. A separate moderator and coolant design is fundamental to the CANDU reactor, as compared to designs operating in the United States in which they are the same. This difference is the most significant contributor to the different behaviors of these two reactor concepts. In a CANDU reactor, the moderator and coolant act relatively independently (different density and temperature) and are separated by the PT and calandria tube walls; in U.S. designs, the moderator and coolant are the same material (same density and temperature).

Removing coolant will soften the neutron spectrum on the outer edges of the PT because the neutrons are thermalized in the moderator, which is at a much lower temperature than the coolant. No possibility of neutron spectrum hardening from up-scattering exists because of the higher temperature of the coolant. The softer neutron spectrum will change the relative absorption rate (capture rate + fission rate) between the fissile isotopes. Because the capture/fission ratio of these isotopes varies with energy, the fission rate will also change. The role of the fertile isotope is particularly important since it controls the resonance absorption in the system. In addition, the lack of internal moderation (inside the PT due to coolant) enhances the high energy part of the neutron energy spectrum in the center of the PT and enhances fast fission in the fertile isotopes. Finally, changes in neutron leakage do cause certain effects. Upon voiding, there are unimpeded flight paths for neutrons along the voided channel, thus decreasing the neutron population. Hence, anisotropic neutron transport is important in determining the magnitude of this effect. Leakage effects are not very large in CANDU reactors because of the size. However, for smaller reactors, this might be an important contributor to the CVR.

AECL has carried out critical experiments over many years at the ZED-2 facility located at AECL Chalk River Laboratories (CRL). The fundamental measurement made at the ZED-2 facility is the critical height of the moderator. This defines the size of a critical assembly for the particular core configuration being investigated. The buckling derived from this experiment is the fundamental quantity used in the code validation process. There are two techniques for determining the buckling—flux mapping fits and substitution measurements. Both techniques

have an associated experimental uncertainty. Thus, for a given ZED-2 core configuration, the buckling determination is repeated several times in order to determine a standard error and a standard deviation for each buckling measurement. Measurements have been carried out in both cooled and voided cases. The difference in buckling gives a measure of the CVR. In addition, each value of the CVR is associated with a standard error and a standard deviation. AECL found the error to be very small (i.e., approximately 1.5 percent of the measured difference) and the deviation was approximately 5.0 percent of the measured buckling leakage effect (see Douglas, et al., November 2001).

Comparing predicted and measured results for the appropriate ZED-2 experiment provides a measure of the ability of the physics codes to accurately predict the void reactivity for a CANDU core. The particular code used in this comparison is WIMS-IST. WIMS-IST either predicts the critical buckling for the cooled and voided configurations, or it can use the measured buckling values to determine the magnitude of the multiplication factor. In summary, the reviewed documents pertaining to conventional CANDUs concluded that the WIMS-IST calculation over predicts the measured (ZED-2 core) effect by 0.5 mill-k (mk) to 2.7 mk, depending on core configuration (see Douglas, et al., November 2001). This reported discrepancy is outside the error estimate made for the experimental results. In order to understand the discrepancy and how this result might be projected to an operating CANDU reactor, AECL carried out comparisons to Monte Carlo calculations. For an infinite lattice of CANDU cells, the agreement between WIMS-IST and Monte Carlo N-Particle (MCNP) was surprisingly good, except for the resonance treatment in *newres*. The above conclusion led to the suspicion that the fundamental nuclear data, which all came from the same source, might have a built-in bias (see Douglas, et al., November 2001). In particular the scattering kernels for D_2O, and the resonance treatment for U-238 (*newres*), as well as the general nuclear data for U-235, U-238, and Pu-239, might need reevaluation.

In order to use WIMS-IST to analyze CANDU reactors, a bias must be introduced in the CVR predictions. Based on the experimental data and comparisons to Monte Carlo calculations, the reported bias varies from 2.0 mk to 1.6 mk depending on the element design, with an error of 1.1 mk. This is a smaller range than mentioned above, primarily because the above range included experiments on fuel elements not currently used in CANDU reactors. The overall CVR for a conventional CANDU reactor fueled with natural uranium is 15 mk to 20 mk (Altiparmakov, January 2002), indicating that the error is comparatively small.

The ACR-700 uses light-water coolant, has a tighter element pitch, has a larger calandria tube diameter, uses SEU fuel and burnable poison, and is expected to triple the fuel burnup compared to current CANDU fuel. All these changes will put greater demands on the nuclear data and the computational models. It is expected that a comprehensive set of experiments will have to be carried out in the ZED-2 facility to validate a revised IST set of computational methods. Section 8 of this report provides a more detailed discussion of the possible changes in WIMS-IST and its validation regarding CVR effects in the ACR-700.

3.5.2.1.2 Coolant Temperature Change

It is physically impossible to have a coolant temperature change without affecting the fuel temperature and the coolant density. Therefore, this discussion of the coolant temperature coefficient includes appropriate changes in the coolant density, fuel temperature, and the coolant temperature. However, potential scenarios exist in which the coolant temperature

changes first, affecting the fuel temperature through heat transfer and power changes brought about by the feedback coefficients. In scenarios that reduce the incoming coolant temperature, such as those that would be precipitated by a large steamline break, the transient response would control the fuel temperature.

A change in the coolant temperature will change the up- or down-scattering adjacent to the fuel rods. In general, higher energy neutrons have a lower probability of being captured by fissile isotopes, except in the case of Pu-239 which has a resonance in the upper end of thermal range. The coolant density will also increase or decrease depending on the temperature change, thereby increasing or decreasing neutron leakage, which in turn will decrease or increase the resonance absorption in the fertile nuclides. Furthermore, an increase in coolant temperature also implies an increase in the fuel temperature, which manifests itself as increased Doppler absorption.

Using the ZED-2 facility, AECL performed benchmark experiments to validate the code calculations of coolant temperature coefficients in conventional CANDUs. Selected channels were heated to various temperatures, ranging from 25 EC (77 EF) to 300 EC (572 EF). The coolant spaces were filled with either water or carbon dioxide. The primary quantity measured was the critical height of the reactor and the corresponding buckling. The buckling was determined by a substitution technique or by flux mapping. Measurements were carried out at various temperatures, and the coolant temperature coefficient determined by the slope of the buckling versus temperature curve (in buckling units). Coolant temperature coefficients were determined for 37-element fresh natural uranium (FNU) and mixed oxide (MOX) fuel bundles. The latter experiments included Dy to simulate fission products. Finally, a set of 43-element FNU experiments was carried out.

A determination of multiplication factor (k_e) using WIMS-IST and the measured critical buckling, as a function of temperature, indicates that the code underestimates the values of k_e (see Douglas, et al, November 2001). However, the overall deviation from unity is between 0.002 and 0.007, and is significantly smaller within one fuel type. This reported result indicates that the calculated slope of the buckling vs. temperature curve should be close to the measured slope. The reported analysis confirmed this result by determining the critical buckling using WIMS-IST and plotting the buckling against the coolant temperature. The experimentally determined slope agreed very well with the calculated slope. The agreement seems to indicate that there is a possibility of cancellation of errors among the three major physical phenomena (up-scattering, reduction in scattering density, and fuel Doppler). For FNU fuel, calculated and measured values of the coolant temperature coefficient agree to within a 95 percent confidence level over most of the temperature range.

Reported comparisons between WIMS-IST and the Monte Carlo code, MCNP, indicate that the agreement is within the statistics of the Monte Carlo calculation. This result indicates that the calculation has no glaring shortcomings. The bias is considered to be essentially zero. AECL recommended an uncertainty of 8 percent for CANDU lattices.

In the case of the ACR-700, the major departure from the current CANDU reactors is the use of light water as a coolant. Because light water is a more efficient moderator of neutrons than heavy water, the up- and down-scattering effects are expected to be more important. In addition, the fuel will be slightly enriched at BOL, and be expected to operate for approximately three times the current CANDU life. This will increase the plutonium and fission product

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

3.5.2.1.3 Moderator Temperature and Density Change

Moderator density and temperature effects cannot be separated experimentally; thus, they are considered together.

Changes in the moderator temperature change the thermal neutron "Maxwellian" distribution. Increases in temperature harden the spectrum, generally reducing the fissions associated with U-235. In the case of Pu-239, the opposite might be true because of its low-lying fission resonance. This component of the coefficient could shift from negative to positive with burnup. An increase or decrease in the moderator pressure has a two-fold effect on the moderator coefficient. First, the lower density implied by this shift will increase the diffusion coefficient, thus increasing neutron leakage and resulting in a negative contribution. Second, the reduced number density of moderating nuclei per unit volume will result in less moderation and a harder neutron spectrum entering the fuel bundle. The net result of this spectral hardening will be to enhance the resonance capture in U-238, which will decrease the reactivity. The reactions in the resonance and thermal energy ranges are important in determining this effect.

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

Comparisons of coefficients determined by WIMS-IST and measured using fresh fuel in the two above mentioned facilities agree to within a one sigma uncertainty of ±2.6 percent. In order to fill in the gaps in the measurement matrix, comparisons to Monte Carlo code predictions of burned fuel were made. In general, the WIMS-IST calculated coefficients and those determined by MCNP for mid-burnup fuel agree to within the statistical accuracy of the Monte Carlo calculation. The agreement for fresh fuel was not as good between the two calculated results. The difference is dominated by the U-235 fission rate calculated by the two codes.

The reported validation of RFSP/WIMS for conventional CANDUs was carried out against measurements taken during warmup of the Point Lepreau power reactor. The measured coefficient is relatively small (approximately 0.085 mk/EC) (see Douglas, et al., November 2001). This calculation indicated that the calculated moderator coefficient was underestimated by approximately 42 percent compared to the measured coefficient. This seemingly large discrepancy derives partly from the difficulty in making measurements of this type on an operating power reactor. Finally, because of the small magnitude of the coefficient, the

reactivity implied by this difference is 0.33 mk for a temperature change of 12 EC (54 EF), which is not very large when compared to other possible reactivity feedback contributors.

In summary, the seemingly good agreement between the measured and WIMS-IST predicted result for fresh fuel, and the disagreement with Monte Carlo calculations (mentioned above) for the same condition, is partially the result of the cross section library used in the MCNP calculation. In particular, the data for U-235 seems to be responsible for the large discrepancy. Computed results on lattice cells simulating exposed fuel, containing plutonium and Dysprosium (fission product simulant), agree much better with the MCNP calculations (within approximately 5 percent). Based on these comparisons, it was reported that WIMS-IST can be used to determine the moderator temperature and the associated density effects to within ±5 percent. In the case of the core estimate based on RFSP/WIMS, the measurement is both difficult and not as accurate as a controlled critical experiment carried out on ZED-2. Thus, the uncertainty was felt to be less than the large difference mentioned above. Based on this argument, the reported uncertainty was fixed at ±10 percent for the core estimate.

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

3.5.2.1.4 Moderator Poison and Purity Effects

The moderator is poisoned under certain operating conditions to make it less effective. This is necessary to ensure reactor shim control or reactor shutdown under certain operating conditions and accident scenarios. During normal operation, the light-water impurity in the moderator acts as a very weak poison (higher absorption than deuterium) with scattering effect and changes the neutron lifetime (more efficient moderation).

The addition of moderator poison in the form of neutron absorbers hardens the neutron energy spectrum. Two poisons are of interest in CANDU reactors. Boron (B-10) is essentially a 1/v (v is neutron velocity) absorber over the entire energy range of interest, and gadolinium (Gd-155 and Gd-157) deviates from 1/v behavior in the range of interest. The effectiveness of the poisons will be dependent on poison concentration, moderator temperature, and fuel burnup. Moderator poison content of the core has a direct and significant effect on the value of the CVR. Thus, in the case of a LOCA, the moderator poison content would affect the scenario outcome. During irradiation, the isotopic ratios in the poison will change and must be taken into account. This effect is more important for the high cross section poisons.

Light-water impurity in the heavy-water moderator will act as both a poison, since hydrogen has a higher absorption cross section than deuterium, and as a more efficient moderator. The more rapid thermalization of the high-energy neutrons and the increased absorption cross section will reduce the neutron lifetime of the core. Changes in moderator purity thus result in changes in neutron lifetime, and secondarily to changes in the CVR.

AECL carried out a series of four critical experiments on the ZED-2 facility, with varying boron amounts in the moderator. A determination of the critical height and corresponding buckling was made for each case (see Douglas, et al., November 2001). In addition, the restart at the

Point Lepreau power reactor yielded poison concentration measurements that could be used to validate the WIMS/RFSP code package.

WIMS-IST calculations of the ZED-2 experiments resulted in a reported bias of !1.2 percent, with a standard deviation of 1.0 percent. The comparison between the WIMS/RFSP calculation and the measurements carried out on the power reactor showed a larger difference. The discrepancy was generally a few percent. The difficulty of the measurement contributed to this discrepancy. Reported code-to-code comparisons between WIMS and MCNP indicate that the MCNP was slightly lower than the WIMS prediction (approximately 2.0 percent). Overall the suggested bias in WIMS-IST was set at !1.5 percent, with an uncertainty of ±2.0 percent (see Douglas, et al., November 2001).

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

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

3.5.2.1.5 Fuel Temperature Change Effects

Changes in the fuel temperature caused by power increase or decrease result in a prompt reactivity insertion. Thus, in the case of a power excursion, a prompt negative reactivity insertion resulting from Doppler broadening of capture resonances can help terminate the excursion.

Changes in fuel temperature can occur extremely rapidly in a power excursion, since they are proportional to fission rate, which takes place inside the fuel. The fuel temperature-based feedback coefficient thus acts very rapidly to mitigate over-power transients and to support reactor control in general. Three physical phenomena influence the magnitude and sign of the fuel temperature-based feedback coefficient:

(1) The well-known Doppler effect is caused by the broadening of either capture or fission cross section resonances. Broadening the resonances of fertile nuclides (U-238) increases the capture of neutrons, thereby introducing a negative contribution to the feedback coefficient. However, there is also a low-lying resonance in Pu-239, which has both capture and fission half-widths. Therefore, as the mean neutron energy changes, the ratio of capture to fission also changes in this resonance. In general, the broadening effect is more pronounced when the temperature change starts at a low temperature and shifts to a higher temperature, than if the starting temperature is high and moves to an even higher temperature.

(2) Thermal absorption cross sections for U-235, Pu-239, Pu-240, and Pu-241 show a non-1/v behavior in the thermal energy range. Any change in the thermal energy spectrum will change the relative magnitude of the absorption reaction rates among these nuclides. Furthermore, for U-235 and Pu-239, the capture-to-fission ratio will change as a function of the thermal energy spectrum. Because CANDU reactors are thermal reactors, even small changes in the thermal energy spectrum can lead to significant changes in the respective reaction rates.

(3) A significant amount of oxygen exists in the fuel, since only oxide fuels are being considered. Thermal neutrons that are scattered off the oxygen nuclei will in general experience an increase in energy, since oxygen bound in the fuel is at a temperature characteristic of the fuel that is much higher than that characteristic of the moderator or the coolant. As the fuel rises in temperature, the effect of the oxygen scattering will be to harden the thermal neutron energy spectrum. This change in the thermal energy spectrum will change the reaction rates of the non-1/v absorbers as described above.

AECL carried out a series of substitution critical experiments on the ZED-2 facility to measure the fuel temperature coefficient (see Douglas, et al., November 2001). These experiments consisted of reference fuel and test fuel channels. In each case, the critical reactor height and corresponding buckling were determined. Thus, the slope of the curve of buckling as a function of fuel temperature results in the feedback coefficient in buckling units. The experimental fuel channels consisted of CANDU fuel that was heated by carbon dioxide located in the volume normally filled with coolant. The temperature of the carbon dioxide that heated the fuel was varied over the range 25 EC (77 EF) to 300 EC (572 EF). Thus, the fuel was at a uniform temperature and the temperature distribution in the fuel did not have the characteristic shape with a maximum in the center. The range is low, considering the maximum fuel temperature is several hundred degrees above the maximum experimental temperature. Despite these shortcomings, these experiments did yield useful data. Two sets of experiments were carried out on 37-element bundles containing FNU fuel and MOX with Dysprosium fuel. These experiments covered a range of simulated exposure from fresh fuel to fuel that has experienced partial burnup. In addition, two sets of measurements were carried out on 43- element fuel bundles containing FNU fuel and low void reactivity fuel (LVRF).

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

During normal operation of a CANDU-type reactor, each bundle is exposed to a unique set of conditions. These conditions include power, temperature, and power shape. In addition, given the unique history of each bundle, the plutonium inventory varies among them. This changes the thermo-physical properties (conductivity, specific heat, etc.). All these characteristics play an important role in the determination of the overall reactor feedback coefficient. Thus, it is very important to validate the analysis code against a set of experiments that encompasses all

the above effects, including a prototypic temperature range. The quality and vintage of the nuclear data play an important role in the ability to determine the appropriate feedback coefficient. This also involves up- and down-scattering off the oxygen bound in the oxide fuel matrix.

Because the ACR-700 will represent a departure in fuel and code design from currently operating CANDU fuel, a very comprehensive set of experiments needs to be carried out to validate the methods and fundamental nuclear data. The increased burnup will generate a larger inventory of fission products, which will also affect the fuel temperature feedback coefficient. It might be necessary to reevaluate fundamental cross-section data for selected nuclides. In addition, an explicit scattering kernel for UO_2 and PuO_2 might have to be developed to accurately account for the potential up-scattering caused by the oxygen in fuel elements containing MOX fuel. Such data already exist for UO_2 and may only need augmentation.

3.5.2.1.6 Coolant Purity Change

The coolant of a current-generation CANDU reactor can be downgraded by the inadvertent addition of light water. The presence of light water in the coolant increases the parasitic absorption of neutrons. It is also a more efficient moderator of high-energy neutrons. The overall effect of degraded coolant on the overall reactivity of the core is small. Its major effect would be on the CVR.

AECL carried out a series of critical experiments on the ZED-2 facility, using 37-element bundles of FNU and MOX fuel. The critical height of the core and the corresponding buckling were measured. In the case of the current CANDU reactors, the purity coefficient is approximately 33.0 mk/wt percent of light water. WIMS-IST calculations of the coolant purity reactivity effect are reported to underestimate the measured value by 12 percent. This agreement is considered acceptable because it is better than 1.5 σ (see Douglas, et al., November 2001).

The ACR-700 will not be cooled by heavy water, but will use light water. The question of heavy-water coolant purity is thus not an issue in this case.

3.5.2.1.7 Flux and Power Determinations

The total power generated in a reactor is a sum of the prompt power caused by fission and capture events and delayed power resulting from delayed neutron emission. Prompt power comprises approximately 93 percent of the total power. Under normal operating conditions, prompt power and delayed power have the same spatial distribution. However, in the event of a fast transient event, they could have different spatial distributions and should be treated separately. The global flux shape is a function of the core geometry; lattice arrangement; distribution of fissile, fertile, and fission product material; control device positions; coolant and moderator temperature; and density, etc. In all CANDU reactors, in-core detectors monitor the spatial flux shape. Channel power is monitored by determining the temperature increases across the channel. Some channels are fully instrumented. A direct measure of the channel power is possible in those cases. Bundle power distribution is not directly measured.

The calculated prediction of reactor flux and power is carried out using the full suite of IST-based codes. The few group parameters are determined using WIMS-IST and DRAGON-IST, which are fed into RFSP-IST for the final reactor flux and power determination. Detailed intercell flux and power distributions can be gleaned from the WIMS-IST or DRAGON-IST calculations. The linkage is illustrated below.

A comparison of predictions with measurements obtained from operating power reactors validated the above code package. These measurements are not as clean and controlled as those made on the ZED-2 or ZEEP critical facilities. They are the only data available for some of the phenomena (fuel management and burnup predictions) (see Douglas, et al., November 2001). Global flux measurements are available from in-core vanadium detectors (102 detectors located in the core) and traveling flux detectors. The traveling detectors are based on small fission chambers that can be moved through the core in the moderator volume. The burnup measurements were determined by following 1 year of operation of the Point Lepreau power reactor. In addition, code-to-code comparisons were reported in which Monte Carlo methods provided the reference solutions.

A reported comparison between the averaged computed flux and the flux measured using the vanadium detectors shows a typical root-mean-square agreement of 3 percent, after the values have been normalized to a common value. On average, the calculated flux is underestimated in the inner core region and overestimated in the outer core region. Global flux tilts (side-to-side, end-to-end, and top-to-bottom) are similarly offset. Comparisons to Monte Carlo calculations carried out using MCNP were also made. In general, the reported agreement between the RFSP and MCNP was within 2.6 percent for channel power and 3.5 percent for bundle power. The difference in channel power ranged from !5 percent to +10 percent. After normalizing to the same power, RFSP overestimated the power on the periphery and underestimated the power in the center. The underestimate in the central region was approximately 1.7 percent, which is consistent with vanadium detector measurements. On the periphery, however, the reported differences were much larger, up to 42 percent in the low-power channels in the core-reflector interface region. Diffusion theory is less accurate in this region and use of that theory could be the reason for the discrepancy.

The time-averaged burnup predictions were validated by following the operation of an operating power reactor. A total of 119 RFSP production runs were made to follow the history of the reactor. During 1 year, a total of 5486 fuel bundles were replaced in the reactor core. The average, minimum, maximum, and standard deviation of each vanadium detector flux measurement were determined for the year of operation. These measurements were then compared to simulated vanadium fluxes from time-averaged calculations. In addition, the exit burnup was predicted and compared to measured exit burnup obtained from the reactor. This comparison yielded the following reported observations:

- The calculated discharge burnup (175.7 MWh/kg(U)) compares well with the measured value (177.8 MWh/kg(U)).

- The time-averaged bundle feed rate was predicted to be 15.345 bundles/full power day (FPD), compared to the actual feed rate of 15.24 bundles/FPD.

- The average difference between the time-averaged vanadium detector response was about 0.18 percent with a standard deviation of 3.1 percent. However, the maximum

positive and negative difference between measured and calculated values were +11.09 percent and !11.74 percent, respectively.

At the lattice level, the reported comparisons between WIMS-IST and experiments carried out on ZED-2 indicate that WIMS-IST is able to predict the flux to within an uncertainty of 1 percent. Whole core calculations carried with MCNP were reported to indicate that the azimuthal variation of the outer elements can be off by approximately 2.5 percent.

The ACR-700 is a smaller reactor than the current CANDU configuration; it has a tighter pitch, less moderator, and a completely new fuel concept. In view of the significant departure from standard CANDU configurations, it is important to revisit all the assumptions made in creating the code/library package during the certification review. In particular, the following might be important:

- The WIMS-IST will have to be validated for the new cell arrangement, as outlined in Sections 3.2.1.1 and 3.3.2.1.8 of this report.

- The RFSP code will have to be validated for acceptable accuracy in those regions where diffusion theory is known to break down. Because the core is smaller, the reflector effects caused by spatial flux gradients may be more significant.

- Spectral changes around the periphery of the core may be more significant, particularly since this core is intended to have a negative value for the CVR. This effect might necessitate rethinking the number of groups needed for RFSP.

3.5.2.1.8 Ability to Treat Checkerboard Coolant Voiding

In the event of a large coolant header break, the coolant in the channels connected to the broken header will void first. In the conventional CANDU designs, the voiding pattern is of secondary importance because the reactivity change is always positive and roughly linear with partial voiding; thus, only the rate of reactivity addition is controlled by the voiding pattern and timing. The most commonly accepted voiding pattern implies a progression, which starts as a fully cooled core (no voiding) to one in which every other fuel channel is voided and finally to a fully voided situation. This progression implies that during the voiding scenario, the resulting core configuration has a “checkerboard” void pattern. In order to analyze the core response to this scenario, it is necessary to recognize the changing value of the CVR with this void pattern.

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

The ACR-700 core lattice has a significantly smaller channel-to-channel pitch and less moderator than the current CANDU designs, the void coefficient is intended to be negative under all operating conditions, and the reactor seems to experience stronger neutron energy spectrum shifts on voiding than the current CANDU design. In view of these differences, it may be necessary to generate few group cross sections for RFSP that are not only dependent on all the variables that are already included in the WIMS input (burnup, position in fuel channel, and fuel channel in core, temperature, and coolant density, etc.), but that are also dependent on the voiding pattern. Thus, it might be necessary to carry out a multi-channel WIMS-type analysis to determine the appropriate cross sections. This would imply that WIMS would have to handle a checkerboard input file recognizing different coolant densities in the neighboring fuel channels, assuming a simple checkerboard voiding pattern. More complicated voiding patterns (e.g., stratified partial voiding) may be identified based on ACR-700 system fluid dynamics calculations.

3.5.3 CVR as an Example of Exercising the Code Package

The accurate prediction of coolant void effects represents a significant challenge to the code package used in the design and analysis of CANDU-type reactors. Voiding the coolant results in shifts in the neutron energy spectrum, which changes the fission-to-capture ratios among the fissile nuclides, resonance reactions and fast fission in the fertile nuclides, and changes in the number of neutrons per fission. In addition, neutron leakage changes with voiding. The effect of leakage will be dependent on the core design. The following sections outline the contributors to the CVR and discuss the code predictions and their validation for the current CANDU design (see Douglas, et al., November 2001 and Douglas, "Void Reactivity Error Assessment for CANDU Reactors," COG-01-030, June 2001). The changes in CVR contributors for the ACR-700 will then be outlined and contrasted with those of the current CANDU (see Altiparmakov, January 2002). Finally, implications for the codes will be discussed.

3.5.3.1 Phenomena Contributing to the CVR in Current CANDUs

A fundamental difference between existing U.S. LWRs and CANDU-type reactors is that, in an existing U.S. reactor design, the coolant and moderator are the same, while in a CANDU-type reactor, they are separate. Thus, in an existing U.S. design, the coolant and moderator have the same temperature, density, pressure, and atomic/nuclear makeup. What happens to the coolant also happens to the moderator. In the case of a CANDU reactor, these two fluids are separated by the PT and calandria tube walls, thus they have different temperatures, pressures, densities, and atomic/nuclear makeup. What happens to the moderator does not necessarily happen to the coolant and vice versa. The thermal neutron energy spectrum in either the coolant or the moderator can be approximated by a Maxwellian distribution at a characteristic temperature close to the temperature of the moderator or coolant. Thus, in the case of a CANDU reactor, these spectra will be different, while in an existing U.S. design, they are the same. This distinction plays an important role in the difference in the CVR for the two reactor types.

Among the most important reactions in a reactor are the fission events taking place in the fissile and fertile materials (U-235, Pu-239, Pu-241, U-238, and Pu-240). During a loss of coolant event, the thermal spectrum shifts from the center of the fuel bundle to the outer edge. This shift changes the capture-to-fission ratio for the individual isotopes and among the isotopes.

Removal of the coolant will have the effect of “softening” the spectrum entering the PT immediately next to the moderator, so that the fuel elements in the outer rings of the bundle will be exposed to lower energy neutrons than before voiding the coolant. This softening occurs because the in-leaking thermal neutrons are characterized by the moderator temperature (relatively cold compared to the coolant). The higher temperature coolant normally “hardens” the incoming spectrum by up-scattering the neutrons in this zone, thus increasing the average neutron energy in the fuel bundle. This spectral softening changes the competition between U-235 and Pu-239 absorption rate (capture + fission) because they are both non-1/v absorbers, but behave differently (Pu-239 has a low-energy resonance). In addition, for each nucleus, the capture/fission ratio has a different dependence on energy. The neutron spectrum leaking in from the moderator will also have fewer resonance range neutrons. Upon voiding the channel, the resonance capture rate in the fertile and fissile nuclides should decrease, which will be a positive effect on the void coefficient since there is a lot more fertile material than fissile material. Furthermore, the lack of coolant to moderate the fast neutrons generated in the fuel elements within the bundle will lead to an increase in the number of fast neutrons in the bundle. More fast neutrons will enhance the fast fission effect in the fertile nuclides (U-238, Pu-240, etc.). This is particularly true of U-238, since it is the dominant heavy nucleus. Finally, the leakage of neutrons out of the reactor is enhanced by removal of the coolant because of the unimpeded flight paths down the fuel channel. This effect is dependent on the relative size of the reactor and the fuel channels. Because it tends to decrease the neutron population, it is a negative effect.

A summary of these effects is shown in Table 3-2, which depicts the magnitude of the changes that occur in the various reactions as a result of voiding. The change is shown in mk. It is representative of a typical midlife core which contains MOX fuel. Fission products are simulated by suitable addition of Dysprosium.

Table 3-2 Reported Changes in Reaction Rate Brought about by Coolant Voiding in a Typical CANDU Core

Phenomena	Relative change following voiding (mk)
U-235 (fission gain + capture loss)	+ 3.3
Pu-239 (fission gain + capture loss)	+ 3.8
U-238 (fission gain + capture loss)	+ 4.3
U-238 (resonance capture change)	+ 7.6
Pu-240 (resonance capture change)	+ 0.6
Loss of coolant (capture change)	+ 1.1
Fission product (absorption)	!0.9
Absorption in structure	+ 2.0
Leakage	!5.0
Total	+ 26.8

The above table is included for illustrative purposes only. It shows that the contributions to the overall CVR are primarily positive, with the largest contribution coming from U-238 (fission gain and resonance capture reduction). The largest negative contribution comes from neutron leakage. The fissile nuclides both contribute positively to the overall effect. In summary, it is concluded that an accurate prediction of the CVR requires an accurate determination of the shift in the neutron spectrum in energy, space, and angle.

3.5.3.2 Experimental Validation of Current CVR Estimates

In order to validate the predictions of the void reactivity in conventional CANDUs , AECL carried out a series of critical experiments on the ZED-2 facility. Basically, ZED-2 is a zero-power reactor facility that consists of a lattice of aluminum alloy tubes that contain fuel elements and coolant (light water, heavy water, or void) immersed in a large tank of heavy water, which acts as the moderator. The depth of the heavy-water moderator defines the height of the reactor core. The pitch and number of fuel-element-containing tubes sets the radial dimension. Experiments were carried out on cores with 28, 37, and 43 elements. In all cases considered below, the lattice had a hexagonal arrangement with a pitch of 31 cm. Measurements were carried out several times to determine an error and a standard deviation corresponding to each configuration.

Using WIMS-IST, AECL calculated the corresponding values of the CVR for the above experimental configurations. In all cases, the calculated values overpredicted the magnitude of the measured CVR in the ZED-2 experiments. The following table summarizes the results.

Table 3-3 Reported WIMS-IST Overprediction of CVR in ZED-2 Experiments for Conventional CANDUs

Number of elements (fuel type)*	Void reactivity discrepancy (mk)
28 (FNU)	+ 0.78
37 (FNU)	+ 1.89
37 (MOX)	+ 1.68
43 (FNU)	+ 1.83

* FNU = Fresh natural uranium
MOX = Mixed oxide

The trend of overprediction by WIMS-IST continues for other fuel arrangements (pitch and number of elements per tube) and types.

In order to check the validity of input choices implied by WIMS-IST, AECL carried out a series of calculations in which more accurate/representative choices in WIMS input were made. These calculations indicate the bias in the magnitude of the CVR as determined by WIMS-IST relative to more accurate determinations made using WIMS. Table 3-4 summarizes the results of these calculations for the most significant contributors of the three FNU cases shown above.

Table 3-4 Reported Effect of Input Choices in WIMS-IST on Magnitude of the CVR for Conventional CANDUs

Modeling approximation (WIMS to WIMS-IST)	28 element (mk)	37 element (mk)	43 element (mk)
White boundary Square to circular (2D)	+ 0.06	! 0.07	! 0.05
Fine hybrid model 2D resonance treatment	+ 0.06	+ 0.06	+ 0.06
Square boundary Reflecting to white (2D)	! 0.22	! 0.22	! 0.22
Fine hybrid model *newres* treatment	+ 1.09	+ 0.33	+ 0.38
Standard model coarse mesh	! 0.20	! 0.22	! 0.16
Imprecise P_f -> P_f Angular integration	+ 0.11	+ 0.11	+ 0.11
Group structure 89 to 33 groups	+ 0.13	+ 0.13	+ 0.13
WIMS-IST (n, 2n) treatment	+ 0.15	+ 0.15	+ 0.15
Total	+ 1.20	+ 0.32	+ 0.48

The above results indicate that resonance treatment, number of groups, spatial treatment, and boundary condition treatment are the most important model-dependent contributors to the bias in determining the CVR. In particular, the calculational algorithm implied by the *newres* treatment of resonance phenomena appears to be the major contributor to the total bias. This deviation results in the largest positive contribution to the total bias, while the treatment of boundary conditions and mesh sizes are the most significant negative contributors to the bias.

In addition to the above comparisons of WIMS-IST to the experimental data and the WIMS intracode model comparisons, WIMS-IST results can be compared to those obtained using a different code. The intercode comparisons are made to Monte Carlo-based calculations carried out using the MCNP code. This code has the advantage that it can model extremely complicated geometrical arrangements in three dimensions, thereby reducing or eliminating the need for most spatial approximations. Furthermore, the neutron cross section energy treatment in MCNP is carried out in a continuous manner, eliminating the need to generate a group structure and the associated average cross sections. Potentially, this code is only limited by the completeness and accuracy of the geometry and material input descriptions, the accuracy of the fundamental nuclear data, and the statistical accuracy to which the calculation is carried out (computer time limits). These studies fall into two categories—those that involve neutron leakage and those that do not (infinite lattice).

Table 3-5 Reported Code-to-Code Comparison of Infinite Lattice CVR for Conventional CANDUs as a Function of Fuel Temperature and Burnup

Element configuration and temperature (C)	Burnup				Code-to-Code Delta
	0 GWd/T	4 GWd/T	8 GWd/T	12 GWd/T	
28 element, 327 EC (621 EF)	19.03	16.93	16.72	16.54	MCNP
	19.35	17.36	17.35	17.33	WIMS
	0.32	0.44	0.63	0.79	Delta
37 element, 327 EC (321 EF)	19.31	16.70	16.76	16.89	MCNP
	18.98	17.11	17.20	17.26	WIMS
	!0.32	0.40	0.44	0.37	Delta
43 element, 327 EC (621 EF)	20.30	18.94	18.20	18.75	MCNP
	20.28	18.30	18.45	18.56	WIMS
	!0.02	!0.64	0.25	!0.19	Delta
28 element, 627 EC (1161 EF)	19.13	16.69	16.79	16.82	MCNP
	19.61	17.60	17.72	17.80	WIMS
	0.48	0.91	0.93	0.98	Delta
37 element, 627 EC (1161 EF)	19.23	17.24	17.47	17.41	MCNP
	19.22	17.33	17.58	17.74	WIMS
	!0.01	0.09	0.10	0.33	Delta
43 element, 627 EC (1161 EF)	20.46	18.52	19.50	19.07	MCNP
	20.54	18.55	18.87	19.10	WIMS
	0.08	0.03	!0.63	0.03	Delta

The three FNU cases listed above were analyzed using both WIMS-IST and MCNP. An extensive comparison was reported in which the burnup, fuel temperature, purity of coolant and moderator, and moderator boron content were varied. Several of these parameters were varied individually and in combinations. This stage of the analysis concentrated on the variation of fuel temperature and burnup, since these are felt to be the most significant variables (fuel comes in fresh, leaves burnt, and the temperature changes in the fuel channel). Table 3-5 shows the variation of the CVR computed by the two codes for changes in temperature and burnup.

The above results indicate the following trends for the magnitude of the CVR in conventional CANDUs:

- The CVR increases with fuel temperature at all burnup levels evaluated.
- The CVR decreases with increasing burnup at all fuel temperatures evaluated. The trend is sharper at and near BOL than later in life.

- The CVR magnitude increases slowly with increasing number of fuel elements, regardless of temperature and burnup.

In general, the magnitude of the CVR in the modeled conventional CANDU lattices is seen to be in the range of 15 mk–20 mk over the entirety of parameters investigated.

A reported comparison between the CVR, determined by WIMS-IST and MCNP, shows the following overall trends:

- The largest discrepancy occurs for the 28-element bundle. In all cases, the MCNP determined values are larger.

- The discrepancy for the other two bundles is smaller and changes sign. For the case with 37 elements, the discrepancy generally starts negative for zero burnup, becomes positive, and reaches a maximum at discharge burnup. The 43-element bundle does not follow this pattern. In this case, the discrepancy changes sign at mid-burnup.

Finally, it is possible for the two codes to estimate the change in reaction rate in the channel upon loss of coolant. Table 3-6 shows the computed change in reaction rate of the major isotopes at a temperature of 900 K for BOL fuel and for discharge fuel. The discharge fuel includes plutonium isotopes and fission products. The fission products are simulated by the addition of an appropriate amount of Dysprosium. The above comparison also indicates that a slight difference exists in the reaction rates, as determined by the two code systems. A slight increase in the case of U-235 fission can be seen. There are slight decreases for Pu-239 fission, U-238 capture, Pu-239 capture, and Dysprosium content. These differences appear to be quite modest.

Table 3-6 Calculated Changes in Reaction Rate upon Voiding in a Conventional CANDU Lattice (Voided Minus Cooled in mk)

Nuclide	WIMS Fresh fuel	MCNP Fresh fuel	WIMS Discharge fuel	MCNP Discharge fuel
U-235 (f)	6.0	6.0	3.4	3.5
U-238 (f)	2.1	2.1	2.0	2.0
Pu-239 (f)	~	~	!1.4	!1.7
U-235	1.0	0.8	0.6	0.6
U-238	!6.0	!5.9	!1.3	!0.9
Pu-239	~	~	!3.4	!3.5
H-1	!0.6	!0.7	!0.5	!0.5
H-2	!0.4	!0.3	!0.3	!0.2
Dy (fission product)	~	~	2.4	2.3

AECL also carried out a study of the effect of axial leakage on the magnitude of the CVR, as determined by the IST package of codes compared to a full-core Monte Carlo simulation using a large number of calculations. The following paragraphs discus the highlights of this work. This study considered calculations involving cores of various lengths, voiding patterns (whole core vs. checkerboard), and fuel compositions. Table 3-7 presents a selection of results for a

core consisting of 37-element fuel bundles, using either FNU, MOX, or SEU fuel. The results assume the core is at operating temperature and the whole core is voided.

Table 3-7 Comparison of Magnitude of the CVR from WIMS/RFSP-IST and MCNP Whole-Core Voiding in a Conventional CANDU Lattice

Configuration	CVR WIMS-Based (mk)	CVR MCNP-Based (mk)	Delta CVR (mk)
FNU	19.22	18.62	0.60
MOX	11.61	11.13	0.48
SEU	19.58	18.68	0.90

The WIMS/RFSP-IST approach slightly overpredicts the magnitude of the CVR relative to the Monte Carlo calculations. The reported discrepancy includes not only the effect of axial streaming, but all other transport theory-related discrepancies. When compared to the intermediate burnup steps for a 37-element infinite lattice at 900 K, the discrepancies shown in Table 3-7 are slightly larger. The finite nature of the reactor contributes to this difference.

To determine the true value of the CVR for a reactor, the bias and uncertainty must be estimated and included. Thus, AECL commonly uses the following relationship to include these effects (see Beuthe and Hanna, "CATHENA MOD-3.5c/Rev.0 Theoretical Manual," COG-00-008, November 2000):

WIMS-IST/RFSP-IST predicted value = True value + Bias ± Uncertainty

Reported comparisons between the CVR values determine by WIMS-IST and experimentally determined values (see Table 3-3) indicate that, for the ZED-2 core, based on a 37-element bundle, the bias is +1.6 mk for MOX fuel and +1.9 mk for FNU fuel. In addition, as the core temperature increases to operating temperatures, the FNU case bias decreases to +1.6 mk. Based on these results, AECL decided to settle on +1.6 mk as the bias for the 37-element bundle case. The uncertainty is based on the uncertainty associated with the experimentally determined CVR. Repeating the measurements several times and estimating the uncertainty from the spread in the measured results determined the uncertainty. Using this method, AECL found that the uncertainty was ±1.1 mk. Thus, for the 37-element bundle, the following applies:

Predicted void worth = True void worth + 1.6 ± 1.1

The behavior of the 43-element bundle fuel is very similar to that of the 37-element bundle fuel. The results shown in Table 3-3 indicate that the bias is slightly higher. Because there is no experimental uncertainty for this case (not enough experiments were repeated), the same value as that used for the 37-element bundle was recommended. Thus, the following applies for the 43-element bundle fuel:

Predicted void worth = True void worth + 1.7 ± 1.1

For the 28-element bundle fuel, the bias is much higher and seemingly more complex. The largest computation-based contributor to the bias is the treatment of resonances, which seemed to be larger in this case than in the other two cases. Therefore, bias was set at

2.0 mk. The uncertainty from the experimental results is very close to that determined for the 37-element bundle fuel and will thus be set at 1.1 mk. In this case, the following applies:

$$\text{Predicted void worth} = \text{True void worth} + 2.0 \pm 1.1$$

AECL concluded that the bias in predicting the magnitude of the CVR has three major contributors. Approximately 66 percent of the bias is attributed to fundamental nuclear data, as highlighted by comparisons between full-core MCNP analyses of ZED-2 experiments. Approximately 25 percent is attributable to modeling choices made in carrying out the WIMS-IST calculations. The remainder of the bias is attributed to experimental errors.

3.5.3.3 Projections and Implications of the CVR for ACR-700 Type Designs

The next generation CANDU reactor is being developed with the purpose of reducing capital costs, improving reactor safety, and decreasing the spent fuel stream (see Altiparmakov, January 2002). The reactor safety improvement goal is to achieve a negative CVR. In order to achieve the negative CVR, the fuel channel pitch will be reduced, the calandria tube diameter will be increased, the fuel will be slightly enriched (SEU), burnable poison (Dy) will be added to the central fuel element, and light water will be used as the coolant. AECL conducted a study in which the fuel composition was varied in a systematic manner to define the most likely combination that satisfies the AECL goal for safety. Table 3-8 summarizes the lattice parameters. The table shows that the pitch is reduced from the 28.75 cm generally used in CANDU reactors, the number of fuel elements is increased to 43 per bundle, and the fuel pin dimension varies between the inner 8 and the outer 35 pins. The burnable absorber will be confined to the central pin.

Table 3-8 Lattice Parameters Used in ACR Design Study by AECL

Parameter	Dimension (cm)
Lattice pitch	22
Calandria tube	
Inner radius	7.50
Outer radius	7.80
Pressure tube	
Inner radius	5.17
Outer radius	5.77
Fuel bundle	
Rod array radius	
Second ring	1.73
Third ring	3.075
Fourth ring	4.384
Clad outer radii	
Eight inner fuel pins	0.675
Thirty-five outer pins	0.575

This study investigated four different fuel types, which are referred to as lattice cell types 1–4, and define the uranium and Dysprosium enrichment as a function of fuel pin ring within the fuel bundle. Table 3-9 gives the range over which the enrichment was varied for the different cell

types. Each lattice cell type has four fuel types, defined by the central pin and the three surrounding fuel pin rings. Fuel 1 corresponds to the central pin, and fuel 4 corresponds to the outer fuel pin ring.

Lattice cell 1 contains no burnable poison and a maximum enrichment of 2.70 percent. Lattice cell 2 has a maximum amount of burnable poison (5 percent) and the possibility of zoned enrichment. Lattice cell 3 uses NU in the central eight pins and a maximum enrichment of 2.70 percent in the outer pins. Lattice cell 4 uses a smaller amount of burnable poison (2.5 percent), NU in the second ring, and enriched uranium in the outer rings. The CVR will determined as a function of burnup for a large variety of proposed fuel bundle designs. The projected burnup is 28 MWd/kgU, which is approximately double the current CANDU burnup. The WIMS-IST code estimates the burnup. AECL conducted a study to determine the most appropriate modeling choices in the WIMS input file.

Table 3-9 Fuel Pin Composition by Lattice Cell Type

	Lattice cell 1	Enrichment and Burnable Poison (%)
Fuel 1 and 2	U-235/U	0.95–2.70
Fuel 3 and 4	U-235/U	0.95–2.70
	Lattice cell 2	
Fuel 1	U-235/U Dy/(U+Dy)	0.95–2.70 5.0
Fuel 2	U-235/U	1.30–2.70
Fuel 3 and 4	U-235/U	0.95–2.70
	Lattice cell 3	
Fuel 1 and 2	NU	100
Fuel 3 and 4	U-235/U	0.95–2.70
	Lattice cell 4	
Fuel 1	NU Dy/(NU+Dy)	97.5 2.5
Fuel 2	NU	100
Fuel 3 and 4	U-235/U	0.95–2.70

In order to summarize this study, lattice cell 2 was used as an example, primarily because it has the best chance of meeting the AECL safety goal of a negative CVR over the entire burnup range. For lattice cell 2, the variation of the CVR with burnup is a function of initial uranium enrichment. The CVR is initially positive for low burnups and low enrichment (1.3 –2.0 percent). For the lowest enrichment considered and at zero burnup (i.e., startup), the CVR is estimated to be +10.0 mk and decreases to zero following a burnup of approximately 7 MWd/kgU. As the enrichment is increased to 2.7 percent, the maximum assumed in the study, the initial value of the CVR decreases and eventually reaches a negative value (approximately -2.0 mk). Assuming the maximum enrichment (2.7 percent), the value of the CVR initially increases slightly with burnup and then starts to decrease. The CVR reaches a minimum at approximately 7 MWd/kgU (approximately !4.0 mk). It then increases slowly to a flat maximum at approximately 23 MWd/kgU (approximately !1.0 mk). This is followed by a steady decrease to the end of the burnup period (final value is approximately !3.0 mk). The undulating nature of the variation of the CVR with burnup results from competing contributions of opposite sign that

vary with burnup in different ways. Several of these contributions are significantly larger than the final value of the CVR. Therefore, their accurate determination is important, not only to the determination of the CVR magnitude, but also its sign. Table 3-10 lists the contributors to the CVR by material and isotope.

The isotopes listed are the most significant. Their contribution due to leakage is determined by adjusting the buckling to achieve criticality. Thus, the leakage effect is caused by global neutron leakage from the core of the reactor. AECL results presented in Table 3-10 indicate that leakage is the main contributor to the value of the CVR. Its contribution starts off negative. Then its value decreases until, at the end of the burnup period, its contribution is slightly positive. The coolant and the outer ring of elements are the next two significant contributors. These two contributors have opposite signs and nearly cancel. The Dysprosium in the central fuel pin is a significant negative contributor, which does not vary significantly with burnup. The fissile isotopes U-235 and Pu-239 are of similar magnitude, but of opposite sign. U-235 starts off as a negative contributor, but then changes to a positive contributor. However, Pu-239 is always negative and, as it builds, it essentially cancels the U-235 contribution. The remaining plutonium isotopes are small contributors, which tend to cancel each other at the end of life. Two fission products (samarium-149 (Sm-149) and Xe-135) are included, but they have relatively minor contribution. Oxygen is also included, but its contribution is also small. U-238 is a significant negative contributor to the CVR, and its contribution does not vary much with burnup.

Table 3-10 Reported Components of the Void Reactivity Lattice Cell 2 (2.7 Percent Enriched Uranium)

Void component	Burnup (MWd/kgU)			
Material	0	7.04	14.08	21.12
Fuel 1	!6.375	!6.739	!7.085	!7.368
Fuel 2	6.066	6.224	7.403	7.984
Fuel 3	1.652	1.288	1.316	0.475
Fuel 4	!30.055	!30.204	!31.155	!31.737
Clad	!1.200	!1.248	!1.274	!1.300
Coolant	21.687	23.532	26.875	31.366
Pressure tube	!1.676	!1.485	!1.130	!0.668
Calandria tube	!0.808	!0.691	!0.460	!0.162
Leakage	!97.300	!56.477	!26.338	3.220
Selected isotopes				
U-235	!11.090	0.638	7.634	11.951
U-236	0	!0.379	!0.555	!0.634
U-238	!9.878	!10.922	!11.481	!12.212
Pu-239	0	!7.767	!10.837	!11.599
Pu-240	0	!1.404	0.345	1.871
Pu-241	0	!0.462	!1.722	!2.977
Sm-149	0	0.407	0.357	0.293
Xe-135	0	0.955	0.385	!0.136
O-16	!0.329	!0.286	!0.250	!0.212
Dy	!7.415	!7.819	!8.433	!9.012

Table 3-11 contrasts the most significant contributors to the CVR for the proposed ACR-700 and the current CANDU designs, as discussed above.

Table 3-11 Comparison of Phenomena Contributing to CVR in ACR-700 and CANDU

ACR-700 Design	Current CANDU
Leakage is a major contributor, starting negative and changing sign to end as a positive contributor.	Fuel region (U-238) is the major contributor to CVR (positive in this case). Other contributors in fuel are lower by approximately an order of magnitude. Leakage contribution is also small and always negative.
Fissile isotope contributors (U-235 and Pu-239) are relatively large, but of opposite sign, thus essentially canceling. U-238 is a negative contributor throughout the cycle.	Fissile isotopes (U-235, and Pu isotopes) are a negative contributor, but relatively small.
Dy is a negative contributor and changes moderately with burnup.	Not applicable

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

3.5.4 Staff Audit Calculations—Reactor Physics

As part of its independent review of the ACR-700 reactor physics codes, the NRC staff requested copies of the WIMS, DRAGON, and RFSP codes used by AECL to model the fuel, reactor physics, and core design of the ACR-700. AECL submitted WIMS-AECL 2-5d, DRAGON Version 3.04 Revision Bc, and RFSP-IST Version 3-03 in executable formats for the staff to use to become familiar with the capabilities and operation of the codes. The staff installed these codes on NRC computers and executed test case inputs to confirm the proper operation of the codes. The staff verified that the test case outputs obtained from the NRC computers accurately reflected AECL test case outputs provided with the codes. Additionally, the staff reviewed the input cases to familiarize itself with the formatting and structure of the inputs into each of the codes. The staff identified various input parameters specific to the ACR-700 fuel design, such as heavy- and light-water purity and density, fuel enrichments, and other key reactor physics parameters.

The staff performed parametric studies of the variations in key reactor physics parameters, such as fuel enrichment, Dysprosium content, reactor coolant voiding, and burnup. The staff varied each of these parameters over limited ranges to study the effects each would have on

key reactor physics parameters, such as the infinite and effective multiplication factors, flux profiles, and buckling factors. The staff reviewed each change to determine whether the modifications made to the input would result in expected changes to the output. For example, the staff changed the voiding of the coolant within the PT to observe the effects this would have on the effective multiplication factor. The staff observed that increased voiding resulted in decreased values of the effective multiplication factor. This would support AECL statements that the ACR-700 is designed with a negative full-core coolant void reactivity. However, it is important to note that the staff performed extremely simplistic parametric studies which were only intended to observe rough trends on a lattice level and cannot be taken as confirmation of a negative coolant void coefficient in the ACR-700 reactor. Additionally, the input cases supplied by AECL and used in the staff's parametric studies, although well developed, may not represent the final design of the ACR-700 fuel or core.

During the audit calculation process, the staff determined that it did not have the latest version of each of the reactor physics codes. For example, the staff had received a copy of WIMS-AECL 2-5d; however, AECL uses a later version, WIMS-AECL 2-6a, which contains significant changes from the previous version essential to proper modeling of the ACR-700 lattice cells. Additionally, the staff has not received copies of other intermediate codes, T16MAC and PROC16, used to format the output cases from WIMS into proper input formats for DRAGON. The staff requested that AECL provide the latest version of each of these codes at the earliest opportunity. The staff also requested that AECL provide the latest fuel and core models to the staff so that it might perform further parametric studies to understand the governing fuel design and operational parameters that affect the reactor physics in the ACR-700. In addition to further parametric studies focused on fuel design parameters, the staff will perform sensitivity studies to confirm that the ACR-700 CVR will be negative. As part of these studies, the staff will perform analyses to confirm a negative CVR under a variety of accident and transient conditions, such as a LOCA resulting in checkerboard voiding of the coolant channels in the ACR-700 core. Finally, the staff will investigate the generation of cross-section libraries and Dysprosium representation in WIMS, as well as the "smearing" or homogenization of poisons in WIMS and RFSP.

3.5.5 Reactor Physics Codes Assessment Summary

Based on the information provided by AECL during the preapplication review, and pending resolution of the issues discussed above, the staff believes WIMS-IST, DRAGON-IST, and RFSP-IST, do not have any fundamental errors or shortcomings that would preclude certification of the ACR-700 design. In addition, these codes seem to have sufficiently flexible algorithms and input options to create versions suitable for analyzing the ACR-700 reactor core. The latter is important; a detailed core analysis, would involve several thousand WIMS-IST calculations and several hundred RFSP-IST calculations. It is therefore necessary to have codes that run very quickly and are validated (bias and uncertainty understood) for the reactor concept being analyzed. The currently used codes have been validated against many critical experiments (ZED-2 facility) and Monte Carlo calculations (MCNP) representative of the current CANDU reactor design.

The ACR-700 differs from the current CANDU in that it has a smaller core, uses SEU, has three times the average discharge burnup, uses light-water coolant at a higher temperature and pressure, and has a larger gap between the PT and the calandria tube. These changes are

intended to increase safety (negative coolant void reactivity), and reduce the heavy-water inventory. However, the changes also represent a significant modification in design and behavior of the core from the current CANDU. Thus, the currently used codes (WIMS-IST, DRAGON-IST, and RFSP-IST) will have to be modified and revalidated for the new conditions. Based on the above review, the staff expects to investigate the following issues during the design certification review:

- WIMS
  - Fundamental nuclear data associated with the fertile, fissile, and burnable poison nuclides should be reevaluated with the extended burnup in mind. In addition, the scattering kernel (S(α,β)) of heavy water and light water should be reevaluated. Finally, it has been pointed out that oxygen bound in the oxide fuel matrix is important in determining the Doppler coefficient. Because the safety performance of the ACR-700 depends significantly on correct treatment in the resonance range, the effect of binding oxygen in the fuel crystal lattice should be evaluated.
  - Resonance treatment in WIMS-IST uses approximations (*newres* treatment) to speed up the calculation. It has been determined that this approximation introduces errors in the calculation of the CVR and other coefficients (Doppler), but these errors are understood within the context of the current CANDU reactors. However, the ACR-700 is a completely new concept, which will demand that a new, fast-running resonance treatment be developed and validated, in a similar way in which the *newres* treatment was validated for the current CANDU design. This validation effort will call for a significant set of experiments (see below), and a large number of Monte Carlo calculations to evaluated gaps where no experimental data exist.
  - The CVR change is determined by the coolant-voiding scenario, which changes depending on location and size of break. The voiding pattern and time dependence are important parameters. Voiding will take place over a timeframe of 1–3 seconds., and the PTs will void in an alternating fashion, creating a checkerboard pattern of voided and filled tubes halfway through the process. Thus, the overall CVR will change as a LOCA event takes place, depending on the fuel burnup on a local level. Thus, the revised version of WIMS-IST will need to treat checkerboard voiding patterns with variable burnup in adjacent fuel channels. This version will also have to be validated against appropriately designed critical experiments.
  - The burnup chains used in WIMS will have to be modified to allow for the proper treatment of burnable poison located in selected fuel pins. In addition, the inclusion of appropriate fission product treatment consistent with the higher burnup and associated higher fission product concentrations will need to be included in the WIMS code.
  - Currently two neutron energy groups are sufficient to carry out the core simulation calculations for conventional CANDUs. Cross sections are prepared by WIMS-IST and used as input to RFSP-IST (see below). Because of the

smaller core dimensions, spectral shifts at the core/reflector interface, importance of the resonance energy range, and importance of leakage effects (see below), a different few group structure might be necessary (more groups with different group boundaries). This would necessitate a change in the WIMS system.

- Leakage from the core is an important contributor to the determination of the CVR for the ACR-700, in contrast to the relatively small effect it has on the CVR for the current CANDU designs. Thus, it is important to correctly simulate the radial reflectors, the core/reflector interface, and axial streaming down the fuel channels. Accordingly, it will be important to prepare the few group reflector cross sections to include all the relevant transport effects. Special care will be necessary to model the edge fuel channels. Both these calculations will be carried out in an appropriate version of WIMS.

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

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

• DRAGON

- DRAGON is a very versatile code. It is primarily used to generate few group cross sections for use in RFSP when the cell geometry is very complicated (partially inserted control elements).

- The discussion regarding nuclear data outlined above also applies to DRAGON.

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

• RFSP

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

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

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

The conclusion of the code validation study indicates that critical experiments will be necessary to validate the revised family of codes for analyzing the ACR-700. These experiments will need to mirror the composition and geometric configuration of the ACR-700 as closely as possible. Thus, the fuel enrichment variation in the fuel channel and the Dysprosium content of the central fuel pins should represent the proposed values for the ACR-700. Briefly, this need results from the fact that resonance absorption and fission reactions control a variety of safety-related feedback coefficients. For example, in the current CANDU reactors, the magnitude of the CVR is dominated by the contribution from U-238 reactions. However, in the case of the ACR-700, the CVR is composed of adding contributions of opposite sign, with a magnitude significantly larger than the final value of the CVR. Thus, contributions from various isotopes vary in the following manner depending on burnup; U-235 (initially negative, eventually positive), Pu-239 (always negative), U-238 (always negative), Dysprosium (always negative), and leakage (initially negative, eventually positive) need to be determined correctly and added. The experiments should reflect the prototypic enrichment at BOL. Ideally, a set of experiments should also be carried out to reflect the end-of-life composition. Suitable MOX fuel should be used to simulate the end-of-life conditions, including the effect of depleting the Dysprosium-containing central fuel pin.

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

The uncertainty associated with a value predicted by the suite of analysis codes (WIMS-IST, DRAGON-IST, and RFSP-IST) is expressed by the following relationship:

$$\text{Predicted value} = \text{True value} + \text{Bias} \pm \text{Uncertainty}$$

The desired true value is related to the calculated value by the bias and the uncertainty. The bias is determined by comparing prototypically similar experiments. The uncertainty in the measurement is determined, in part, by repeating the experiment many times (see Douglas, June 2001). In addition, the uncertainty associated with a parameter can be estimated by statistically varying input quantities in the calculation of the parameter, repeating this process for all input parameters, and then propagating the uncertainty, which yields a statistical distribution for the parameter of interest. The calculation used in the determination of the parameter should clearly be more accurate than the suite of codes being validated (i.e., Monte Carlo method). Thus, it might be necessary to follow both paths, since it might be impossible to carry out experiments for all core conditions.

In the case of current CANDU reactor designs, the reported uncertainty in the CVR is estimated to be ±1.1 mk, which is approximately 5 percent of the magnitude of the CVR (+15 mk to +20 mk). Projected values of the full-core CVR for the ACR-700 are in the range of !1.0 mk to !7.0 mk, depending on burnup (see Altiparmakov, January 2002).

The NRC staff has requested the applicant to address the issues discussed in this section in a series of requests for additional information (RAIs). These RAIs suggest that the applicant follow the orderly process for preparing applications for approval of computer codes outlined in

DG-1120. The staff will revisit these conclusions as part of the design certification review after the additional requested information has been submitted.

3.6 The CATHENA Thermal-Hydraulic Computer Code

3.6.1 Code Description and Theory Manual

The purpose of this section is to present the preliminary evaluation of the CATHENA T-H computer code used by AECL for the ACR-700 safety analysis. CATHENA is a general purpose T-H computer code designed for the analysis of single- and two-phase flow in piping networks. The code provides for the two-fluid, one-dimensional nonequilibrium treatment of steam and water mixtures with separate accounting for the mass, momentum, and energy of the gas and liquid phases. Relationships dependent on flow regime describe the transfer of mass, momentum, and energy across the interface between the gas and liquid phases. The gas phase may include noncondensable gas. The code contains thermodynamic properties for heavy and ordinary water, as well as the properties of several noncondensable gases. AECL has submitted a description of CATHENA Mod-3.5c/Revision 0 for NRC staff review (see Beuthe and Hanna, November 2000), and indicated that the safety analyses for the ACR-700 will be performed using Mod-3.5d of CATHENA. In order to review an application for design certification, the staff will need documentation of the differences between Mod-3.5c/Revision 0 and the code that will be used to perform the ACR-700 design certification safety analyses.

In addition to formulations that allow the user to describe general piping networks, specialized component-specific models are also provided for the pressurizer, abrupt area changes, valves, orifices, accumulator tanks, and the reactor core. The horizontal core of the ACR-700 is described by a heat transfer package referred to as the Generalized Heat Transfer Package (GENHTP). The GENHTP package includes the following features:

- radial and azimuthal conduction
- zirconium-water reaction heat generation
- radiation heat transfer
- PT to calandria tube conductive heat transfer if contact is made
- fuel rod, PT, and calandria tube wall to fluid heat transfer

Reactor power can be calculated using an input table, by point kinetics, or by coupling CATHENA to a more detailed reactor physics computer code.

Mass, momentum, and energy transfer between the phases and momentum loss to the piping walls is calculated using flow regime mapping techniques. Flow is described as being horizontal stratified, dispersed bubble, slug, or dispersed droplet. Weighting factors are used to determine the transition between flow regimes. A significant consideration is the transition between horizontal stratified and dispersed flow regimes. Separate transition criteria are provided for circular pipes, such as the inlet and outlet headers, and for the PTs containing fuel bundles. The circular pipe weighting factor is determined based on separate effects testing. An equation developed by AECL determines the fuel bundle weighting factor, which includes geometric considerations for either 7- or 37-pin bundles. For the design certification review, the NRC staff will need information on the validation of the models for the horizontal to mixed flow transition for the 43-pin CANFLEX fuel bundles that are to be used in ACR-700.

For vertical and inclined flow, a simplified version of the flow map of Weisman and Kang is used. Flow is described as dispersed bubble, slug, dispersed droplet, annular, and annular mist.

The prediction of limiting conditions for countercurrent flow of steam and water is significant for the ACR-700 because, following a LOCA, ECC water that is injected into the inlet headers flows against the rising steam within the feeder pipes to reach the fuel channels. At the flooding limit, separated flow will no longer occur so that any incoming ECC water will be carried out with the rising steam. CATHENA uses weighting factors to provide a smooth transition between countercurrent separated flow and mixed concurrent flow. For horizontal flow, such as would occur within the fuel channels, CATHENA determines the flooding limit using the correlation of Ardron and Banerjee in "Flooding in an Elbow Between a Vertical and Horizontal or Near-Horizontal Pipe," International Journal of Multiphase Flow, 1986. For inclined and vertical flow, such as would occur in the feeder tubes, the flooding limit is determined using a modification by Popov and Rohatgi to the Ishii entrainment criterion described in "Effect of Interfacial Shear and Entrainment Models on Flooding Predictions," presented at the AIChe Symposium Series—Heat Transfer in 1986.

For nonstratified flow, the code provides five options for calculating momentum transfer as a result of frictional losses to the piping wall for two-phase flow. The recommended choice is the heat transfer fluid flow service model (see Chaxton, et al., "H.T.F.S. Correlation for Two-Phase Pressure Drop and Void Fraction in Tubes," AERE-R7162, 1972), which is the code default. For PTs which have undergone creep, a model by Friedel is included in the code to calculate the two-phase pressure drop. The NRC staff has requested additional information concerning the significance of this model and its validation. For stratified flow, friction between the individual phases and the pipe wall is determined by the code for circular pipes using the methodology of Solbrig described in "Heat Transfer and Function Correlations to Describe Steam-Water Behavior in Nuclear Safety Studies," AIChe Symposium Series 74, Heat Transfer: Research and Applications, 1978). For noncircular channels and the core channels containing fuel bundles, the wall contact friction is based on tabulated geometric data. The NRC staff has requested AECL to provide the source of this data.

Interface heat and momentum transfer is determined as a function of flow regime. Separate models are programmed into CATHENA for dispersed bubble, slug, dispersed droplet, horizontal stratified, and vertical annular flow. These models utilize published correlations based on experimental determinations.

Interface heat transfer is determined for metastable and stable thermodynamic states. Superheated liquid and subcooled vapor are metastable states. Large heat transfer coefficients minimize any deviation from saturation of either the liquid or vapor phases. The NRC staff has requested additional information concerning the validity of this assumption.

The stratified flow model includes a level swell model for the conditions in which bubbles reside below the liquid surface. This model is important for determining the extent of fuel element coverage in a partially voided fuel channel. The fuel bundle heat transfer calculations use the levels calculated by this model. Models are also available for calculating the temperature stratification in the steam space above the two-phase level. Using these models, the fuel pins closer to the top of the bundle within the steam space will be exposed to higher temperatures

than the fuel pins at lower elevations within the steam space. The NRC staff has requested additional information concerning both of these models.

Heat structures in CATHENA include piping and vessel walls, steam generator tubes, fuel channels, calandria tubes, and fuel bundles. The code offers flexible options for describing the heat structures. A T-H component may contact a number of solid surfaces, each with a different temperature. The heat flux to or from a solid surface may be apportioned between the liquid and steam phases of the fluid node, in accordance with the wetted area. The heat flux may be further apportioned between the bulk phase and that going to the interface between the phases. By this modeling, heat transfer can produce changes in the sensible heat of the bulk fluid or evaporation/condensation at the interface. The fraction of heat going to each process is determined internally by the code.

The orientations between the heat structures and the fluid nodes are also flexible. The simplest model is to assume constant temperature along the length of each segment of the heat structure. The boundaries of the heat structure nodes do not have to coincide with those of the fluid nodes. For cases in which the noding boundaries do not coincide, the code automatically apportions the heat flow according to subtended areas.

A second method for modeling heat structures linearizes the temperatures of the structure nodes and performs the heat transfer calculations using the linearized temperatures. This method produces greater stability than the constant temperature method and is useful whenever there are large changes along the axis of a heat structure. With this model, as with the constant temperature model, the heat structure and fluid node boundaries do not need to coincide.

A third method for determining effective heat structure temperatures for use in a heat transfer calculation is called the quench-inferred temperature method. This method is useful in calculating heat transfer from a fuel bundle undergoing reflood following a LOCA. This model also linearizes temperatures between the nodes of the heat structures and adds additional nodal points corresponding to the location of the onset of nucleate boiling, the occurrence of critical heat flux (CHF), and the location where the heat structure rewets. The NRC staff has requested more information concerning how these locations are determined and the application of this model to the prediction of CHF for the ACR-700.

In addition to convective heat transfer to the surrounding coolant, CATHENA can calculate direct heat transfer from the surfaces of the fuel pins to the surface of the surrounding fuel channels by radiation or by direct contact. To determine radiation heat transfer view factors, AECL has developed a separate utility called MATRIX. CATHENA can calculate heat transfer by direct contact between the fuel-bundle-bearing pads and the PT, fuel pin contact with the PT as a result of bundle slumping, and PT contact with the calandria tube as a result of PT ballooning. The user supplies the direct contact conductance coefficient.

CATHENA includes two PT deformation models. These models calculate PT ballooning as a result of excessive heating and predict PT failure when the local creep strain becomes excessive. The theory manual does not include equations for calculating contact area between a ballooned PT and the surrounding calandria tube. The NRC staff has requested that the applicant provide this methodology.

Once contact between a PT and its calandria tube is calculated by the code, deformation of the calandria tube has begun. For conditions of continued fuel channel heating without adequate cooling, calandria tube failure will eventually occur. For the ACR-700, the PTs are thicker and the gap between the PTs and calandria tubes wider than for existing CANDU reactors. For the ACR-700, spacer garter springs are placed about the PTs at intervals to maintain the gap spacing. AECL needs to demonstrate that the pressure deformation models in CATHENA, developed to evaluate performance of existing CANDU PTs and calandria tubes, are appropriate for the ACR-700.

The CATHENA technical manual discusses two sources of heat generation within the core channels, (1) the nuclear heat generation within the fuel pellets, and (2) heating by the zirconium-water reaction that would occur if temperatures become sufficiently elevated. Nuclear heat can be entered either as table input or by using the CATHENA point kinetics model. The NRC staff understands that CATHENA can also be coupled to three-dimensional neutronics codes for calculating nuclear heat and has requested additional information concerning this methodology.

The zirconium-water reaction is calculated using the equation of Prowse and Vandenberghe. This is a different correlation from the Baker and Just equation required by Appendix K to 10 CFR Part 50 for LOCA analysis performed in accordance with Appendix K. The NRC staff has requested additional information concerning the conservatism of the Prowse and Vandenberghe equation for performing the ACR-700 safety analyses.

The radiation heat transfer model for calculating radiation heat transfer between the surfaces within a fuel channel is designed to account for the effect of geometry and temperature changes. With this model, the user will input a series of emissivity-view factor matrixes as a function of an independent variable. The emissivity of the heat structures within a fuel channel will change with temperature, and the view factors will change as a result of creep-strain of the PT and changes in fuel geometry from overheating. The user selects an appropriate independent variable for the type of analysis being performed. For conditions in which the PT radius is changing, the user might select total PT strain as the independent variable. For conditions in which emissivity and geometry are both changing, the user might select an internal fuel channel temperature as the independent variable. The theory manual does not discuss how the code would treat radiation for a fuel channel that is partially filled with water. For a partially filled fuel channel, the shape factor matrix would be considerably different from that of a steam filled channel. The NRC staff has asked AECL to explain how it will apply these models to the ACR-700 safety analysis and how it will validate them for specific ACR-700 conditions.

The CATHENA code includes a number of special models for various plant components which need individual treatment by the code. Special models are provided for analysis of the following components:

- abrupt area change
- accumulator tank
- adjacent-node mixing
- boiling length average CHF model
- break discharge
- choked junction

- delay line
- Fisher valve
- generalized tank
- heat balance model
- heat exchanger
- junction resistance
- point-reactor neutron kinetics
- pump
- separators
- T-junction
- valve/orifice

The following paragraphs discuss the staff's evaluation of these models.

3.6.1.1 Abrupt Area Change

As fluid flows through an abrupt change in area, alterations in the fluid momentum affect the local pressure. The area change causes a nonrecoverable pressure loss from turbulence to be imposed on the fluid. In addition, recoverable changes occur in the pressure from the area change itself and from changes in fluid density. CATHENA contains correlations for computing nonrecoverable pressure losses for either single-phase or two-phase flow. The code can also compute recoverable pressure losses from area changes as a user option. The code ignores recoverable pressure effects from density changes. Because the fluid density might change significantly during the course of a LOCA event or safety/relief valve opening, the staff has asked for more information concerning the application of this component model to the ACR-700.

3.6.1.2 Accumulator Tank

CATHENA contains three accumulator tank models. The first is a simplified generic model that the NRC staff assumes will be used for the safety analyses of the ACR-700. Additional models describe plant-specific features of the RD-14 test facility and the MAPLE-X10 isotope production reactor, which are designed to enhance the accuracy of analyses of those facilities.

The generic accumulator model makes several simplifying assumptions, including the neglect of hydrostatic heads, heat flow from the tank walls, and flow of the cover gas into the reactor system once the accumulator has emptied. The NRC staff has requested additional information concerning this model.

3.6.1.3 Adjacent-Node Mixing Model

The adjacent-node mixing model attempts to account for heat transfer within the length of stagnant pipes. When piping is divided into nodes for analyses using CATHENA, the one-dimensional flow model does not provide for heat transfer between adjacent nodes unless there is also fluid flow down the length of the pipe. This is a weakness of all one-dimensional fluid models. CATHENA attempts to overcome this weakness using empirical relationships to model the following:

- free convection transport

- transport resulting from turbulence
- heat diffusion associated with molecular conduction

The heat transferred by these three mechanisms is determined for each phase using an equivalent conductivity across the distance between the thermal centers of adjacent nodes. The equivalent conductivity from natural convection is determined by the code based on empirically based equations that are a function of the pipe angular orientation. The equivalent conductivity for turbulent transport is calculated from the difference in nodal velocities and a constant input by the user. The code invokes the adjacent-node mixing model by default, using default input for the turbulent mixing contribution. The NRC staff has asked for additional information on how this model will be implemented for the ACR-700 and the relationship of this model with the numerical mixing that occurs in CATHENA as a result of the nodal treatment of components.

3.6.1.4 Boiling Length Average Critical Heat Flux Model

The boiling length CHF is determined by the code used in the "Groeneveld Table Lookup" CHF model, which is the code default. The CHF is calculated for CANDU fuel bundles of both 37-element and 43-element CANFLEX design in both heavy water and ordinary water. Tables of CHF are provided as a function of pressure, mass flux, and flow quality at the inlet to the fuel channel. A boiling length multiplier is used to account for differences in local channel conditions from those of the test section from which the table was derived. The boiling length multiplier is valid only for unidirectional flow and not for flow reversals. The NRC staff understands that AECL plans to develop a CHF correlation that is specific to the ACR-700 fuel at some time in the future. The staff will review the ACR-700 specific correlation in detail when AECL submits it.

3.6.1.5 Break Discharge Model and Generalized Discharge Model

The break model in CATHENA provides several critical flow formulations from which the user may select. Flow rates are derived from the thermodynamic conditions in the upstream node. For subcritical flow, the Bernoulli equation can be selected. For critical flow of single-phase liquid, the Henry-Fauske correlation is available. For two-phase critical flow, the choices are constant enthalpy, Moody, or Henry-Fauske. For pure steam, a constant entropy model can be used. These critical flow model options should provide the user with adequate flexibility to provide conservative results when it is appropriate to either maximize or minimize the critical flow calculated by the code. The user may select to use static or stagnation conditions in the upstream node for input to the selected critical flow correlations, again giving the user the option to maximize or minimize the calculated flow rate.

For breaks in the reactor system, the reactor system pressure will eventually decrease so that critical flow will no longer be possible. This occurs whenever the velocity of fluid exiting the break is less than the speed of sound or whenever the minimum pressure in the break nozzle, based on critical flow conditions, deceases below the pressure in the downstream node. In CATHENA, it is assumed that critical flow no longer exists when the upstream pressure deceases to less than twice the downstream pressure. Although this assumption is roughly correct, the NRC staff believes a more rigorous method should be investigated by which the speed of sound and the minimum throat pressure are used to determine the transition from critical flow to subcritical flow.

A generalized discharge model provides the user with additional formulations for calculating critical flow. Basic principles have been used to determine equations for critical flow rate, and derivatives. Critical flow, using the formulations of the generalized discharge model, can then be determined, along with the semi-implicit time step advancement of the code, giving increased stability over the direct flow rate computations using the break discharge models. For two-phase flow, constant entropy expansion at thermodynamic equilibrium can be assumed for breaks at the end of long pipes. For breaks at the end of short pipes, constant entropy at constant flow quality is assumed. A more generalized treatment of nonequilibrium is also provided based on the work of V.H. Ransom and J.A. Trapp in "A Choked-Flow Calculation Criterion for Nonhomogeneous, Nonequilibrium, Two-Phase Flows," International Journal of Multiphase Flow, 1992. The deviation from equilibrium is determined using an empirically derived constant. The generalized discharge model can be used to check for critical flow between any two junctions in the nodalization scheme, as well as to calculate break flow from the system.

3.6.1.6 Choked Junction Model

CATHENA uses the choked junction model to provide a critical flow check through components such as an orifice or a valve. Unlike the generalized discharge model, the sonic velocity for the choked junction model is calculated from the thermodynamic conditions in the upstream node. In the generalized discharge model, the change in fluid properties from expansion down the length of a pipe is considered. Using the choked junction model, if the fluid velocity, as calculated by the conservation of momentum equation, is greater than the sonic velocity, then the junction velocity is set to the sonic velocity. Two options are available for calculating sonic velocity. One assumes homogeneous equilibrium. The other is a homogeneous frozen model in which liquid flashing to steam is not permitted. Both these models would be expected to overpredict sonic velocity because they are based on upstream nodal conditions. The frozen model would have the greater overprediction.

3.6.1.7 Delay Line Model

The delay line model is used to simulate temperature-time delays associated with the transit time for flow through long pipe lengths, such as feedwater and condensate piping. This model is valid only if the flow is unidirectional. Use of the delay line model avoids the need to provide detailed noding for sections of the plant only indirectly affected by the transient or accident being analyzed.

3.6.1.8 Fisher Valve Model

CATHENA contains detailed modeling of Fisher valves based on the manufacturer's valve specifications. Correlations for single-phase liquid flow, single-phase steam, and two-phase flow are provided. The calculated flow rates are compared against maximum flow rates provided by the manufacturer. The use of actual valve data to compute flow rates provides for increased accuracy over more generic valve models.

3.6.1.9 Generalized Tank Model

The generalized tank model in CATHENA provides code users with a flexible sink volume for collecting discharges from the reactor system. A tank volume is divided into two regions, one

containing predominantly liquid and the other containing predominantly gas. The code continually calculates the level of the interface between the two regions. Bubble formation and separation are evaluated in the lower region, and rain out and spray addition are evaluated for the upper region. Multiple flow paths may be connected to a tank volume to model gas and liquid inflows and outflows. Direct heating may be added to the lower region.

Mass transfer between the upper region and the lower region is calculated by summing the contributions from rain out, bubble rise, and condensation. Heat transfer to internal heat structures is calculated based on user-specific heat transfer options. The same options are available for evaluating the heat flow to or from any CATHENA heat structure. The selection of the appropriate heat transfer option will depend on the experience and training of the user. Interregional direct heat transfer between the upper and lower region is not described, nor is direct condensation to the heat structures. This is a possible source of error in the general tank model. The NRC staff will revisit the general tank model when AECL defines its use in the ACR-700 safety analysis.

The generalized tank model may be used to analyze the containment building. The NRC staff understands that AECL will utilize the GOTHIC code for containment analysis of the ACR-700. The staff, therefore, does not plan to review the CATHENA generalized tank model for containment analysis. The generalized tank model could also be used to model the ACR-700 pressurizer. The NRC staff intends to review a comparison of CATHENA predictions with applicable experimental data for pressurizer in-surges and out-surges if the model is used for pressurizer analysis.

3.6.1.10 Heat Balance Model

The heat balance model provides users with a means of tracking the power transfer to the fluid within a user-defined section of CATHENA T-H components. From this balance, users can determine if, at a particular instance in time, the fluid within a group of components is gaining or losing energy.

3.6.1.11 Heat Exchanger

The heat exchanger model provides the ability to transfer energy between the fluid within a component to a user-defined sink. The user defines the sink temperature, heat transfer area, and heat transfer coefficient. The model provides the user with additional options and the capability of performing sensitivity studies. The text recommends use of multiple fluid nodes and the CATHENA heat structure models, if a more detailed analysis of heat exchanger performance is needed.

3.6.1.12 Junction Resistance Model

The junction resistance model provides code users with options to input form loss coefficients. These would be used in place of the form loss coefficients calculated by the code to account for area changes. Reversible loss coefficients can be input as well. By inputting reversible form loss coefficients, the user can partially account for the lack of momentum flux computation by CATHENA. The NRC staff will review user input form loss coefficients for the ACR-700 as part of the methodology review.

3.6.1.13 Point Neutron Kinetics Model

The CATHENA point neutron kinetics model considers reactivity feedback from changes in the following parameters:

- temperature of the fuel
- temperature of the water
- density of the water
- Xe-135 concentration

Because the reactivity coefficients are not constant with the independent property, provision is made to input the reactivity coefficients in either quadratic or tabular form.

The point kinetics model provides two options for calculating reactor power. The first method calculates the reactivity feedback using a flux squared weighted average of the independent variables to determine the reactivity change. The second method looks up the reactivity change in each core segment separately and then finds the net change using flux squared weighting. This second method gives more accurate results and is better able to accurately predict test data.

CATHENA can also be coupled to AECL three-dimensional neutronics codes to evaluate cases in which changes in the core power shape are expected to affect the results. The NRC staff will review each safety application of CATHENA to ensure that the appropriate methodology is used to calculate the nuclear power. At the present time, the staff understands that the point kinetics models will not be used for ACR-700 analysis, but that CATHENA will be coupled to three-dimensional reactor physics models for all transient reactor power determinations.

Decay heat may be calculated by a user-supplied table or using modifications to the 1979 American Nuclear Society (ANS) 5.1 decay heat standard. The code has options for NU CANDU fuel or enriched fuel for the MAPLE isotope production reactor. Decay heat for the ACR-700 may differ from either of these designs because of differences in the fraction of fissions coming from the various fissionable isotopes. For most safety analyses, conservative results are obtained when decay heat is maximized. For some analyses, such as evaluation of return to power following a main steamline break (MSLB), more conservative results may be obtained by minimizing decay heat. Return to power following an MSLB is not expected for ACR-700. The NRC staff will review the decay heat model used for each ACR-700 safety application.

3.6.1.14 Pump Model

CATHENA calculates pump heads and torques from input pump curves that are a function of volumetric flow and impeller speed or from one of eight pump models that are built into the code. The pump affinity laws are utilized for nonrated conditions. The code has the capability to determine degraded pump conditions for two-phase flows based on user input. Pump coastdown can be evaluated for analysis of loss-of-flow (LOF) transients. The heat generated by pump operation is added to the coolant. Pump curves for the ACR-700 pumps are not currently included in the code. These can be input by the user when the plant pump characteristics are determined.

3.6.1.15 Separator Model

CATHENA provides two types of steam separation models. The first describes the steam generator separator, and the second describes a horizontal piping connection. The steam generator separator model determines the moisture carried out into the steamlines. Liquid entrainment to the steamlines is undesirable because damage to the turbine blades may occur. For analysis of MSLBs, it is conservative to assume a high efficiency for the steam generator separation equipment, since this assumption results in the greatest cooldown of the reactor system and the greatest amount of energy release to the containment for containment pressure and temperature analysis. The NRC staff will investigate the assumptions used for steam separation efficiency for design-basis analyses of the ACR-700 when AECL submits them.

Determining the steam and liquid fractions for a takeoff from a horizontal pipe is important for determining the flow conditions for a postulated break flow and for determining the refill of fuel channels during recovery from a LOCA. Special models are provided to calculate flows at the high velocities that would exist for postulated feeder tube breaks either at an inlet or outlet header or at the PT connection. Only horizontal takeoff pipes are modeled, even though the feeder pipes connect to the headers and PTs at a variety of angles. The NRC staff has requested additional information concerning the adequacy of these models.

3.6.1.16 T-Junction Model

The T-junction model in CATHENA is used to calculate hydraulic losses through tee connections as a function of friction and momentum changes. The formulations consider the fluid at the T-junction to be fully mixed and incompressible. The code user may select from several formulations for determining pressure drop in the main line and for the branch off-take. The use of tee junctions should aid in predicting flow to and from the headers and feeder pipes servicing the various fuel channels. The assumption of mixed flow might not always be adequate to describe header flows, especially during the reflooding period following a LOCA. The staff has requested more information on this subject and on the range of the pressure drop correlations selected for the analysis of the ACR-700 as compared to the conditions predicted for the plant.

3.6.1.17 Valve/Orifice Models

Flow losses through orifices and valves are calculated for single-phase flow with the imposition of two-phase multipliers. Two models are available for calculating single-phase loss coefficients. The first is an ASME model and the second is a component-specific model for thin-plate, square-edge orifices used in the feeder pipes for flow balancing of CANDU reactors. The CANDU orifice models, which are flow dependent, have been specifically correlated from hydraulic loss experiments for orifices used in CANDU reactors.

Code users have three options for determining two-phase multipliers to apply to the single-phase loss coefficients. These are the homogeneous flow assumption and correlations by Chisholm and Rooney, James, and Lin. The NRC staff will review the models selected for the ACR-700 analysis as part of the methodology review.

Options are available to model four types of valves. The “controllable valves” category is the most versatile, allowing the valves to be opened and closed as a function of other, user-

specified variables. In addition, a simple unidirectional check valve model is available, as well as a simple relief/safety valve model. Detailed models are available for modeling check valves unique to the MAPLE isotope production reactor, but are not within the scope of this review.

3.6.1.18 System Control Models

These models provide the user with considerable flexibility in modeling plant control systems, as well as control of the code operation and code output. The NRC staff will review modeling of control systems as part of the review of the ACR-700 input models which are used for the CATHENA safety analyses in the design control document.

3.6.2 Code Assessment—Experimental Data Comparisons

The purpose of this section is the evaluation of code assessment, experimental data comparisons, and PIRT. DG-1120 outlines an orderly process for establishing and evaluating evaluation models. One important step is to identify and rank the physical processes and phenomena that are being modeled. This ranking is important to ensure that significant features and attributes that a computer code should have are identified so that available resources may be appropriately allocated. The principal product of this process is a PIRT. The PIRT should be used to determine the resource utilization for physical model development, scaling, validation, sensitivity studies, and uncertainly analyses.

AECL has begun the process of demonstrating the adequacy of CATHENA for the ACR-700 analysis, in accordance with the guidance of DG-1120, by developing a PIRT for the T-H processes and phenomena that would occur during a large-break LOCA (see H.E. Sils and N. Popov, “PIRT for Critical Inlet Header Break LOCA in ACR-700,” 108US-03500-LS-001, Revision 0, February 2004). AECL chose a postulated break equivalent to 25 percent of the double-ended area of an inlet header for this study. For this break size, the inlet header pressure is reduced to approximately that of the opposite outlet header so that flow in the affected core channels stagnates, which leads to core channel voiding and fuel cladding (sheath) heatup. AECL determined that the 25 percent break size would produce the greatest degree of stagnation through a series of analyses with the CATHENA code. AECL recognizes that the PIRT is not the end result, but rather a tool to guide and focus subsequent efforts. These efforts include provisions for physical model development, scaling, validation, sensitivity studies, and uncertainly analyses, as described in DG-1120.

The NRC staff views the development of the T-H PIRT for large-break LOCA for the ACR-700 as a significant step in the staff’s process for reviewing CATHENA. The staff believes that the applicant should use the PIRT process to evaluate other transient and accident scenarios for which CATHENA will be used in the safety analyses for the ACR-700.

As discussed in Section 3.5 of this report, the NRC staff also assembled PIRT panels to investigate processes and phenomena important to three categories of ACR-700 safety analysis. The NRC PIRT panels investigated severe accident, nuclear analysis, and T-H aspects. The NRC T-H PIRT panel evaluated the same 25 percent inlet header break size as did AECL PIRT panel. In some instances, the importance rankings of the NRC panel differed from those for the same processes and phenomena investigated by AECL PIRT panel. The

staff will consider the differences in importance rankings between the NRC and AECL PIRT panels in future discussions with the applicant.

Before requesting preliminary review of CATHENA for the ACR-700, AECL embarked on a program similar to the PIRT process described in DG-1120 for validating the CATHENA code for the physical processes important to the safety analyses of operational CANDU reactors. The physical processes for the following seven accident categories were evaluated and prioritized.

(1) large LOCA
(2) LOCA with loss-of-emergency coolant injection (LOECA)
(3) small LOCA
(4) loss of flow (LOF)
(5) loss-of-regulation (LOR)—loss of reactivity control
(6) loss-of-feedwater (LOFW)
(7) MSLB

Phenomena were then grouped according to whether they were principally T-H or relevant to fuel-channel, thermal-mechanical behavior. Twenty-three thermodynamic phenomena were identified as having significance for the seven accident categories (see Mallory, "Validation Plant for CATHENA Mod-3.5c," RC-2240, 2000). Table 3-12 ranks these phenomena into three classifications of importance.

To determine the phenomena significant to fuel-channel thermal-mechanical behavior, AECL evaluated the seven accident categories used in the T-H evaluations and included the following accident categories:

- fuel-handling accidents (FH)
- flow blockage to the fuel channels (FB)
- end fitting failure
- spontaneous PT and calandria tube rupture (CTR)
- stagnation feeder break

Twenty-five fuel-channel, thermal-mechanical phenomena were identified. Ten of the phenomena involve the determination of fuel behavior, so that analyses using the applicant's ELOCA fuel behavior code is necessary. The applicant decided to validate both codes together for these phenomena at a later time. Because the ELOCA code is not part of the preapplication review, the NRC staff will address the phenomena associated with ELOCA during design certification.

Six of the fuel-channel, thermal-mechanical phenomena are not modeled by CATHENA. With the omission of the non-CATHENA and CATHENA-ELOCA phenomena, nine fuel-channel thermal-mechanical phenomena remain. Table 13-3 lists the remaining nine phenomena for which the NRC reviewed the CATHENA validation during preapplication.

J.P. Mallory and K.A. Arsenault in "CATHENA Mod-3.5c/Rev 0 Systems Thermal-Hydraulic Validation Manual," RC-2701, 2001, discuss validation of CATHENA for the 23 T-H phenomena by comparison to test data, Kowalski, et al. in "CATHENA Mod-3.5/Rev. 0 Fuel and Fuel-Channel Thermal-Mechanical Effects Validation Manual," RC-2702, 2001, discuss the validation

of CATHENA for the 9 fuel-channel, thermal-mechanical phenomena. These documents demonstrate that, for some of the processes, CATHENA performs well in comparison to the test data. For some of the phenomena, the documentation reports that CATHENA is less than adequate and that additional model improvement and validation need to be done. The NRC staff has requested AECL to address these deficiencies and explain the code validations that were performed for existing CANDU designs to ensure their relevance to the ACR-700.

The applicant's PIRT for a large-break LOCA ranked processes and phenomena at the component level (i.e., inlet and outlet headers, feeder tubes, core channels) (see Sils and Popov, 2004). As reported in Mallory and Arsenault, 2001, and Kowalski, et al, 2001, current CATHENA validations, with some exceptions, have not extended to the component level. The NRC staff has requested additional information concerning validation of CATHENA for processes and phenomena at the component level. The staff has also requested that the applicant develop additional PIRTs to include all events for which CATHENA will be used to perform the ACR-700 safety analysis. The code should also be validated at the component level in accordance with these additional PIRTs.

Table 3-12 Validation Priorities for CANDU Accident Analysis Thermal-Hydraulic Phenomena

ID No.	Phenomena	Large LOCA	LOECA	Small LOCA	LOF	LOR	LOFW	MSLB
T-H1	Break Discharge Characteristics and Critical Flow	P	P	P			P	P
T-H2	Coolant Voiding							
T-H3	Phase Separation	P	P	P	P		P	P
T-H4	Level Swell and Void Holdup	S	S	P				
T-H5	HT Pump Characteristics (Single & 2-Phase)	P	P	P	P			
T-H6	Thermal Conduction	P	P	P	S	S		
T-H7	Convective Heat Transfer	P	P	P	P	P	P	P
T-H8	Nucleate Boiling			P	P			
T-H9	CHF & Post Dryout Heat Transfer	S	S					
T-H10	Condensation Heat Transfer	P		P	P			
T-H11	Radiative Heat Transfer	P	P	P	P			
T-H12	Quench/Rewet Characteristics	P		P	P	P		
T-H13	Zirc/water Thermal-Chemical Reaction	P	P	S				
T-H14	Reflux Condensation			S	P		S	S
T-H15	Counter Current Flow	P		P	P		P	S
T-H16	Flow Oscillations			S	P	P	P	P
T-H17	Density Driven Flows: Natural Circulation	S	S	P	P		P	P
T-H18	Fuel Channel Deformation	P	P	S				
T-H19	Fuel String Mechanical-Hydraulic Interaction			S				
T-H20	Waterhammer			S			S	
T-H21	Waterhammer: Steam Condensation Induced	S						S
T-H22	Pipe Thrust	S	S				S	S
T-H23	Noncondensable Gas Effect	S	S	S	S			

P—Primary Phenomena S—Secondary Phenomena

Note: Light shading indicate high-priority accident categories; dark shading indicates high-priority phenomena within those accident categories.

Table 3-13 Validation Priorities for CANDU Accident Analysis Fuel a[illegible] Thermal-Mechanical Phenomena to Be Validated as Part of the CATHE[illegible]

ID NO	Phenomena	large LOCA	LOCA w/o ECC	Small LOCA	EFF	CTR	FB	SFB	FHA	LOF	[illegible]	[illegible]	[illegible]
FC9	Sheath Oxidation Zirc. - water Reaction												
FC13	Sheath-to-Coolant and Coolant-to Pressure Tube Heat Transfer												
FC16	Pressure Tube-to-Calandria Tube Heat Transfer												
FC17	Calandria Tube-to-Moderator Heat Transfer												
[illegible]	Pressure Tube Deformation												
[illegible]	Calandria Tube Deformation or Failure												
[illegible]	Pressure Tube Oxidation												
[illegible]	Element-to-Pressure Tube Radiative Heat Transfer												
[illegible]	Element or Baring Pad-to-Pressure Tube Contact Heat Transfer												

[illegible]nd:

[illegible] Primary Relevance During at Least One Phase

[illegible] Secondary Relevance

Notes:

FC9 is the same as T-H13
FC13 is the same as T-H7, T-H9, T-H9 and T-H10
FC18 is the same as T-H18
FC21 is the same as T-H11

3.6.3 NRC Staff Audit Calculations—Thermal-Hydraulics

This section presents the NRC staff's audit calculations in the T-H area. The NRC staff contracted with Information Systems Laboratories, Inc., to have RELAP5 input developed for the ACR-700. DLC-001-2003, "Development of RELAP5 Input for the ACR-700 Reactor," dated January 7, 2004, documents this work. The model was developed from information supplied by AECL. The RELAP5 Mod.3.3 code used by the staff has a horizontal fuel channel model for CANDU plant analysis that was included in the code as part of an international cooperative

program (see Chung, et, al., “Improvements of RELAP5/MOD3.2.2 Models for the CANDU Plant Analysis,” NUREG/AI-0189, October 2000). The staff was able to make direct comparisons between predictions by RELAP5 and those of CATHENA by obtaining the CATHENA code in executable form from AECL, with a preliminary CATHENA input description of the plant. The NRC staff uses RELAP5 as an aid in understanding and evaluating the sequences and phenomena in postulated reactor accidents. RELAP5 is not a design-basis licensing tool. Conclusions on the acceptability of the results for postulated transient and accident analyses for the ACR-700 will be based on the applicant’s calculations using CATHENA or other AECL methodology and not on results from RELAP5.

The ACR-700 reactor core at the time that the CATHENA and RELAP5 models were developed employed a total of 284 fuel channels. The ACR-700 core has since been increased in size to 292 channels. The coolant makes two core passes, each containing half the total flow channels, through the ACR-700 core. In the preliminary ACR-700 CATHENA model, the channels of each core pass are lumped together. Thus, the model is incapable of evaluating differences in channel elevation and feeder tube orientation. Reactor power is put into the model in table form. Power is assumed to be constant until reactor trip, thus neglecting any reactivity effect on power before reactor trip. The first RELAP5 model was made to be similar to the CATHENA model; it described only two lumped core channels and used the same tabular power input as CATHENA. In a second RELAP5 model, six individual core channels were broken out from one of the lumped channels. The six single channels were located at three core elevations (top, middle, and bottom). At each elevation, one channel was modeled with feeder off-takes at the bottom of the adjacent headers. A second channel at the same elevation was modeled with the feeder off-takes at the sides of the headers. Both the RELAP5 and CATHENA input models describe the core channels in one dimension. Multidimensional flow effects that would be expected in horizontal flow channels under conditions of partial voiding would be lost in these simulations. Therefore, further validation of fuel channel modeling for the ACR-700 is needed.

The two reactor inlet headers (RIHs) receive flow from two coolant pumps each. Each header dispenses coolant flow to half the core channels. The two reactor outlet headers receive flow from half the core channels and dispense flow to the steam generator inlets. The RIHs and outlet headers are modeled as single nodes in both the CATHENA and RELAP5 input models. During a LOCA, flow patterns of steam and water within the inlet and outlet headers may be highly complex. The headers themselves will be 12.4 meters long for the ACR-700 and will have 146 off-takes all along the lengths. The modeling of the headers using a single node has not been shown to be adequate at this time. AECL should undertake further validation of header phenomena using applicable experimental data.

At this time, most of the analytical effort by the staff has been to evaluate the postulated break of an inlet header that would produce partial flow stagnation in the core. The stagnation condition occurs if the pressure in the inlet header is reduced by the break so that it is approximately equal to that in the outlet header. By this action, the effect of the coolant pumps is negated and flow in the affected core channels stops. AECL predicts the greatest degree of stagnation for a break of 25 percent of the double-ended area of an inlet header or 0.0868 square meters. Both RELAP5 and CATHENA predict similar plant responses for this break size, with the channels connected to the broken inlet header voiding rapidly, causing a temperature excursion in the fuel bundles. The fuel temperature is reduced first by small oscillatory coolant flows in the core channels and then by the effect of opening of the large

interconnect pipe between the outlet headers, allowing water from the other core pass to flow into the affected channels. Cooling is continued as water from the ECI accumulators is injected into both RIHs. The water from the ECI accumulator connected to the broken inlet header is mostly lost and is not effective in cooling the core. The ECI accumulator that flows into the unbroken inlet header supplies water to both core passes by means of the feeder pipes and the large interconnect line.

Both CATHENA and RELAP5 show the core to be refilled with water and to remain filled as long as the ECI accumulators are discharging. After the ECI accumulators are exhausted, both codes show the core entering an unstable stage with highly oscillatory flows and voiding. Both codes predict adequate core cooling during this stage. Eventually the core flow is predicted to become stable and the core channels no longer undergo intermittent voiding.

The results from these calculations have raised the following T-H concerns:

- Opening of the large interconnect line between the outlet headers plays a significant role in providing core cooling. At the present time, CATHENA has not been validated for modeling the large interconnect line. This validation will need to be completed for the review of CATHENA for the ACR-700 LOCA analysis by the NRC staff.

- In the core channels adjacent to the break, complete voiding occurs immediately (i.e., within 2 seconds). Then the channels are refilled by the ECI flow. When the ECI flow is exhausted at approximately 260 seconds, the core channels adjacent to the break void again. They void and refill intermittently until 813 seconds. Channel flow is in an oscillatory state during this time period. The NRC staff has not seen validation of CATHENA for this period of unstable flow. AECL will need to complete this validation.

- For stagnation header breaks, the ACR-700 fuel sheath temperatures are predicted to increase early in the transient. Analyses by CATHENA for the period before the interconnect line opens predict small flows in the affected channels driven by small pressure differences across the channels. These small flow rates are predicted to mitigate the rise in sheath temperature during the stagnation period. The NRC staff has not seen validation of CATHENA for this period of very small core flows approaching stagnation. AECL will need to complete this validation.

- Occurrence of flow stagnation in the individual core channels following an inlet header break will depend on the flow resistance of each individual channel and its connected feeder tubes, including the effect from the alignment pattern of the fuel bundles in each core channel. Results from a multiple channel representation of the core might vary significantly from the lumped channel approach currently taken by CATHENA. The orientation of feeder pipes to the inlet and outlet headers will be different for the various channels and might affect the result. The NRC staff has not yet seen the result from multiple channel analyses of the ACR-700 using CATHENA. AECL should submit these analyses to support the ACR-700 review with appropriate code validation.

- The reactor inlet and outlet headers are modeled as single volumes. Actually, the headers are long (11 m) and have multiple connections for the feeder tubes and coolant piping. Two-phase flow patterns within the headers during postulated LOCAs may be highly complex and in multiple space dimensions. The applicant needs to validate, by

comparison to experimental data, that the CATHENA analytical model adequately represents processes that will occur in full-scale reactor inlet and outlet headers of the ACR-700.

3.6.4 CATHENA Code Assessment Summary

The NRC staff has reviewed a considerable amount of material concerning the CATHENA code, including the theory manual, validation documents, and the preliminary plant description provided by AECL. CATHENA is a flexible T-H tool with a significant number of user options. Special component models are available to describe specific features of CANDU reactors. The staff believes that with proper selection of user options, the code should be capable of providing safety analyses for DBEs for the ACR-700 that would bound the actual consequences. To ensure that the calculated results are indeed bounding, AECL should compare code predictions to experimental data that are adequately scaled to reproduce the processes and phenomena that would occur during postulated reactor events. The applicant has provided the staff documents describing past validations of CATHENA. Much of the past validation efforts were related more to operating CANDU reactors than to the ACR-700. The applicant is currently performing additional experimental tests to collect data more descriptive of the ACR. The staff has suggested that this additional code validation be implemented using the PIRT process discussed in DG-1120.

At this time, the applicant has not provided complete documentation of how the CATHENA code will be applied to the ACR-700 safety analysis. Such documentation should include plant input models, justification for how plant components are described, and justification for the code options selected. It would be expected that modeling and selection of input assumptions would be commensurate with the type of transient or accident being analyzed. AECL submitted a preliminary CATHENA input model for a large-break LOCA. This model makes several simplifications, including describing the reactor inlet and exit headers as single nodes and the core channels in two lumped, one-dimensional paths. Thus, the multidimensional flow fields that are expected in both the headers and the core channels cannot be described, and differences between the individual channels cannot be investigated. The staff compared the calculated results for a large-break LOCA using the CATHENA model with those from the NRC staff's RELAP5 T-H code. The predicted results using RELAP5 were similar to those from CATHENA. The staff input model also described the reactor inlet and outlet headers with single nodes. Additional validation will have to be performed before the CATHENA results can be accepted or the results from the staff's audit calculations verified.

The staff has transmitted to AECL a list of the additional information that will be needed for approval of CATHENA for the safety analysis of ACR-700 (see letter from J. Kim (NRC) to V. Langman, "Requests for Additional Information—CATHENA Code for ACR-700 Application," dated May 14, 2004). In this transmittal, the NRC staff suggested that AECL follow the orderly process for preparing applications for approval of computer codes, as outlined in DG-1120. The staff will revisit the conclusions in this report after the additional requested information has been submitted.

3.7 NRC STAFF PIRT PANELS

This section discusses the activities of the NRC's PIRT process. The NRC staff assembled panels of internationally recognized experts to investigate the importance of processes and phenomena that might occur during postulated transients and accidents at the ACR-700. Plant behavior is not equally influenced by all processes and phenomena. By categorizing the importance of processes and phenomena, resources are better allocated for computer code validation and review. The end result of the panels' efforts was development of preliminary PIRTs, in which the phenomena and processes important for the ACR-700 safety analysis are ranked in importance against figures of merit (FoM) significant to the safety of the plant. These tables are considered to be preliminary at this time and will be the subject of further discussions with AECL.

The PIRT approach was applied separately in each of three areas of interest—nuclear analysis, T-H analysis, and severe accident analysis. The NRC staff assembled three PIRT panels of individuals who are expert in the three disciplines. Panel members did not need to be expert in CANDU technology, although several are. Each of the panels met three times, twice for 2 days, and once for 3 days. Each panel had a facilitator who led the discussions in the meetings and was responsible for compiling the information from the panel members. Each panel also had one to two NRC staff members who provided guidance. The panels for nuclear analysis and T-H relate directly to the nuclear and T-H computer codes being reviewed under FT3 for the preapplication review and will be discussed in more detail below. The following paragraphs summarize the work and findings by these panels. The NRC staff will ask that AECL address the implications of the findings of the PIRT panels on the validation of the computer codes that the staff is reviewing under FT3. The work of the NRC-sponsored PIRT panels was limited by time and resource restrictions. The staff has requested AECL to provide more comprehensive PIRTs to identify all processes and phenomena that are important to the computer code reviews under FT3.

3.7.1 Makeup of the PIRT Panels

The following individuals make up the nuclear analysis panel:

- David Diamond, Brookhaven National Laboratory
- Thomas Downar, Purdue University
- Ronald Ellis, Oak Ridge National Laboratory
- Farzad Rahnema, Georgia Institute of Technology
- Paul Turinsky, North Carolina State University

David Diamond was also the facilitator/editor for the panel, and Donald Carlson was the NRC representative.

The following individuals made up the T-H panel:

- Samin Anghaie, Florida State University
- Sanjoy Banerjee, University of California, Santa Barbara
- Peter Griffith, Massachusetts Institute of Technology
- Yassin Hassan, Texas A&M University

- Pradip Saha, Massachusetts Institute of Technology
- Novak Zuber, Consultant

Brent Boyack was a consultant who served as the panel's facilitator, and David Bessette was the NRC representative.

3.7.2 Summary of Evaluations and Conclusions by the Nuclear Analysis PIRT

Two scenarios were selected for consideration. The first scenario was a large LOCA caused by either an inlet or outlet header break. The break is large enough to initially void half the channels that flow directly into or out of the affected header. The channels not connected to the affected header will not void immediately. Thus, a checkerboard pattern of voiding is initially created. These scenarios are of interest in the CVR analysis and are discussed in more detail in Section 8.3.2 in this report. Checkerboard voiding will happen in various large-break scenarios. Preliminary simulations provided by AECL show that pronounced checkerboard voiding happens in about 1 second, with reactor trip at 2 seconds, and followed seconds later by full-core voiding (see"Scenario Description for ACR-700 Inlet Header Critical Break LOCA," AECL, Public, 2004). A successful trip makes the CVR no longer relevant.

The FoM of interest is the calculation of the reactivity effect of the aforementioned voiding under LOCA conditions, namely the CVR. The CVR of interest is obtained from steady-state calculations of the core multiplication factor (k_{eff}) at operating conditions, and then again at the same conditions, but with either half or all the channels voided. The core is assumed to be at equilibrium fueling conditions, as the transition core has not yet been finalized. This FoM is considered important for two reasons. First, GDC 11, "Reactor Inherent Protection," provides for a negative power coefficient. The coolant density feedback is an important component of the power coefficient. In existing CANDU reactors, the CVR worth is positive, and the ACR-700 is specifically designed to have a negative void worth. The confirmation of negative void reactivity is the NRC's "Preapplication Focus Topic 9." Second, the CVR in existing CANDU designs has been one of the key factors in determining whether the energy deposition during a large LOCA (because of a positive CVR) is within the safety limit by which the reactors are licensed in Canada.

The calculation of the CVR was broken into three components—reactor conditions, core simulation (neutronics and T-H), and lattice physics. The PIRT phenomena for each component were developed with ranking for importance and knowledge level. The results show gaps in all three components of the calculation of the CVR that need to be filled by a combination of experimental data and analytical support. The phenomenon with the largest discrepancy between importance and knowledge level relates to spatial homogenization of cross sections for use in core simulations.

The experiments that are recognized as being important are those that are already planned by AECL, particularly postirradiation examinations (PIEs) that are to be done for CANFLEX fuel and experiments on the ZED-2 facility. The latter are particularly important because they would acquire not only data directly related to the CVR, but also data relevant to other reactor physics parameters. The primary value of the ZED-2 experiments is to validate computational methods over a variety of conditions. They can also be used to directly calculate the CVR for an isolated fuel bundle and provide information on its uncertainty. The PIRT also shows that understanding

the leakage effects would be particularly useful. Neutron leakage between ACR channel nodes is significant, especially when the void is in a checkerboard pattern, and from peripheral channels to the adjacent reflector regions. Because the checkerboard situation exacerbates the homogenization problem, which was identified as particularly important, the ZED-2 facility may be also be very useful if it can represent such a configuration. In general, it would be very useful for the NRC to follow the ZED-2 work so that results can be used to validate NRC calculational methods and to resolve the leakage issues brought up in the PIRT.

Two types of calculations were suggested by the PIRT. One type consists of sensitivity calculations, using appropriate methods, to understand the effects of different parameters (e.g., sensitivity studies to look at the effect on the CVR of more spatial detail for fuel temperature, changes in geometry caused by PT sagging, and manufacturing tolerances for the fuel element containing Dysprosium). The second type of calculation is a benchmark calculation to check the validity of less rigorous methods. For lattice physics analysis, the obvious choice for the benchmark is a continuous Monte Carlo method. Examples of applications are to check cross section libraries, resonance treatment, and the effect of transverse control devices. For questions that arise on the core level, Monte Carlo should be used, as well as a lattice physics code in a color set configuration (multiple fuel cells represented). Among other things, running these codes would help resolve some of the leakage issues. The latter is particularly important in determining homogenized cross sections and, as stated above, this is one of the key nuclear analysis that needs to be addressed.

The second scenario of interest to the nuclear analysis panel was also a LOCA. However, in this case, the panel was not interested in particular scenarios (i.e., the panel was interested in all LOCAs because they all have in common the need to know the decay heat level throughout the core and hence the initial power distribution). The FoM of interest is the calculation of (1) the bundle power distribution throughout the core at full power, and (2) the corresponding peak fuel element power density. The former helps determine the hydraulic conditions during the event, and the latter is used specifically to calculate the peak-clad temperature. The core is assumed to be at equilibrium fueling conditions because insufficient information exists at this time to consider transition cores. This FoM is considered important because it directly impacts one of the ACs for this event—the peak clad temperature.

The PIRT was again developed according to phenomena related to reactor conditions, core simulation, and lattice physics. The experimental work is from ZED-2 and PIEs; the latter provides validity for calculations of isotopic composition. The most important uses of the ZED-2 experiments for power distributions are to compare measurements with calculated power distributions within the fuel bundle and to provide measurements at a simulated core-reflector interface to validate the core simulator modeling of these boundaries.

The calculational support suggested by the PIRT includes sensitivity studies using appropriate methods. Examples of such studies are the sensitivity of power distribution to the uncertainty in control device position, to changes in xenon distribution as a result of refueling, to different cross section libraries, to cross section group structure, and to the displacement of heavy water with structural material. Benchmark calculations using methods more rigorous than those expected to be the norm are also suggested. The examples that were identified as being most urgent to resolve relate to calculating the bundle power near the core-reflector interface and the fuel element power everywhere, the latter using dehomogenization (flux reconstruction). Interstitial effects, end effects, and core-reflector interface effects make the reconstruction

difficult. To resolve these issues, one suggestion is to conduct benchmark calculations using Monte Carlo methods to represent three-dimensional subregions of the core in detail. Other examples for which benchmarks would be useful include consideration of full and partial rod insertion, axial reactor boundary conditions, and the application of assembly discontinuity factors.

3.7.3 Summary of Evaluations and Conclusions by the Thermal-Hydraulic PIRT

The T-H PIRT was developed for a 25-percent RIH LOCA, which results in stagnation or near-stagnation conditions in the broken pass, with consequent loss of Class IV power. AECL provided a scenario description of this LOCA, including the results of the CATHENA code calculation. The scenario was divided into three phases—(1) the blowdown phase beginning at time zero and ending at 53 seconds, (2) the late blowdown/ECI/refill period between 53 and 250 seconds, and (3) the LTC period for times greater than 250 seconds.

During the PIRT process, each phenomenon was assessed relative to a two-fold FoM. The primary FoM was associated with the temperature response of the fuel cladding. Specifically, each phenomenon was assessed for its impact on the cladding temperature during each phase of the accident scenario. The secondary FoM was associated with the performance of the PT during the accident scenario. Specifically, each phenomenon was assessed for its impact on PT high-temperature strain leading to maximum high-temperature plastic strains of less than 2 percent.

In addition to the CATHENA code calculation for the selected LOCA, the T-H panel members reviewed the experimental results obtained during the 25-percent RIH break Test B9401 conducted by AECL in the RD-14M integral test facility. The CATHENA calculation of this test was also reviewed based on the limited results presented to the panel.

The summary of the T-H PIRT results shows that the most important components are the headers and the heat transport circuit below the headers. This includes the fuel bundles, fuel channels, and the feeders. The most important phenomena are the multi-channel flow distributions, flow reversal/stagnation inventory/void distribution, multi-dimensional effects, including flow regime in headers and fuel channels, and rewetting of fuel bundles. The large outlet header interconnect line, a new feature in the ACR-700, is also very important for determining the maximum cladding temperature for the chosen scenario. From the structural point of view, the deformation of a fuel bundle, fuel bundle to PT contact, and the PT to calandria tube contact are the most important items. Therefore, future research should be directed toward better understanding and predicting these T-H and structural phenomena.

3.8 Assessment of the Thermal-Hydraulic Experimental Database for the ACR-700

3.8.1 Introduction

This section presents the staff's assessment of the T-H experimental database for the ACR-700. Section 3.5 and Section 8 of this report provide review insights on the neutronics experimental database needs. State-of-the-art T-H modeling does not permit the extrapolation of computer codes to new applications for which applicability has not been determined and for

which the codes have not been assessed. The behavior of T-H systems is geometry dependent. Moreover, the codes contain numerous models that are empirically based and scale dependent, and for which applicability is limited to the range of scales and conditions for which they were developed. Experiments are necessary to validate the codes and models for new applications.

The Commission's regulations in 10 CFR 52.47(b)(2)(i)(A) provide that certification of standard designs that differ significantly from light-water designs of plants that have been licensed will be ranged only if:

> The performance of each safety feature of the design has been demonstrated through either analysis, appropriate test programs, experience, or a combination thereof;
>
> Interdependent effects among safety features of the design have been found acceptable by analysis, appropriate test programs, experience, or a combination thereof;
>
> Sufficient data exist on the safety features of the design to assess the analytical tools used for safety analysis over a sufficient range of normal operating conditions, transient conditions, and specified accident sequences, …

The following documents provide further guidance:

- RG 1.157, "Best-Estimate Calculations of Emergency Core Cooling System Performance," issued May 1989
- DG-1120, "Transient and Accident Analysis Methods"

According to 10 CFR 50.46 and Appendix K to 10 CFR Part 50, LOCAs are DBEs which may be analyzed either using conservative assumptions or best-estimate analysis. The requirements for analysis of LOCAs include a spectrum of break sizes and break locations sufficient to assure that ECCS cooling performance is calculated for the most severe LOCA.

The adequacy of the experimental database supporting the ACR-700 depends upon several concerns and parameters. First, the facilities used to obtain the data should be designed such that the processes of primary interest take place during the experiments, and the experiments should be conducted over the range of conditions appropriate for the ACR-700. Second, the experimental test facility should be correctly scaled to the full-scale prototype. Major distortions or nonprototypicalities in facility design should be assessed and accounted for when using the data. Finally, the test facility should be sufficiently well instrumented such that the experimental results are useful for model development and code assessment.

This section discusses the T-H experimental database and facilities likely to play a major role in supporting the ACR-700 design for LOCA analysis. The experimental database for neutronics is important in LOCA analysis for some transient scenarios. The data are necessary for validation of ACR-700 neutronics codes and methods, as discussed previously in FT3 and as discussed later for FT9.

The PIRT for T-H guided the adequacy review of AECL experimental data. As described in Section 3.7.3 of this report, the T-H PIRT was developed for a critical RIH break which results in stagnation or near-stagnation in fuel channels fed by the broken header. The PIRT panel, which was composed of experts in reactor safety and T-H, identified and ranked phenomena using two FoMs—cladding temperature and PT strain. In addition, the panel ranked the knowledge base for those processes with high or medium importance rankings in each of three periods of the scenario (blowdown, refill, and LTC). The knowledge base represented the PIRT panel's estimate of the cumulative set of experimental data that could be applied to developing models and understanding the phenomena. As part of the PIRT development, AECL presented results from its experimental programs and provided the panel with documentation on the test programs and test facilities.

Based on the information presented during the PIRT meeting, the expert panel identified several highly ranked processes for which the knowledge level was insufficient. This finding suggests that additional data are needed to develop the models and correlations necessary to analyze ACR-700 transients. The following table provides a brief summary of these phenomena or processes.

Table 3-14 T-H Processes Requiring Additional Data

Component	Thermal-Hydraulic Process
Fuel Bundle	Flow pattern and post-CHF heat transfer (including CHF, rewet, and dryout)
Fuel Bundle	Deformation, embrittlement, and fuel element failure
Fuel Bundle	Conditions leading to flow stagnation (including flow reversal, end fitting CCFL, and processes such as wall friction, form loss, and interfacial drag that make it difficult to predict flows with little or no driving head)
Fuel Channel	PT-calandria tube contact heat transfer
Fuel Channel	Deformation
Heat Transport System	Inlet and outlet header void distribution and phase separation (including inventory distribution, mixing/splitting, multidimensional effects, and gravity- and pressure-driven flows)
Heat Transport System	Feeder tube condensation and flashing
Heat Transport System	Break flow orientation (LTC period)
Heat Transport System	HTS pump performance (blowdown period)
Heat Transport System	Inventory/void distribution, flow reversal/stagnation, multichannel flow effects in HTS below headers
ECI System	Header interconnect flashing, refill, and flow
LTC and Containment	Sump performance and debris

For each of these processes, AECL should demonstrate an adequate experimental database with appropriate modeling. Not all the processes involve the CATHENA code, which is being reviewed under this FT. Sump performance and debris evaluations involve input to CATHENA

but are not directly calculated by the code. The degree to which CATHENA will evaluate fuel bundle deformation, embrittlement, and failure or fuel channel deformation CATHENA for the ACR-700 has not been established at this time.

AECL has performed a significant amount of experimental work in support of conventional CANDU plants that was intended to provide data for the highly ranked processes identified as having an inadequate database for the ACR-700. It is useful to note that several of the processes in Table 3-14 for the fuel bundle and fuel channel components were present in the cold water injection test (CWIT) experiments that we conducted for the current CANDU plants, and that others associated with the headers and feeder tubes were present in the large-scale header test (LASH) series for such plants. Integral tests in the RD-14M facility have been conducted to investigate natural circulation which may help resolve multichannel flow effects in the heat transport system (HTS). The following sections discuss these test facilities in further detail.

3.8.1.1 Integral Test Data

The ACR-700 core is substantially different from that of currently operating reactors in the United States. Analysis of core cooling of PWRs and BWRs can be performed, for the most part, on a one-dimensional basis. The ACR-700 geometry with horizontal orientation of the fuel channels raises issues associated with coolant distribution within channels and among channels. The most significant design feature from the T-H perspective is the use of a header-feeder arrangement to distribute coolant to the individual reactor channels.

The RCS flow in the ACR-700 is in a figure-eight arrangement. Each of two headers feeds 146 channels. The channels form five rows of offtakes from the header. Each row is 45 degrees apart and contains approximately 30 feeder offtakes.

Such an arrangement poses inherent scaling issues when considering a reduced-size integral test facility. The channels must be modeled essentially as full scale in the reduced-scale experimental facility, since any significant reduction in the size of the feeders would cause distortions in flow regimes. The remainder of the coolant circuit has a different scaling basis.

The RD-14 facility is full height and full pressure, and it is based on a reduced-volume scale of approximately 1:100. Only a limited number of channels can be represented. There have been two facility configurations. The RD-14 facility modeled only a single figure-eight pass. Therefore, the T-H system response was strictly one-dimensional. No multidimensional flows could be represented in the header.

The RD-14M facility was a modified version of the RD-14 facility. Instead of representing the core using 2 full-power, full-scale channels in a single figure-eight arrangement, the core was modeled by 10 reduced-scale channels. The RD-14M facility maintained the same total core power as was used for RD-14.

Plans are underway to return to a two-channel figure-eight arrangement similar to RD-14 to conduct RD-14/ACR tests. This program may be expected to provide information with respect to changes in the ACR-700 system design from the earlier CANDU design, including (1) experiments involving the outlet-header interconnection, which is responsible for reducing flow

stagnation in the broken part of the RCS, (2) experiments with accumulators instead of pumped high-pressure injection, (3) and emergency cooling injection only into the outlet headers.

This review is based on the limited information currently available:

- Appendix A, “Scaling: RD14M Design Rationale,” to AECL Report 108-126410-470-001, Revision 0

- Wright, D., et al., AECL Report 109US-03532-225-002, Revision 0, February 2004, on CATHENA calculations for an ACR-700 critical inlet header break

- Ingham, P., et al., AECL Report 108US-03500-225-003, Revision 0, February 2004, on RD-14M experimental results for a critical inlet header break, but for a CANDU system different from the ACR-700

- McGee, G., et al., AECL Report 108US-03532-225-001, Revision 0, on a comparison of CATHENA to the RD-14M experiment

AECL has submitted only a single RD-14M experiment (B9401), which was a 25-percent inlet header break, to the NRC staff for review. Therefore, the staff will request additional experimental data from the RD-14M test series.

The staff has two specific concerns with the RD-14M experiments that have been run to date. First, the applicant has not explored a spectrum of break locations. It has run only a single break location, at the end cap of a header. It has not run any experiments for breaks at other locations, such as feeders. The NRC staff generally finds it very useful to have integral test data at beyond-design-basis conditions. These tests generally examine multiple failure scenarios, and they demonstrate the robustness of the design and safety system. The staff has performed these tests in previous design certification reviews and found them to be instrumental in addressing facility distortions and code uncertainties.

Second, AECL has not measured break flow. Absent such information, it is difficult to perform code assessment. It is accepted practice that, when performing code calculations, the code calculation of break flow and the experiment be in close agreement. Otherwise, important local parameters cannot be compared, and the code calculation loses meaning because of the overriding importance of the ability to predict total system energy and inventory to enable the representation of local parameters.

Appendix A to AECL Report 108-126410-470-001, Revision 0, describes the RD-14 scaling. The description indicates that the facility is scaled primarily to maintain a likeness in total loop flows by considering driving force and loop resistance under natural circulation conditions. Certain local phenomena need to be considered, such as counter-current flow limitation (CCFL) in the feeder pipes and scaling of the end fittings to the channels.

The foremost scaling concern with RD-14M is the header-feeder geometry. The ACR-700 core consists of many parallel channels at different elevations feeding from different locations in the headers. On each end of the parallel flowpaths is a common pressure condition represented by the header. Under conditions of low driving force, such as postblowdown or natural circulation, this arrangement is inherently unstable and difficult to predict.

The ECC is introduced into the headers. When two-phase conditions develop in a header, cooling of a fuel channel is influenced by the elevation and location along the horizontal of its feeder connection on the header. When flow is out of the header, feeders connected near the bottom of the header receive water, while those connected higher up on the header may receive steam.

The dependence of flow resistance on void fraction introduces inherent, destabilizing feedback to flows in individual channels. Each channel can contain its own phase composition. Flow oscillations among channels, as well as reversals and flow stagnation, are likely, even when averaged channel conditions show net flow in one direction. At any moment in time, flow in some channels may be positive, some negative, and some passing through flow reversal. The possible combinations based on the axial and azimuthal location of the offtake, the length of the feeder tubes, the elevations of the channels, and the channel powers make each channel nearly unique.

It may nevertheless be possible to demonstrate that adequate core cooling can be maintained, but it is not clear that the RD-14M database is adequate at this time. Some means are necessary to show that the channel-to-channel range of variation can be bounded. Code calculations alone should not be relied upon. The PIRT panel ranked all phenomena associated with such effects high, and the state of knowledge low.

In particular, the relationship between the conditions in the header and channels is important in considering all those phenomena that can potentially affect the inflow and outflow of liquid to the channels. This integration is essential when the phenomena observed or considered in a local component or set of components have the potential for affecting the system FoM. This means that local phenomena that are related to generic systemwide conditions determine the outcome.

In the ACR-700, this functional integration is very important because experimental data (see AECL Report 108US-03500-225-003, Revision 0) show that flow stagnation in the channel will quickly result in fuel heatup and potential channel deformation. The results on page 5 of the RD-14M test results show that, in the critical 30 mm break, a short-lived flow stagnation of 3 second duration from 12 to 15 seconds on the plot results in a temperature increase in the cladding from 330 EC to 500 EC. This represents a heatup rate of almost 60 EC/s. With this temperature excursion, and keeping in mind that 600 EC temperatures may lead to the deformation of the channel, one can easily conclude that this issue of the scaling will need significant effort to resolve because of its impact and the margin available.

The integral facility does not address this fundamental aspect of the scaling problem because of objective constraints, as stated in the scaling evaluation presented (see Appendix A to AECL Report 109-126410-470-001, Revision 0). The RD-14M facility has 10 horizontal sections (i.e., channels) arranged in a figure eight with the four headers and two steam generators. The scaling of the individual channels is well documented, and the actual size of the facility approaches prototypical scale. This ensures a reasonable representation of the bottom-up channel phenomena.

Additional information is needed on the coupling of the channels with the header because little is said of the ability of the header to represent prototypical conditions. In principle, fast transients would be characterized by homogenous header conditions, as is the case with the

blowdown phase of large-break LOCAs. For very slow transients, stratification can develop, and uniform conditions may occur across the length of the header. For either of these two extremes, the RD-14M header may provide an adequate representation of the prototype. It is in the intermediate range of transients for which the header conditions in the RD-14M facility become questionable. Indeed, it is in the intermediate range that the conditions conducive to significant heatup occur, and in that range where the RD-14M facility might yield the least amount of useful information.

There appears to be a concern with RD-14M scaling. This concern relates to the duration of the blowdown phase, which was much shorter in the experiments than in the code calculations for a full-size plant. It is suspected that this may result from distortions of local volumes because of the binary nature of the facility scaling. This makes suspect direct inferences drawn from maximum clad temperatures in the RD-14M experiments, as the emergency core coolant comes in much earlier than in the full-scale system. AECL should address this discrepancy in a scaling study.

It is premature to draw conclusions on the adequacy of the RD-14M database because AECL has performed limited testing. A broad test matrix with particular consideration of the full range of boundary conditions at the header-feeder interface could provide a mapping of the range of conditions expected to occur in the full-scale prototype. It could provide information necessary to make the connection between system conditions (top-down) and local effects (bottom-up). This effort is yet to be undertaken, and its outcome is necessary to determine the adequacy of the experimental assessment basis.

In Appendix A to AECL Report 108-126410-470-001, Revision 0, the statement in the introductory remarks, “in any facility there will always be certain scaling parameters which must be compromised,” should be weighed against the importance of the phenomena related to the compromised scaling parameters. The channel heatup is the FoM. Scaling compromises that directly distort the FoM should be assessed and taken into account.

In Appendix A, Section 3, the section describing the headers outlines the importance of the header in the blowdown refill transient. The only scaling rationale offered is the fill or drain time. This is indeed necessary but not sufficient, because the distribution of the inventory in the full-size header needs to be accounted for by means of an appropriate range of testing conditions exploring the whole span of flow conditions possible in the full-size header. The statement, “lower flow velocities…enhance[s]…flow separation…is conservative in terms of feeder-connection uncovering,” is too broad.

AECL properly addressed additional detailed considerations, such as the analysis of the endpieces of the channel and their heat capacity. As previously mentioned, the scaling of the feeders and the channel is also well documented. Nonetheless, the boundary conditions between the feeders and the header are unresolved issues.

AECL should consider using the facility to better understand the range of conditions in the header that are conducive to sharp cladding heatups. It is through a significant effort in mapping that AECL can acquire the information necessary to draw conclusions as to the adequacy of the facility and, more generally, as to the inherent safety margins. The assessment of the safety margins is directly related to the level of uncertainty that can be

tolerated in the code calculations, and it introduces more or less stringent requirements on the overall code assessment process.

The use of the code to extract information from experimental results should be reviewed, since the basis for the assessment of the code needs to be established. The NRC currently knows of no basis to assess the code capability in predicting both the header-feeder interactions as well as the phenomena in the fuel channel. The RD-14 and RD-14/ACR facilities allow only one degree of freedom for the flow to pass though the core. Only one fuel channel is represented for each of the two core passes. However, flow through the ACR-700 core can be multidimensional. The RD-14M facility contains multiple channels. AECL should establish the basis for assuming that RD-14 and RD-14M simulate multidimensional effects of the ACR-700.

Predictions of LTC are expected to be particularly difficult for the code to model, especially at low pressures with large density ratios from fluid to vapor. The T-H codes tend to be unstable at low pressure because of rapid changes in void caused by flashing and condensation.

The scaling of the RD-14M facility needs to be better understood, in particular with regard to the ACR-700 tests that may be planned for the future. The blowdown phase scaling appears to give rise to a transient much shorter in duration before ECC injection begins in RD-14M than in the full-scale system. This gives rise to much milder clad temperature transients (in the sense of lower peak fuel clad temperature) in the experiments than in the full-scale system.

For these reasons, and because the peak cladding temperature for the ACR-700 approaches the region where PT deformation could be obtained, it is possible that some channels will experience significant fuel deformation and possibly PT deformation and sagging. This raises questions related to the coolability of the deformed geometry, both from internal flows between the headers and because of moderator cooling of the calandria tubes, which would ultimately come into contact with the PTs. The effects of flow stagnation in a channel in the various phases of an accident (e.g., blowdown, ECC injection and refilling, and LTC) warrant exploration.

3.8.1.1.1 Integral Test Assessments

The NRC cannot make a final conclusion with regard to the adequacy of existing RD-14 and RD-14M data at this time, pending completion of a comprehensive facility scaling evaluation. Instead, RD-14 and RD-14M scaling should be evaluated on the following bases.

<u>Fractional (pi-group) scaling</u>. Fractional scaling involves an evaluation of the global system response in terms of several dimensionless parameters (pi-groups) that are derived from mass, momentum, and energy conservation equations. The system is evaluated by transient period, taking into account parts of the system that are active. The objective is to ensure that the transient in the facility is appropriately represented in the full-scale prototype. Power-to-volume ratios should be considered, and system volume versus elevation needs to be examined to determine if gravitational driving heads are preserved.

<u>Preservation of flow regimes</u>. The adequacy of the piping diameters to preserve flow regime transitions as well as CCFL should be determined. This includes consideration of the Froude number.

Facility heat structures. Full-height, full-temperature facilities less than 1:100 volume scale suffer from significant distortions caused by heat loss or heat flux. Fast transients may be distorted by excessive heat flux from structures to the fluid. Slow transients may have the opposite problem of excessive heat loss to the environment. The scaling of heat addition from the metal structures to fluid in terms of heat capacity and area which affect the extent of heat flux distortions should be determined, as well as the distortions caused by heat loss to the environment.

Preservation of geometric similitude. How closely the isometric layout of the ACR-700 is maintained should be determined.

Preservation of loop flow. How well the facility preserves the combination of gravity driving forces and loop flow resistance for single- and two-phase conditions should be determined. Particular emphasis should be given to the multitude of competing flowpaths between headers.

Additional data may be needed to map the range of conditions that can occur in the header and translate this range to maintaining adequate coolant flows to all the channels. Some basis should be provided to determine the channel-to-channel variability that may occur. The following sections discuss facilities that may address this deficiency.

3.8.1.2 Separate Effects Test Data

AECL has performed a significant amount of experimental testing in support of conventional CANDU reactors. Revision 1.0 of AECL Report RC-1517, “A Phenomenology-Based Matrix of Tests for Use in Validation of Thermal-Hydraulics Codes Employed in CANDU Safety Analysis,” by J. Pascoe, et al., issued November 1995, summarizes many of these tests. The following section assesses the separate effects test data.

3.8.1.2.1 Header-Feeder Flow Distribution

The prediction of phase separation within the headers and of the flow to the fuel channels presents difficulties in analyzing transients in the ACR-700. Channels in which the flow stagnates, or which become starved of liquid from the headers, may rapidly dry out and heat up. Modeling the void distribution and phase separation at the feeder tubes is complex. This complexity is captured by the number of individual processes ranked high by the T-H PIRT panel for the header component. The PIRT panel ranked each header component as a highly significant process about which little is known at least once during the transient.

The NRC understands that AECL has data that were unavailable to the PIRT panel, including data from the LASH facility (see Buell, et al., 1999). The LASH facility was designed to provide experimental information for model development and code assessment. The LASH facility consists of 2 horizontal headers, each with 30 feeder tubes. The inlet and outlet headers are identical except for the axial nozzles. The headers are full scale, half length, and representative of headers in the Pickering Nuclear Generating Station. The LASH facility has been used to obtain experimental information on the variations in flows and void distributions among the feeders under steady-state and transient two-phase conditions (see Shin, 1987 and 1991).

Each header in the LASH facility has a total of 60 nozzles in 12 banks of 5 nozzles each. However, for the experiments, only 30 feeder tubes were connected to the nozzles, and 30

nozzles were blanked. The inlet and outlet headers were connected through feeder tubes and simulated fuel channels. The combined K-factor for each feeder and simulated fuel channel was representative of a 600-MW reactor feeder/channel system. Twenty of the channels were instrumented with turbine flow meters, and 20 single-beam gamma densitometers were installed on the inlet feeders close to the header. Fluid level in the headers was inferred using combined level probe assemblies consisting of a differential pressure rake, a thermocouple rake, and a conductivity rake.

It may be possible to address some of the data concerns of the PIRT panel using existing LASH experimental data or by performing additional tests in the LASH or other similar facility. It is our understanding, however, that a predictive modeling capability for header flow distribution does not currently exist. Scaling of the LASH facility to the ACR-700 should be demonstrated. Although the LASH facility is full scale, the K-factors used in the design of the facility were for a conventional CANDU system. In addition, the ACR-700 will include an outlet header interconnect. The effect of this interconnect was highly ranked for the ECI system, with low knowledge level. Existing LASH experiments do not include the effect of the interconnect. LASH only represented top header breaks, which is the most benign location. Other break locations should be included. AECL should consider additional tests in a modified LASH facility as a means to address these concerns.

3.8.1.3 Horizontal Fuel Bundle Data

The CWIT facility was designed to provide experimental data on the T-H behavior of a simulated CANDU fuel channel/feeder system. AECL has performed various experiments with this facility to study the behavior over a wide range of postulated accident conditions, such as ECI following a LOCA, and fuel cooling in the absence of forced flow. Thus, the facility has essentially been used to study the last two phases of a LOCA transient, as listed in Table 3-14 in this report.

The CWIT facility consists of two horizontal headers (inlet and outlet) connected to two vertically separated, horizontal fuel channels by inlet and outlet feeder piping. An injection system directs liquid flow into one or both headers, and a blowdown system simulates header breaks. The facility is designed with 5-m vertical separations between the top and bottom channels and also between the top channel and the header centerline. These vertical distances correspond to the elevation differences between the bottommost channels, the uppermost channels, and the header centerlines in CANDU reactors. In performing experiments, single-channel tests may be conducted by blocking off one of the parallel channels.

The headers are nonprototypical of CANDU reactors and are made from 1200 mm (48 in.) lengths of 254-mm (10-in.) schedule 120 pipe. Feeder connections are 45 degrees downward. Vertical feeder connections are present but have not been used. The feeders themselves consist of a combination of nominal 50-mm (2-in.) and 75-mm (3-in.) schedule 80 pipe. The pipes are arranged as horizontal and vertical pipe sections connected by 90-degree elbows, which is nonprototypical of CANDU feeders. Electrically heated fuel element simulators (FES) have been manufactured to closely represent the geometric and physical properties of CANDU-6 reactor fuel. To this end, the fuel bundle consists of 37 elements and utilizes wear pads, interelement spacers, and simulated endplates. The facility uses an actual reactor PT and end fittings prototypical for the CANDU-6. The PT is surrounded in a square channel enclosure that can be flooded with water to simulate the moderator heat sink.

With regard to instrumentation, the CWIT uses more than 300 instrument channels. These channels are divided among four major types of instruments—thermocouples, absolute and differential pressure transducers, orifice and venturis flow meters, and gamma ray densitometers. These instruments are used to measure temperatures, pressures, flow rates, and fluid densities, respectively.

A concern in ACR-700 reactor transients is the integrity of the fuel element cladding and the PT. Thus, phenomena occurring in the fuel channels and feeder tubes during reactor transients have a high significance when they contribute to temperature-induced cladding or PT failure. Moreover, to the extent that such highly ranked phenomena are poorly understood, their effects are difficult to quantify, and increased safety margins are therefore warranted. To identify such phenomena, the PIRT panel ranked both the significance of and the knowledge level about various processes occurring in the three main periods of a LOCA transient (i.e., blowdown, refill, LTC). Since insignificant processes are not safety concerns, and since well known processes are not research topics, safety-related engineering research naturally focuses on significant processes which are poorly understood. Accordingly, processes ranked high by the PIRT panel represent an information need to support the experimental database, and additional work is likely necessary.

In determining the adequacy of the database obtained from the CWIT facility, consideration should be given to the primary and secondary FoMs. The primary FoM in development of the T-H PIRT was the temperature response of the cladding, and the secondary FoM was the high-temperature strain of the PT. Accordingly, in various components of the system, phenomena are deemed important based upon their potential to induce high-temperature cladding or PT failure.

The applicability of the data from CWIT for application to the ACR-700 depends on the facility scaling and range of T-H conditions. Therefore, geometric differences between CWIT components and those in the ACR-700 are relevant. The CWIT is intended to be a full-scale feeder/fuel channel test facility for CANDU reactors. As such, it is scaled for the CANDU-6, not for the ACR-700. Since the ACR-700 is anticipated to run under different operating conditions than CANDU reactors, the test matrix of previous experiments may not be sufficient for the ACR-700. Assuming that the goal is still to simulate separate effects at full scale, physical dimensions and operating conditions should both be adjusted to bring the facility scaling in line with the ACR-700. The CWIT was not designed to simulate the blowdown period of a LOCA, but only the two later periods (i.e., refill and LTC). As a result, existing CWIT data are not adequate for the blowdown period and for the several processes identified by the PIRT as having a low knowledge level.

Scaling considerations of CWIT components follow.

<u>Feeder tubes</u>. According to AECL Report WNRE-773, the quenching and refilling of the feeder tubes provide the most significant time delay for the cold ECI fluid to reach the fuel channels. In terms of the primary and secondary FoMs, this phenomenon raises cladding and PT temperatures by delaying the arrival of the cooling fluid. As the incoming liquid quenches the feeder tubes, the generated steam runs counter to the downflowing liquid, thereby impeding its progress. Accordingly, the feeder tubes in the test facility should be of prototypical size, shape, thickness, and material so that the feeder heat storage and the CCFL are properly scaled. Moreover, since this heat storage is a function of temperature, the tubes and the rest of the

system should be preheated to prototypical temperatures to yield a correct value of the refill time delay.

Headers. According to AECL Report WNRE-773, the total refill times for parallel-channel tests are generally shorter than or equal to those of single-channel tests, and the two parallel channels usually refill in opposite directions. Since shorter refill times positively impact the cladding and PT temperatures, this occurrence is significant in terms of the primary and secondary FoMs. AECL Report WNRE-773 offers an explanation for this phenomenon on page 14. It is speculated that the commencement of refill in one of the channels causes a pressure upset in the headers, which induces an oppositely directed refilling of the second channel. Assuming the plausibility of such an explanation, the question arises as to whether the use of nonprototypical headers distorts this phenomenon.

Feeder end fittings. According to AECL Report CANDEV-85-07, two-phase natural circulation flows through the fuel channel are marked by periodic oscillations. As boiling commences, the flow stratifies. However, because the water level is initially above the end fitting/feeder connection, the generated steam cannot escape from the fuel channel. As a result, the bubble grows until the water level drops below the end fitting/feeder connection level. At this point, the two-phase flow is flushed out of the system, and subcooled liquid enters to begin a new cycle. Since such cyclic drying out of a particular fuel region impacts cladding and PT temperatures, prototypical end fitting/feeder connections should be used to capture this effect.

Fuel bundles. Since the fuel design of the ACR-700 differs from that of the CANDU-6, the need for geometric similarity necessitates changes in the facility's FESs. After all, the bundle geometry will impact flow patterns and heat transfer. Moreover, since CHF is anticipated to occur by way of dryout rather than DNB, the flow pattern influence of the bundle geometry is most germane to the primary and secondary FoMs. Furthermore, since deformation (e.g., slumping, sagging, ballooning) of the fuel bundles or PT would alter the flow patterns and heat transfer, such changes have great potential to impact the cladding and PT temperatures. Therefore, a means should be established of simulating this deformed geometry if such conditions are to be analyzed for the ACR-700 using CATHENA. For instance, because fuel element slump and deformation are of concern, the test rod bundle should be designed so that rods can be reconstituted and the bundle rebuilt. The reconstituted bundle should include rods with an assumed distortion (i.e., with some rods slumped or in a more compact array shifted lower in its PT) to estimate the potential degradation of heat transfer within the reconstituted bundle, change in CHF relative to the undeformed bundle, and effect on flow patterns. With regard to the stored energy of the rods, the PIRT panel deemed this effect to be significant during the blowdown phase alone, and not for the subsequent phases of a LOCA. Hence, if the facility is intended for the ECI and LTC phases alone, there is greater flexibility in choosing the rod materials because it is then unnecessary to scale the stored energy. However, if the facility is also intended to simulate the blowdown phase, then the stored energy should be properly scaled, or the scaling analysis must address its distortion. Regardless of which option is chosen, the rods should be able to operate at anticipated steady-state and transient temperatures. At these elevated temperatures in a liquid deficient environment, the radiant heat transfer between the rods and the PT becomes significant and alters the temperature distribution within the fuel channel.

Pressure tube. With respect to the primary and secondary FoMs, a prototypical PT is needed for several reasons. First, as mentioned above, radiant heat transfer between the fuel bundle

and the PT will become significant during the refill (ECI) phase when the temperatures are elevated and an intervening liquid is absent. Thus, a PT of the proper material and thickness should be used to provide a prototypical temperature response from which to assess the PT strain. Second, since the energy stored by the PT during the ECI phase will be released to the fluid during the LTC phase, a prototypical PT tube is needed to match the temperature response and flow patterns of the coolant at the commencement of natural circulation. Finally, since PT deformation (i.e., sagging or ballooning) will impact the flow patterns and hence the dryout heat transfer, a means of simulating deformed PT geometry is needed if these conditions are to be analyzed for the ACR-700 using CATHENA.

3.8.2 Experimental Database Assessment Summary

While AECL has performed a significant amount of testing that supports conventional CANDU reactor systems, the adequacy of that database for successful completion of design certification for the ACR-700 has not been established. The NRC staff's review of AECL test programs and database, supported by an independent study by a panel of outside experts, determined that the experimental database proposed by AECL for the ACR-700 needs enhancement in several areas.

Header-feeder inventory and flow distribution. The NRC does not consider the distribution of inventory within the headers, and the separation of phases that occurs as the flow enters the feeder tubes, adequate to support model development, code assessment, or identification of limiting flow conditions, based on information provided during the preapplication review. Of particular concern is the need for a scaling rationale so that results from facilities such as the LASH and RD-14M can be extended to the full-scale ACR-700.

Horizontal fuel bundle T-H. Horizontal fuel bundle T-H refers to several physical processes that are important in determining the two-phase flow pattern, void distribution, and heat transfer in fuel channels undergoing a LOCA transient. It may be necessary to obtain additional information on channel void formation and void distribution during the first couple of seconds during blowdown to evaluate the affect of void coefficient on local core power prior to control rod insertion. AECL must establish the applicability of the CWIT facility data using a 37-element bundle to 43-element CANFLEX fuel during the design certification review. The NRC did not identify any information as appropriate for heat transfer and two-phase flow in slumped or degraded fuel channels. This may become important in assessing code performance for some LCDA transients, such as SFBs and partial blockage events.

Integral test data. The RD-14M facility and its variations (RD-14 and RD-14/ACR) provide data useful to understanding the behavior of critical header breaks in a CANDU reactor system. Data from these tests are useful for code assessment and may provide information sufficient for model development for some phenomena. AECL should establish a scaling rationale demonstrating that the RD-14M results apply to the ACR-700 during the design certification. In addition, AECL did not make break flow measurements. Without an accurate estimate of the break flow, code assessment simulations of RD-14M tests will be difficult. The tests conducted in RD-14 and RD-14M did not include a wide range of break locations or scenarios that involve multiple failures, such that robustness of the design can be demonstrated. The experimental database does not extend to long-term cooling. The RD-14/ACR design is a return to a single

pass configuration. The minimum number of channels necessary to properly represent the range of core behavior has not been established.

3.9 Quality Assurance

Appendix B to 10 CFR Part 50 deals with QA criteria for nuclear power plants. Section 2 of DG-1120 discusses the application of Appendix B to computer code development. The NRC staff has requested that the applicant address Appendix B requirements and DG-1120 guidance and show that they are met for the computer codes being reviewed under FT3. Section 5.6 of this report provides a more detailed discussion of the staff's assessment of AECL Quality Assurance Program.

3.10 Focus Topic 3 Conclusions

The ACR-700 preapplication review provides feedback, staff observations, requests for additional information, and identification of any regulatory, policy, and technical issues that AECL should address during the design certification phase to allow the NRC to complete the ACR-700 design safety determination.

The NRC staff has issued RAIs to AECL that address the technical issues needing resolution for the computer code review (see letter from J. Kim of the NRC to V.J. Langman, dated May 14, 2004). In this transmittal, the NRC staff requested the applicant to follow the orderly process for preparing applications for approval of computer codes as outlined in DG-1120. The staff has also asked for specific information concerning computer code user options and modeling assumptions that will be used in the safety analysis of the ACR-700. These responses will best be addressed as part of the evaluation model documentation that is described in the DG. The staff asked for additional information concerning the validation of the computer codes against experimental data. The applicant has already performed extensive code validation. Some of the applicant's code validation indicated the need for additional experimental data comparisons. This additional validation can best be addressed as part of the code assessment documentation described in the DG. A significant feature of the code assessment process is the development of PIRTs. The applicant has already developed a PIRT for large-break LOCA. This PIRT should be reconciled with the somewhat different conclusions from an NRC staff-sponsored panel. The applicant should also develop PIRTs for other code uses anticipated for ACR-700 safety analysis. The PIRTs can then serve as the basis for a comprehensive process of code validation as discussed in the DG.

At this time, the staff has not identified any regulatory issues, such as rules, rulemaking, or exemptions that need resolution before the staff's completing review of the FT3 computer codes. In addition, the staff has not identified any policy issues that warrant upper management or Commission guidance for resolution in completing the review for the FT3 computer codes.

4. SEVERE ACCIDENT DEFINITION AND ADEQUACY OF SUPPORTING RESEARCH AND DEVELOPMENT (FT4)

4.1 Review Scope

In this chapter, the staff performed its preapplication review of the 700 MWe ACR-700 design for the severe accident definition and adequacy of supporting severe accident research and development (R&D) programs. The staff identified key severe accident technical issues, in the appropriate regulatory context, which should be addressed in the ACR-700 standard reactor design certification review. AECL should provide a proposed severe accident R&D program to assist the staff in performing its review of the ACR-700 standard reactor design certification.

AECL has provided information on the ACR-700 design, expected behavior during hypothetical accidents, information on its current experimental database, and its proposed future reactor accident research programs applicable to the ACR-700. The staff reviewed the following documents provided by AECL:

- ACR 108-126810-LS-001, "Phenomenology for Limited and Severe Core Damage Accidents in an ACR," September 2003
- ACR 108-126810-LS-002, Revision 0, "ACR Limited and Severe Core Damage Accidents: Supporting R&D," November 2003
- ACR-700, 10810-03660-ASD-005, "Review of Design Features for Severe Accident Management," January 2004
- AECL Presentation, "Introduction to PIRT Expert Panel," October 2003
- AECL Presentation, "PIRT Expert Panel Meeting on ACR-700," December 2003

4.2 Regulatory Issues

In its review of the ACR-700 preapplication severe accident definition and the adequacy of supporting severe accident R&D for the ACR-700 design, the staff considered the following regulatory guidance:

- "NRC Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants," August 8, 1985
- SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light Water Reactor Designs," and corresponding staff requirements memorandum (SRM) dated July 21, 1993, as they discuss the staff technical and policy issues pertaining to evolutionary and advanced LWR design certification, including severe accident preventative and mitigative feature issues
- SECY-96-128, "Policy and Key Technical Issues Pertaining to the Westinghouse AP-600 Standardized Passive Reactor Design," June 12, 1996, as it relates to external reactor vessel cooling

An SCDA (severe accident) is a low probability event beyond the most limiting DBA. For light-water-cooled nuclear power reactors, the most limiting DBA is defined as a major reactor accident that would result in potential hazards not exceeded by those from any accident considered credible. In 10 CFR Part 100, such accidents have generally been assumed to result in subsequent meltdown of the core followed by the release of appreciable quantities of fission products. In SECY-94-302, the staff assumed that the core remains in the reactor vessel and containment is intact for DBA radiological consequence analyses. The DBA does not assume either reactor vessel failure or core-concrete interactions. In contrast, severe accidents involve sequences of successive multiple failures resulting in substantial core melt with and without the failure of reactor vessel and containment.

For severe reactor accidents, the NRC has requirements to address anticipated transients without scram (10 CFR 50.62), station blackout (10 CFR Part 63), and combustible gas control (10 CFR 50.44) for light-water-cooled nuclear power reactors; however a definitive set of regulatory requirements for addressing specific severe accident phenomena does not exist. Instead, the Commission has developed guidance and goals for addressing safety issues related to reactor accidents more severe than DBAs (see SECY-93-087).

4.3 Potential Policy Issues

An interoffice working group is developing a recommendation for probabilistic event selection. The staff has proposed a severe accident research plan that would support adequate technical review of LCDAs regardless of the FSAR classification. The plan is intended to develop independent capabilities for the staff to analyze the ACR-700 for Chapter 15 DBA and Chapter 19 severe accident scenarios, and to provide the basis for making sound regulatory decisions involving the review of the ACR-700 standard reactor design certification.

4.4 Technical Issues

The ACR-700 design is unique compared to conventional LWRs. Some of the most notable characteristics of the ACR-700 design are modular horizontal fuel channels, on-power fueling, SEU fuel with light-water cooling, and the separate low-pressure, low-temperature, and heavy-water moderation with relatively low neutron absorption. These unique design characteristics pose a challenge to the staff's current ability to analyze postulated ACR-700 reactor accident scenarios.

To support its identification of the physical processes most important in the analysis of ACR-700 reactor accidents, the staff conducted a severe accident analysis PIRT. The PIRT process identified a number of key severe accident technical issues that must be addressed, including the following:

- melt progression through pressure and calandria tubes in horizontal geometry
- creep of PTs during whole-core degradation
- pressurized expulsion of melts from the PT into the calandria vessel through the calandria tube

- flowpaths, flow splits, and flow instabilities during severe accident progression
- containment loading and failure modes
- source terms during fuel interaction with pressure and calandria tubes

The PIRT process also identified potential deficiencies in the experimental database used to validate the analysis codes.

Melt progression in horizontal geometry is different from that in the vertical geometry found in LWRs. Initial and long-term melt progression will cause pressure and calandria tubes to fail through sagging and other contact modes, as opposed to candling and clad ballooning modes in LWRs. The melt progression will allow slumping of rows of severely degraded PTs onto nondegraded tubes vertically below. Slumping causes flow reduction and creates a hot spot on PT and calandria tube surfaces, leading to severe core degradation. Extended melt progression information in this geometry is not well characterized in comparison with the database for melt progression in LWRs. Heat transfer processes and T-H, in particular those associated with flow stratification and flow splits in headers and manifolds, as well as CHF in horizontal geometry, need better characterization. Fundamental heat transfer processes, in principle, are not complicated. In addition, creep of PTs affects cooling and can bring these tubes into contact with calandria tubes. Limited data exist on heat transfer from creeping tubes during whole-core degradation. International research products could be reviewed for possible application.

Pressurized expulsion of melts from the PT into the calandria is the key phenomenon that may take a single channel failure event and propagate to whole-core failure. AECL is actively performing experimental research on forced fuel-coolant interaction (FCI) modes with chemical augmentation. AECL has standalone parametric unqualified models to address the phenomena, which will require verification based on the results of the planned experiments. Moreover, mechanistic models may be needed to provide scaling of loads and energetics.

Flowpaths dictate the ability to remove heat and to carry fission products through the RCS and into the containment or, in the case of bypass accident sequences, to the environment. The complicated geometry of the ACR-700 leads to uncertain flow splits in a parallel flow regime, with possible flow instabilities, complex flow patterns, and possible PT failures.

The ability to retain molten core debris within the calandria vessel is an important aspect of accident progression in the ACR-700 design. This is an area of significant uncertainty. Similar to the position taken with regard to the AP600 and the AP1000, and articulated in SECY-96-128, the staff will require AECL to use a balanced approach, involving reliance on in-vessel retention of the core complemented with limited analytical evaluation of ex-vessel phenomena.

Containment failure modes influence the severe accident fission product source terms and, in that sense, a reliable assessment of containment loadings and failure modes is important to estimate the source terms. Existing NRC code (ORIGIN) should be adequate to predict inventories of fission products in the fuel and fuel cladding gap. Most of the phenomena affecting releases of fission products from the fuel during reactor accidents in the ACR-700

reactor will be very similar to phenomena affecting fission product release in reactor accidents at LWRs. Fission product release associated with core debris interactions with concrete should also be similar, at least qualitatively, to that pertaining to LWRs. Existing fission product release models in NUREG-1465 can be modified to predict fission product release during the gap phase release of an ACR-700 reactor accident and the continued progression of an accident up to the point that core debris interacts with PTs and calandria tubes.

Fission product release prediction during periods of fuel interaction with pressure and calandria tubes is complicated by a lack of data. No data or models exist that are well suited for source term prediction during coolant injection into a channel with extensively degraded fuel. For source term perspective, the staff believes that it is useful to consider the point of extensive core debris interactions with PTs and calandria tubes as marking the boundary between DBAs and severe accidents. Finally, iodine chemistry in the containment, particularly the formation of gaseous organic iodides and the gas phase decomposition of gaseous iodides, must be considered. The cesium iodide aerosol behavior and transport in the ACR-700 containment should be similar to that in LWRs.

Aside from the above key issues, minor to moderate modifications to MELCOR will be needed in some areas, including steam oxidation kinetics of PTs, air oxidation kinetics of PTs, fuel and cladding creep, graphite/CANLUB interactions with fuel and cladding, calandria vessel failure caused by thermomechanical loadings, T-H modeling of PT headers and manifolds with flow stratification, post-CHF heat transfer in distorted geometries, fuel channel boiloff, bubble dynamics, condensation of steam in moderator, flow splits in a parallel flow regime, empirical correlations for heat transfer from molten core debris to the adjacent steel vessel, and CHF from the steel vessel to surrounding water in the shield tank. Some of the latter ones will require data from T-H experiments and, as such, should be considered within the scope of T-H analysis of the ACR-700.

4.5 Conclusions

Based on the gaps between the current knowledge base and the information required to address pertinent regulatory issues for the ACR-700, AECL should address the following ACR-700 severe accident technical issues in order for the NRC staff to reach satisfactory conclusions for design certification.

- Deterministic treatment of severe accident phenomena including melt progression in a horizontal geometry, pressurized melt ejection, in-vessel (calandria vessel) retention, and ex-vessel coolability, fuel-coolant interaction, and core-concrete interaction.

- Fission product and aerosol behavior in primary heat transport system and containment including transport, deposition, and retention mechanisms in ACR-700 specific geometry.

- Iodine chemistry in containment, particularly the formation of gaseous iodides.

- Containment loading and performance under severe accident conditions.

- Severe accident code development, verification, and validation.

The staff plans to perform the following activities in order to develop independent capabilities for safety analysis of the ACR-700 design.

The severe accident code, MELCOR, will be modified as part of the RES ACR-700 severe accident research program to model the unique ACR-700 configuration. Current flowpath models in MELCOR will be modified to account for complex flowpaths, with guidance for the flow splits to be provided by T-H analysis. Sagging and slumping models may exist in open literature or in AECL-developed codes. These models will be reviewed for their efficacy and for possible adoption in MELCOR. Modifications will be made also to account for PT creep, noting, in particular, the new cladding material. Canadian data on cladding may be used for material properties data. Minor to moderate modifications to MELCOR will be performed to account for steam/air oxidation kinetics of PTs, graphite/CANLUB interactions with fuel and cladding, calandria vessel failure caused by thermomechanical loadings, and fission product interaction with catalytic recombiners. The MELCOR development effort will be supplemented by T-H modeling of PT headers and manifolds with flow stratification, post-CHF heat transfer in distorted geometries, fuel channel boiloff, bubble dynamics, condensation of steam in moderator, and flow splits in a parallel flow regime. The scope of T-H analysis of the ACR-700 includes these latter ones.

AECL will perform pressurized melt expulsion experiments (MFMI), and the resulting data will be used to investigate the FCI energetics. This work is the most important, in that it will determine whether a single channel failure has the potential to propagate to multiple channels leading to whole-core failure. The FCI codes used by the NRC (e.g., TEXAS) will be validated with the MFMI experimental data so that they can subsequently be used for independent verification of the ACR-700 FCI calculations. Although the FCI calculations will be performed outside the scope of MELCOR assessment, MELCOR results will be used as initial and boundary conditions for such calculations.

The modified MELCOR code will be used for analysis of two types of accident scenarios. The first type is an SCDA in which a global power/cooling mismatch initiated by a LOCA with loss of ECC and other cooling systems may lead to multiple channel failure, calandria, and possibly containment failure. The MELCOR calculations will focus on demonstrating the efficacy of ACR-700 mitigative design features on core debris coolability. The second type is a LCDA in which a local power/cooling mismatch at full power may lead to flow blockage in a single channel and consequent heatup and melting of fuel in the channel.

The ACR-700 mitigative design features are expected to prevent the single channel failure from propagating to adjacent channels. The MELCOR code will analyze this particular accident event using T-H output from TRACE single channel calculations. In another class of LCDA events, the feeder stagnation breaks while the intact channel geometry is maintained. This class of events will be handled within the scope of the ACR-700 T-H analysis using the TRACE code.

5. CANADIAN DESIGN CODES AND QUALITY ASSURANCE STANDARDS (FT6)

5.1 Review Scope—Design Codes

AECL identified the issue of codes and standards as one of the FTs for preapplication review. As a part of the ACR-700 preapplication plan, AECL submitted several reports to the NRC staff related to the use of Canadian codes and standards in the ACR-700 design. These reports included 108US-03621-LS-002, Revision 0, "Canadian Codes and Standards Used in CANDU Plants," issued March 2004. In addition, two key FTs—On-Power Fueling and Class 1 Pressure Boundary—included some discussion on the use of Canadian codes and standards, albeit to a much lesser extent.

The staff does not have any substantive information at this time to reach a final conclusion regarding the acceptability of the use of Canadian codes and standards. However, in its limited review of the list of Canadian codes and standards used in CANDU plants and the ACR-700 Class 1 pressure boundary and on-power fueling machine reports, the staff has identified several issues involving the use of Canadian codes and standards in the design of the ACR-700. This document discusses the technical, regulatory, and policy issues identified to date related to the use of Canadian codes and standards. The NRC staff has also addressed the use of Canadian codes and standards for the design of the ACR-700 Class 1 pressure boundary and on-power fueling machine in those respective FTs.

5.2 Regulatory Issues—Design Codes

In its preapplication review, the NRC staff identified several regulatory issues that involve use of Canadian codes and standards in the design of the ACR-700 plant. The first issue is the use of Canadian codes and standards for component designs that are not addressed by the ASME Code. The ACR-700 Class 1 pressure boundary design and on-power fueling machines are unique to CANDU reactors and are not found in U.S. nuclear power reactors. For the design of these CANDU-unique components, AECL plans to supplement the ASME Code, Section III, design rules with Canadian codes and standards when the ASME Code rules are not applicable, might not exist, or might be insufficient to address the ACR design. The staff has considered various options for its review of these Canadian codes and standards. For example, the staff considered reviewing each Canadian code and standard for technical adequacy. This approach, however, would involve a significant amount of time and technical staff resources. The staff also considered requesting ASME (or another U.S. standards development organization) to review and adopt the Canadian codes and standards. This approach would also entail a significant amount of time because of the relatively slow code consensus process. The NRC staff also considered requesting AECL to provide justification and a detailed comparison of Canadian codes and standards with U.S. codes and standards, or perhaps take credit or partial credit for the review and approval of Canadian codes and standards by the Canadian Nuclear Safety Commission (CNSC). This last option would likely raise a policy issue requiring Commission approval.

The second issue is the extent to which design and construction use Canadian codes and standards. The use of Canadian codes and standards is not limited only to the Class 1 pressure boundary and on-power fueling machines. The extent to which AECL plans on using

Canadian codes and standards for other design areas, such as building structures, electrical systems, and safety classification of components, is not clear to the staff. The NRC's SRP and RGs reference many design codes and standards (e.g., Institute of Electrical and Electronics Engineers (IEEE) standards, American Concrete Institute (ACI) codes, ANS standards). If Canadian codes and standards are used for design, the design requirements must be compatible with construction practices. It is unclear to what extent AECL will propose Canadian codes and standards for the construction, fabrication, inspection, and testing of plant SSCs and whether they will be compatible with U.S. design codes and standards.

To address the challenges described in the above issues, the NRC staff proposes to establish an approach for the review of Canadian codes and standards that would result in an efficient and effective process without sacrificing the need to ensure that Canadian codes and standards are technically sufficient and provide an acceptable level of quality and safety. The review would involve a tiered approach to ensure that the extent of review is commensurate with the regulatory significance of the particular code or standard. The approach would use existing processes established in the regulations and is consistent with the approach previously used in the review of nuclear power plant applications for operating licenses and design certifications. The approach, as presented in the following paragraphs, also discusses the type of information the NRC staff will need from AECL in order to review the acceptability of using Canadian codes and standards in the design and licensing of the ACR-700 in the United States, as well as the extent to which the NRC staff will review this information to determine its acceptability.

The top tier codes and standards are those that are incorporated by reference into 10 CFR 50.55a, "Codes and Standards." These codes and standards include the ASME Boiler and Pressure Vessel Code (Sections III and XI), the ASME Code for Operation and Maintenance of Nuclear Power Plants (ASME OM Code), and IEEE Standard 603, "Criteria for Protection Systems for Nuclear Power Generating Stations." Those ACR-700 systems and components that fall within the scope of these U.S. codes and standards must meet the requirements specified in the ASME Code and IEEE standard. However, AECL may propose alternatives to the design requirements of 10 CFR 50.55a, pursuant to 10 CFR 50.55a(a)(3). These alternatives may use some or all of the provisions in Canadian codes and standards. In proposing alternatives, AECL must submit the alternative, including the proposed Canadian code and standard, to the NRC and describe which portions of the ASME Code or IEEE standard it will not meet, as well as which provision(s) in the Canadian code and standard it proposes to use in lieu of meeting the ASME Code or IEEE standard. AECL must demonstrate that the alternative provides an acceptable level of safety, or that compliance with the specified ASME Code requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. The staff's review would identify and resolve any problems in the use of the Canadian standard and ensure that the proposed alternatives to the regulations provide an acceptable level of quality and safety.

Similarly, when an ACR-700 system or component is found to be within the scope of the ASME Code or IEEE standard, but the requirements in the ASME Code or IEEE standard do not apply, might not exist, or might be insufficient to address the particular aspect of the ACR-700 design, then AECL should provide the applicable portions of the Canadian code or standard that it used for the design, including any specific technical design methods and acceptance criteria it plans to use. AECL should provide an evaluation of the acceptability of the system or component in conjunction with its design certification application. The NRC staff will review this information and evaluate the adequacy of the Canadian code or standard, including the design

method and acceptance criteria, to the extent necessary for the staff to reach a final conclusion on all safety questions associated with its use. The staff may require the applicant to complete certain aspects of the design using the Canadian code or standard and have that information available for audit if such information is necessary for the NRC staff to make its safety determination.

A Canadian code, standard, or design method might be found to be within the scope of 10 CFR Part 50, other than 10 CFR 50.55a, but not meet the specified U.S. regulatory requirement. If such exceptions to the regulations exist, then AECL must submit to the NRC a request for an exemption pursuant to 10 CFR 50.12 in conjunction with its design certification application. The staff will review the exemption request as part of the design certification review.

There are many areas in a nuclear power plant design for which U.S. regulations do not specify the use of a particular code or standard. Although the regulations might not require a specific code or standard, one might be referenced in NRC staff guidance documents. For example, U.S. regulations do not incorporate codes and standards addressing the design of nuclear power plant buildings by reference or explicitly require them. However, RGs reference them and the NRC SRP specifies them as methods acceptable to the NRC staff for satisfying regulatory requirements. When submitting the design certification application for the ACR-700, AECL should identify the Canadian equivalent codes and standards to those U.S. codes and standards that are referenced in NRC guidance documents, submit those Canadian codes and standards to the NRC, describe how the Canadian code and standard deviates from NRC guidelines, and justify why the Canadian code or standard is acceptable. The NRC staff will review the submitted information, including the Canadian codes and standards, to the extent necessary to reach a safety determination of the technical adequacy and sufficiency of the Canadian code or standard and to reach a final conclusion on whether the use of the Canadian code or standard will result in an acceptable design.

Lastly, numerous codes and standards are used in nuclear power plant designs that are not required by regulations nor referenced in NRC guidance documents. Because no regulatory requirements or guidance exist, the Canadian equivalent to these codes and standards may be used without the need for NRC staff review and approval. The applicant should identify these Canadian codes and standards in design and procurement specifications, as well as in construction and installation specifications.

5.3 Potential Policy Issues—Design Codes

In SECY-03-0047, “Policy Issues Related to Licensing Non-Light Water Reactor Designs,” dated March 28, 2003, the staff discussed in Issue 3 how NRC requirements for non-LWR designs relate to the use of international codes and standards. Although the ACR-700 is not considered a non-LWR design, the policy issue discussed in this paper directly applies to the use of Canadian codes and standards for the ACR-700. In SECY-03-0047, the staff described two options for the NRC’s involvement in international codes and standards. In one option, the staff would review international codes and standards only as part of an application or preapplication review. This option would have NRC review international codes and standards only as necessary to evaluate a licensing or preapplication submittal which incorporates such standards. The staff, however, recommended to the Commission a second and more proactive approach that would involve NRC staff participation in the identification, development, and

endorsement of international codes and standards where applicants or preapplicants have identified such standards as necessary for use in their submittals. The Commission, in its SRM dated June 26, 2003, disapproved the staff's proactive approach to participate in the development and endorsement of international codes and standards and, instead, directed the staff to review international codes and standards only as part of an application or preapplication review of non-LWRs, then apply the lessons learned from these reviews to their activities involving U.S. domestic codes and standards committees.

The staff's approach to reviewing Canadian codes and standards, as described in the regulatory issues above, is consistent with the Commission's SRM. The staff has identified no new policy issues in the use of Canadian codes and standards for the ACR-700 design.

5.4 Technical Issues—Design Codes

For the design of safety-related piping systems in the ACR-700, AECL proposed to use the 2001 edition up to and including the 2003 addenda of ASME Code, Section III. It should be noted that the NRC staff has not yet incorporated by reference the 2001 edition up to and including the 2003 addenda into 10 CFR 50.55a. The proposed rule issued in the Federal Register (69 FR 879) included several modifications and limitations on the use of the 2001 edition up to and including the 2003 addenda for piping seismic design in paragraph NB-3000 of the ASME Code. AECL must meet any modifications and limitations on the 2001 edition up to and including the 2003 addenda related to piping design when the final rule is issued later this year. In addition to using the ASME Code, AECL noted that it will use Canadian standards where ASME Code rules do not apply or have not been specifically developed to address certain unique features of the CANDU plant design. The staff finds the use of ASME Code, Section III, meets the requirements of paragraphs (c), (d), and (e) of 10 CFR 50.55a. However, at this time, AECL must address several issues during the design certification stage regarding the proposed use of the 2001 edition up to and including the 2003 addenda. First, if AECL plans to design the ACR-700 to a single-earthquake loading (i.e., the safe-shutdown earthquake (SSE)) as permitted by Appendix S to 10 CFR Part 50, AECL must ensure that the piping design criteria adequately account for earthquake cycles for fatigue evaluation and seismic anchor motions resulting from an SSE, consistent with the design criteria used for the previously certified advanced light-water reactor (ALWR) standard plant designs when the operating basis earthquake is eliminated. This might require a modification to the piping design rules in the 2001 edition of ASME Code, Section III. Secondly, in order to meet GDC 2, the piping functional capability must be assured using the new code rules in the 2001 edition by evaluating the effect of the new B2 indices (see NUREG-1367).

5.5 Canadian Design Codes Assessment Conclusions

In light of the issues discussed above, the review of Canadian codes and standards will have a significant impact on the time and technical resources of the NRC staff during the design certification stage. The staff's review of Canadian codes and standards in conjunction with the ACR-700 design certification will involve significantly more time than the staff time expended on previous reviews of design certification applications (e.g., the advanced boiling-water reactor (ABWR), System 80+, and the AP600). In those previous advanced LWR reviews, the NRC required the applicants to meet the ASME Code incorporated by reference in 10 CFR 50.55a. The applicants also met the codes and standards identified in the SRP and RGs or justified any

deviation from those codes and standards. The review of Canadian codes and standards will present a major task for the NRC staff during the design certification stage.

5.6 Review Scope—Quality Assurance Standards

Focus Topic 6 was designated by AECL to be, in part, the preliminary review of QA controls applied to the design and testing for the ACR-700. As part of the ACR-700 preapplication review of FT6, AECL requested the staff to review a report comparing Canadian standards CSA-N286 series to Appendix B to 10 CFR Part 50; AECL report 108US-01910-ASD-001, "Comparison of 10 CFR Part 50, Appendix B, and ASME NQA-1-1004 Requirements Versus CSA-N286 Series of Standards," issued February 2003.

5.7 Regulatory Issues—Quality Assurance Standards

An application for design certification must contain the technical information which is required of applicants for construction permits and operating licenses by 10 CFR Part 50 and its appendices. Appendix B to 10 CFR Part 50 establishes the regulatory basis for determining the adequacy of QA programs for nuclear power plants licensed in the United States. Pursuant to 10 CFR 50.34, every applicant for a construction permit must include in its preliminary safety analysis report a description of the QA program to be applied to the design, fabrication, construction, and testing of the SSCs of the facility.

In the United States, applicants for construction permits generally provide license-specific information in the QA program, while committing to standard QA practices described by consensus standards. When endorsed by RGs, these standards are conditionally approved as providing a method acceptable to the NRC staff for complying with the provisions of Appendix B to 10 CFR Part 50. The NRC must approve the use of alternatives or exceptions to these methods before implementation. The NRC has endorsed, through RG 1.28, Revision 2, the 1983 edition of ANSI/ASME NQA-1, "Quality Assurance Requirements for Nuclear Facilities," as a method acceptable to the NRC staff for complying with the provisions of Appendix B for the design and construction of nuclear power plants. The NRC has similarly endorsed the 1994 edition of NQA-1 through the provisions of 10 CFR 50.54(a).

In 2001, the CNSC issued a license condition that established the CSA-N286 series of standards as a regulatory requirement for licensee QA programs in Canada. These standards include the following:

- CSA-N286.0-92, "Overall Quality Assurance Program Requirements for Nuclear Power Plants"
- CSA-N286.0.1-92, "Commentary on the Principles for Quality Assurance Programs of CSA N286 Series Standards"
- CSA-N286.1-00, "Procurement Quality Assurance for Nuclear Power Plants"
- CSA-N286.2-00, "Design Quality Assurance for Nuclear Power Plants"
- CSA-N286.3-99, "Construction Quality Assurance for Nuclear Power Plants"

- CSA-N286.7-99, “Quality Assurance of Analytical, Scientific, and Design Computer Programs for Nuclear Power Plants”

AECL requested, by letter from V.J. Langman dated December 18, 2002, that the NRC staff conduct a comparative evaluation of CSA-N286 and Appendix B. As the basis for comparison, the staff used the U.S. consensus standard NQA-1-1994, which is similar to the N286 standards in the level of process detail. A matrix comparing these U.S. and Canadian standards follows.

Table 5-1 Comparison of NQA-1-1994 to CSA-N286

Appendix B to 10 CFR Part 50 (Reference)		NQA-1 for 1994	CSA-N286 for 1992	Notes/Assessment of Differences
CRITERION I: ORGANIZATION				
I-1 Responsibility for Establishment and Execution of a Quality Assurance Program				
a.	Assigns responsibility for establishment of a QA program to the applicant (the term "applicant" in Appendix B and "supplier" in ISO are implicitly synonymous).	The organization invoking this Part (Part I) shall be responsible for specifying which Basic Requirements (BR); and, Supplements, or portions thereof, apply… (Intro)	The owner shall ensure that overall and second tier measures are established for identifying person(s) responsible for implementation of the QA program. (N286.0(5.1.d))	Equivalent quality elements. *NQA-1 applies to anyone contractually obligated to meet this standard. N286.0 is addressed to the owner, who is responsible to ensure that all participants meet its requirements.*
b.	Allows delegation of responsibility for establishment and execution of the QA program to others as long as responsibility is retained by the applicant.	The individual(s) or organization(s) responsible for establishing and executing a QA program may delegate any or all work to others but shall retain responsibility therefor. (2.2)	This standard is addressed to the owner and is applicable to procurement activities carried out by both the owner and participants designated by the owner. (N286.1(1.2))	Equivalent quality elements. *The supplementary requirements of the N286 standards only apply when and to the extent specified by the organization.*
I-2 Responsibility and Authority for Attaining and Verifying Quality				
a.	Requires identification and documentation of persons and organizations affecting safety-related functions.	The organizational structure, functional responsibilities, levels of authority and lines of communications for activities affecting quality shall be documented. (BR1)	The owner shall identify the person(s) responsible for the QA program. (N286.0(5.1.d)) The owner shall define and document the organizational structure. (N286.0(5.1.a))	Equivalent quality elements.

Appendix B to 10 CFR Part 50 (Reference)		NQA-1 for 1994	CSA-N286 for 1992	Notes/Assessment of Differences
I-3 Independence of Personnel Performing Verification Activities				
a.	Requires organizational freedom and authority to identify quality problems; initiate, recommend, or provide solutions; and verify implementation of solutions.	…have organizational freedom to: (a) identify quality problems; (b) initiate, recommend, or provide solutions…(c) verify implementation of solutions. (BR1)	Shall identify the person(s) responsible for implementation and effectiveness of the overall QA program. (N286.0(5.1.d))	Requirements not similar. *No direct discussion of initiating, recommending, or providing solutions for QA personnel. However, appears to be addressed under "Nonconformance."*
CRITERION II: QUALITY ASSURANCE PROGRAM				
II-2 Items Services and Organizations Covered by the Quality Assurance Program				
a.	Requires identification of items to be covered by the QA program.	The program shall identify the activities and items to which it applies. (BR2)	The owner shall describe the overall program in a QA manual in sufficient detail to demonstrate that the requirements of this Standard are met. (N286.0(3.2))	Requirements not similar. However, not an issue. *Does not specifically require identification of items to be covered by the QA program.*
II-3 Determination of Appropriate Quality Requirements				
a.	Requires identification of items controlled by the program and control only to a degree consistent with the item's importance to safety.	The program shall provide control over activities affecting quality to an extent consistent with their importance. (BR2)	The owner shall clearly establish and document their responsibilities. (N286.0(5.1.c))	Requirements not similar. However, not an issue. *Also, see note II-2a.* *Does not clearly state this. However, N286.0 does have a very detailed definition of "graded safety-related."*

Appendix B to 10 CFR Part 50 (Reference)		NQA-1 for 1994	CSA-N286 for 1992	Notes/Assessment of Differences
II-4 Controlled Conditions for Activities Affecting Quality				
a.	Requires activities affecting quality to be accomplished under controlled conditions.	The program shall provide for the planning and accomplishment of activities affecting quality under suitably controlled conditions. (BR2)	*No direct requirement for activities affecting quality to be accomplished under controlled conditions.*	Requirements not similar.
b.	Provides examples of controlled conditions as "appropriate equipment, suitable environmental conditions."	Controlled conditions include the use of appropriate equipment, suitable environmental conditions for accomplishing the activity… (BR2)	*No direct requirement for activities affecting quality to be accomplished under controlled conditions.*	Requirements not similar.
c.	Requires control of prerequisites.	Controlled conditions include…assurance that prerequisites for the given activity have been satisfied. (BR2)	*No direct requirement for control of prerequisites.*	Requirements not similar.

Appendix B to 10 CFR Part 50 (Reference)		NQA-1 for 1994	CSA-N286 for 1992	Notes/Assessment of Differences
II-6 Indoctrination and Training of Personnel				
a.	Specifies extent as "suitable proficiency is achieved and maintained." (Implicitly requires a program for retraining or proficiency maintenance.)	The program shall provide for…personnel performing activities affecting quality to assure that suitable proficiency is achieved and maintained. (BR2)	Personnel qualification criteria shall be established. These criteria shall include a definition of the minimum education, experience, initial training, and continuing training requirements. (N286.2(3.2.1))	Requirements not similar. *The N286 standards do not have a specific requirement that personnel performing inspections and tests have the necessary experience or training commensurate with the scope, complexity, or special nature of the activities. Also, not as specific in the requirements for indoctrination, evaluation, reassessment or recordkeeping of personnel.*

Appendix B to 10 CFR Part 50 (Reference)		NQA-1 for 1994	CSA-N286 for 1992	Notes/Assessment of Differences
II-7 Management Review of Quality Assurance Program Status and Adequacy				
a.	Specifies the required review interval as "regularly."	Management…shall regularly assess the adequacy and effective implementation of the QA program. (BR2)	Formal reviews of the effectiveness of the overall program shall be conducted by, or on behalf of, the owner at least annually. (N286.0(18.3))	Equivalent quality elements.
CRITERION III: DESIGN CONTROL				
III-1 Translation of Design Requirements into Design Documents				
a.	Requires inclusion of regulatory requirements and design basis into working documents.	Applicable design inputs such as design bases,…regulatory requirements shall be identified and documented, and their selection reviewed and approved by the responsible design organization. (SR3S-1(2))	Design inputs, such as…regulatory requirements, and applicable codes and standards, shall be considered in defining design requirements. (N286.2(5.2.1))	Requirements not similar. *N286.2 does not directly require inclusion of regulatory requirements into working documents.*
III-2 Inclusion of Quality Standards in Design Documents				
a.	Defines the type of quality standards as "appropriate quality standards."	Appropriate quality standards shall be identified and documented, and their selection reviewed and approved. (SR3S-1(3))	Design activities shall be identified and controlled. (N286.2(4.2))	Requirements not similar. *Does not specifically define the type of quality standards as appropriate quality standards.*
III-3 Control of Deviations from Quality Standards				
a.	Requires control of deviations from the standards.	The design input shall…permit the design activities to be carried out in a correct manner and to provide a consistent basis for making decisions, accomplishing design identification measures, and evaluating design changes. (SR3S-1(3))	Changes shall be subject to a process of review and approval similar to that applied to the original design. (N286.2(5.9.1)) *No specific mention requiring control of deviations from the standards.*	Requirements not similar.

Appendix B to 10 CFR Part 50 (Reference)		NQA-1 for 1994	CSA-N286 for 1992	Notes/Assessment of Differences
III-4 Review of Materials and Processes for Suitability				
a.	Limits the materials, parts, equipment, and processes selected for review to those that are essential to the safety-related function.	*Does not specifically limit the materials, parts, equipment, and processes selected for review to those that are essential to the safety-related function.*	*Does not specifically limit the materials, parts, equipment, and processes selected for review to those that are essential to the safety-related function.*	Requirements not similar. However, does not appear to be an issue.
III-6 Control of Design Documents				
a.	Specifies control of documents involving design interfaces only. This implies both internal and external organizations participating in the design.	Design interfaces shall be identified and controlled and the design efforts shall be coordinated among the participating organizations. (SR3S-1(6))	Design information, including changes, shall be communicated from one organization to another, and within an organization, by controlled documents. (N286.2(5.8))	Requirements not similar. *Neither is specific to including external organizations*.
III-7 Independent Verification of Design Adequacy				
a.	Requires qualification testing of specific design features to be performed under the most adverse design conditions.	Testing shall demonstrate adequacy of performance under conditions that simulate the most adverse conditions. (SR3S-1(4.2.3))	*N286.2 has no provision requiring qualification testing of specific design features to be performed under the most adverse design conditions.*	Requirements not similar.
B.	Provides examples of items applicable to design control measures, such as reactor physics, material comparability, and acceptance criteria for inspection and testing. As stated in quality element III-4, the application of design control measures is limited to items essential to safety-related functions.	In all cases, the design verification shall be completed prior to relying upon the component, system, structure, or computer program to perform its function. (SR3S-1(4))	The nature and extent of verification of design document shall be determined based on the following criteria: (a) impact on safety; (b) complexity of design; (c) degree of standardization; (d) state of the art; (e) similarity to previously proven designs. (N286.2(5.6.1))	Requirements not similar. *However, does not appear to be an issue.*

Appendix B to 10 CFR Part 50 (Reference)		NQA-1 for 1994	CSA-N286 for 1992	Notes/Assessment of Differences
III-8 Control of Design Changes				
a.	Requires approval by the organization that performed the original design.	Changes shall be approved by the same affected groups or organizations which reviewed and approved the original design documents… (SR3S-1(5))	Changes shall be subject to a process of review and approval similar to that applied to the original design. (N286.2(5.9.2)) Design verification shall be carried out by suitably qualified persons (including the designer's supervisor), provided they did not perform the activities or make the design decisions that are being verified. (N286.2(5.7.1))	Requirements not similar. *Does not specifically state approval should be from the organization that performed the original design.*
CRITERION IV: PROCUREMENT DOCUMENT CONTROL				
IV-1 Inclusion of All Applicable Requirements in Procurement Documents				
a.	Extends purchase document requirements to subcontractors.	To the extent necessary, procurement documents shall require Suppliers to have a QA program consistent with the applicable requirements of this part… (BR4)	When items or services are procured from non-participant organizations and a QA program is required, an appropriate QA Standard shall be specified. (N286.1(4.2.4))	Requirements not similar. *However, does not appear to be an issue.*
IV-2 Extension of Quality Requirements to Subcontractors and Suppliers				
a.	Requires procurement documents to impose the pertinent requirements of Appendix B on subcontractors.	Where necessary, these requirements shall be specified by reference to specific drawings, specifications, codes, standards, regulations, procedures, or revisions… (SR4S-1(2.2)) *No specific requirement to impose Appendix B on subcontractors.*	When items or services are procured from nonparticipant organizations and a QA program is required, an appropriate QA Standard shall be specified. (N286.1(4.2.4)) *No specific requirement to impose Appendix B on subcontractors.*	Requirements not similar. *However, does not appear to be an issue.*

Appendix B to 10 CFR Part 50 (Reference)		NQA-1 for 1994	CSA-N286 for 1992	Notes/Assessment of Differences
CRITERION V: INSTRUCTION, PROCEDURES, AND DRAWINGS				
V-1 Documented Description of Activities Affecting Quality				
a.	Requires documented instructions, procedures and drawings.	Activities affecting quality shall be prescribed by and performed i.a.w. (in accordance with) documented instructions, procedures or drawings... (BR5)	The owner shall ensure that second-tier measures are established so that only specified and accepted items, processes and practices are used. (11.1) *Not specific, but appears equivalent, when document taken as a whole.*	Requirements not similar. *However, does not appear to be an issue.*
b.	Defines the detail of the instructions, procedures, and drawings as appropriate to the circumstances.	Activities affecting quality shall be prescribed by and performed i.a.w. documented instructions, procedures or drawings of a type appropriate to the circumstances. (BR5)	Processes and practices shall meet the requirements of applicable Standards. (N286.0(11.1)) *Not specific, but appears equivalent, when document taken as a whole.*	Requirements not similar. *However, does not appear to be an issue.*
V-2 Compliance with the Documented Description of Activities Affecting Quality				
a.	Requires activities affecting quality to be accomplished in accordance with the instructions, procedures, and drawings.	Activities affecting quality and services shall be prescribed by and performed i.a.w. documented instructions, procedures or drawings… (BR5)	Processes and practices shall meet the requirements of applicable Standards. (N286.0(11.1))	Equivalent quality elements. *Not specific, but appears equivalent, when document taken as a whole.*
V-3 Inclusion of Acceptance Criteria in Prescribed Documents				
a.	Requires inclusion of appropriate quantitative and qualitative acceptance criteria.	…appropriate quantitative or qualitative acceptance criteria for determining that prescribed activities have been satisfactorily accomplished. (BR5)	The owner shall ensure that second-tier measures are established so that only specified and accepted items, processes and practices are used. (N286.0(11.1))	Equivalent quality elements. *Not specific, but appears equivalent, when document taken as a whole.*

Appendix B to 10 CFR Part 50 (Reference)		NQA-1 for 1994	CSA-N286 for 1992	Notes/Assessment of Differences
b.	Defines the objective of acceptance criteria as determining that important activities have been satisfactorily accomplished.	…appropriate quantitative or qualitative acceptance criteria for determining that prescribed activities have been satisfactorily accomplished. (BR5)	The owner shall ensure that second-tier measures are established so that only specified and accepted items, processes and practices are used. (N286.0(11.1))	Equivalent quality elements. *Not specific, but appears equivalent, when document taken as a whole.*
CRITERION VI: DOCUMENT CONTROL				
VI-1 Release and Control of Documents				
a.	Defines the documents to be controlled as those which prescribe activities affecting quality.	The preparation, issue and change of documents that specify quality requirements or prescribe activities affecting quality shall be controlled… (BR6)	*No specific reference to activities affecting quality*.	Requirements not similar. *Does not appear to be an issue.*
VI-2 Review and Approval of Documents by Authorized Personnel				
a.	Requires distribution to and use at the location where the prescribed activity is performed.	…documents are reviewed for adequacy, approved for release by authorized personnel, and distributed to and used at the locations where the prescribed activity is performed. (SR6S-1(1))	The owner shall ensure that current documents are available to users. (N286.0(16.1.c)) Such information…made available at location where it is to be used. (N286.0.1-92)	Equivalent quality element.
VI-3 Control of Review and Approval of Changes to Documents				
a.	Requires changes to documents to be reviewed and approved.	Such documents, including changes, shall be reviewed for adequacy and approved for release by authorized personnel. (BR6)	The owner shall ensure review and approval of changes to accepted processes are made and changes documented. (N286.0(15.1.b))	Equivalent quality elements. *Need to ensure "processes" would cover procedures*.

Appendix B to 10 CFR Part 50 (Reference)		NQA-1 for 1994	CSA-N286 for 1992	Notes/Assessment of Differences
b.	Requires changes to be reviewed and approved by the same organizations that performed the original review and approval.	Changes to documents, other than those defined as minor changes in para. 3.2 below, are considered as minor changes and shall be reviewed and approved by the same organizations that performed the original review and approval unless other organizations are specifically designated. (SR6S-1(1))	The owner shall ensure review and approval of changes to processes are made by persons who have full knowledge of the original intent and requirements.	Equivalent quality elements. *Need to ensure that someone with "full knowledge" is the same as the Appendix B "same organization" requirement.*
CRITERION VII: CONTROL OF PURCHASED MATERIAL, EQUIPMENT, AND SERVICES				
VII-1 Conformance of Purchased Items and Services to Procurement Documents				
a.	Requires control of purchased material, equipment, and services.	The procurement of items and services shall be controlled to assure conformance with specified requirements. (BR7)	Procurement processes and practices shall be specified and documented. (N286.1(3.9))	Equivalent quality elements. *(The N286 standards provide general requirements for specifying all items, but does not provide specific requirements for dealing with commercial-grade items.)*
b.	Includes controls of purchased material, equipment, and services from contractors and subcontractors.	The Purchaser of items and services shall establish measures to interface with the Supplier and to verify Supplier's performance as deemed necessary by the Purchaser. (SR7S-1(4))	This Standard is addressed to the owner and is applicable to both the owner and participants designated by the owner. (N286.1(1.2))	Equivalent quality elements. *Need to ensure "designated by the owner" will translate to the owner taking responsibility for the contractor.*

Appendix B to 10 CFR Part 50 (Reference)		NQA-1 for 1994	CSA-N286 for 1992	Notes/Assessment of Differences
VII-5 Documented Evidence of Conformance Prior to Installation				
c.	Requires evidence of conformance to be at the site prior to installation and use.	Where required by code, regulation, or contract requirement, documentary evidence that items conform to procurement documents shall be available at the nuclear facility site prior to installation or use. (SR7S-1(8))	Verification may be performed at the contractor's facility or upon receipt of the item(s), as determined by the verification plan. (N286.1(4.8.4))	Requirements not similar. *No specific requirement for evidence of conformance to be at the site prior to installation and use.*
VII-6 Documented Evidence of Conformance After Installation				
a.	Requires documentation of evidence of specific requirements met by the purchased material and equipment.	These controls shall provide for the acquisition, processing, and recorded evaluation of the quality assurance, technical, inspection, and test data against acceptance criteria. (SR7S-1(6))	*No specific requirement for the acquisition, processing, and recording evaluation of the QA, technical, inspection and test data against acceptance criteria.*	Requirements not similar.
b.	Requires retention of evidence at the site.	Where required by code, regulation, or contract requirement, documentary evidence that items conform to procurement documents shall be available at the nuclear facility site prior to installation or use. (SR7S-1(8)) *Implied, but not a requirement to retain documents*.	Verification may be performed at the contractor's facilities or upon receipt of the item(s), as determined by the verification plan. (N286.1(4.8.4)) *No specific requirement of retention of evidence of purchase requirements at the site*.	Requirements not similar. *Implied, but not a requirement to retain documents.*

Appendix B to 10 CFR Part 50 (Reference)		NQA-1 for 1994	CSA-N286 for 1992	Notes/Assessment of Differences
VII-7 Assessment of Supplier Quality-Related Activities				
a.	Requires assessment at intervals consistent with the importance, complexity, and quantity of the product or services.	The extent of verification activities, including planning, shall be a function of the relative importance, complexity, and quantity of the item or services procured and the Supplier's quality performance. (SR7S-1(5.1))	Contractor's performance shall be monitored and the results recorded. (N286.1(4.7.6)) *No specific requirement to assess at intervals consistent with the importance, complexity, and quantity of the product or services.*	Requirements not similar. *No requirement to periodically assess. Looks at history and current performance*.
CRITERION VIII: IDENTIFICATION AND CONTROL OF MATERIALS, PARTS, AND COMPONENTS				
VIII-2 Lineage Traceability and Duration of Identification Control				
a.	Requires duration of identification maintenance to be throughout fabrication, erection, installation, and use of the item.	Identification shall be maintained on the items or in documents traceable to the items, or in a manner which assures that identification is established and maintained. (BR8) *Not specific on the requirement for duration of identification*.	The owner shall ensure that essential records are valid and traceable to the items and activities to which they refer. (N286.0(17.1.d)) *No specific requirement for duration of identification*.	Requirements not similar.
VIII-3 Prevention of Use of Incorrect Items				
a.	Requires design of control measures to prevent use of incorrect or defective materials, parts, and components.	Controls shall be established to assure that only correct and accepted items are used or installed. (BR8)	Measures for the control of storage and handling shall be such as to preserve items for the time of their receipt and prevent their abuse, misuse, damage, etc. (N286.3(3.9.3.3.1))	Equivalent quality elements. *No specific requirement for the control of incorrect or defective materials and parts. Covered implicitly elsewhere.* (see N286.0(11.1.a,b))

Appendix B to 10 CFR Part 50 (Reference)		NQA-1 for 1994	CSA-N286 for 1992	Notes/Assessment of Differences
CRITERION IX: CONTROL OF SPECIAL PROCESSES				
CRITERION X: INSPECTION				
X-1 Establishment and Execution of Inspection Program				
a.	Requires establishment and execution of a program for inspection.	Inspections required to verify conformance of an item or activity to specified requirements shall be planned and executed. (BR10)	Inspections and tests shall be carried out i.a.w. approved procedures. (N286.3(3.10.3.1))	Equivalent quality elements. *The N286 series of standards does not have specific requirements for inspections. It considers inspections a form of verification.*
X-3 In-process Inspections				
a.	Requires examination, inspection, or test.	Inspection of items in-process or under construction shall be performed for work activities where necessary to verify quality. (SR10S-1(6))	*Does not have specific requirements regarding items that are impossible or disadvantageous to inspect.*	Requirements not similar. *Does not require in-process testing.*
b.	Requires performance of examinations, measurements, and tests for operations that are necessary to assure quality.	Inspection of items in-process or under construction shall be performed for work activities where necessary to verify quality. (SR10S-1(6))	Inspect and identify products or services as required by the Inspection and Test Plan(s). (Z299.1(3.5.8))	Requirements not similar. *Does not require tests for operation.*
X-4 Indirect Inspection by Monitoring				
a.	Requires monitoring when inspection is impossible or disadvantageous.	If inspection of processed items is impossible or disadvantageous, indirect control by monitoring of processing methods, equipment, and personnel shall be provided. (SR10S-1(6))	Monitor special process methods. Hold products or services until the required inspections and tests are completed or otherwise verified. (Z299.1(3.5.8.a,b))	Requirements not similar. *Do not have specific requirements to perform both inspections and process monitoring when necessary.*

Appendix B to 10 CFR Part 50 (Reference)		NQA-1 for 1994	CSA-N286 for 1992	Notes/Assessment of Differences
b.	Specifies monitoring of processing methods, equipment, and personnel.	Both inspection and process monitoring shall be provided when control is inadequate without both. (SR9S-1(6))	*Does not have specific requirement for both inspection and process monitoring when control is inadequate without both.*	Requirements not similar.
X-5 Recognition of Hold Points				
a.	Defines hold points as points beyond which work may not proceed until inspections are completed.	If mandatory inspection hold points are required beyond which work shall not proceed without the specific consent of the designated representative… (SR9S-1(4))	*These standards refer only to inspection requirements, which would include hold points. Specific reference to hold points is made in manufacturing QA standards such as Z299.1.*	Equivalent quality elements.
b.	Requires indication of hold points in appropriate documents if hold points are used.	…the specific hold points shall be indicated in appropriate documents. (SR9S-1(4))	*See above*.	Equivalent quality elements.
CRITERION XI: TEST CONTROL				
XI-1 Establishment and Execution of Test Program				
a.	Requires assurance that SSCs will perform satisfactorily in service.	Tests required to verify conformance of an item or computer program to specified requirements and to demonstrate satisfactory performance for service shall be planned and executed. (BR11)	The purpose of qualification testing a prototype or initial production unit is to verify the design or specific design features. In the case of the latter, the other features of the design shall be verified by other means. (N286.2(5.6.6)) *No specific discussion of construction, preoperational or operational tests.*	Requirements not similar. *N286.4 is dedicated entirely to commissioning and tests performed during commissioning.*

Appendix B to 10 CFR Part 50 (Reference)		NQA-1 for 1994	CSA-N286 for 1992	Notes/Assessment of Differences
b.	Requires the test program to include preinstallation proof tests, preoperational tests, and operational tests.	Required test, including as appropriate, prototype qualification tests, proof tests prior to installation, construction tests, pre-op tests, and op tests shall be controlled. (SR11S-1(2))	The purpose of qualification testing a prototype or initial production unit is to verify the design or specific design features. In the case of the latter, the other features of the design shall be verified by other means. (N286.2(5.6.6)) *No specific discussion of construction, or preoperational or operational tests.*	Requirements not similar. *N286.4 is dedicated entirely to commissioning and tests performed during commissioning.*
XI-2 Inclusion of Test Parameters in Test Documents				
a.	Requires test procedures to assure completion of test prerequisites.	Test procedures shall include or reference provisions for assuring that prerequisites for the given test have been met. (SR11S-1(2))	Written test procedures shall be issued. These instructions shall define: test preparation. (N286.2(6.7.3))	Equivalent quality elements.
b.	Requires testing to be performed under suitable environmental conditions.	Prerequisites shall include the following, as applicable: suitable environmental conditions… (SR11S-1(3))	*No specific requirements for testing to be performed under suitable environmental conditions.*	Requirements not similar.

Appendix B to 10 CFR Part 50 (Reference)		NQA-1 for 1994	CSA-N286 for 1992	Notes/Assessment of Differences
CRITERION XII: CONTROL OF MEASURING AND TEST EQUIPMENT				
CRITERION XIII: HANDLING, STORAGE, AND SHIPPING				
XIII-2 Provisions for Special Product Requirements				
a.	Requires specification and provision of special protective environments when required for particular products.	When required for particular items,…special protective environments (such as inert gas atmosphere, specific moisture content levels, and temperature levels) shall be specified, provided and their existence verified. (SR13S-1(2))	When special handling tools and equipment are required, they shall be inspected and tested at specified times to verify that they are adequately maintained. (N286.3(3.9.3.3))	Requirements not similar. *Does not specify special equipment (such as containers, shock absorbers, and accelerometers).*
b.	Provides examples of types of protective environments.	…special protective environments (such as inert gas atmosphere, specific moisture content levels, and temperature levels) shall be specified… (SR13S-1(2))	*Does not specify special protective environments (such as inert gas atmosphere, specific moisture content levels and temperature levels).*	Requirements not similar.
CRITERION XIV: INSPECTION, TEST AND OPERATING STATUS				
CRITERION XV: NONCONFORMING MATERIALS, PARTS, AND COMPONENTS				
XV-2 Identification, Documentation, Segregation, and Notification				
a.	Requires notification to affected organizations.	Controls shall provide for notification to affected organizations. (BR15)	The owner shall ensure that second-tier measures are established so that items, documents, etc., which do not conform requirements are: (b) reviewed and remedial action determined, executed, verified and recorded. (N286.0(13.1.b))	Requirements not similar. *No specific direction to notify affected organizations.*

Appendix B to 10 CFR Part 50 (Reference)		NQA-1 for 1994	CSA-N286 for 1992	Notes/Assessment of Differences
XV-3 Disposition of Nonconforming Items				
a.	Requires nonconforming items to be accepted, rejected, repaired, or reworked.	The disposition, such as use-as-is, reject, repair, or rework, of nonconforming items shall be identified and documented. (SR15S-1(4.4))	The owner shall ensure that second-tier measures are established so that items, documents, etc, which do not conform requirements are: (c) controlled to prevent unauthorized use or implementation. (N286.0(13.1.c))	Equivalent quality elements.
CRITERION XVI: CORRECTIVE ACTION				
XVI-1 Identification and Corrections of Condition Adverse to Quality				
a.	Provides examples of applicable conditions, (e.g., failures, malfunctions, deficiencies, deviations, defective materials, and nonconformances).	*Discusses conditions adverse to quality and significant conditions adverse to quality. No specific examples are given.*	*Discusses "recurring" or "serious" nonconforming conditions. No specific examples are given.*	Requirements not similar. *No specific examples are given.*

Appendix B to 10 CFR Part 50 (Reference)		NQA-1 for 1994	CSA-N286 for 1992	Notes/Assessment of Differences
CRITERION XVII: QUALITY ASSURANCE RECORDS				
XVII-2 Identification of Record Types				
a.	Provides a listing of the minimum types of records to be maintained (e.g., operating logs, results of reviews, inspections, tests, audits, monitoring of work performance, and material analyses, qualifications of personnel, procedures, and equipment). Inspection and test records shall identify the inspector or data recorder, the type of observation, the results, the acceptability, and action for any deficiencies.	Quality assurance records shall be specified… (BR17) The applicable design specifications, procurement documents, test procedures, operational procedures, or other documents shall specify the record to be generated, supplied or maintained by or for the Owner. *Does not provide a listing of the minimum types of records to be maintained.*	Two categories of essential records shall be established: permanent and nonpermanent. (N286.1(3.15.3)) *Does not provide a listing of the minimum types of records to be maintained.*	Equivalent quality elements.

	Appendix B to 10 CFR Part 50 (Reference)	NQA-1 for 1994	CSA-N286 for 1992	Notes/Assessment of Differences
	XVII-3 Special Requirements for Inspection and Test Records			
a.	Requires identification of the inspector, type of observation, inspection results, and acceptability.	See above. *Does not specify identification of the inspector, type of observation, inspection results, and acceptability.*	The owner shall ensure that overall and second-tier measure are established to: (d) ensure essential records are valid and traceable to the items and activities to which they refer. (N286.0(17.1))	Requirements not similar. *Does not specify identification of the inspector, type of observation, inspection results, and acceptability.*
	CRITERION XVIII: AUDITS			
	XVIII-2 Audit Performance, Documentation, and Review			
a.	Requires trained auditors who are independent of the activity being audited.	…by personnel who do not have direct responsibility for performing the activities being audited. (BR18) *Does not specify having trained auditors. However, see II-6a.*	Audits shall be carried out by personnel who neither performed nor verified activities being audited. (N286.0(18.2.e))	Requirements not similar. *Does not specify having trained auditors and does not discuss lead auditors. However, see II-6a.*
	XVIII-3 Audit Followup Requirements			
a.	Includes reaudit of deficient areas in followup action.	*Does not specify reaudit of deficient areas in followup action.*	*Does not specify reaudit of deficient areas in followup action.*	Requirements not similar.

AECL conducted a comparative evaluation of CSA-N286 and NQA-1-1994 in AECL Assessment Document 108US-01910-ASD-001, "Comparison of 10 CFR Part 50, Appendix B, and ASME NQA-1-1004 Requirements Versus CSA N286 Series of Standards," issued February 2003. The staff has reviewed AECL report and, while agreeing with the general approach, has identified additional differences that AECL should address as part of its application for licensing the ACR in the United States. Table 5.7-1 identifies all significant differences identified by the NRC's comparative evaluation. The staff documented the issues that resulted from its comparison in the form of an RAI, which the NRC sent to AECL by letter dated May 5, 2004.

For each of the potential issues identified in the RAI, AECL responded, in AECL letter 108US-01321-021-001 dated May 31, 2004, by citing commitments to additional CSA standards, existing AECL or ACR programs and procedures, or by committing to impose additional requirements on the ACR-700 program. Collectively, the references and

commitments made by AECL supplement the N286 requirements such that AECL has effectively committed to comply with all NQA-1-1994 requirements for reactors of the ACR-700 design to be installed in the United States.

5.7.1 Safety Classification—Safety Related

The term "safety related," as defined by 10 CFR Part 50, paragraph 50.2, refers to those SSCs that are relied upon to remain functional during and following DBEs to assure the integrity of the RCPB, the capability to shut down the reactor and maintain it in a safe-shutdown condition, or to prevent or mitigate the consequences of an accident which could result in potential offsite exposure comparable to defined regulatory guideline exposures.

The staff requested AECL to cite the Canadian regulation(s) that define "safety related" for the ACR-700. The staff asked AECL to provide references to specific Canadian regulatory guidance and consensus standards that amplify this classification. The staff requested AECL to provide a matrix of all items and activities classified as safety related and to identify the quality standards (e.g., N286) imposed on these items and activities.

AECL would apply the term "safety related" to ACR-700 design and testing, as defined in 10 CFR Part 50. AECL would organize the list of systems into classes which clearly identify safety-related systems and systems important to safety. The applicant will identify these classes in a table in the ACR-700 design control document.

Canadian regulations require that all safety-related systems comply with N286 quality requirements and any additional quality requirements invoked by applicable technical standards, such as ASME. The Canadian definition of "safety related" applies to all systems that perform a safety function, including all backup systems, rather than just those that prevent or mitigate an accident. The definition of "safety related" in the N286 standard combines the 10 CFR Part 50 definitions of "safety related" and "important to safety."

5.7.2 Safety Classification—Important to Safety

The criteria of Appendix A to 10 CFR Part 50 apply to all SSCs important to safety. According to GDC 1, "structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed." Examples of systems important to safety include the following:

- fire protection (10 CFR 50.48)
- environmental qualification (10 CFR 50.49)
- anticipated transients without scram (10 CFR 50.62)
- station blackout (10 CFR 50.63)
- pressurized thermal shock (10 CFR 50.61)

AECL cited the Canadian regulation(s) that pertain to "important to safety" for the ACR-700 program. The staff requested specific information regarding applicable regulatory guidance and consensus standards that amplify this classification. The staff also asked for a matrix of all

items and activities classified as important to safety identifying the quality standards (e.g., N286) imposed on these items and activities.

As discussed in the previous section, the Canadian definition of “safety related” applies to both safety-related and important-to-safety SSCs. Examples of important-to-safety SSCs cited by AECL include fire protection systems, the second shutdown system (that addresses anticipated transients without scram), and any additional power supplies needed to maintain the plant in a safe state. The staff expects AECL to comply with the acceptability criteria as described by Section 3.2.2, “System Quality Group Classification,” of the SRP for items and activities important to safety. The list of systems will be organized into classes which clearly identify the systems that are safety related and systems that are important to safety according to U.S. regulatory definitions. The applicant should identify these classes in a table in the ACR-700 design control document.

5.7.3 International Standards Organization Quality Programs

The staff has not approved the use of International Standards Organization (ISO) quality programs for design, procurement, or testing of safety-related or important-to-safety SSCs. The NRC documents the staff’s assessment of the adequacy of international quality standards, such as ISO, for licensing in the United States in SECY-03-0117, “Approaches for Adopting More Widely Accepted International Quality Standards,” dated July 23, 2003. The staff requested that AECL identify how it applied ISO quality programs to ACR-700 design, procurement, and testing.

AECL responded that all design, procurement, and testing activities supporting the ACR-700 design certification will be conducted in accordance with N286 requirements.

5.7.4 Conducting Activities Under Controlled Conditions

Appendix B to 10 CFR Part 50 requires that activities affecting quality be accomplished under controlled conditions. Basic Requirement 2 of NQA-1 states, in part, that the program must provide for the planning and accomplishment of activities affecting quality under suitably controlled conditions. Controlled conditions include the use of appropriate equipment, suitable environmental conditions, and assurance that prerequisites for the given activity have been satisfied. The N286 standards do not appear to have a similar requirement.

The staff requested AECL to describe the QA controls used to ensure that activities affecting quality related to ACR-700 design and testing activities are accomplished under controlled conditions.

AECL responded that N286.0 had equivalent wording (Clause 11 states, “the right items, processes, procedures shall be used”; Clause 3.9 states, “only specified processes and practices shall be used”). The response also referred to a ACR-700 QA Manual statement that activities are performed using qualified items, processes, and procedures. Prerequisites for a given activity are identified and addressed during the performance of that activity, as required by the ACR-700 QA Manual.

5.7.5 Nondestructive Examination Auditor Qualifications

Supplement 2S-2 of NQA-1 provides amplified requirements for the qualification of personnel performing radiographic, magnetic particle, ultrasonic, liquid penetrant, electromagnetic, and other testing, also referred to as nondestructive testing (NDT). The N286 standards do not appear to have similar requirements.

The staff requested AECL to describe the QA controls used to ensure that personnel who perform NDT are adequately trained and qualified for activities affecting quality related to ACR-700 design and testing activities.

AECL responded that these requirements are covered through the CSA-Z299 manufacturing standards. For pressure boundary components, N285.0 and CSA-B51 cover these requirements. For nondestructive examination personnel qualification, AECL followed Canadian Government Specification Board standards, which meet or exceed the American Society of Nondestructive Testing requirements that are specified by Supplement 2S-2.

5.7.6 Lead Auditor Qualifications

Supplement 2S-2 of NQA-1 provides amplified requirements for the qualification of lead auditors. Amplified requirements include communication, training, audit participation, examination, maintenance of qualification, requalification, and record of certification. The N286 standards do not appear to have similar requirements.

The staff requested AECL to describe the QA requirements that ensure that lead auditors are adequately trained and qualified for activities affecting quality related to ACR-700 design and testing activities.

AECL responded that AECL Procedure 00-904.5, “Qualification of QA Audit Personnel,” defines the requirements for qualification and training of auditors and lead auditors. AECL has evaluated additional requirements for auditor and lead auditor qualification and records of qualification and stated that it will develop a specific procedure to address NQA-1 requirements and apply it to ACR-700 activities.

5.7.7 Audits Conducted by a Third Party

Appendix B to 10 CFR Part 50 requires management to review the status and adequacy of the QA program on a regular basis. Basic Requirement 2 of NQA-1 states, in part, that management must regularly assess the adequacy and effective implementation of the QA program. In part, N286.0 states that formal reviews of the effectiveness of the overall program must be conducted by, or on behalf of, the owner at least annually. NQA-1 does not allow for the use of third-party audits.

The staff requested AECL to describe the controls applied to QA program reviews for ACR-700 design and testing activities. The staff requested clarification on whether “on behalf of the owner” allows for third-party audits.

AECL responded that the ACR-700 QA Manual requires management to regularly (annually) assess the adequacy of the QA program. AECL performs quality surveillances and audits on supplier programs. For supplier qualification, AECL performs some recognized third-party audits.

5.7.8 Documentation of Regulatory Requirements in Working Documents

Appendix B to 10 CFR Part 50 requires inclusion of regulatory requirements and design bases into working documents. Supplement 3S-1 of NQA-1 states, in part, that applicable design inputs such as design bases and regulatory requirements must be identified and documented. In part, N286.2 states that design inputs, such as regulatory requirements and applicable codes and standards, must be considered in defining design requirements.

The staff requested AECL to describe the QA controls that ensure identification and inclusion of regulatory requirements, applicable codes, and design bases information in working documents related to ACR-700 design and testing activities.

AECL responded that Section 5.2.1 of its corporatewide QA manual specifies requirements for addressing and documenting applicable design inputs. The ACR-700 QA program requires the identification and documentation of all applicable design inputs such as design bases and regulatory requirements. Design requirements documents specify these requirements.

5.7.9 Design Controls

Supplement 3S-1 of NQA-1 describes critical reviews that provide assurance that the final design is correct and satisfactory. Consideration is given to the following issues:

- Were the design inputs correctly selected?
- Are assumptions necessary to perform the design activity adequately described and reasonable, and are the assumptions identified for subsequent reverification when the detailed design activities are completed?
- Was an appropriate design method used?
- Were the design inputs correctly incorporated into the design?
- Is the design output reasonable compared to the design inputs?
- Are the necessary design input and verification requirements for interfacing organizations specified in the design document or in supporting procedures or instructions?

The requirements of N286.2 do not appear to be as rigorous in defining critical reviews that provide assurance that the final design is correct and satisfactory.

The staff requested AECL to describe the QA controls that ensure design controls of a similar nature are applied to ACR-700 design activities.

AECL responded that AECL Procedure 00-531.2, “Design Review,” requires a design review to address these Supplement 3S-1 requirements.

5.7.10 Control of Design Documents Related to External Organizations

Appendix B to 10 CFR Part 50 specifies the control of documents involving design interfaces, including both internal and external organizations. Supplement 3S-1 of NQA-1 states, in part, that design interfaces must be identified and controlled and that design efforts must be coordinated among the participating organizations. N286.2 does not specifically discuss design information that crosses organizational interfaces.

The staff requested AECL to describe the QA controls that ensure that control of design documents includes communications with external organizations pertaining to design and testing activities associated with the ACR-700 program.

AECL responded that it would strengthen Section 3.4 of the ACR-700 QA Manual to define controls required for identification and control of design interfaces, as well as coordination of design efforts among participating organizations.

5.7.11 Qualification Testing

Appendix B to 10 CFR Part 50 requires qualification testing of specific design features to be performed under the most adverse design conditions. Supplement 3S-1 of NQA-1 states, in part, that testing must demonstrate adequacy of performance under conditions that simulate the most adverse conditions. N286.2 has no provision requiring qualification testing of specific design features under the most adverse design conditions.

The staff requested AECL to describe the QA controls that ensure qualification testing of specific design features of the ACR-700 under the most adverse design conditions.

AECL responded that it would include clarification in the ACR-700 QA Manual.

5.7.12 Design Approval

Appendix B to 10 CFR Part 50 requires design approval by the organization that performed the original design. Supplement 3S-1 of NQA-1 states, in part, that changes must be approved by the same affected groups or organization which reviewed and approved the original design documents. N286.2 states, in part, that changes must be subject to a process of review and approval similar to that applied to the original design. It is not clear that the N286.2 and NQA-1 requirements are equivalent.

The staff requested AECL to describe the QA controls that ensure approval by the organization that performed the original design applies to ACR-700 design activities.

AECL responded that the ACR-700 QA Manual would further clarify this requirement.

5.7.13 Documents in Use at Prescribed Activity

Appendix B to 10 CFR Part 50 requires documents to be distributed to and used at the location where the prescribed activity is performed. Supplement 6S-1 of NQA-1 requires that documents be reviewed for adequacy, approved for release by authorized personnel, and distributed to and used at the location where the prescribed activity is performed. In part, N286.0 states that the owner must ensure that current documents are available to users. It is not specific about use of an approved document at the location where the prescribed activity is being performed.

The staff requested AECL to describe the QA controls that ensure the use of approved documents where the prescribed activity is performed, as it relates to design and testing activities related to the ACR-700 program.

AECL responded that the ACR QA program meets this Supplement 6S-1 requirement. Section 4.3 of the ACR-700 QA Manual, as supplemented by Sections 3.14, 5.4, and 5.8 of AECL companywide design manual, describe the requirements for document control. Procedures 00-414.3 and 108-414.3.1 specify the detailed requirements.

5.7.14 Document Review and Approval

Appendix B to 10 CFR Part 50 requires document changes to be reviewed and approved by the same organization that performed the original review and approval. Supplement 6S1 of NQA-1 states, in part, that document changes other than those defined as minor must be reviewed and approved by the same organizations that performed the original review and approval unless other organizations are specifically designated. N286.0 states that the owner must ensure that review and approval of changes to processes are made by persons who have full knowledge of the original intent and requirements. It is unclear that the process implemented under N286.0 would be equivalent to that required by Supplement 6S-1.

The staff requested AECL to describe the QA controls that ensure document changes are reviewed and approved by the same organization that performed the original review and approval as related ACR-700 design and testing activities.

AECL responded that the ACR-700 QA Manual would clarify this requirement.

5.7.15 Commercial-Grade Items

Supplement 7S-1 of NQA-1 states, in part, that where the design utilizes commercial-grade items, the following requirements are an acceptable alternative to other requirements of this supplement, except as noted in the following, and in the requirements of Supplement 4S-1:

- The commercial-grade item is identified in an approved design output document. An alternate commercial-grade item may be applied, provided the cognizant design organization verifies that the alternate commercial-grade item will perform the intended function and will meet design requirements applicable to both the replaced item and its application.

- Source evaluation and selection, where determined necessary by the purchaser based on complexity and importance to safety, must be in accordance with paragraph 3.1 of Supplement 7S-1.

- Commercial-grade items must be identified in the purchase order by the manufacturer's published product description (e.g., catalog number).

- After receipt of a commercial-grade item, the purchaser must determine the following:

  – Damage was not sustained during shipment.

  – The item received was the item ordered.

  – Inspection and/or testing is accomplished, as required by the purchaser, to assure conformance with the manufacturer's published requirements.

  – Documentation, as applicable to the item, was received and is acceptable.

The N286 standards do not provide specific requirements for commercial-grade items.

The staff requested AECL to describe the QA controls that ensure the design and testing activities would properly control commercial-grade items and services procured for the ACR-700 program.

AECL responded that new plants generally do not use commercial-grade items. For replacement items, AECL developed a process for evaluating alternate commercial-grade items. The process would be applied to the ACR-700 program if commercial-grade items or services were used.

5.7.16 Contractor Performance Monitoring

Appendix B to 10 CFR Part 50 requires that contractor performance be assessed at intervals consistent with the importance, complexity, and quantity of the product or services. Supplement 7S-1 to NQA-1 states, in part, that the extent of verification activities, including planning, must be a function of the relative importance, complexity, and quantity of the item or services procured and the supplier's quality performance. In part, N286.1 states that a contractor's performance must be monitored and the results recorded. No specific requirement exists to assess at intervals consistent with the complexity, and quantity of the product or services.

The staff requested AECL to describe the QA controls that ensure that contractor performance is monitored at intervals consistent with the relative importance, complexity, and quantity of the product or services provided for ACR-700 design and testing activities.

AECL responded that AECL corporatewide procurement QA manual meets Supplement 7S-1 requirements.

5.7.17 Inspection Activities

Appendix B to 10 CFR Part 50 requires monitoring when inspection is impossible or disadvantageous. Supplement 10S-1 of NQA-1 states, in part, that inspection of items in process or under construction must be performed for work activities, where necessary, to verify quality. If inspection of processed items is impossible or disadvantageous, indirect control by monitoring of processing methods, equipment, and personnel must be provided. Both inspection and process monitoring must be provided when control is inadequate without both. The N286 standards do not appear to have similar requirements.

The staff requested AECL to describe the QA controls that ensure monitoring of items in process or under construction is performed for work activities where necessary to verify quality when inspection is impossible or disadvantageous for activities related to the design and testing of the ACR-700 program.

AECL responded that the CSA Z299 manufacturing (or equivalent) standards cover this requirement under “Special Processes” and “Inspection and Test Plans.” AECL Procedure 00-912.1, “Selection of Quality Program Categories,” also contains these requirements.

5.7.18 Protective Environments

Appendix B to 10 CFR Part 50 mandates specification and provision of special protective environments when required for particular products. Supplement 13S-1 of NQA-1 states, in part, that when required for particular items, special protective environments (such as inert gas atmosphere, specific moisture content levels, and temperature levels) must be specified, provided, and their existence verified. The N286 standards do not address special protective environments for the storage of items.

The staff requested AECL to describe the QA controls that ensure that special protective environments (such as described by Supplement 13-1) are specified, provided, and their existence verified, when necessary for design and testing activities related to the ACR-700 program.

AECL responded that the CSA Z299 manufacturing (or equivalent) standards cover this requirement under handling and storage. AECL manufacturing specifications, which are incorporated into purchase orders, also cover these requirements.

5.7.19 Notification of Nonconformances

Appendix B to 10 CFR Part 50 requires notification of nonconformances to affected organizations. Basic Requirement 15 of NQA-1 states, in part, that controls must provide for notification of nonconformances to affected organizations. In part, N286.0 states that nonconforming processes, practices, and documents must be (1) identified, documented, and reported; (2) reviewed, resolved, verified, and results documented; and (3) controlled to prevent unauthorized use or implementation. There is no specific requirement for notifying affected organizations.

The staff requested AECL to describe the QA controls that ensure notification of nonconformances to affected organizations for design and testing nonconformances related to the ACR-700 program.

AECL responded that it would develop and implement a procedure to meet this basic NQA-1 requirement.

5.7.20 Trained Auditors

Appendix B to 10 CFR Part 50 requires trained auditors. In part, N286.0 states that audits must be carried out by personnel who neither performed nor verified activities being audited. N286.0 does not require trained auditors.

The staff requested AECL to describe the QA controls that ensure that audits are conducted by trained auditors for activities related to the design and testing of the ACR-700 program.

AECL responded that auditors will be trained, as detailed in AECL response to Section 5.7.6 above.

5.7.21 NQA-1-1994, Part II

AECL assessment of these standards (see AECL Assessment Document 108US-01910-ASD-001) did not report a comparison of N286 to NQA-1, Part II, “Quality Assurance Requirements for Nuclear Facility Applications.” Part II of NQA-1 establishes QA requirements for the planning and implementation of activities during the fabrication, construction, modification, repair, maintenance, and testing of SSCs for nuclear facilities.

According to NQA-1, contractual documents must specify a QA program, based on the nature and scope of the work to be performed and the relative importance of the items or services, by selective applications of portions of Part I, “Basic and Supplemental Requirements,” for programmatic activities and of Part II for work-oriented activities. These activities include attainment of quality objectives and verification that activities affecting quality have been correctly performed. These activities include planning, subsurface investigations, fabrications, fabricating, handling, shipping, storing, cleaning, erecting, installing, inspecting, testing, maintaining, repairing, and modifying.

The staff requested AECL to describe the quality controls for activities addressed by Part II of NQA-1 and its methods for incorporating Part II provisions in written contracts, policies, procedures, and instructions related to design and testing activities of the ACR-700 program.

AECL responded that the N286 standards do not have requirements similar to those described in Part II of NQA-1. However, AECL procurement documents include technical specifications for fabrication and construction. AECL procedure covering engineering quotation requests and related AECL Procedures 00-852.1, 00-852.2, and 00-852.1 delineate these technical specifications.

5.7.22 Legacy Issues

During preapplication review meetings in March 2004, AECL stated that it based about 85 percent of the ACR-700 design on testing and design completed before beginning ACR-700 design activities. Consequently, some tests supporting the ACR-700 design certification may not have been conducted under a quality program equivalent to N286 or NQA-1.

The staff requested AECL to provide a matrix of testing that supports the ACR-700 design, a brief description of each test, the date of the test, the quality controls applied, and the specific design area supported by the test.

AECL responded that CANDU validation matrices are structured on a discipline basis (i.e., fuel and fuel channel, physics, system thermal hydraulics, containment). These matrices summarize all data constituting the technology bases for the CANDU reactor design. Most of this information is generally applicable, and much of it is specifically applicable to the ACR-700 design. AECL will update the matrices to reflect validation for recent testing. AECL will identify the legacy data that support the ACR-700 technology base and the QA programs and/or procedures in effect when these data were generated. The information will be available for NRC review subsequent to submittal of the ACR-700 design certification application.

5.7.23 Procurement

Procurement regulations apply to both hardware and services. Pursuant to 10 CFR Part 21, suppliers must report defects and noncompliances that have the potential for creating a substantial safety hazard. Licensing of an ACR-700 plant in the United States would require an applicant for a construction permit to impose 10 CFR Part 21 requirements on the nuclear steam system supplier. For design, 10 CFR Part 21 particularly applies to procurement of engineering and testing services.

The staff requested AECL to discuss how its application for certifying the ACR-700 will address 10 CFR Part 21, particularly for procurement of engineering and testing services.

AECL responded that 10 CFR Part 21 requirements would be imposed as part of the ACR-700 QA program for ACR-700 reactors to be installed in the United States. AECL would develop a procedure for complying with 10 CFR Part 21 requirements and impose these requirements through the procurement process on all suppliers. AECL will revise the ACR-700 QA Manual to incorporate 10 CFR Part 21 requirements.

5.7.24 Control of Subcontractors

During preapplication review meetings, AECL discussed how participants, partners, and subcontractors had contributed to the ACR-700 design. AECL identified Babcock and Wilcox Canada and Hitachi as contributing participants.

The staff requested that AECL identify all participants, partners, and subcontractors that contributed to the ACR-700 design, the scope of services provided, the information exchanged, and the quality controls for documenting, verifying, and validating this information.

AECL responded that it will provide a list of participants, partners, and subcontractors that contributed to the ACR-700 design and testing, the scope of services provided, and the information exchanged to the NRC during the design certification review.

5.8 Potential Policy Issues—Quality Assurance Standards

The staff has identified no issues concerning the ACR-700 QA Program that would need Commission guidance.

5.9 Technical Issues—Quality Assurance Standards

The staff has identified no technical issues concerning the ACR-700 QA Program based on the information submitted during the preapplication review.

5.10 Quality Assurance Assessment Conclusions

By comparing U.S. and Canadian QA regulations and standards, the staff sought to identify those differences between the licensing requirements that could present obstacles to licensing the ACR-700 in the United States. The staff communicated the potential issues that resulted from its comparative review to AECL by letter dated May 5, 2004. Based on the staff's comparative evaluation, AECL responses, and additional information provided by AECL during the preapplication review, the staff concludes that there is reasonable assurance that the differences identified in this report should not preclude licensing the ACR-700 in the United States. However, the staff will need to verify AECL has adequately implemented the quality assurance measures committed to in their RAI responses.

It is noted that this review focused on differences in QA requirements between the United States and Canada. The staff's review of the design certification application will evaluate the ACR-700 QA Program documentation in its entirety, including those documents incorporated by reference. After completing its initial review of quality-related documents, the staff plans to conduct an implementation audit of the quality controls for the ACR-700 design and supporting test programs. The scope of the staff's review and audit activities will include the issues identified in this report, as well as the responsive commitments by AECL.

6. DISTRIBUTED CONTROL SYSTEMS AND SAFETY CRITICAL SOFTWARE (FT7)

6.1 Review Scope

This report is based on the review of information in Section 7, “Instrumentation and Control,” of AECL document 10810-01371-TED-001, Revision 0, “ACR-700 Technical Description,” dated July 3, 2003. This report describes the instrumentation and controls (I&C) (including alarms, displays, and annunciation) for safety-related and nonsafety-related systems.

The safety-related I&C are those which, by virtue of their failure to perform safety functions in accordance with their design, could cause regulatory dose limits for the plant to be exceeded in the absence of mitigating systems action. The systems identified in AECL 10810-01371-TED-001, Revision 0, are those associated with the proper functioning of the associated safety support systems, safety-related process systems, and status displays and annunciators.

6.1.1 ACR-700 Safety Systems

- Shutdown System 1
  - (1) includes shutoff rods
- Shutdown System 2
  - (1) includes liquid injection shutdown system
- emergency core cooling
  - (1) includes ECI system
  - (2) includes LTC system
- containment system
  - (1) includes the reactor building, containment penetrations and appurtenances, airlocks, and containment isolation

6.1.2 ACR-700 Safety Support Systems

- reserve water system
- electric power system
- service water systems
  - – includes raw service water and recirculated cooling water
- instrument air
  - (1) includes post-LOCA instrument air

The I&C for the following systems are also safety-related systems:

- systems required for safe shutdown
- safety-related process systems

Instrumentation, controls, and displays required for the operation, testing, and status monitoring of these systems are considered safety related.

6.2 Regulatory Basis

The GDC in Appendix A to 10 CFR Part 50 establish high-level minimum requirements and the principal design criteria for nuclear power plants. These criteria set forth the necessary design, implementation, construction, testing, and performance requirements for SSCs important to safety. IEEE Standard (Std.) 603-1991, "Criteria for Safety Systems for Nuclear Power Generating Stations," which is incorporated in 10 CFR 50.55a(h), provides more specific requirements for the design of safety-related I&C systems. The criteria of IEEE Std. 603 address considerations such as I&C system design bases, single failures, redundancy, independence, integrity, reliability, qualification, status indication, and testing. Appendix B to 10 CFR Part 50 establishes QA requirements for the design, construction, and operation of these systems.

The GDC that apply to I&C systems include GDC 1, "Quality Standards and Records"; 2, "Design Bases for Protection Against Natural Phenomena"; 4, "Environmental and Dynamic Effects design bases"; 13, "Instrumentation and Control"; 19, "Control Room"; 20, "Protection System Functions"; 21, "Protection System Reliability and Testability"; 22, "Protection System Independence"; 23, "Protection System Failure Modes"; 24, "Separation of Protection and Control Systems"; 25, "Protection System Requirements for Reactivity Control Malfunctions"; and 29, "Protection Against Anticipated Operational Occurrences." Chapter 7 of the NRC SRP sets forth detailed guidance for the review of the plant I&C system description, provided in the plant safety analysis report, for conformance to these requirements.

The ACR-700 has two safety-related shutdown systems, SDS1 and SDS2, that operate independently to automatically shut down the reactor when needed. SDS1 terminates reactor power operation and maintains the reactor in a safe-shutdown condition by releasing 20 gravity shutoff rods into the reactor core. These shutoff rods are made of neutron-absorbing material and will terminate reactor operation whenever they are dropped into the reactor core. In addition, SDS1 employs a logic system with three independent channels, which detect the need for a reactor trip by SDS1. If two of three channels sense the requirement for SDS1 action, then a reactor trip is initiated. The shutoff rod drop logic of SDS1 allows the shutoff rods to drop into the reactor core by removing the direct current from the clutches that hold the shutoff rods in their parked, out-of-core position. SDS1 fails safe on the loss of electrical power.

Eleven measured neutronic and process variables, plus a manual trip and trip computer (TRC) Watchdog Timer, can initiate a trip on SDS1. During startup after a prolonged reactor shutdown, special startup instrumentation may be used to provide a trip on high count rate until the fission chambers demonstrate an onscale rational power signal. Trip parameters include the following:

- high neutron power (regional overpower)
- high rate of logarithmic neutron power (log rate)
- HTS low pressure*
- HTS high pressure (immediate and delayed)*
- HTS low flow*
- moderator low level
- moderator high level
- reactor building high pressure
- steam generator low level*
- TRC Watchdog Timer
- manual trip

(*Note—These trip parameters have operational bypasses that condition out the trip when reactor power is less than a set value. Low-pressure and low-flow HTS trips are bypassed when reactor power is less than 0.1 percent of full power; HTS high-pressure delayed trip is bypassed when reactor power is less than 70 percent of full power; steam generator low-level trip is bypassed when fission chambers are less than 1 percent of full power and flux detectors are less than 10 percent of full power.)

SDS2 causes reactor shutdown by the rapid injection of concentrated gadolinium nitrate solution into the bulk moderator through six horizontally distributed nozzles. SDS2 employs an independent triplicated logic system, which senses the requirement for this emergency shutdown and opens fast-acting helium pressure valves to inject the gadolinium poison into the moderator. SDS2 fails safe on the loss of electrical power or instrument air.

The neutron absorbing gadolinium nitrate solution (called “poison”) is stored in six identical pressure vessels (poison tanks) located in an accessible part of the reactor building. Each poison tank feeds one injection line, which passes through the calandria vault to an injection nozzle passing horizontally on the top and bottom of the reactor core. The poison solution is prepared in a mixing tank, from which it is transported under moderate pressure to the poison tanks. A single helium supply tank contains high-pressure helium, which forces the liquid poison from the poison tanks into the core.

SDS2 trip parameters are the same 11 as those of SDS1 using separate sensors. The three trip channels (D, E, and F for SDS1, and G, H, and J for SDS2) each have completely independent and physically separated power supplies, trip parameter sensors, instrumentation, trip logic, conditioning, and annunciation. When any two of the three channels trip, the shutdown system is actuated. General coincidence logic is used (i.e., an entire channel trips when any measurement of any parameter within that channel reaches its trip setting). The trip logic in each channel is implemented using TRCs. These are industrial-grade, field-proven computers that implement all trip parameters, conditioning, and setpoints that are functions of reactor power and/or pump configuration. The digital outputs of the TRCs drive relays in the channel trip logic. The only trip logic external to the computers is that related to manual trips, TRC watchdog trips, and the temporary startup instrumentation trip.

If a trip condition exists for longer than the seal-in delay, it becomes sealed in. The operator can only reset the tripped channel, once sealed in, after all trip parameters in that channel have cleared.

The following provides the interlocks for interface with other systems:

- The condition of SDS2 unavailable (fewer than five of six tanks in service or helium tank pressure low) will inhibit moderator liquid poison removal (by the moderator purification system) and also shut off rod and control absorber withdrawal.

- The tripped condition of SDS2 will inhibit withdrawal of shutoff rods and control absorbers and will inhibit moderator liquid poison removal.

- The tripped condition of SDS2 inhibits withdrawal of the shutoff rods.

- The condition of SDS1 unavailable (less than 18 of 20 shutoff rods fully withdrawn) inhibits moderator poison removal and mechanical control absorber withdrawal.

The fuel-handling and storage system provides on-power fueling capability at a rate sufficient to maintain continuous reactor operation at full power. The fueling control system (FCS) handles the control of fuel changing and fuel transfer operations in automatic, semiautomatic, and manual modes. The FCS contains the fuel-handling distributed control system (FHDCS), the protective interlock system, and the seismic trip system. The interlock system is divided into operational interlocks and safety interlocks. In this context, safety interlocks are all interlocks that protect against malfunctions that have a nuclear safety, personnel safety, or high economic impact. All safety-related interlocks are implemented through hardwired logic, and only the operational interlocks are implemented in software. Some of the safety interlocks can be defeated by a set of bypass hand-switches located on the backup control console, which can be operated if necessary with the appropriate authorization. Unless defeated, the protective system interlocks are operative in both automatic and manual modes of operation.

6.3 Potential Policy Issues

The staff has identified no policy issues or issues requiring Commission guidance with regard to the ACR-700 I&C assessment.

6.4 Technical Issues

The following regulatory and technical issue comments on the information in AECL 10810-01371-TED-001, Revision 0, Sections 7 and 9.3, are based on existing regulations and guidance that is documented in Chapter 7 of the NRC SRP:

- Separate sensors are provided for monitoring the parameters for SDS1 and SDS2. It appears that the trip setpoints for both SDS1 and SDS2 are the same. The type of sensors used may also be the same. This may raise the issue of how the design complies with the NRC position on defense-in-depth and diversity contained in the SRM on SECY-93-087. Branch Technical Position (BTP) 19 (in Appendix 7-A to Chapter 7 of the NRC SRP) describes in detail the regulatory basis, material to be reviewed, acceptance criteria, and review process. In the case of SDS1 and SDS2, the question will be raised as to whether both systems are developed to meet the same functional and software requirements.

- The sharing of HTS pressure instrumentation between SDS1 and LRVs will also be reviewed in detail. In reviewing how the design meets the single failure criterion, the NRC expects the safety analyses to address the following:

  – If a single failure causes an LRV to open and this leads to a condition that requires SDS1 actuation, then an additional failure of a second channel and its consequences is to be considered.

- The use of the same sensor to perform two opposing trip functions raises the question of the fail-safe mode of the channel. The HTS high-pressure and low-pressure trips, and the moderator high- and low-level trips are derived from the same sensor. What is the failure mode of these channels upon loss of power to the sensor?

- The design of SDS1 and SDS2 operating bypasses of trip parameters must meet the requirements of IEEE Std. 603-1991, Section 6.6. Section 6.6 states that whenever the applicable permissive conditions are not met, a safety system must automatically prevent the activation of an operating bypass or initiate the appropriate safety function(s). If plant conditions change so that an activated bypass is no longer permissible, the safety system must automatically accomplish one of the following actions:

  (1) remove the appropriate active bypass(es)
  (2) restore the plant conditions so that the permissive conditions once again exist
  (3) initiate the appropriate safety function(s)

- Manual trip in SDS1 and SDS2 is initiated by pressing channelized pushbuttons located in the main control room and secondary control building. These pushbuttons are mechanically linked, enabling a single operation action to trip all channels. This design is not in complete compliance with the requirement in IEEE Std. 603, Section 5.2.1, which states that means must be provided in the control room to implement manual initiation at the division level of the automatically initiated protective actions.

- AECL 10810-01371-TED-001, Revision 0, states that manual trips are external to the channel trip computers. However, the trip signal apparently passes through an Or-gate located in the computer cabinet. It is not clear that a failure in the computer cabinet will not affect the channel manual trip signal.

- RG 1.152, “Criteria for Digital Computers in Safety Systems of Nuclear Power Plants,” and BTP 14 provide guidance for software development processes that are expected to produce reliable software. The NRC believes that the concept of quantitative reliability goals are not sufficient by themselves to meet NRC regulatory requirements for the reliability of digital computers used in safety systems. An AECL presentation to the Advisory Committee on Reactor Safeguards (ACRS) indicated that the reliability of the safety critical software is demonstrated through particular quantitative reliability goals (assessed by trajectory-based random testing of the software). This may raise an issue in this design. The current NRC position on quantitative reliability goals for digital computers does not provide for the use of digital reliability goals, and AECL should determine if it wishes to request that this information be used. Additionally, the NRC

has not taken a position on these particular testing methods and will have to develop a regulatory basis to support the review.

- Section 7.1.1.1 of AECL 10810-01371-TED-001, Revision 0, does not designate the protective interlock system or seismic trip system of the FCS as safety systems. The systems perform functions that meet the definitions of safety systems in 10 CFR 50.2 and IEEE Std. 603. As safety systems they would be required to meet the single failures, redundancy, independence, integrity, reliability, qualification, status indication, and testing requirements. There is a question as to whether the hardware interlock systems for this design can meet the single failure and redundancy requirements.

6.5 I&C Assessment Conclusions

The comments above provide an example of the technical and regulatory issues that need to be considered in the more detailed design of the ACR-700 that would be submitted for review.

6.6 Human Factors Engineering Review Scope

The NRC limited the scope of this review to AECL document, "Plant Performance Specification—ACR-700 Human Factors Engineering Program Plan," (HFEPP) Revision 1, dated June 23, 2003. This document "identifies the Human Factors Engineering (HFE) principles, standards, requirements and processes necessary to integrate HFE throughout the design of the entire plant."

The staff based its review criteria on an adaptation of existing NRC review guidance for human factors engineering (HFE) as found in the following documents:

- Revision 1 of the NRC SRP
- NUREG-0711, Revision 2, "Human Factors Engineering Program Review Model," 2004
- NUREG-0700, Revision 2, "Human-System Interface Design Review Guideline," 2002
- Information Notice (IN) 97-78, "Crediting of Operator Actions in Place of Automatic Actions and Modifications of Operator Actions, Including Response Times," 1997
- NUREG-1764, "Guidance for the Review of Human Actions, Final Report," 2004
- RG 1.174, "An Approach to Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," 1998
- RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decision Making: Technical Specifications," 1998
- IN 91-18, "Information to Licensees Regarding Two Inspection Manual Sections on Resolution of Degraded and Non-Conforming Conditions and on Operability," 1991

- ANSI/ANS Standard 58.8, "Time Response Design Criteria for Safety-Related Operator Actions," 1994 and 2001

Title 10, Part 52, of the Code of Federal Regulations also applies to the review of HFE aspects of nuclear power plants for design certification.

6.7 Regulatory Issues

The HFEPP does not identify the NRC regulations that apply to design certification of the ACR-700 design. In particular, it does not address the requirement for a safety parameter display console (10 CFR 50.34(f)(2)(iv)). In addition, all previous NRC design certifications for advanced reactors designs have required compliance with the staffing requirements contained in 10 CFR 50.54(x). Although AECL has indicated that station staffing is one of several purchasing/operating utility responsibilities (e.g., in addition to training and procedures development) and that "the purchasing/operating utility should document the development methodology," the AECL position relative to this NRC requirement is unclear. This NRC staffing requirement also relates to the application of the on-power fueling technology that is characteristic of CANDU reactor designs. Specifically, it is unclear from the HFEPP how the function of on-power fueling is integrated into the overall operation and maintenance activities of the ACR-700 design from the perspective of human performance. As the function of on-power fueling is not a typical process used in conventional LWR designs, the NRC will likely give added attention to understanding the engineering and human factors details of this activity.

The HFEPP does not include as an HFE activity the element of minimum inventory of alarms, controls, and displays (minimum inventory) or an explanation for excluding it as an HFE activity. The minimum inventory has been an element of the HFE programs for all previous applicants for design certification. This element is designated as Tier 1 information in the applicant's design control document.

6.8 Potential Policy Issues

The staff has identified no policy issues or issues requirement Commission guidance with the ACR-700 HFE program.

6.9 Technical Issues

The HFEPP provides a top-level identification of the HFE activities and design processes which AECL proposes to "result in effective ACR-700 human-system interface (HSI) designs, which, in turn, can be safely operated and maintained in the operational environment, and which are compatible with human information requirements, capabilities and limitations." The document neither provides a level of detail to determine if the proposed plan ("HFE activities and design processes") will lead to products that meet the NRC's design certification acceptance criteria for the HFE program requirement, nor provides finished products of AECL efforts to complete the elements of an NRC-approved design process. The HFEPP does refer to two future reports, the Human Factors Engineering Summary Report (HFESR) and the ACR-700 Human Factors Engineering Verification and Validation Plan. The HFESR will provide more detail on the HFE activities performed as part of the ACR-700 design work. The ACR-700 Human Factors Engineering Verification and Validation Plan will outline the design verification methods that will

be used to determine that the ACR-700 HFE design meets the design requirements, once sufficient information is available to determine the appropriate scope and activities. According to AECL, verification will take place through the companywide design verification, design review, and document review and comment processes, discussed in References 17, 18, and 19, respectively, of the HFEPP. The ACR-700 Human Factors Engineering Verification and Validation Plan will also outline the HFE validation plan to be used to confirm that individual designs and the design as an integrated whole meet the overall performance objectives of the system and the plant. The HFEPP indicates that, "as a minimum, the main control room design will be mocked up with both static and dynamic features. This mock up will serve as both an ongoing design tool and a platform for formal HFE validation."

As one of the program source documents (documents that govern the format, content, and structure of the HFE program), the HFEPP references NUREG-0711, the NRC's principal review guidance for evaluating the design of applicant HFE programs. However, NUREG-0711, Revision 2, dated February 2004, has superseded the NUREG-0711 citation used in the HFEPP. In addition, the HFEPP references NUREG-0700, Revision 1, as one of its guidance documents (documents that will be used as primary sources of HFE design criteria for incorporation into ACR-700 documentation and design guidance), which has been revised. The current version is Revision 2, issued May 2002.

In general, the HFE activities that comprise the proposed HFEPP (as depicted in Figure 5-1 of the HFEPP) appear to be in accordance with NUREG-0711, the NRC's principal HFE program review guidance document. However, the staff has questions related to AECL process of determining the human factors level of effort described on pages 4-1 and 4-2 of the HFEPP. AECL has proposed a multi-level-of-effort approach.

This approach initially focuses the human factors design team and directs human factors design support—

> "to areas more dependent on high levels of human-system performance towards the achievement of overall safety and production…. Level A design areas are designated systems, subsystems, equipment and layouts in receipt of direct and detailed human factors design team support. Level A design areas typically represent areas where human involvement is known to be intense and important to the achievement of plant safety and production goals."

Determining the level of human factors review effort appears to emphasize the degree of human interaction required by the design areas rather than whether the design areas are safety related. For example, AECL states on page 4-2 that, "Level B design areas presently include, for example: Remote panels or interfaces for controls and displays which warrant little human interaction, located outside the control room…." However, it is possible that although certain design areas may require little human interaction (e.g., "systems and equipment located outside of the Control Centers areas") these areas could (1) still be associated with safety-related equipment/functions and (2) because they require "little human interaction" (and hence are not frequently accessed), they may require closer attention from HFE to ensure reliable performance. The staff will give added attention to AECL's proposed strategy for HFE level of effort.

In addition, AECL identified that the development of operating procedures (and training programs) “will normally be the responsibility of the purchasing/operating utility, and as such, beyond the scope of Reference 1.” Though operating procedures have been identified as the responsibility of the purchasing/operating utility, it is unclear whether purchasing/operating utility procedures (emergency operating procedures (EOPs) in particular) are developed from, for example, a generic analysis of transients and accidents or from plant-specific analyses. The generic guideline concept has been used in U.S. conventional light-water plants, whereby various nuclear steam supply system vendors developed generic analyses of transients and accidents for PWRs and BWRs (emergency response guidelines and emergency procedures guidelines, respectively) from which individual plants prepared their EOPs. Plants customized the generic technical guidelines to accommodate individual plant differences, such as plant-specific setpoints and various plant-specific operating parameters as necessary. AECL should explain the process ACR-700 plants will use to develop their procedures with special attention to EOPs (using guidance in NUREG-0711 and the NRC SRP, Chapter 13, “Conduct of Operations,” Section 13.5.2.1, “Operating and Emergency Operating Procedures”).

6.10 Human Factors Assessment Conclusions

The staff has concluded, based on the considerations discussed previously, that AECL document, “Plant Performance Specification—ACR-700 Human Factors Engineering Program Plan,” (HFEPP) Revision 1, dated June 23, 2003, provides a reasonable framework for preparing an effective HFE program. The HFEPP, however, does not provide a level of detail to determine whether the proposed framework, if implemented, will lead to products that meet the NRC design certification acceptance criteria for the HFE program requirement. Neither does the document provide finished products of AECL efforts to complete the elements of an NRC-approved design process. As stated previously, the document also does not address certain NRC regulatory and technical issues that must be satisfied for the design certification of a proposed advanced reactor’s HFE program.

7. ON-POWER FUELING (FT8)

7.1 Review Scope

7.1.1 Introduction

Pursuant to 10 CFR 52.48, the NRC will review any application filed for a design certification under 10 CFR Part 52 for compliance with the standards set out in 10 CFR Part 20, 10 CFR Part 50 and its appendices, and 10 CFR Parts 73 and 100 as they relate to applications for construction permits and operating licenses for nuclear power plants, and as those standards are technically relevant to the design proposed for the facility. AECL has requested a preapplication review of the preliminary design of the ACR-700. This design incorporates on-power fueling capability. Power reactors currently licensed to operate in the United States have not been designed with this capability. Therefore, on-power fueling is a process outside the scope of current NRC regulatory guidance and may conflict with existing NRC regulations that have been developed in conjunction with the construction and licensing of U.S. LWRs.

The preapplication review of FT8, "On-Power Fueling," allows the staff to determine the feasibility of successfully completing the standard design licensing review of the on-power fueling process through early identification of technical and regulatory issues. This effort involved the review of the basic design of the ACR-700 on-power fueling process and the systems involved in that process, and a comparison of information developed from that review with design-related NRC regulations and associated regulatory guidance (e.g., SRP and regulatory guides). The scope of review includes the system interactions necessary to accomplish on-power fueling and the effects these interactions have on required functions, such as RCPB integrity, containment integrity, decay heat removal, and reactivity control.

The staff reviewed documents provided by AECL during the preapplication review, including system descriptions, design requirements, safety design guides, initiating event analyses, analysis of NRC generic issues, preliminary piping and instrumentation drawings, and AECL response to an NRC RAI related to FT8. From these documents, the staff identified component functions and system interfaces important to safety and credible failure modes with potentially safety-significant effects. The staff also used information from the documents to identify technically relevant, design-related regulations in 10 CFR Parts 50 and 52. However, the staff reviewed the issues related to application of codes and standards, including application of 10 CFR 50.55a, separately for application to the on-power fueling process and related systems. The staff also used the documents to identify technically relevant principal design criteria based on Appendix A to 10 CFR Part 50.

The comparison of design information with the regulations resulted in the identification of technical, regulatory, and policy issues. The staff considered technical issues to be those issues for which either specific, detailed NRC guidance does not exist or for which the review of the issue involves a novel interpretation of existing regulations and guidance. For these issues, further data and analyses will be necessary to establish appropriate acceptance criteria. The staff considered regulatory issues to be those issues for which a regulation is applicable but unnecessary to satisfy the underlying purpose of the regulation, such that an exemption may be appropriate, or issues for which no regulation applies but the issue has substantial safety significance such that rulemaking would be appropriate. Finally, the staff considered policy

issues to be those issues of substantial safety significance where Commission guidance would be appropriate in establishing acceptance criteria.

7.2 System Descriptions

7.2.1 Description of On-Power Fueling Systems and Components

On-power fueling for the ACR-700 involves two fueling machines, one operating at each end of the fuel channel during fueling operations. The major fueling machine components are the head, carriage, bridge, and catenary. Mechanical components within the head assembly include a ram, separators, a magazine, and a snout, which contains homing and locking mechanisms for connection to the fuel channel. A catenary system is used to connect support services, such as electrical power and control, cooling water, and instrument air to the fueling head and bridge. The following provides a functional description of the fueling machine components and other support and control systems to clarify the regulatory aspects of the on-power fueling process.

7.2.1.1 Fueling Machine Head

Each fueling head is a pressure vessel designed to ASME Code, Section III, Class I. The snout is located at the front of the fueling machine. It attaches and locks the fueling machine to the fuel channel or a fuel transfer port to form a leak-tight seal. When the fueling machine is not attached to either a fuel channel or transfer port, the end of the snout is sealed with a plug. Separators are located between the snout and magazine rotor and are used to sense the passage of the fuel channel end fitting shield plug and fuel bundles into the magazine. The separators are also used to separate the fuel bundles that are required to be discharged from the remaining fuel string. The magazine is a multichamber rotor with 6 chambers that can hold up to 12 fuel assemblies and additional chambers to receive and temporarily store the fuel channel hardware, such as the fuel channel shield plug and closure adapter. Suitable latches that can be released by operation of the ram hold the components securely. The magazine is supported on a cantilevered shaft. It is rotated using a spur ring gear driven by a pinion on the magazine input drive that is in turn driven by an externally mounted indexing mechanism to ensure positive location at each rotor location. The ram is located directly behind the magazine and provides the motive force to transfer fuel and hardware in and out of the fuel channel. In addition, the ram provides the articulation necessary for the installation and removal of the fuel channel closure, shield plug, and snout plug. Redundant electric servo motors drive the snout, magazine, and ram mechanisms. Connections for local manual operation of these components are also provided.

The drive gearbox is mounted towards the rear of the head. Two reversible, variable speed, variable torque electric motors on the gearbox provide the motive power for all seven drives through a differential gear unit and are powered from inverters with adjustable frequency and voltage outputs. The differential gear unit is coupled through a spur gear train to each of the output drive shafts. Each spur gear is connected to its output drive shaft through two electromagnetic clutches so that the drive shafts are only rotated when the clutches are energized. The output speed of the differential gear unit is controlled by the combined speed of the two motors, while the torque is limited by the voltage applied to the motors. Speed of the motors is frequency controlled. Electromagnetic brakes on each of the drive systems are

designed to be locked at any selected position. Interconnecting shafts transmit the power from the gearbox to the driven component. Both the clutches and brakes on the output drive shaft are disengaged when power is off. This permits manual operation of all head drives with the head at any location in the event of a loss of power. Drive systems for the snout, magazine, and the ram penetrate the head pressure boundary. The boundary is maintained through rotary shaft seals on the input drive units. These seals limit leakage past the drive shafts with minimal torque loss. Shaft encoders are provided on all of the drive systems except the snout lock to provide continuous position indication of the driven components. The open and closed positions of the snout lock have indication.

7.2.1.2 Bridge and Carriage Assemblies

The bridge spans the reactor face and is supported by four fixed columns with ball screws on each that drive an elevator up and down. The bridge holds the carriage, which in turn supports the fueling head. The vertical motion of the bridge provides the coarse Y motion of the fueling machine.

Lateral motion of the carriage on the bridge provides the coarse X motion. The carriage also has a rotation drive that allows it to rotate the fueling machine 90 degrees to fit inside the maintenance lock. Motion of the fueling machine towards and away from the reactor or transfer port end fittings is accomplished with an axial drive (Z motion). A very slow short stroke vertical drive is provided for Y position correction during the homing operation. A homing position sensor stops the forward movement of the head (Z) towards an end fitting. Operation of the X and Y fine drives in response to this signal will cause the snout of the head to be aligned in incremental steps to the end fitting center line. Gimbals on the carriage allow limited rotational movement of the head in pitch and yaw within the carriage, which permits the head to move during homing and also accommodates some relative movement of the head and end fitting after the head is locked on the end fitting. The rotational motion is spring centered to maintain the head level in pitch and aligned in yaw. During the time that the snout is locked onto an end fitting, the fine X and Y drives are inhibited and the control system only allows the Z drive to back away from the reactor face. The pitch centering spring is locked out to minimize loads on the end fitting that may be caused by excessive minor corrective movements. Electro servo motors drive all motion associated with the bridge and carriage mechanisms. The carriage traverses along the bridge (Y movement) by an electric drive that has an encoder associated with it to transmit carriage position to the control system. Two sets of electric drives and encoders are associated with carriage positioning for redundancy and reliability.

7.2.1.3 Catenary

One main catenary per fueling machine connects electric power, control signals, cooling water, and other auxiliary systems that originate in the fueling machine maintenance lock area. Each catenary consists of steel chains that support clamps running horizontally from one chain to the other. The clamps in turn carry the cables and hoses. These clamps carry the catenary tensile loads and also prevent loads from being transmitted to the hoses and cables. A load sensor is located at the upper end of each chain to detect abnormal or excessive tension. If this occurs, the control system stops all X and Y motion.

7.2.1.4 Fueling Machine Maintenance Area Shield Doors

The fueling machine maintenance area shield doors are large doors that can be closed to allow personnel access to the fueling machines when the machines are in the maintenance lock. Electric motors position the doors. A switch in the main control room controls the movement of these doors, and indicator lights display the door position. Since the catenary extends from the maintenance lock when the machine is in transit to or attached to the reactor face, this door must remain open when the fueling machine is in these positions. If the door were to close while the catenary was extended, the doors have the potential to damage redundant cables or hoses, which could cause a loss of vital control or cooling functions. Safety interlocks inhibit these doors from closing unless the FMs are completely in the maintenance locks. A seismic motion interlock that is part of the seismic trip system will remove power to the door drives if the ground motion is equivalent to an SSE in magnitude.

7.2.1.5 Fueling Machine Process System

Each fueling machine requires a reactor grade supply of water at a typical temperature of 105 EF, which is delivered at various pressures according to the function performed. These functions include controlling the magazine housing pressure, cooling the spent fuel in the magazine, filling the void in the ram housing created when the ram is extended, and providing injection flow into the fuel channel when the fueling machine is attached to the channel end fitting with the channel closure plug removed. This keeps the fueling machine cool and minimally contaminated. During on-power fueling, the water flow from both machines is directed into the reactor channel. Both fueling machines provide a combined maximum injection flow of 60 gallons per minute (gpm). This water system can be divided into the fueling machine water supply subsystem and the fueling machine water control subsystem.

The fueling machine water supply subsystem consists of a high-pressure supply valve station, a heat exchanger, centrifugal booster/circulation pumps, filters, a tank, and return pumps. Provisions exist for supplying alternate coolant flow to the fueling machine when off reactor, if the booster/circulation pumps fail to operate.

High pressure is required only for on-power fuel changing operations. Pressurized water is supplied from the discharge of the RCS pressure and inventory control pump. The water temperature is reduced to near ambient, and the booster/circulation pump increases its pressure to slightly above the reactor channel end fitting pressure to provide injection flow. When the fueling machine is not coupled to a reactor channel or transfer port, it is operated at park pressure. Park pressure is accomplished by supplying pressurized water from the RCS pressure and inventory control pump, reducing its temperature to near ambient, and using the booster/circulation pump to provide flow.

When the fueling machine is coupled to the spent fuel transfer port, the return flow is circulated in a closed loop of the fuel-handling system. The fueling machine booster/circulation pump provides flow to the fueling machine at low pressure for spent fuel transfer. When the fueling machine is coupled to the new fuel transfer port, circulating cooling water flow is not required.

The fueling machine water control subsystem consists of the fueling machine water valve station, related pumps and filters, and fueling machine head-mounted equipment. The fueling

machine water valve station contains process instrumentation and valves that control the flow of water to the fueling machine.

Piping that connects the fueling machine to the water valve station consists of SS tubing and high-pressure hoses. The equipment mounted on the head includes pumps, process instrumentation, valves, and other flow-restricting devices that limit the leakage in the event of a hose failure.

7.2.1.6 Fuel-Handling Control Console

The fuel-handling operator performs all normal, abnormal, and emergency control functions from a console located in the main control room. It is made up of video display units that allow data entry and display of system conditions. Alarm displays and hard copy printers are also provided. These same video displays can display the images from the closed circuit TV monitors located in the reactor vault and fuel transfer areas. There is also a dedicated alarm interrogation panel for fuel-handling alarms. The normal control mode of the fuel-handling process is the automatic mode. The fuel-handling operator can also control operations in either a semiautomatic or manual mode. Manual mode allows the operator to exercise control over individual mechanisms such as the positioning of components or singular movements associated with a fueling machine.

7.2.1.7 Fuel-Handling Backup Control Panel

The backup control panel is hardwired and seismically qualified for safety-related fuel-handling control and monitoring functions. It allows the fuel-handling operator to put the fuel-handling system in a safe state if there is a failure of normal control.

7.2.1.8 Fuel-Handling Control System

The fuel-handling control system controls all the functions of fuel changing and fuel transfer operations in automatic, semiautomatic, and manual modes. It is made up of three subsystems, including the FHDCS, the seismic trip system, and the protective interlock system.

The FHDCS configuration is based on distributed control and is independent of the plant operational control system. Each fuel-handling subsystem has a separate controller. Each controller handles all subsystem functions, including motion control, logic control, and proportional integral derivative control. A dedicated controller provides supervisory sequential control of the fuel-handling operations. This sequential controller coordinates the motion of the mechanisms through the subsystem controllers, while individual motion controllers perform the actual motion control. Redundant communication links connect both the sequential controller and subsystem controllers. Control data and fuel system status are sent to the fuel-handling video displays that provide the operator interface.

The seismic trip ensures that no undesirable control actions are promulgated during a seismic event. This system continually monitors for seismic activity through motion sensors. If a seismic event is detected, this system removes power to the various drive mechanisms on the fueling machines to negate any spurious operation.

The protective interlock system contains operational and safety interlocks. Safety interlocks protect against malfunctions that have a nuclear safety, personnel safety, or high-loss economic impact. The safety interlocks are all implemented through hardwired logic. Only operational interlocks are implemented through programming software. Some of the safety interlocks can be defeated by a set of bypass hand switches located on the backup control console, and operation of these switches requires procedural authorization. Unless manually defeated, all protective interlocks are operable in both the automatic and manual modes.

7.2.1.9 Positioning Control Signal Instrumentation

Signals from this instrumentation control the positioning mechanisms on the various components of the fueling machines. On the earlier CANDU plants, traditional potentiometers provided the basis for this position sensing instrumentation. The ACR-700 will use resolvers and linear variable displacement transducers.

7.2.1.10 Snout Homing and Locking Mechanism

The homing and locking mechanism is basically a three-jaw clamping device which preloads a seal ring and clamps the head to the channel end fitting. An “Acme” screw draws the jaws together, and a high-tensile stud passes through the center of the screw for backup protection. This mechanism contains a self-locking worm gearset which is driven from the gearbox mounted at the rear of the head. A guide ring is mounted in front of the jaws. This ring contains four coarse-position sensor pads and has a tapered bore that guides the head over the end fitting. Fine homing sensors are mounted on plates in the jaw assembly and provide feedback to the fine homing control system. Actuation of either the coarse or fine positioning sensors provide signals to the fine X and fine Y drives mounted on the carriage/bridge. These drives make the appropriate correction to the head position to align it on the end fitting. Coarse position sensors also control the advance of the head towards the end fitting (Z drive) and protect the end fitting from damage should the initial positioning of the head fall outside acceptable limits. An air nozzle in the seal face area cleans the end fitting seal face of loose debris before clamping. The gap between the seal ring flange and end fitting face is monitored to confirm that an adequate degree of tightening has been achieved to ensure a leak-tight joint. A mechanical safety lock prevents inadvertent operation of the clamping mechanism while the head is pressurized above a predetermined, nominal operating pressure.

7.2.2 Description of Reactor Fueling Sequence

Under one proposed fueling scheme for the ACR-700, up to six channels with coolant flow in the same direction are refueled successively with new fuel from a single new fuel port visit before transferring the spent fuel to the spent fuel storage pool. Other fueling schemes are possible where new fuel is carried in both machines so that fueling can be performed successively on channels with coolant flow in either direction. The following describes a nominal fueling of four channels with two new fuel assemblies each, in succession.

The fuel changing sequence begins with one of the fueling machines accepting four pairs of new fuel assemblies from its new fuel port, where new fuel is pushed into the fueling machine magazine from the transfer mechanism under remote control from the main control room. If not already open, the shielding door is opened and the fueling machine bridge is lowered until it is

at the same elevation as the maintenance lock tracks. The carriage travels along the maintenance lock tracks and transfers to the fueling machine bridge. The fueling machine is rotated 90 degrees to face the reactor. The bridge is raised until the fueling machine snout is at the correct elevation. The bridge brakes are engaged and the power to the bridge is removed so that inadvertent bridge movement is prevented. The carriage moves until the snout is in front of the upstream end fitting of the target fuel channel. The fueling machine is advanced until the front of the snout is over the end fitting. Misalignment of the snout and end fitting will cause the fueling machine to pitch and/or yaw. Any misalignment is corrected using the fine position drives. At the same time, an identical fueling machine, which has at least four empty fuel chambers in its magazine, is located at, and aligned with, the downstream end of the same channel.

The fueling machines are advanced until they contact the end fittings. The machines attach and lock themselves to the end fittings forming a leak tight connection. The plugs in the snouts of the fueling machines are removed and the machines are pressurized with cooling water to about 345 kPa (50 psi) higher than the channel pressure in order to facilitate injection flow into the channel. The temperature of the fueling machine process water is maintained at a lower temperature than that in the fuel channel, and the injection flow is maintained to thermally isolate the fueling machine from the higher temperature of the RCS and to prevent crud from the channel entering the fueling machine. The channel closures that provide an in-bore seal inside the end of each channel are removed by the fueling machines and stored in the magazines. In this state, with the ends of the fuel channel open, the fueling machines become part of the RCS boundary containing coolant at RCS pressure. Reactor coolant continues to flow uninterrupted through the channel to cool the fuel. In addition, coolant is injected into the channel from both fueling machines.

The shield plugs are removed. As the downstream shield plug is withdrawn, the fuel string follows because channel coolant flow pushes the fuel string downstream. The separator mechanism separates the fuel string from the shield plug and prevents it from entering the fueling machine magazine while the shield plug is being stored. The fueling machine allows the fuel string to enter an empty fuel chamber in a controlled manner, separates the first pair of bundles from the fuel string, and stores this pair in the magazine. The fueling machine then reinstalls the shield plug in the end fitting, which locates the 10-bundle fuel string correctly in the core. With the downstream shield plug installed, the other fueling machine pushes a pair of new fuel bundles into the channel at the upstream end, and channel flow moves these two bundles to the fuel string. The upstream fueling machine installs the shield plug, and the pressure differential across the channel is measured to verify channel flow. Both fueling machines install the channel closures and snout plugs, restoring the fuel channel to normal operation and isolating the fueling machines from the RCS. After leak testing the channel closures and fueling machine snout plugs, the fueling machines are unclamped and retracted away from the end fittings.

The fueling machines are relocated to the next fuel channel that has to be fueled. Fueling is repeated until four channels have been visited, at which point the fueling machine containing spent fuel moves to the spent fuel port. Since the two fueling machines are identical, they can be used for either function. The choice depends on the direction of coolant flow, which is in opposite directions in adjacent channels. The new fuel is always inserted at the upstream end and spent fuel discharged at the downstream end of the channel. Once the fueling machine is returned to the maintenance lock, the spent fuel is discharged into the spent fuel port.

Operators in the control room monitor the fueling machines using closed circuit TV cameras located in the reactor building.

7.2.3 Description of Safety Features Associated with On-Power Fueling Systems

The design of the fueling machines have built-in safety features for normal operations, as well as special features to deal with equipment breakdown. Understanding the safety aspects designed into the fueling machines provides insights into the need for specific design requirements.

For routine automatic operation, the control system provides a permissive for each step at the proper point in the sequence, and a separate, hardwired protective system backs up the permissive in important areas. If the machines are operated under manual control, the protective system is still effective. Should operating conditions require the bypassing of any part of the protective system, procedures require strict supervision of the bypass switches.

Detaching from a reactor channel before replacing the closure and shield plugs would probably be the most serious fueling machine-related failure because it results in a rapid discharge of coolant from the channel. Since the machine forms an extension of the pressure boundary when sealed to the fuel channel, the housing of the fueling machine is designed to the ASME Pressure Vessel Code (Section III, Class I Nuclear Vessels). Additionally, the housing is attached to the end fitting by a snout lock designed to safety factors as stringent as those required for the fueling machine head pressure vessel. The snout lock is self-locking, has a backup in the clamping screw, has a safety lock to prevent it being driven off when the head is pressurized above 5.5 kPa, and is also prevented from being driven by the protective system if the head pressure is above 340 kPa. Therefore, this type of accident is unlikely and would require multiple failures of the control and protective systems combined with a failure of the snout safety lock. The operating data support this.

The staff evaluated damage to active fuel because of excessive fuel ram force for the ACR-700 design. The inverter voltage setting controls the force of the ram when pushing fuel. The protective system would recognize excessive inverter voltage and shut down the drive. The inverter adjustment is accessible from the control room, and acceptable forces can be periodically verified on the machines when not in use. Unauthorized adjustment of the maximum permissible setpoint in the protective system could result in loadings of greater than 5000 pounds to the fuel string, but interlocks and operational procedure limit forces of that magnitude.

The probability of overheating spent fuel from loss of cooling is small because of the arrangement of the cooling system. Duplicated centrifugal pumps with separate electrical power sources maintain cooling flow circulation, so that adequate flow would be provided even if one pump fails. Multiple pumps with separate electrical power sources provide cooling flow to the system heat exchanger. Multiple hose connections supply cooling flow to the head. Each hose is equipped with either a check valve or remote operating valves that would isolate any individual hose that failed. Therefore, cooling flow would be maintained in the event of a single hose failure. The Class III bus, which has onsite backup power, supplies all cooling system electrical power, so the short duration of any power outage would not result in significant overheating. During traversing to the spent fuel port at half level, the decay heat would produce

boiling of the water in the head with steam escaping through the open snout of the head. The normal action of level correction would automatically continue refilling the head to the required level to keep the fuel covered.

The reactor bridge is powered to raise and lower the head and suspension and to traverse horizontally along the bridge. The failure of a drive to operate or unwanted operation of a drive would constitute an abnormal operation. Failure of one of the drive motors to operate would not prevent the bridge from moving, and failure of both drive motors is unlikely since they are fed from different power sources through separate wiring. Failure of one brake to release would be overcome by the combined power of the two drive motors when lowering. The capacity of the brakes would prevent the inadvertent operation of one drive motor downward or upward while attached to the fuel channel. Inadvertent release of the two of the four brakes would not cause downward movement because two brakes alone will resist gravitational forces. All brakes are prevented from releasing if the head is forward of its retracted position in the Z direction. The control system would detect failure of an encoder on one elevator drive. The protective system compares signals from the two encoders to detect any loss of synchronism between the two elevators. The carriage drive also contains two electric motors, two brakes, and two encoders powered by separate power supplies. The brake capacity is sufficient to prevent motion through inadvertent powering of one drive motor. The output of the two motors is sufficient to overcome a failure of one brake to release. The drives are inhibited in the Z direction if the head is forward of the Z retract position.

Since the fueling machines operate within the reactor containment, the containment provides an additional layer of protection for the public in the event of failures that result in fuel damage. To deal with minor malfunctions, the equipment has been designed to facilitate breakdown operations and maintenance such that the man-rem absorbed by maintenance personnel is kept to a minimum. All remote drives are designed to allow manual operation. In addition, the gear box is provided with redundant motors and clutches. Failure of one of these devices will not disable the fueling machine. Therefore, personnel exposure to radiation and tritium from recovery operations is minimized, since it is possible to vary the settings of the component speeds and applied forces remotely from the control room. Additionally, local monitors will indicate abnormal fields by alarming in the fueling machine maintenance areas.

7.3 Regulatory/Policy

7.3.1 Regulatory Review

The staff found that design-related regulations contained in 10 CFR Parts 50 and 52 that are applicable and technically relevant to on-power fueling, as described in Section 7.2.4 of this report, address valid safety concerns. The staff did not find other design-related regulations to be applicable and technically relevant. All design-related issues concerning on-power fueling with potentially substantial safety significance have related regulations or GDC that apply. Therefore, the staff determined that existing regulations are adequate to support design certification of on-power fueling for the ACR-700.

The staff considers the use of Canadian codes and standards for on-power fueling component design, as discussed in the codes and standards section of this report, presents an issue which

may require rulemaking, and possibly, exemptions or relief requests from ASME Code requirements adopted by reference in 10 CFR 50.55a.

7.3.2 Potential Policy Issues

The on-power fueling process introduces three initiating events that are unlike those events previously evaluated for existing licensed LWRs. These events are local reactivity excursions resulting from on-power fueling, fuel channel pressure boundary failure resulting from on-power fueling, and inadequate heat removal for recently irradiated fuel in the fueling machine. The preliminary design relies on active control systems and interlocks to reduce the frequency of these initiating events because essential actions for on-power fueling are precursors to those events. Nevertheless, AECL review of initiating events indicates that the probability of an on-power fueling-induced LOCA and the probability of inadequate heat removal for recently irradiated fuel in the fueling machine are relatively high (approximately 4.9×10^{-2} events per year and 8.5×10^{-2} events per year, respectively). These two events also have a relatively high conditional probability of fuel damage to 12 fuel assemblies (the maximum amount of fuel in one channel or a filled fueling machine magazine, which is about one-half of 1 percent of the core). Consequently, the mitigation provided by an intact containment is important in minimizing the consequences of these types of fuel damage events.

The policy issue involves application of the Commission's safety goals and the principle of defense-in-depth to these initiating events. These events pose little risk of damage to a significant portion of the core or of a large early release of radioactive material. However, these fuel damage event sequences may have higher frequencies than more severe fuel damage event sequences evaluated for existing licensed LWRs. The staff recommends developing acceptance criteria for these potential limited fuel damage events provided that the overall risk is equivalent in terms of frequency times consequences.

Based on its review of the referenced documents provided by AECL, the staff identified a potential policy issue associated with the use of Canadian codes and standards for the design of components and structures of the ACR-700 on-power fueling technology as discussed in Section 5.3 of this report.

7.4 Technical Issues

AECL report, ACR 108-35000-LS-001, Revision 0, "The Technology of On-Power Fueling," issued September 2003, provides a very informative and useful description of the systems operation, equipment function, and design considerations for the on-power fueling system. Although this document provides information necessary for overall familiarization with this important FT, it does not supply the level of detailed technical information necessary to demonstrate how the design of the on-power fueling system components meet the NRC requirements for design certification.

The ACR-700 design must meet the NRC regulations for design certification specified in 10 CFR Part 52. The type of design information required is generally described in 10 CFR 52.47. This regulation states, in part, that the information submitted must include design information sufficiently detailed to permit the preparation of acceptance and inspection requirements by the NRC, and design, procurement, and installation specifications by a

combined license applicant. In addition to the regulations that are contained directly in 10 CFR Part 52, certain sections of the ASME Code, Division 1, are incorporated by reference in 10 CFR 50.55a. 10 CFR 50.55a incorporates the RCPB component design rules of ASME Section III, the ISI rules of ASME Section XI, and the inservice testing rules of the ASME OM Code by reference. Furthermore, to provide guidance to applicants, the NRC has developed RGs which describe the following: (1) the methods acceptable to the NRC staff for implementation of specific parts of the CFR; and (2) the methodology used by the staff in evaluating specific design and analytical criteria, including the mitigation of postulated accident conditions. These RGs are not substitutes for regulations, and compliance with them is not required. Methodology and design or analytical solutions other than those discussed in the RGs are acceptable if they provide an adequate basis for the findings requisite to the issuance of a design certification or license by the Commission. In addition to these RGs, the NRC has an SRP which the NRC staff uses to assist in the review of design certification applications. The SRP format is arranged by developing detailed guidelines for review of the implementation of the general technical issues described in the GDC in Appendix A to 10 CFR Part 50. The individual SRP sections are organized by related technical topics, and the organization of applications for LWR design certifications typically follow the format of the SRP.

Regarding the design of on-power fueling components and their support structure, including associated ACR-700 reactor assembly components when the fueling machine is a part of the Class 1 pressure boundary, ACR 108-35000-LS-001 does not provide technical information necessary to demonstrate conformance with the regulations described above. The following information, although not intended to be a complete listing of requirements, is typical of the level of technical information necessary for review of the design of on-power fueling mechanical equipment in order to meet the design certification requirements outlined in 10 CFR 52.47(a)(2):

- GDC 1, 2, 14, and 15 in Appendix A to 10 CFR Part 50; Appendix S to 10 CFR Part 50; and SRP Section 3.9.1 require review of the following:

  – operational and seismic design transients used in the design and fatigue analyses of all ASME Code, Section III, Class 1 and Class CS (core support) components, component supports, and reactor vessel (or equivalent) internal structures

  – description and verification of all computer codes used in the analyses of seismic Category I components

  (1) description of experimental stress analysis programs used in lieu of theoretical stress analyses

- GDC 1, 2, 4, 14, and 15 in Appendix A to 10 CFR Part 50; RGs 1.20, 1.68, 1.61, 1.92; and SRP Section 3.9.2 require review of the following:

  (1) a description of the startup test program for piping vibration, thermal expansion, and dynamic effects testing for all ASME Code Class 1, 2, and 3 systems, and those non-code systems whose failure could have an adverse effect on safety-related systems

(2) seismic analysis methods (response spectra, time history, equivalent static load) used for design of seismic Category I systems, piping, and components, and their support structures

(3) analysis of dynamic responses of structural components within the reactor vessel (or equivalent) caused by steady-state and operational flow transients

(4) a description of the preoperational and startup test program for flow-induced vibration testing of the reactor vessel (or equivalent) core/fuel assembly support and internal structures

(5) dynamic system analysis to confirm structural integrity with no loss of function of the reactor vessel (or equivalent) internals when subjected to combined loading from a LOCA and the SSE

- GDC 1, 2, 4, 14, and 15 in Appendix A to 10 CFR Part 50; RG 1.124, "Service Limits and Loading Combinations for Class 1 Linear-Type Component Supports"; RG 1.130, "Service Limits and Loading Combinations for Class 1 Plate-and-Shell-Type Component Supports"; and SRP Section 3.9.3 require review of the following:

  (1) design and service loading combinations and stress limits and allowable deformation criteria specified for design of ASME Code Class 1, 2, and 3 components, and Class CS structures (or equivalent)

  (2) design of pressure relief devices for overpressure protection of ASME Code Class 1, 2, and 3 components

  (3) design and structural integrity of supporting structures, including deformation criteria compatible with component operability requirements, for ASME Code Class 1, 2, and 3 components, including piping support systems

  (4) design considerations and inspection and testing requirements for mechanical and hydraulic snubbers used as support structure for ASME Code Class 1, 2, and 3 components

- GDC 46 in Appendix A to 10 CFR Part 50, 10 CFR 50.55a(g), and SRP Section 3.9.6 require review of the description of the inservice testing program specified for ASME Code Class 1, 2, and 3 pumps and valves.

- GDC 1 in Appendix A to 10 CFR Part 50; RG 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive Waste-Containing Components of Nuclear Power Plants (for Comment)"; and SRP Section 3.2.2 require review of the quality group classification system for pressure-retaining components in fluid systems important to safety, including piping and instrument diagrams defining the boundaries for system safety classes and quality group classifications.

The types of information described above are necessary for the staff to determine whether the proposed design satisfies the regulatory requirements contained within 10 CFR Part 52, Subpart B, and Appendix A to 10 CFR Part 50. The staff also requires basic design parameters

of the system (e.g., design pressure and temperature, component dimensions, component arrangement drawings, system piping and instrumentation diagrams, etc.) for its review. AECL report ACR 108-35000-LS-001 does not provide this type of detailed technical information.

In some cases, the applicant could address these issues by indicating that the on-power fueling structures, systems, or components would be designed, fabricated, erected, inspected, and tested consistent with applicable 10 CFR Part 50, Appendix A, GDC, including more detailed discussions of specific design criteria based on guidance included in associated RGs and SRP sections. The applicant may use methods other than those discussed in these regulatory guidance documents; however, the applicant should justify their use. This will require detailed information from the applicant to demonstrate that the use of alternative design, fabrication, inspection, and testing criteria will provide an acceptable level of quality and safety.

Based on review of the above referenced documents, the staff has determined that AECL should further address certain technical and regulatory issues. Sections 7.4.1, 7.4.2, and 7.4.3 discuss these issues.

7.4.1 Use of Canadian Codes and Standards

The design of the ACR-700, including portions of the on-power fueling system, presents some unique features, compared to LWRs designed in the United States, affecting the design of safety-related pressure boundary components. These unique design provisions require supplementary rules for pressure-retaining components, in addition to the design rules contained in the applicable portions of the ASME Codes.

Design certification applicants must meet the requirements of the ASME Codes as specified in 10 CFR 50.55a. Consequently, if the ACR-700 design does not satisfy certain ASME Code rules, the applicant must explain why the ASME Code rules cannot be satisfied, and either request an exemption from the regulations pursuant to 10 CFR 50.12, or request relief from the code rules by proposing alternatives to the ASME Code pursuant to 10 CFR 50.55a(a)(3). For those cases where no ASME Code rules exist, or where they are insufficient to address a portion of the ACR-700 design, the applicant must submit the alternative code rules to be used, including the technical basis for the acceptability of the proposed alternative rules, for NRC review and approval. Although Section 5.2 of ACR 108-35000-LS-001 identifies the alternative codes relevant to the design of the on-power fueling system components, it does not provide the required technical justification for acceptability, pursuant to 10 CFR 50.55a(a)(3)(i). Technical justification of these alternative codes is necessary, because the NRC staff will need to incorporate the use of these supplemental codes and standards into the regulatory framework as requirements in the ACR-700 design certification rule.

7.4.2 Inservice Testing Program

AECL report ACR 108-35000-LS-001 does not specifically address the inservice testing requirements for pumps and valves in the on-power fueling system. A Canadian standard, CAN/CSA-N285.4-94, has been developed to address the periodic inspection of nuclear plant components, but the requirements for inservice testing, pursuant to 10 CFR 50.55a(f), are not evident in this standard. AECL should to address these requirements for periodic testing of safety-related pumps and valves in the on-power fueling system. The NRC staff will need to

determine whether the design of the pumps and valves in the ACR-700 plant provides sufficient access to enable the performance of inservice testing in accordance with the ASME OM Code.

7.4.3 Classification of System Components

The description of the on-power fueling system does not contain sufficient detail for the NRC staff to determine how the system components were classified. The NRC staff was unable to determine where the system quality group classification and ASME Code classification boundaries are specified. This information is required to verify the adequacy of the system quality group classification boundaries with respect to NRC guidelines contained in SRP Section 3.2.2, and in RG 1.26.

RG 1.26 and the related SRP Section 3.2.2 provide the basis for the staff review of classification of various plant piping safety systems and components that have a safety-related function. The general classification is in terms of meeting certain requirements of the ASME Code, Section III. RG 1.26 establishes a method for classifying safety-related piping components into quality groups which are required to meet the requirements for ASME Code, Section III, Class 1, 2, or 3 components, or other requirements. Therefore, it is important to determine which of the plant safety-related systems and components must be so designed and which specific classification is appropriate.

The staff has reviewed the information provided by AECL in the referenced documents which relate, in part, to safety system classification. The description of the ACR-700 design and AECL proposed classification indicates that the safety systems will be generally constructed to ASME Section III, Class 1, 2, and 3 requirements, with some exceptions where the ASME requirements are supplemented or replaced by certain Canadian and/or other standards which the staff has not reviewed or approved. However, there are some differences between the current AECL classification system, which has six classes, and the RG 1.26 guidance, which has three classes for ASME Code, Section III, components, and a fourth quality group (i.e., D, for non-ASME Code, Section III, components). During the detailed review of the ACR-700 design for certification, in order to determine that the safety systems and related components have been adequately classified, the staff will need to determine that AECL has verified that all of the necessary ASME Code, Section III, and other requirements are met, either directly or as a result of meeting requirements equivalent to RG 1.26 guidance in all important aspects. On a case-by-case basis, where it is determined that ASME Code, Section III, requirements apply, but certain components do not meet such requirements, the staff may authorize alternatives to these requirements in accordance with 10 CFR 50.55a(a)(3).

The ACR-700 design is similar in many regards to currently licensed LWRs in the United States. The ACR-700 technical description and “Classification of Safety-Related Systems and Structures” (AECL Report No. 108US-03621-LS-001, 2003) generally describe the reactor (or equivalent) coolant system and the systems interfacing with it, in addition to other balance-of-plant systems. In general, the proposed safety system classification appears to be appropriate, with one major exception identified with the currently available level of detail. For some systems interfacing with the RCS, there is only one Class 1 isolation valve at each of the interfacing system class boundaries. Two Class 1 isolation valves in series are required at these interfaces, in accordance with 10 CFR 50.50a(c), in order to ensure the integrity of the Class 1 RCS and to prevent possible overpressurization of lower pressure systems. The staff

identified no other specific deficiencies from the descriptions provided in the above-mentioned documents. After a detailed review of the ACR-700 piping systems and related components during the design certification review, the staff may identify additional systems and individual components which will need to be classified as ASME Code, Section III, Class 1, 2, or 3.

In its review of the information provided by AECL during the preapplication review of safety system classification for the ACR-700 design, the staff did not identify any issues in the submitted information related to this topic which would preclude certification of the on-power fueling process and which cannot be resolved with additional technical clarifications and exchanges.

7.4.4 Review of NRC Regulations for Applicability

The staff reviewed the ACR-700 design documentation obtained from AECL and compared the design of the ACR-700 on-power fueling systems to the design-related regulations contained in 10 CFR Parts 50 and 52. Table 7-1, which follows, lists the NRC regulations the staff found applicable and technically relevant to the design of on-power fueling or interfacing systems and which serve as the basis for the staff's conclusion.

Table 7-1 NRC Design-Related Regulations Applicable to On-Power Fueling

10 CFR Parts 50 and 52	Design-Related Issues
50.2 Definitions	*Reactor coolant pressure boundary*—When locked onto a fuel channel of the reactor and with the closure plug not fully engaged in the fuel channel end fitting, the fueling machine head and snout, the fueling machine head relief valve, and all connected piping and hoses not isolated by two normally closed valves up to and including the outboard containment isolation valve are part of the RCPB. This encompasses the hoses in the catenary system that provide for fueling machine head cooling and pressurization. However, these hoses need not meet the requirements for ASME Code, Section III, Class I components when the requirements of 10 CFR 50.55a(c)(2) for isolation capability and makeup water are satisfied. *Safety-related structures, systems, and components*—Equipment associated with on-power fueling systems that forms part of the RCPB and that ensures the integrity of the RCPB is maintained (e.g., interlocks and control systems), and systems necessary to maintain coolant inventory and effectively cool irradiated fuel within the fueling machine head, must be safety related. RG 1.29, "Seismic Design Classification," includes a function-oriented list of generic LWR systems that have safety-related functions. By this guidance, fueling machine cooling and coolant inventory control systems and RCPB components should be safety related.
50.34 Contents of Applications; Technical Information	This regulation invokes Appendices A, B, and S to 10 CFR Part 50 and contains technical requirements related to Three Mile Island (TMI), which must be addressed for design certification per 10 CFR 52.47. Applicable GDC of Appendix A are addressed individually. Appendices B and S apply to safety-related components involved in on-power fueling. The following TMI requirement applies to on-power fueling-related systems of the ACR-700. Pursuant to 10 CFR 50.34(f)(2)(xxvi)), leakage control and detection must be incorporated into systems outside containment that contain (or might contain) accident source term radioactive materials following an accident to limit radiological exposures to workers and the public. This would affect the design of the coolant inventory control and cooling systems for the fueling machine because these systems are located outside containment and connected through the fueling machine to the RCS during on-power fueling. Review of conformance with the SRP specified in 10 CFR 50.34(h) is generally not relevant to on-power fueling systems because the concept of on-power fueling was not considered in development of the SRP.

10 CFR Parts 50 and 52	Design-Related Issues
50.46 Acceptance Criteria for ECC Systems for Light-Water Nuclear Power Reactors	This regulation invokes Appendix K to 10 CFR Part 50 and requires analysis of postulated LOCAs. Accidents initiated by on-power fueling would be a subset of the accidents involving fuel channel pressure boundary failure LOCAs.
50.48 Fire Protection	The required fire protection plan must address the possibility of a fire affecting equipment important to safety (i.e., safety-related equipment and nonsafety-related equipment whose failure could affect the integrity of the RCPB, prevent achieving or maintaining safe shutdown of the reactor, or result in potential offsite exposures comparable to 10 CFR 50.34(a)(1) guidelines) related to on-power fueling. Potential damage to equipment associated with on-power fueling from a design-basis fire should be limited such that the capability to safely shut down the reactor and maintain the integrity of the RCPB is ensured. Damage to the fueling control panel in the main control room or damage to control or power circuit cables serving the fueling machine could threaten this capability in that damage to this equipment could result in spurious operation of important devices. Mechanical interlocks that operate independently from control and power circuits to block spurious operation that could result from fire damage may be credited to reduce the effects of fire damage.
50.49 Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants	This regulation requires electrical equipment important to safety that must function following exposure to a harsh environment resulting from a DBE to be qualified for that environment. It would apply to controls and motors needed to engage/disengage the fueling machine as part of the RCPB since any unwanted disengagement could result in a small LOCA or possibly affect the ability to implement or maintain safe shutdown. It also applies to the electrical equipment that could affect the coolant inventory control and cooling functions for the magazine containing up to 12 irradiated spent fuel elements in the fueling machine head.
50.55a Codes and Standards	This regulation is design related and applicable to on-power fueling systems, but Section 5.2 of this report addresses the application of codes and standards in general.
50.60 Acceptance Criteria for Fracture Prevention Measures for Light-Water Nuclear Power Reactors for Normal Operation	This regulation invokes Appendices G and H to 10 CFR Part 50 and requires fueling machine heads constructed from ferritic material to demonstrate adequacy of fracture toughness.

10 CFR Parts 50 and 52	Design-Related Issues
50.63 Loss of All Alternating Current Power	This rule requires that the core be cooled and containment integrity be maintained for the specified duration using systems independent of the normal alternating current power distribution. The effect of on-power fueling on cooling of the core could be addressed by either evaluating the effect of the fueling machine attachment, including fuel and coolant inventory, on the ability to adequately cool the fuel within all fuel channels and attached fueling machine(s) for the required duration, or by demonstrating that the probability of a station blackout coincident with having a fueling machine attached to the reactor face is sufficiently small that it need not be addressed.
50.68 Criticality Accident Requirements	The criticality accident prevention requirements of this rule apply when fresh fuel and/or spent fuel elements are in the fueling machines. Although the quantity of fuel is similar to that handled in previously licensed U.S. LWRs, the configuration of the elements in a compact array within the fueling machine head makes analysis to the specific criteria in 10 CFR 50.68(b)(3) and (b)(4) appropriate. With justification, performance of a single bounding evaluation would be acceptable.
52.47 Contents of Applications	Pursuant to 10 CFR 52.47(a)(iv), the application must address technically relevant unresolved safety issues as well as medium- and high-priority generic safety issues. Generic Issues A17 (systems interactions) and B-63 (isolation of low-pressure systems connected to the RCPB) apply directly to on-power fueling systems. Design of the online fueling systems should not impair the operation of SSCs needed for DBA mitigation. The design should also not create any safety-significant events resulting from any subtle interactions with safety-related SSCs. Since the cooling system for the online fueling system interfaces through the fueling machine with the RCPB, the design of these interfacing systems should comply with the intersystem LOCA mitigation capability guidance specified in SECY-93-087. Other generic issues apply more generally to the ACR-700 design.

7.4.5 Evaluation of Operational Events and Failure Modes with Significant Consequences

Operating CANDU reactors have accumulated significant operating experience with on-power fueling operations. Although the preliminary design of the ACR-700 differs in several important aspects from the earlier designs of on-power fueling systems, operating experience provides relevant information about what equipment failures can occur, how likely a specific equipment failure is to occur, and what consequences can be expected from each type of equipment failure. The staff reviewed operating experience information provided by AECL for relevance to the ACR-700 preliminary design of on-power fueling systems. The staff also independently reviewed hundreds of international events in the International Incident Reporting System for

relevant operating experience from CANDU-type reactors, which confirmed AECL data and supplemented that data with additional relevant events from Indian reactors.

The staff also reviewed the preliminary design of on-power fueling systems to identify possible failure modes and initiating events that have the potential to result in those failure modes. The staff compared the results of this review with failure modes and initiating events identified by AECL to establish the completeness of the information.

The staff prepared Table 7-2, which follows, to document the results of the comparison and relevant operating experience for each failure mode. The staff developed the first two columns primarily from the event groups and event types identified in AECL Assessment Document 108-3660-ASD-001, Revision 1, "Systematic Review of Plant Design for Identification of Initiating Events," issued January 2004. The staff identified one additional event type not clearly described in this document involving the potential for on-power fueling to effect channel coolant conditions such that a regional core power excursion results. The significance of this event type depends on the final core design characteristics. In the third column, the staff listed relevant operating experience data, which AECL provided in Licensing Submission Document 108-35000-LS-001, Revision 0, "The Technology of On-Power Fueling," issued September 2003; and AECL letter 108US-01321-021-001, "Responses to NRC's RAIs on On-Power Fueling," dated May 17, 2004. Finally, in the fourth column, the staff listed design issues that had a significant potential to affect the frequency or consequences of the identified event types. The staff identified one issue not clearly described in the assessment document involving control of fueling machine maintenance door motion that could affect the frequency of damage to fueling machine cooling hoses and control cables.

Table 7-2 Analysis of Failure Modes and Operating Experience for On-Power Fueling

AECL Grouped Event Definition	Identified Event Types	Applicable Operating Experience	Specific Issues
Regional Core Power Excursion	Incorrect fuel loading (too many fresh assemblies)	Incorrect fueling sequence—precursor event (India, 1987)	Human factors, software, and FCS design
	Inadequate local flux compensation or measurement during fueling	Inadequate flux compensation during reverse fueling caused regional power excursion (India, 1996)	Neutron flux detection and reactivity control system design

AECL Grouped Event Definition	Identified Event Types	Applicable Operating Experience	Specific Issues
	Reactivity effects resulting from coolant conditions during fueling (e.g., void collapse caused by excess cold water injection)	No identified events	Fuel design, coolant channel flow, and fueling machine water supply system design
Loss of cooling to fuel while in the fueling machine head (while machine is away from reactor face)	Loss of coolant circulation to fueling machine	Coolant hose failures (multiple examples) (AECL)	Hose design and qualification, isolation capability, redundant flowpaths, and protection from external damage (catenary design and shield door design and motion controls)
		Circulation pump trip (AECL)	Coolant circulation system design for reliability
	Loss of coolant inventory from fueling machine	False water level indication during draining evolution (AECL)	Human factors, controls for draining fueling machine coolant, and reliability of monitoring instrumentation
		Incorrect indication of irradiated fuel inventory (AECL)	Human factors, controls for draining fueling machine coolant, and reliability of monitoring instrumentation
		Pressure boundary failure (AECL)	Design and qualification of pressure boundary components and seals
		Failed purge or drain valve (AECL)	Design and qualification of fueling machine valves and their control systems
	Loss of heat sink	No identified events	Design and qualification of cooling water systems

AECL Grouped Event Definition	Identified Event Types	Applicable Operating Experience	Specific Issues
Fueling machine-induced LOCA (while machine clamped to reactor)	Fueling machine releases from channel end fitting with channel closure incorrectly installed or not installed	Precursor events—fueling machine ram failures and jammed components have prevented proper installation of channel closures, but fueling machine remained attached to end fitting through reactor shutdown and recovery (AECL; India, 1990; and Korea, 1996)	Design of channel closure, reliability of monitoring to ensure proper installation of channel closure, and reliability and diversity of interlocks and controls that permit fueling machine release from channel
Fueling machine-induced EFF (while machine unclamped)	Combined axial and transverse motion of fueling machine causes damage to channel end fitting	No identified events	Reliability and diversity of controls and interlocks for fueling machine motion, and design of fueling machine drive mechanisms to limit impact forces
Mechanical damage to fuel while in transfer to spent fuel reception bay	Damage from operation of fuel transfer line valves	No identified events	Human factors and design reliability of controls and interlocks for fuel transfer line valves
	Damage from fuel transfer mechanisms	Broken fuel pins (AECL)	Human factors, design reliability of controls and interlocks for fuel transfer mechanisms, and design of fuel transfer mechanisms to limit applied forces
Fueling machine-induced LOCA (while machine clamped)	Failure of bridge brakes allows fueling machine motion that damages channel end fitting	No identified events	Design reliability of bridge brakes and elevator drives

AECL Grouped Event Definition	Identified Event Types	Applicable Operating Experience	Specific Issues
	Failure of control system allows improper or uneven bridge motion that damages channel end fitting	Permanently deformed end fitting when fueling machine drifted down after control system problem permitted brake release (AECL)	Human factors and design reliability of controls and interlocks for fueling machine bridge travel
Loss of heat transport coolant because of fueling machine water system failures (reactor operating at full power)	Seal leaks or pressure boundary failures in the fueling machine or connected components	Fueling machine to end fitting seal failure and hose failures (AECL)	Design and qualification of pressure boundary components and seals, isolation capability, and protection from external damage (catenary design, shield door design, and motion controls)
Leaks during throughflow defueling (reactor shutdown)	Seal leaks or pressure boundary failures in the fueling machine or connected components	Fueling machine to end fitting seal failure and hose failures (AECL)	Design and qualification of pressure boundary components and seals, isolation capability, and protection from external damage (catenary design, shield door design, and motion controls)
LOOP (reactor operating at full power)	Loss of drive mechanisms and coolant circulation until onsite power supplies start	Loss of offsite power events (AECL)	Design reliability of onsite power supplies and redundancy of electrical distribution systems
Partial channel flow blockage (reactor operating at full power)	Flow blockage caused by interference of on-power fueling components	No identified events	Design of on-power fueling components to maintain adequate channel coolant flow
	Flow blockage caused by foreign material entry	Foreign material fouling (AECL)	Design provisions for inspection of new fuel and fueling machine internal components

AECL Grouped Event Definition	Identified Event Types	Applicable Operating Experience	Specific Issues
Mechanical damage to fuel during on-channel fueling (inside containment—screened out of safety assessment because of low offsite radiological consequences)	Fuel transfer mechanisms apply excessive force to mispositioned or jammed fuel assembly	Broken fuel pins (AECL), precursor event involving mispositioned fuel assembly (India, 1999)	Design of fuel transfer monitoring instrumentation to detect mispositioned or jammed fuel assembly and design of fuel transfer mechanisms to limit applied force

7.4.6 Design Criteria Applicable to the Design of On-Power Fueling Systems

Because the GDC of Appendix A to 10 CFR Part 50 are sufficiently flexible and the design of the ACR-700 is sufficiently close to previously licensed U.S. LWRs, many of the GDC apply to the preliminary design of on-power fueling systems for the ACR-700. As specified in 10 CFR 50.34(a)(3)(i), applicants for design certification must (1) address conformance to the GDC as minimum requirements for the principal design criteria for water-cooled nuclear power plant designs similar to previously licensed reactors and (2) use the GDC as guidance for the development of principle design criteria for other types of nuclear power plants. Table 7-3, which follows, includes the staff's determination of the minimum applicability of specific GDC to the preliminary design of on-power fueling systems for the ACR-700.

The staff identified one additional concern that should be encompassed with the principal design criteria of the plant, namely, that design features that protect irradiated fuel from mechanical damage during fuel handling should be addressed. Fuel-handling systems must be designed to protect the integrity of the fuel cladding and fuel assembly structure, preferably by limiting by design the maximum forces that could be applied to the fuel assembly during handling, such that the potential for release of substantial quantities of radioactive material and the potential for damage to the reactor core from fuel debris is minimized.

Table 7-3 Application of General Design Criteria to On-Power Fueling Systems

Applicable GDC	Applicability to ACR-700 On-Power Fueling
Criterion 3—Fire protection	Fire damage or fire protection system actuations could affect fueling machine power and control systems such that safe shutdown of the plant or the integrity of the RCPB may be threatened. The safe-shutdown analysis should incorporate these potential effects to ensure the potential adverse effects of fires are minimized.

Applicable GDC	Applicability to ACR-700 On-Power Fueling
Criterion 4—Environmental and dynamic effects design bases	On-power fueling-related equipment important to safety must be designed to withstand the environmental and dynamic effects of DBAs. This includes protection of important control systems from the high temperatures and wetting associated with high-energy fluid system failures, as well as protection of RCPB components from large forces generated by high-velocity fluid impingement or impacts with moving heavy equipment. Issues specific to the ACR-700 include protection of RCPB components, including fueling machine catenary and reactor PT end fittings, from unacceptable damage resulting from motion of the maintenance shield door and fueling machine.
Criterion 10—Reactor design	The effects of on-power fueling must be factored into the design of the fuel and the fuel channel to ensure that fuel design limits are not exceeded for both normal fueling and AOOs, such as fueling machine hose ruptures.
Criterion 13—Instrumentation and control	Instrumentation for on-power fueling must be capable of monitoring variables and systems that can affect reactor core integrity (e.g., channel coolant flow and fuel power), RCPB integrity (e.g., channel closure position, RCPB leakage, fueling machine position, and fueling machine motion), and containment integrity (e.g., valve positions) during normal, upset, and accident situations. Control systems and interlocks must reliably ensure that the variables and systems function within necessary operating ranges. Important issues specific to the ACR-700 include defining parameters that must be measured to assure the establishment of adequate RCPB integrity during on-power fueling, and the quality and reliability of interlocks that assure these parameters are at acceptable values before the pressure boundary is shifted between the fueling machine and the channel closure.
Criterion 14—RCPB	The design of the fueling machine head and associated systems that form part of the RCPB during the on-power fueling process must be addressed under this criterion. In particular, the fueling machine to end fitting latch mechanism and the closure plug to PT latch mechanism must be designed to have a very low probability of abnormal leakage or rapidly propagating failure when properly seated, as determined by the associated control systems and interlocks.
Criterion 15—RCS design	The on-power fueling-related systems must be designed to assure the design limits for RCPB components (e.g., thermal cycles, temperature, and pressure) are not exceeded during normal operation, including AOOs.

Applicable GDC	Applicability to ACR-700 On-Power Fueling
Criterion 17—Electric power systems	The design of electrical power systems must reliably provide electric power to on-power fueling systems necessary to maintain the RCPB components, the fuel within the fuel channel, and the fuel within the fueling machine within design limits during normal operation and following AOOs. The electrical power system must also reliably provide electric power to systems and components necessary to maintain vital functions, such as maintaining adequate cooling of irradiated fuel in the fueling machine and preventing degradation of RCPB integrity, in the event of postulated accidents.
Criterion 19—Control room	Control panels for on-power fueling must be located in the main control room. Equipment controls and instrumentation must be provided at locations outside the control room to maintain adequate cooling of irradiated fuel in the fueling machine and maintain the reactor in a safe-shutdown condition, including maintaining RCPB integrity.
Criterion 26—Reactivity control system redundancy and capability	The control rods must be designed with the capability to reliably control local reactivity changes resulting from normal operation or AOOs, such as loading of fresh fuel or injection of cooling water during on-power fueling, such that acceptable fuel design limits are not exceeded. Based on the design of on-power fueling systems, the limiting reactivity addition that the control rods must control without exceeding specified acceptable fuel design limits should be defined. The preliminary design of the fueling machine would allow replacement of all the fuel within a single channel with new fuel without reloading the fueling machine.
Criterion 30—Quality of RCPB	Fueling machine components that are part of the RCPB must meet this criterion.
Criterion 31—Fracture prevention of RCPB	Fueling machine components that are part of the RCPB must meet this criterion.
Criterion 32—Inspection of RCPB	Fueling machine components that are part of the RCPB must meet this criterion.
Criterion 33—Reactor coolant makeup	The reactor coolant makeup system must be designed to assure fuel design limits are not exceeded as a result of water loss caused by fueling machine cooling water hose failure, fueling machine to channel seal failures, and other credible failures of small fueling machine pressure boundary components.
Criterion 44—Cooling water	This criterion requires suitable redundancy in the provision of cooling water for SSCs important to safety, such as systems necessary for adequate cooling of irradiated fuel within the fueling machine.

Applicable GDC	Applicability to ACR-700 On-Power Fueling
Criterion 45—Inspection of cooling water systems	The cooling water systems must be designed to permit periodic inspection of important components to assure the integrity and capability of the system.
Criterion 46—Testing of cooling water systems	The cooling water systems must be designed to permit periodic pressure and functional testing to assure the following: (1) the structural and leaktight integrity; (2) the operability and performance of active components, such as valves and interlocks; and (3) the operability of the system as a whole in providing emergency cooling to irradiated fuel.
Criterion 54—Piping systems penetrating containment	On-power fueling-related systems penetrating containment, including the fueling ports and cooling water lines, must be designed with leak detection, isolation, and containment capability for all configurations used during the on-power fueling process. If the fueling machine pressure boundary is credited as part of the containment boundary, such as by forming a closed system inside containment during fuel transfer, the fueling machine pressure boundary must be designed with a capability for testing its operability as a containment boundary and its leakage.
Criterion 55—RCPB penetrating containment	On-power fueling related systems penetrating containment that are an extension of the RCPB when the fueling machine is connected to the reactor face, such as the cooling water lines, must be designed with isolation valve capability specified by this criterion.
Criterion 56—Primary containment isolation	The spent fuel and new fuel ports, which are open to the containment atmosphere when the fueling machines are not connected, must be designed with isolation valve capability specified by this criterion.
Criterion 57—Closed system isolation valves	On-power fueling systems that are isolated from the RCS at all times and the configuration for fuel transfer where the fueling machine is connected to the spent fuel or new fuel ports may be treated as closed systems within containment. Configurations that provide a closed system within containment must be designed with the isolation valve capability specified by this criterion.
Criterion 61—Fuel storage and handling and radioactivity control	On-power fueling systems must be designed with the capability for periodic inspection and testing, shielding for radiation protection, and systems to control the release of radiation. The fueling machine must be designed with reliable, redundant methods for decay heat removal and with the capability to keep irradiated fuel located within the machine covered with water under normal or accident conditions.

Applicable GDC	Applicability to ACR-700 On-Power Fueling
Criterion 62—Prevention of criticality in fuel storage and handling	Criticality within the fueling machine must be prevented by keeping the fuel in a geometrically safe configuration. An acceptable standard is defined by 10 CFR 50.68.
Criterion 63—Monitoring fuel and waste storage	The fueling machine must be designed to monitor the water level, temperature, and radioactivity level of the cooling water within the fueling machine head to detect conditions indicative of a loss of residual heat removal or excessive radioactivity levels indicative of mechanical damage to fuel.

7.5 Conclusions

Based on the preapplication review of reference documents for the on-power fueling FT, the staff has identified a number of technical issues that will require more detailed information and development of appropriate acceptance criteria to facilitate a design certification review in accordance with the requirements of 10 CFR Part 52, Subpart B. The staff finds that, based on the level of general information included in the preapplication documentation provided to date, the design of the on-power fueling system components and support structures do not present any technically insurmountable issues. The staff identified one potential policy issue regarding the increased frequency of fuel damage events involving a small fraction of the core that could result from the concept of on-power fueling.

8. CONFIRMATION OF NEGATIVE VOID REACTIVITY (FT9)

8.1 Review Scope

This PASAR chapter summarizes the insights from the staff's review activities for preapplication FT9, "Confirmation of Negative Void Reactivity." As discussed below, significant additional work is needed and planned to support the expected design certification review. In a letter dated July 30, 2003, from V.J. Langman to B. Sosa of the NRC, AECL requested that FT9 be prioritized as one of the four critical areas to be addressed in the ACR-700 preapplication review. Because nuclear analysis validation issues figure prominently in the staff's assessment of CVR for FT9, this PASAR chapter also addresses the review of related code validation aspects of AECL FT3, "Computer Codes and Validation Adequacy."

All nuclear design information provided to date by AECL relates only to the ACR-700 equilibrium core design. Because AECL has provided no information on the designs of the startup and transitional cores, these and other nonequilibrium core fueling configurations are excluded from the scope of the preapplication review activities discussed in this chapter. The staff's review activities for design certification will address void reactivity, and other safety-related nuclear characteristics, for all anticipated fueling configurations and operating conditions of the ACR-700 core.

8.1.1 Regulatory Basis

The NRC regulations do not include a licensing requirement that the inherent reactivity feedback from coolant voiding must be negative. The only requirements on reactivity coefficients are in GDC 11, "Reactor Inherent Protection" (Appendix A to 10 CFR Part 50) which states, "the reactor core and associated coolant systems shall be designed so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity."

Associated guidance in the NRC SRP (NUREG-1800) is excerpted below:

> The only directly applicable GDC in the area of reactivity coefficients is GDC 11, which states "…the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity," and is considered to be satisfied in light water reactors by the existence of the Doppler and negative power coefficients. There are no criteria that explicitly establish acceptable ranges of coefficient values or preclude the acceptability of a positive moderator temperature coefficient such as may exist in pressurized water reactors at beginning of core life.
>
> The acceptability of the coefficients in a particular case is determined in the reviews of the analyses in which they are used, e.g., control requirement analyses, stability analyses, and transient and accident analyses. The use of spatial effects such as weighting approximations as appropriate for the individual transients are included in the analysis reviews. The judgement to be made under this SRP section is whether the reactivity coefficients have been assigned suitably conservative values by the applicant. The basis for that judgment

> includes the use to be made of the coefficient, i.e., the analyses in which it is important; the state of the art for calculation of the coefficient; the uncertainty associated with such calculations; experimental checks on the coefficient in operating reactors; and any required checks of the coefficient in the startup program of the reactor under review.

While the NRC has no regulatory requirement for negative void reactivity, the staff did raise a licensing policy issue concerning the existence of positive CVR effects in two of the reactor designs that were under NRC preapplication review in the early 1990s. The Commission-endorsed ACRS comment on this issue, as discussed in "Issues Pertaining to the Advanced Reactor (PRISM, MHTGR, and PIUS) and CANDU 3 Designs and Their Relationship to Current Regulatory Requirements" (SECY-93-092, 1993), is quoted below:

> We agree with the staff that the existence of a positive void reactivity coefficient is a significant concern, but that it should not necessarily disqualify a reactor design. The burden of showing that the consequences of those accidents that would be aggravated by a positive void reactivity coefficient are either acceptable or could be satisfactorily mitigated by other design features surely falls on the pre-applicant. On the other hand, the staff should state the criteria it will use to judge "acceptable" or "satisfactorily."

In an SRM dated February 13, 2001 ("Staff Requirements—COMJSM-00-0003—Staff Readiness for New Nuclear Plan Construction and the Pebble Bed Reactor"), the Commission directed the staff to assess its technical, licensing, and inspection capabilities and identify any enhancements that would be necessary to effectively carry out the agency's responsibilities in licensing new reactors. The Commission also directed the staff to critically assess the regulatory infrastructure supporting 10 CFR Parts 50 and 52 and other applicable regulations that may require updating.

While it is the responsibility of the applicant to demonstrate the safety of a proposed reactor, the NRC will conduct, as necessary, research activities to support the licensing efforts by applying existing knowledge and tools and creating new knowledge and tools as needed. This enhances the level of independence in the staff's review of information provided by the applicant.

For a design certification review of the ACR-700, the NRC staff will conduct specific review activities and audit analysis support efforts involving several major technical areas, including reactor systems analysis, probabilistic risk assessment (PRA), and reactor materials engineering. The primary goal in the area of reactor systems analysis is to enable the staff to independently predict and understand how the ACR-700 system responds under normal, transient, and accident conditions. The prediction and understanding of reactor system responses employ specific computational tools and expertise from the fields of nuclear analysis, T-H analysis, and severe accident analysis. AECL-requested review of CVR mainly involves the field of nuclear analysis.

Of particular significance to the staff's anticipated design certification review of ACR-700 are the nuclear analysis efforts needed for the following:

- the evaluation of CVR and other inherent reactivity feedback characteristics of the reactor design in relation to GDC 11

- the prediction of global and spatial reactivity transients in evaluating the reactor system in relation to applicable reactivity control design criteria (e.g., from Section III of Appendix A to 10 CFR Part 50)

- the prediction of detailed fuel burnup as well as static and transient core power distributions (and the associated evolution of fuel temperatures, stored heat, and decay power) for evaluating the reactor system in relation to GDC 35, "Emergency Core Cooling"

8.1.2 Overview of Coolant Void Reactivity in the ACR Nuclear Design

Section 4.3.1.2 of the ACR-700 technical description (AECL 10810-01371-TED-001, Revision 0), which is quoted below in its entirety, provides a useful introductory overview of this FT. (To enhance the introductory value, two paragraphs that originally appear in the middle of the quoted section are moved to the beginning. Figures and tables mentioned in the quoted text are not reproduced here.):

> The coolant-void reactivity (CVR) coefficient affects the nature of the reactivity transient in the early stages of a postulated accident. Whatever the sign of this coefficient in a nuclear power plant, one can identify transients and accidents which increase reactivity and others which decrease reactivity. For example for a reactor with a negative CVR, a loss of coolant would cause a reactivity decrease; whereas a steam main break would cause a reactivity increase. In a reactor such as ACR, as long as the coefficient is small in absolute magnitude, the safety of the plant is relatively insensitive to the sign or value of the CVR.
>
> In order to meet licensing requirements in some national jurisdictions, the ACR has been designed with a negative coefficient of coolant void reactivity. Since there is no safety advantage in making it overly negative, the design requirement is simply to have high confidence that it is negative.
>
> A CANDU reactor with H_2O coolant would have a high positive coolant-void reactivity (CVR) at the CANDU 6 lattice pitch of 286 mm. The key parameter that determines the coolant-void reactivity is the moderator-to-fuel volume ratio in the lattice cell. For the purpose of this discussion, moderator is defined as the D_2O between the fuel channels and coolant is the heat removal medium inside the fuel channels. The CANDU 6 reactor lattice has a large moderator-to-fuel ratio of 16.4, resulting in a well-moderated lattice that is optimized for the natural uranium (NU) fuel cycle. An effective way of reducing the coolant-void reactivity, for a CANDU reactor using H_2O coolant and SEU fuel, is to reduce the lattice pitch until an under-moderated lattice condition is reached. The lattice becomes under-moderated when the D_2O moderator alone cannot provide sufficient moderation to achieve maximum reactivity. The H_2O inside the fuel channel then functions as both coolant and moderator. Coolant-void reactivity is determined by the net result due to the loss of absorption (a positive reactivity change) and

the loss of moderation (a negative reactivity change). Spatial and spectral changes of the neutron flux in the lattice cell due to voiding of the coolant, as well as the nuclide composition of the fuel, also affect the coolant-void reactivity.

The space requirements between feeders on adjacent fuel channels in a CANDU reactor determine the minimum lattice pitch that is permissible. A detailed engineering assessment recommended a minimum lattice pitch of 220 mm and maximum calandria tube outside radius of 78 mm. However, the moderator-to-fuel volume ratio is still too high to achieve a slightly negative coolant-void reactivity.

Using a small quantity of burnable poison in the central pin of a CANDU fuel bundle can further reduce the coolant-void reactivity. There is an increase in the thermal neutron flux towards the center of the fuel bundle upon voiding of the coolant. Depending on the amount of burnable poison incorporated in the central fuel pin, this increase in neutron absorption could generate a negative reactivity component strong enough to reduce the overall coolant-void reactivity from a slightly positive value to a slightly negative value.

The CVR has been calculated for the ACR using widely accepted tools (WIMS, RFSP, MCNP) validated against a large range of experiments in heavy-water moderated channel geometry. These tests do not however cover the ACR conditions exactly, and therefore a substantial uncertainty allowance has been incorporated into the design, such that the predicted design-center whole-core CVR is about -7 mk[1]. This margin provides confidence that the design requirement should be met.

An experimental program is underway at AECL Chalk River Laboratories which will measure CVR in the ZED-2 reactor, in an ACR geometry with CANFLEX fuel. The results from the experiments are expected to reduce the size of the uncertainty in code predictions, and to enable a reduction in the uncertainty allowance in the ACR design. Such a reduction can be achieved by small changes to the enrichment, the dysprosium content or both. Since the design changes would be small, and since the plant behavior is insensitive to CVR, there would be little change in safety characteristics.

The reference ACR fuel design uses uniform 2.1% SEU fuel elements except the central element, which uses natural uranium (NU) fuel containing 7.5 wt% of the burnable poison dysprosium. This fuel design gives a core-averaged discharge burnup of 21 MWd/kgU. The geometrical specifications of the CANFLEX fuel bundle for lattice-cell calculations are presented in Table 4.3-1. The design data of the fuel channel is presented in Table 4.3-2, while Table 4.3-3 gives the

---

[1]For CANDU reactors, CVR effects have traditionally been evaluated in terms of the whole-core CVR, which is simply the change in the neutron multiplication factor, k, caused by the complete voiding (i.e., removal or steaming) of all liquid coolant in all of the fuel channels. More formally, CVR = 1/k(cooled) - 1/k(voided). Note that k is unity when a reactor is at steady power. The unit milli-k (mk) denotes a change in k of 0.001.

design data for the ACR fuel. Dysprosium is the best candidate to use as burnable poison in the ACR fuel because its burnout rate is most compatible with that of ^{235}U. Also, its chemical and physical properties are similar to those of gadolinium, which is widely used as burnable poison in LWR fuel. NU was chosen as the dysprosium carrier rather than graphite or depleted uranium.

There are two consequences of using burnable absorbers for coolant-void reactivity reduction:

(1) Higher fuel enrichment is required to overcome the parasitic load of the absorber, which also increases the fuel-fabrication cost, and

(2) The relative power in the central element that contains neutron absorber is reduced; however, the contribution of the central element to the total power production in the bundle is small, and the impact is reduced by using natural uranium (rather than depleted uranium) in the central element.

The fuel enrichment and the burnable poison concentration can be tailored to meet the design targets of fuel burnup and coolant-void reactivity.

Figure 4.3-1 shows the dimensions of the lattice pitch (LP), pressure tube (PT), calandria tube (CT), and the moderator-to-fuel volume ratio (VM/VF) in the NU CANDU and ACR lattices. The small ACR lattice results in a highly compact reactor core. The savings in D_2O cost is clearly demonstrated in Figure 4.3-2, which compares the size of this compact ACR core with those for other CANDU designs. The reference ACR-700 core has 284 fuel channels producing 731 MWe inside a 5.20-metre-diameter calandria shell. This is much smaller than the 7.6-metre-diameter calandria shell that is required to accommodate 380 fuel channels in the current NU CANDU 6 reactor, which produces 728 MWe. The ACR-700 core characteristics are presented in Table 4.3-4.

The compact lattice and slightly negative coolant-void reactivity result in moderately negative power feedback, exceptional stability, and other benign neutronic characteristics for all sizes of ACR reactors currently under consideration.

Although not mentioned above, a point worth emphasizing is that measurements of CVR in operating CANDU reactors are inherently difficult, have never been performed, and are not presently planned by AECL in the ACR-700 reactor core. The remainder of this chapter discusses the NRC staff's review comments and analysis insights related to AECL statements noted above.

8.1.3 Review Activities

The NRC staff developed a preliminary review plan to provide the early review of FT9 requested by AECL and to outline the continuation of related review and infrastructure efforts that would be needed through the design certification phase. The review plan for FT9 is closely coordinated with the related nuclear aspects of FT3.

The nuclear analysis review and infrastructure efforts needed for evaluating CVR and nuclear code validation adequacy are difficult to separate from those needed for other nuclear aspects of the anticipated ACR-700 design certification review. The following outlines the four coordinated elements of the staff's nuclear analysis efforts for reviewing the ACR-700 design:

(1) Nuclear analysis PIRT for the ACR-700—The NRC staff completed an initial PIRT process for ACR-700 reactor systems analysis, which was broken down into three coordinated PIRTs on nuclear analysis, T-H analysis, and severe accident analysis. The nuclear analysis PIRT for the ACR-700 considered two FoMs:

- CVR in large-break LOCAs
- steady-state core power distributions as initial conditions for fission power, stored energy, and decay heat in various LOCAs

(2) Independent predictions of CVR and other inherent reactivity responses in the ACR-700—Using existing higher-order methods, the staff has made significant early progress on providing independent predictions of ACR void reactivity effects and the competing neutronic phenomena that govern them. These efforts use rigorous models and available higher-order methods of steady-state neutron transport theory and nuclear transmutation analysis to help develop the modeling insights needed for establishing and testing practical audit analysis tools that adequately predict the effects of CVR and other safety-significant nuclear phenomena arising in ACR-700 operations, transients, and accidents.

(3) Evaluation of differential and integral nuclear data for validating ACR-700 nuclear analysis codes—These NRC activities include the following:

- the review and evaluation of experimental benchmark measurement techniques in ZED-2 and other applicable critical facilities (e.g., the Deuterium Critical Assembly in Japan)
- the development and uses of important nuclear cross-section covariance data
- the adaptation and application of sensitivity and uncertainty analysis (S/U) methods to evaluate the applicability and adequacy (i.e., phenomenological prototypicality) of benchmark validation experiments in ZED-2 and other facilities and to help quantify, from the set of integral benchmark results, the biases and uncertainties in the computed predictions of CVR and other nuclear characteristics of the ACR-700 core

(4) Establishing and testing of independent lattice physics and core simulator models for the ACR-700—These efforts entail the following:

- adaptation, testing, and application of two- and three-dimensional SCALE/Triton (see "SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation," NUREG/CR-0200, Revision 7, ORNL/NUREG/CSD 2/R7, June 2004, draft)) lattice models of ACR fuel and transverse absorber rods to provide the lattice-smeared and energy-collapsed

nuclear data needed by the Purdue Advanced Reactor Core Simulator (PARCS) code (see T.J. Downer, et al., 2002)

– adaptation, testing, and use of PARCS code models, coupled with TRACE (see "TRAC-M/FORTRAN 90 (Version 3.0) Theory Manual," NUREG/CR-6724, 2001) or RELAP (see "RELAP5/MOD3.3 Code Manual, Volume 1: Code Structure, System Models, and Solution Methods," NUREG/CR-5535, Revision 1, 2004) system T-H models, for simulating the operating and transient behavior of the ACR-700 core

These activities include technical exchanges with AECL, cooperative meetings with the staff of the CNSC, and the review and supporting audit analysis of related technical information provided by AECL. The following sections summarize early results and insights from these review and audit analysis activities.

8.1.4 Information Reviewed

The reference core design that AECL described when the preapplication review started in September 2002 had an equilibrium reload fuel design with an enrichment of 2.0 weight percent (wt%) uranium-235 (U-235), an unspecified loading of Dy burnable poison in the central NU fuel pin, and an average discharge burnup of 20 GWd/t. According to AECL, the associated whole-core CVR was -3 mk ("ACR Workshop—Core Design and Reactor Physics," AECL Presentations at NRC, September 25, 2002).

In June 2003, AECL changed the fuel design to make the calculated CVR more negative and thereby increase confidence that the actual CVR will still be negative after accounting for calculation uncertainties and variable operating conditions. The revised fuel design specifications, as indicated above and in the ACR-700 technical description (i.e., enrichment of 2.1 wt% U-235, burnable poison loading of 7.5 wt% Dy, average discharge burnup of 21 GWd/t), gave an AECL-calculated whole-core CVR of -7 mk.

AECL's latest reference core design, as presented in March 2004 (see "ACR-700 Technical Description," AECL 10810-01371-TED-001, Revision 1), increases the number of fuel channels from 284 to 292 without changing the core lattice dimensions or fuel design and retains the same AECL-predicted whole-core CVR value of -7 mk.

Listed below are the documents reviewed to date that are relevant to FT9.

- WIMS-AECL Theory Manual, COG-00-077, Proprietary, December 10, 2002
- System Validation Manual for WIMS-IST/DRAGON-IST/RFSP-IST Reactor Physics Code Suite, COG-01-144, Proprietary, December 10, 2002
- WIMS-AECL Release 2-5d Users Manual, COG-94-052/RC-1176, Proprietary, December 10, 2002
- DRAGON Theory Manual—Part 1: "Collision Probability Calculations," IGE-236, Proprietary, December 10, 2002

- “Development of Simple-Cell Model for Performing History-Based RFSP Simulations with WIMS-AECL,” FFC-RCP-005, Proprietary, December 10, 2002
- “A Phenomenon-Based Matrix of Texts for Use in Validation of Reactor Physics Codes Employed in Nuclear Safety Analysis,” TTR-623, Proprietary, December 10, 2002
- RFSP-IST User’s Manual, TTR-734, Proprietary, December 10, 2002
- RFSP-IST Theory Manual, TTR-738, Proprietary, December 10, 2002
- “Technical Basis for the Validation of Computer Programs Used for Safety Analysis of the ACR Design,” 108US-03500-TBD-001, May 20, 2003
- “ACR-700 Reactor Physics Design,” 10810-03300-ASD-001, Proprietary, June 15, 2003
- Data on physics and fuel sent with responses to RAIs 1–12, Proprietary, June 15, 2003
- Nonproprietary DRAGON-IST code, input deck and reports, Public, July 22, 2003
- WIMS-AECL Release 2-5d, RFSP-IST Version 3-03, Proprietary, July 25, 2003
- DRAGON-IST Reports, Proprietary, July 25, 2005
- “Safety Analysis Computer Code Qualification: Status and Plan,” 108-03510-225-001, Public, August 5, 2003
- “Summary of Verification Tests for WIMS-IST,” COG-00-226, Proprietary, August 12, 2003
- “Initial Conditions and Standard Assumptions Safety Analysis Basis,” 10810-03510-AB-001, Public, August 13, 2003
- “A Phenomenon-Based Matrix of Tests for Use in Validation of Reactor Physics Codes Employed in Nuclear Safety Analysis,” TTR-623, Proprietary, October 17, 2003
- “ACR Physics Experiments in ZED-2 using NU Fuel,” 108-123110-440-001/FFC-RRP-4541, Proprietary, November 13, 2003
- “Planned Physics Experiments in ZED-2 in Support of ACR,” 108-123110-440-002/FFC-RRP-464, Proprietary, November 13, 2003
- “Analysis of Full-Core and Substitution Experiments with 28-Element UO2 Fuel at 20–22.86 cm Pitch,” 108-123110-440-003/FFC-RRP-478, Proprietary, November 13, 2003
- “Substitution Experiments on 37-Element LVRF Fuel into a 21.59-cm Pitch ZED-2 Lattice,” 108-123110-440-004/FFC-RRP-497, Proprietary, November 13, 2003

- "Fine-Structure Measurements on 37-Element LVRF in a 21.59-cm Pitch ZED-2 Lattice," 108-123110-440-005/FFC-RRP-516, Proprietary, November 13, 2003

- "Analysis of Substitution Experiments on 37-Element LVRF Fuel into a 21.59-cm Pitch ZED-2 Lattice," 108-123110-440-006/FFC-RRP-523, Proprietary, November 13, 2003

- "Validation of WIMS-AECL on Tight Pitch ZED-2 Experiments Using Light Water Cooled 28-Element NU Fuel," 108-119190-440-006/FFC-RRP-507, Proprietary, November 13, 2003

- RFSP-IST User's Manual (hard copy), TTR-734, Proprietary, November 18, 2003

- "Void Reactivity Error Assessment for CANDU Reactors," COG-01-030, Proprietary, December 4, 2003

- "Calculations of Coolant-Void-Reactivity Components in Lattices of NG CANDU Type," FFC-RRP-398, Proprietary, December 16, 2003

- "Recommended Composition for Simulated Burned-up ACR Fuel for ZED-2 Measurements," 108-119190-440-005, Proprietary, December 16, 2003

8.2 Regulatory Issues

As noted previously, NRC regulations do not include a licensing requirement that the reactivity feedback from coolant voiding must be negative. The only requirements for reactivity coefficients are those in GDC 11. Nevertheless, in SECY-93-092, the staff raised a licensing policy issue concerning the existence of strongly positive CVR effects in two of the reactor designs that were under NRC preapplication review in the early 1990s.

There may be a need for more specific guidance on the confidence levels to be considered in assessing the impact of reactivity coefficient uncertainties in the contexts of (1) establishing GDC compliance and (2) best-estimate accident analysis methods.

8.3 Technical Issues

8.3.1 Summary of Evaluations and Conclusions by the Nuclear Analysis PIRT

The NRC PIRT process for the ACR-700 (see the NRC PIRT Report for ACR-700 Reactor Systems Analysis, to be issued) was organized into the technical discipline areas of nuclear analysis, T-H analysis, and severe accident analysis. Of particular interest for CVR are the PIRT activities for nuclear analysis.

The nuclear analysis PIRT subpanel selected two scenarios for consideration. The first scenario is a large LOCA resulting from either an inlet or outlet header break. This LOCA is large enough to initially void half the channels, namely those that flow directly into or out of the affected header. The channels not connected to the affected header will not void immediately. Thus, a checkerboard voiding of alternate fuel channels is initially created. These scenarios are of interest vis-à-vis the CVR. Checkerboard voiding will happen in various large-break

scenarios. In the inlet header stagnation break, flow stagnation results when the pumped flow forces are balanced by the opposing break-induced flow forces in all channels fed by the broken inlet header. The stagnation break area corresponds to about 25 percent of the header cross section. Simulations provided by AECL show that pronounced checkerboard voiding happens in about 1 second, with reactor trip assumed at 2 seconds, and followed seconds later by full-core voiding (see "Scenario Description for ACR-700 Inlet Header Critical Break LOCA," AECL, Public, 2004)). A successful trip makes the CVR no longer relevant.

The FoM of interest is the calculation of the reactivity effect of the aforementioned voiding under LOCA conditions—namely, the CVR. The CVR of interest is obtained from steady-state calculations of the core multiplication factor, k or k_{eff}, at cooled operating conditions and then at the same conditions but with either half or all of the channels voided. The core is assumed to be at equilibrium fueling conditions, as the transition core has not yet been designed. This FoM is considered important for two reasons. In existing CANDU reactors, the CVR is positive, and the ACR is specifically designed to have a negative void reactivity. The confirmation of negative void reactivity is one of the critical FTs proposed by AECL for the NRC's preapplication review of the ACR-700.

The calculation of the CVR was divided into three components, including reactor conditions, core simulation (neutronics and T-H), and lattice physics. The PIRT with phenomena for each component was developed with ranking of importance and knowledge level. The results show gaps in all three elements of the calculation of CVR that need to be filled by a combination of experimental data and analytical support. The phenomenon with the largest discrepancy between importance and knowledge level relates to spatial homogenization of cross sections for use in core simulations.

The experiments recognized as important are those that are already planned by AECL; namely, PIEs that are to be done for CANFLEX fuel and experiments in the ZED-2 facility. The latter are particularly important as they would acquire not only data directly related to the CVR but also data relevant to other reactor physics parameters. The value of the ZED-2 experiments is primarily to validate computational methods over a variety of conditions. They can also be used to directly calculate the CVR for an isolated fuel bundle and provide information on its uncertainty. The PIRT also shows that understanding of leakage effects would be particularly useful. Leakage is significant between ACR channel nodes, which is especially important when the void is in a checkerboard pattern, and from peripheral channels to the adjacent reflector regions. Since the checkerboard situation exacerbates the homogenization problem, which was identified as particularly important, ZED-2 may also be very useful if it can represent such a configuration. In general, it would be beneficial for the NRC to follow the ZED-2 work so that results can be used to validate the NRC's calculational methods and to resolve the leakage issues brought up in the PIRT.

The PIRT suggested two types of calculations. One type consists of sensitivity calculations, using appropriate methods, in order to understand the effect of different parameters. Examples of this include sensitivity studies to look at the effect on CVR of more spatial detail for fuel temperature, changes in geometry caused by PT sagging, and manufacturing tolerances for the fuel element containing Dy. The second type of calculation suggested is a benchmark calculation to check the validity of less rigorous methods. For lattice physics analysis, the obvious choices are continuous-energy Monte Carlo methods for use as the benchmark. Examples of applications are to check multigroup cross-section libraries, resonance treatment,

and the effect of transverse control devices. Questions that arise on the core level should be resolved by the use of Monte Carlo as well as a lattice physics code in a color set configuration (multiple fuel lattice cells represented). Among other things, this would help resolve some of the issues in treating cell-to-cell leakage effects. The latter is particularly important in determining homogenized cross sections and, as stated above, this is one of the key nuclear analysis issues.

The second scenario of interest to the nuclear analysis panel is also a LOCA. However, in this case, the panel was not interested in particular scenarios. Rather, it was interested in all LOCAs as they all have in common the need to know the decay heat level throughout the core and hence, also the initial power distribution. The FoM of interest is the calculation of (1) the bundle power distribution throughout the core at full power and (2) the corresponding peak fuel element power density. The former helps determine the hydraulic conditions during the event and the latter is used specifically to calculate the peak-clad temperature. The assumption is that the core is at equilibrium fueling conditions, as there is insufficient information at this time to consider initial or transitional cores. This FoM is considered important because it directly impacts one of the acceptance criteria for this event, the peak clad temperature.

The PIRT was again developed according to phenomena related to reactor conditions, core simulation, and lattice physics. The experimental work is from ZED-2 and PIEs; the latter provided validity for calculations of isotopic composition. The most important uses of the ZED-2 experiments vis-à-vis power distributions are to compare measurements with calculated power distributions within the fuel bundle (form factor) and to provide measurements at a simulated core-reflector interface in order to help validate the core simulator modeling of these boundaries.

The calculational support suggested by the PIRT includes sensitivity studies using appropriate methods. Examples of this include the sensitivity of power distribution to the uncertainty in control device position, to changes in xenon distribution as a result of refueling, to different cross-section libraries, to cross-section group structure, and to the displacement of heavy water with structural material. Benchmark calculations using methods more rigorous than those expected to be the norm are also suggested. The examples identified as being most urgent to resolve relate to calculating the bundle power near the core-reflector interface and the fuel element power everywhere, the latter requiring dehomogenization (flux reconstruction). Interstitial effects, end effects, and core-reflector interface effects make the reconstruction difficult. To resolve these issues, one suggestion is to do benchmark calculations (e.g., using Monte Carlo methods) to represent three-dimensional subregions of the core in detail. Other examples of cases where benchmarks would be useful include consideration of full and partial rod insertion, axial reactor boundary conditions, and the application of assembly discontinuity factors.

8.3.2 Staff Insights from Independent Analysis of Coolant Void Reactivity

As previously discussed, the NRC staff has performed a preliminary analysis of ACR-700 neutronics behavior during full-core voiding and partial voiding. In general, the staff found reasonable agreement with the applicant's conclusion regarding a negative CVR for full-core voiding (not accounting for uncertainties). However, as a result of the PIRT process, the NRC recognized that checkerboard voiding should be considered as an important FoM for reactivity

effects in large-break LOCAs. Pursuant to insights emerging from the nuclear analysis PIRT process, the staff completed best-estimate neutronic calculations that predict the CVR to be positive during the initial checkerboard voiding of alternate fuel channels in large-break LOCAs. Such a positive reactivity response would cause reactor power to initially rise during the first few seconds before shutdown rods can be inserted. The calculations showing a positive checkerboard void reactivity used the same models and methods that show a negative whole-core CVR.

The NRC calculations used static calculations with the HELIOS code (see J.J. Casal et al., "HELIOS: Geometric Capabilities of a New Fuel-Assembly Program," 1991), a lattice physics tool similar in function and rigor to the WIMS-IST code used by AECL, in conjunction with preliminary static calculations by the PARCS core simulator code. These NRC analyses include detailed evaluations of the competing neutronic phenomena that contribute to the net CVR effects. Initial work has also started on employing existing higher-order methods in the MCNP/MonteBurns code package (see Los Alamos National Laboratory, "MCNP—A General Monte Carlo N-Particle Transport Code," Version 5, Volume 1: Overview and Theory and Volume II: Users Guide, LA-UR-03-1987 and LA-CP-03-0245, 2003; and D.L. Poston and H.R. Trellue, "User's Manual, Version 2.0 for MonteBurns," LA-UR-99-4999, Los Alamos National Laboratory, 1999) to help evaluate and qualify the use of practical lattice physics codes, like HELIOS, WIMS-IST, or SCALE/TRITON, in predicting burnup-dependent isotopic compositions, neutron fluxes, and nuclear reaction rates in the ACR-700 fuel and for the essential smearing and collapsing of lattice cross-section data for use in dynamic core simulator codes like PARCS (see "PARCS: Purdue Advanced Reactor Core Simulator," 2002) or RFSP-IST. Greater detail on these independent analysis efforts and the emerging technical insights on CVR appears in T.J. Downar, C.A. Cotton, and D. Lee, "Assessment of PARCS Code Modeling Capabilities Needed for Simulating Coolant Void Reactivity Effects in ACR-700," issued 2004; and D. Diamond, "ACR-700 CVR Calculations," issued 2004. The following three subsections provide highlights of these efforts.

8.3.2.1 Analysis of ACR Nuclear Phenomena Contributing to the Net Coolant Void Reactivity

To test the physics codes used in the analysis and gain a better understanding of the physics of ACR-700 CVR, the staff performed simple lattice calculations for a conventional NU CANDU as well as for the ACR-700 using the HELIOS lattice code with a 47-group cross-section library. The fuel lattice specifications for the conventional CANDU core appear in J.J. Whitlock, W.J. Garland and M.S. Milgram, "Effects Contributing to Positive Coolant Void Reactivity in CANDU," issued 1995. Figure 8-1 shows the layout of fuel channels and moderator in the respective core lattices.

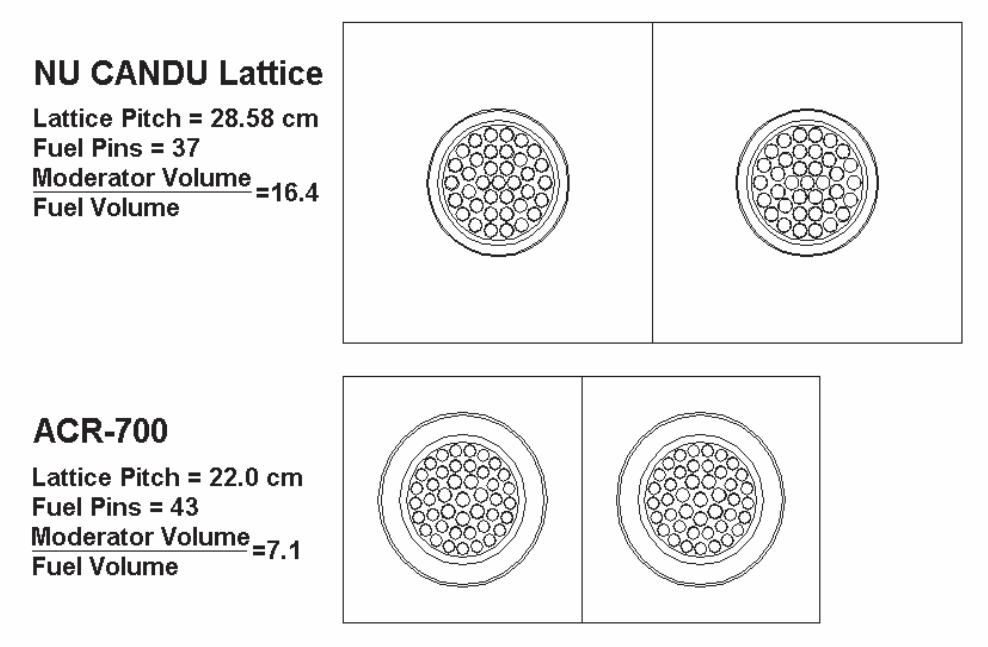

Figure 8-1 Fuel Channel Lattice Geometry in Conventional NU CANDU and ACR-700 Cores

The reactivity caused by complete coolant voiding was calculated as CVR = 1/k(cooled) - 1/k(voided). An initial sequence of calculations was performed in order to understand the impact on CVR of the various design changes between conventional CANDUs and the ACR-700. As indicated in Table 8-1, HELIOS predicted a CVR of +17.1 mk for the NU CANDU lattice, which agrees well with the published WIMS-AECL results at the same conditions (Whitlock, 1995).[20] The next four calculations show the effect of the remaining design changes leading to the ACR-700 reference design. First, the conventional CANDU lattice pitch was retained, but with CANFLEX fuel pins at 2.1 percent enrichment and light water replacing the heavy-water coolant. This increased the CVR to +39.5 mk. From this design, the lattice pitch was then reduced to the ACR-700 specifications, resulting in a significant decrease in the CVR from +39.5 mk to +4.7 mk. The Dy burnable absorber was then added to the central pin which made the lattice CVR negative at -1.3 mk, which agrees reasonably well with AECL results. In one final calculation, the lattice fuel was depleted to 10 GWd/t, which made the CVR even more negative at -5.6 mk. AECL has indicated that core leakage will add an additional few mk of negative reactivity to the CVR.

Table 8-1 Effect on CVR from Core Design Choices in Conventional CANDUs and Reference ACR-700 (Nominal Calculated Values)

Coolant	Channel Lattice Geometry	Fuel Bundle Type	Uranium in Outer Pins and Center Pin	Center-Pin Burnable Absorber	Burnup	Lattice Whole-Core CVR
D_2O(CANDU)	CANDU	37-Pin (CANDU)	NU and NU (CANDU)	None (CANDU)	0 GWd/t	+17.1 mk
H_2O (ACR)	CANDU	CANFLEX (ACR)	2.1% SEU and NU (ACR)	None (CANDU)	0 GWd/t	+39.5 mk
H_2O (ACR)	ACR	CANFLEX (ACR)	2.1% SEU and NU (ACR)	None (CANDU)	0 GWd/t	+4.7 mk
H_2O (ACR)	ACR	CANFLEX (ACR)	2.1% SEU and NU (ACR)	7.5% Dy (ACR)	0 GWd/t	-1.3 mk
H_2O (ACR)	ACR	CANFLEX (ACR)	2.1% SEU and NU (ACR)	7.5% Dy (ACR)	10 GWd/t	-5.6 mk

The sequence of calculations here confirms that the dominant design changes contributing to the negative CVR are the change in the pitch to the ACR CANFLEX lattice and the addition of the Dy burnable absorber. The “Assessment of PARCS Code Modeling Capabilities Needed for Simulating Coolant Void Reactivity Effects in ACR-700” (2004) includes a detailed analysis of the negative CVR contribution by the isotopes of Dy.

The computed neutron energy spectra in Figure 9-2 show that voiding the NU CANDU fuel lattice results in energy shifts for the fast, epithermal, and thermal parts of the neutron energy spectrum. The effects of these spectral shifts upon voiding can be understood qualitatively in terms of spectral components of the classic four-factor formula for neutron balance in multiplying systems. Upon voiding in NU CANDUs, the local loss of neutron downscattering does two things. First, it increases the fast flux in the fuel region, thereby increasing the fast-neutron-induced (greater than1.0 MeV) fission of U-238, and second, it shifts the slowing down of neutrons away from the fuel and its epithermal U-238 capture resonances and into the moderator, thereby increasing the resonance escape probability. The loss of thermal-upscatter by hot coolant of the neutrons coming from the lower temperature moderator shifts the thermal energy spectrum in the fuel to a lower energy range corresponding to a larger thermal fission cross-section of U-235 and an increase in the thermal utilization. As this discussion will soon

show in quantitative terms, these spectral factor changes are all positive contributors to void reactivity in NU CANDUs.

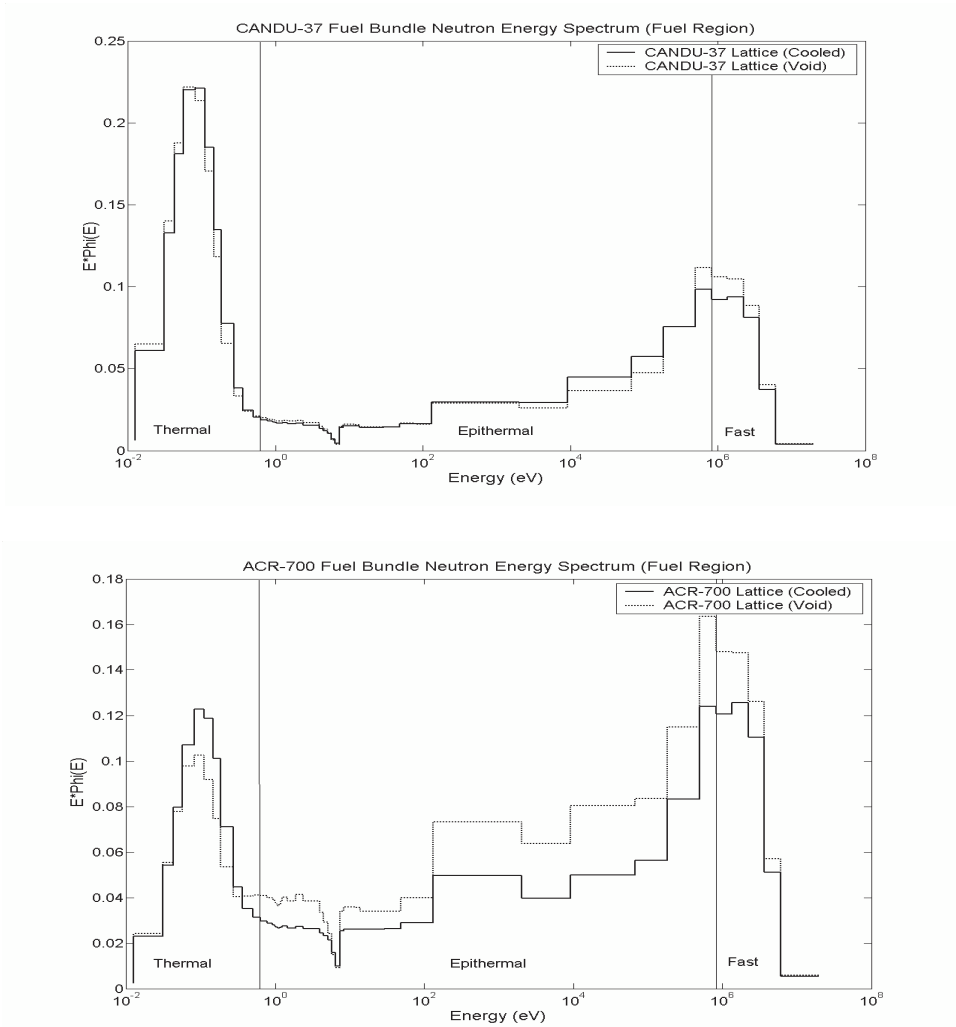

Figure 8-2 HELIOS-Computed Neutron Energy Spectra in Fuel Region for Cooled and Voided Core Lattices in Conventional CANDU (top) and ACR-700 (bottom) (0 GWd/t)

The voiding-induced spectral shifts in NU CANDU are relatively subtle because most of the neutron moderation takes place within the much larger moderator region that remains in place during coolant voiding. By contrast, the ACR spectra in the lower part of Figure 9-2 show a pronounced reduction in the thermal flux and an increase in the epithermal flux in the fuel lattice upon coolant voiding. This flux shift occurs primarily because of the reduced lattice pitch in the ACR lattice and the loss of light-water coolant which has a better downscattering ability (i.e., a higher scattering cross section with more energy loss per scatter) than the heavy-water coolant in the NU CANDU lattice.

The specific contributions to the reactivity changes for both the NU CANDU and ACR lattices were analyzed using the reaction rate edits from HELIOS to construct each of the factors in the familiar four-factor formula. This approach is consistent with the work reported by Whitlock, Garland, and Milgram (1995) for a typical CANDU fuel lattice. Consistent with Whitlock's approach, the resonance escape probability in the four-factor formula was separated into an epithermal and fast factor to more clearly isolate the physics of coolant voiding. The reaction rates were edited with the following energy group structure:

- 0 eV to 0.625 eV (thermal)
- 0.625 eV to 0.821 MeV (epithermal)
- 0.821 MeV to 10 MeV (fast)

Table 8-2 first compares the HELIOS results for NU CANDU to those reported by Whitlock, Garland, and Milgram (1995). The positive CVR for the NU CANDU core lattice is composed of a summation of relatively small contributions from each of the four-factor terms. In contrast, as shown in the remaining columns of Table 8-2, the four-factor CVR contributors in the ACR-700 lattice are much larger in magnitude, and the small negative void reactivity is the result of differences in large positive and negative four-factor spectral components. Small uncertainties or variations in these competing large contributors will therefore clearly have a comparatively large effect on the net CVR in the ACR-700.

Table 8-2 Calculated CVR Components (mk) in Core Lattices of Conventional CANDU and Reference ACR-700

Four-Factor Spectral Component of Full-Voiding Lattice CVR	WIMS-AECL*	HELIOS-1.8			
	Conventional CANDU			Preliminary ACR-700 Core Design**	
	0 GWd/t	0 GWd/t (Fresh)	0 GWd/t (Equil-Xe)	0 GWd/t (Fresh)	0 GWd/t (Equil-Xe)
fast fission factor, $\Delta\rho_g$	+4.4	+4.0	+4.3	+35.7	+38.4
fast resonance escape probability, $\Delta\rho_{PF}$	-3.2	-2.4	-2.5	-3.8	-3.9
epithermal resonance escape probability, $\Delta\rho_{PE}$	+9.9	+10.2	+10.6	-68.3	-70.6
thermal utilization factor, $\Delta\rho_f$	+3.3	+4.3	+4.5	+40.0	+40.1
reproduction factor, $\Delta\rho_\eta$	+1.8	+1.4	+1.1	-5.2	-6.1
TOTAL = Full-Voiding Lattice CVR	+16.3	+17.5	+17.9	-1.6	-2.0

*From Whitlock, Garland, and Milgram, 1995
** From "ACR-700 Technical Description," AECL 10810-01371-TED-001, Revision 0

8.3.2.2 Analysis of Partial Voiding Effects within Channels

The single-channel HELIOS lattice model employed in the preceding analysis was also used to analyze partial voiding of the ACR lattice. Two sets of calculations were performed in order to compare the homogeneous versus heterogeneous voiding of coolant within channels. In the first set of calculations, the coolant was removed from the channel by uniformly reducing the coolant density from 0 percent to 100 percent in the lattice. In the second set of calculations, heterogeneous voiding of the lattice was studied by removing coolant from the channel in sequential one-seventh pie slices of the lattice. Even though such a sector-wide voiding scenario is not physical, the results of these calculations suggest a potential need to consider stratified and other nonuniform voiding patterns within channels and provide physical insights that contribute to the understanding of the actual checkerboard voiding arrangements which will be discussed shortly. As in the previous section, reaction rate edits were extracted from HELIOS to construct each factor in the four-factor formula for both the homogeneous and heterogeneous cases. All calculations were performed for fresh fuel (0 GWd/t). Figure 8-3

shows the total reactivity changes observed in the single-channel model for reducing the coolant homogeneously and heterogeneously (sequential one-seventh slices).

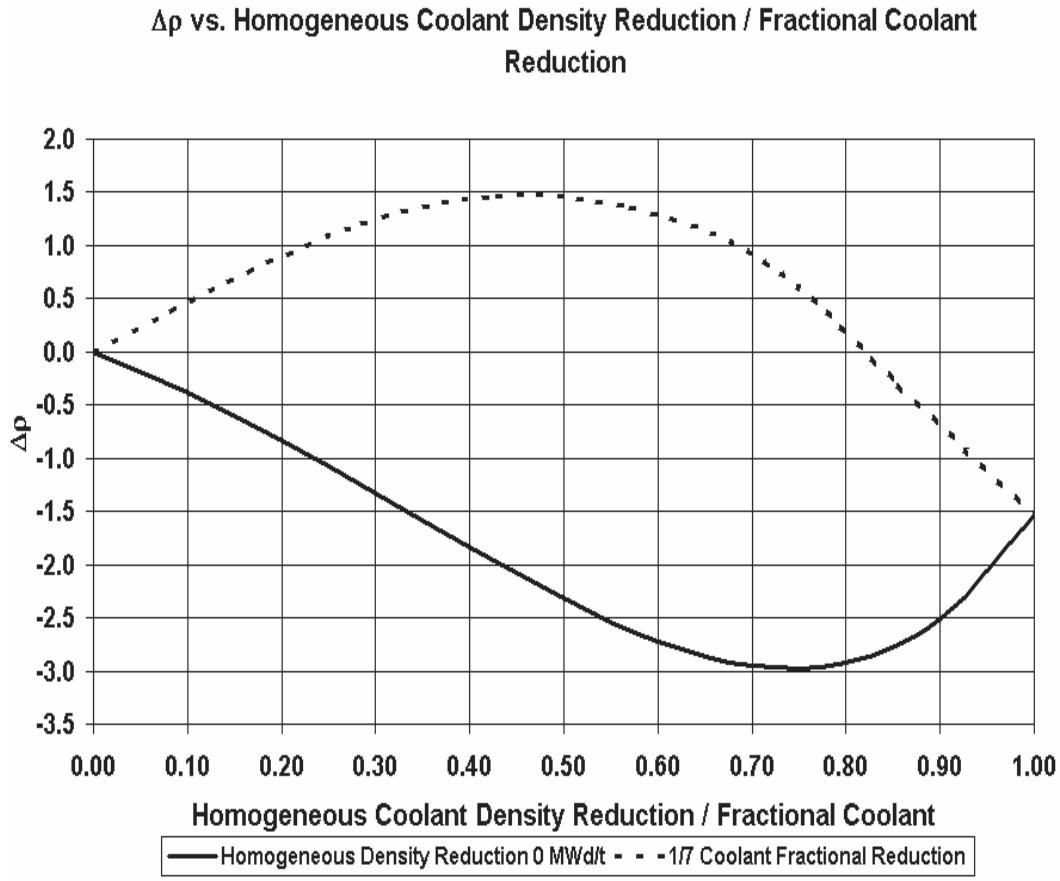

Figure 8-3 HELIOS CVR Results for Uniform or Homogeneous Coolant Density Reduction Versus Sector-Wide Heterogeneous Removal of Coolant within Channels (0 GWd/t)

As indicated in the plotted results, the CVR from homogeneous coolant voiding is always negative. As the coolant density is reduced, the CVR becomes more negative until about a 75 percent reduction in coolant density has taken place, after which the reactivity change increases but remains negative. Conversely, CVR in the sectored heterogeneous voiding case is initially positive and becomes negative only after about 80 percent of the coolant is removed. The results of this plot study suggest that there are significant coolant heterogeneity effects within a traditional single-channel lattice model that can lead to a positive reactivity change upon partial voiding of the lattice.

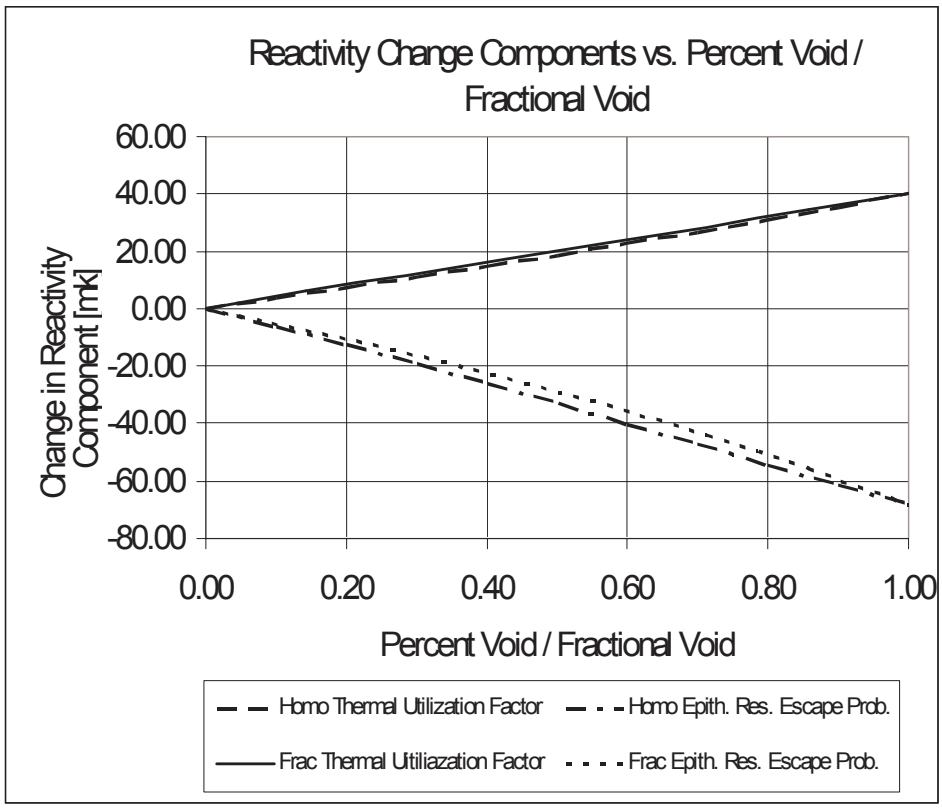

Figure 8-4 HELIOS-Computed Four-Factor Spectral Components of ACR-700 Void Reactivity for Uniform Coolant Density Reduction Versus Sector-Wide Removal of Coolant within Channels

The four-factor analysis was applied to both the homogeneous and heterogeneous partial voiding cases. As before, two of the most dominant components are the thermal utilization and the resonance escape probability. Figure 8-4 shows these two components plotted separately for the homogeneous and heterogeneous channel voiding cases. While the change in thermal utilization is similar for both homogeneous and heterogeneous cases, there is an important difference in the change in the resonance escape probability for the homogeneous and heterogeneous cases as the coolant is removed. The change in the resonance escape probability in the heterogeneous case is less negative during intermediate voided states and, as shown in Figure 8-3, leads to an overall positive CVR during partial voiding. The physical reasons for this behavior can be attributed to a spatial self-shielding effect in the heterogeneous voided lattice that will be more clearly demonstrated using the actual checkerboard voiding configuration in the 2x2 channel cases discussed next.

8.3.2.3 Analysis of CVR for Checkerboard Core Voiding Patterns

The next sequence of calculations involved extending the mirror-reflected single-channel ACR lattice model to a 2x2 array in order to investigate the impact of the “checkerboard” voiding of alternate fuel channels during large LOCAs. Figure 8-5 shows the construction of the 2x2 HELIOS model.

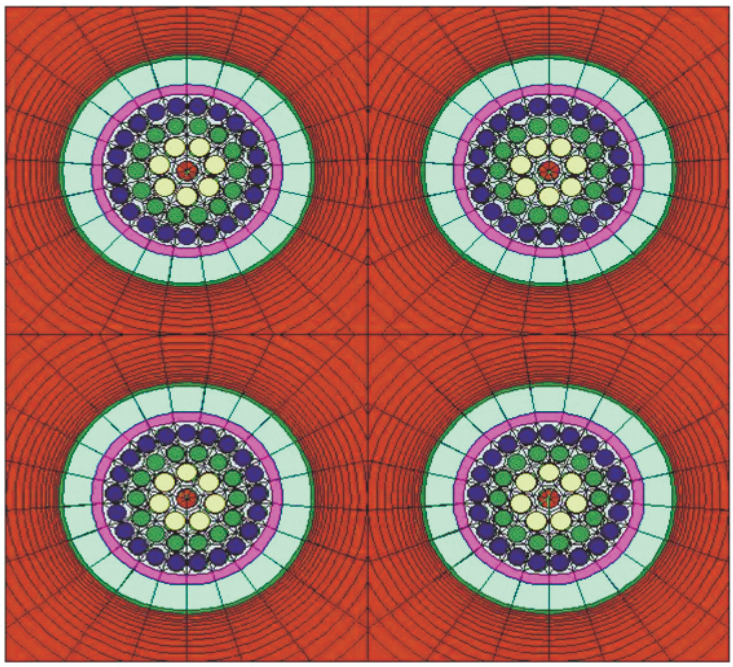

Figure 8-5 ACR-700 2x2 Core Lattice Modeled by HELIOS with Periodic Boundary Conditions

1.6 GWd/t	24.4 GWd/t
24.4 GWd/t	1.6 GWd/t

Figure 8-6 2x2 Lattice Array with Mixed Fuel Burnup of 1.6 and 24.4 GWd/t

Several different burnup distributions were analyzed for the 2x2 configurations. In the first set of cases, the burnup of all channels is the same (homogeneous or uniform) at 12.3 GWd/t, a typical burnup near midcore in the ACR reference design. The second set of 2x2 cases uses two different fuel burnups to represent an ACR burnup map near either end of the reactor. Figure 8-6 shows the burnup pattern schematically. For all cases, the analysis considered both complete voiding and checkerboard half-voiding configurations.

As indicated in the first row of Table 8-3, the CVR for uniform burnup in the fully voided case is again negative, whereas it is positive for the checkerboard voided case. The analysis for mixed burnup cases used two different checkerboard voiding configurations for mixed burnup cases. In the CB-1 case, the low-burnup fuel (1.6 GWd/t) was voided, and the high-burnup fuel (24.4 GWd/t) remained cooled. In the CB-2 case, the high-burnup fuel was voided, and the low-burnup fuel remained cooled.

The first checkerboard result worth noting in Table 8-3 is that HELIOS predicts a fully voided CVR of -3.4 mk for the heterogeneous burnup case, which is similar to the -3.4 mk for the uniform burnup case. This is consistent since the average burnup of the heterogeneous case is similar to the uniform burnup case. The next result worth noting is that the CVRs predicted for the two checkerboard voided cases are both positive, but the CVR for the CB-2 case in which the low burnup fuel remains cooled is considerably more positive.

Table 8-3 HELIOS Results for Checkerboard Voided Core Lattices with Uniform and Mixed 2x2 Burnup Arrays

2x2 Burnup Case	Lattice Neutron Multiplication Factor, k_{eff}				Coolant Void Reactivity (mk)		
	Cooled	CB-1 Voiding	CB-2 Voiding	Full Voiding	CB-1 Voiding	CB-2 Voiding	Full Voiding
Uniform Burnup 12.7 GWd/t	1.03073	1.03571		1.02717	+4.7		-3.4
Mixed Burnup 1.6/24.4 GWd/t	1.03505	1.03719	1.04205	1.03140	+2.0	+6.5	-3.4

8.3.2.4 Continuing Independent Analysis Efforts

The staff's continuing nuclear analysis efforts will use established lattice physics and Monte Carlo codes to further evaluate the reactivity effects of distributed and partial voiding within channels and of checkerboard and other realistic core-wide voiding patterns. Efforts are now underway to build upon the PIRT recommendations by using detailed models and existing Monte Carlo codes (e.g., MCNP/MonteBurns described in "MCNP—A General Monte Carlo N-Particle Transport Code, Version 5," 2003, and "User's Manual, Version 2.0 for MonteBurns," 1999;) MONK described in "Current Status and Future Direction of the MONK Software Package," N. Smith, M. Armishaw, and A. Cooper, 2003; and SCALE/KENO described in draft NUREG/CR 0200, Revision 7, 2004) with supplemental cross-section data as needed to perform parametric modeling studies of radial and axial (i.e., end-effect) fuel pin depletion zoning, multicell models for checkerboard CVR calculations, reactivity device calculations and/or color-set fuel depletion, and, ultimately, static full-core Monte Carlo calculations. Such calculations should evaluate CVR, static spatial power distributions, and other safety-significant nuclear effects. Results from such analyses will support recommendations on the modeling approximations and methods to be used in the staff's audit code models for ACR-700 fuel lattice physics, reactivity device lattice physics, operational core neutronics, and core spatial kinetics. The modeling studies will provide detailed models of the ACR-700 core and core lattices to help qualify the more approximate models and methods to be implemented in the

NRC's audit code suite for nuclear analysis of reactor operating states, transients, and accidents.

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

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

As needed, the staff's nuclear analysis efforts will also support the NRC's review activities for severe accidents by providing an independent analysis of potential recriticality and the potential for positive void reactivity effects in credible severe accident configurations.

8.3.3 Summary of Staff Insights and Plans for Evaluating the Coolant Void Reactivity Validation Program

As noted previously, AECL does not foresee conducting in-reactor tests to directly measure CVR effects in the first or subsequent ACR-700 cores. Without direct measurements, the evaluation of CVR effects must rely solely on calculations that simulate or predict the neutronic behavior of the ACR core. The validation efforts in this area will quantify the bias and uncertainty in the calculated CVR values by benchmarking the calculation methods against a set of applicable measurements from critical experiment facilities.

As shown in the previous subsections, the net CVR in the ACR-700 results from a combination of large positive and large negative reactivity contributions that vary with fuel burnup isotopics and core operating conditions. Specifically, the overall reactivity response to coolant voiding in the ACR-700 includes spatially dependent effects from voiding-induced changes in fast fission, neutron thermalization, resonance absorption, thermal upscatter by coolant, thermal absorption by coolant, thermal absorption by various fuel actinides and fission products, thermal absorption by Dy burnable poison, thermal fission of major fissile actinides, and leakage of epithermal and

thermal neutrons. AECL predicts the resulting value of the whole-core CVR to be negative (-7 mk), with potentially significant sensitivities to design and operating parameters of the transitional and equilibrium cores, core modeling methods, and nuclear data uncertainties.

Given the nominal values of the predicted negative CVR and the lack of planned CVR measurements in the ACR-700 core, it is expected that modeling issues, data uncertainties, and the benchmark evaluation and treatment of prediction biases and uncertainties (i.e., code validation) will figure prominently in the staff's eventual review conclusions in this area. For example, an eventual review conclusion that ACR-700 coolant void feedback is indeed negative over a spectrum of core conditions should be supported by independent analysis confirming, with reasonable assurance, that the predicted void reactivity effects are still negative after adjusting for potential prediction biases and uncertainties as evaluated through application-specific code validation against relevant experimental benchmarks and supplemental modeling studies.

The staff's continuing review activities for the ACR-700 will include the assessment and independent analysis of AECL's existing and planned benchmark databases for nuclear analysis validation. Depending on the prototypicality and ranges of parameters covered by AECL's benchmark experiments and tests in ZED-2 and elsewhere, the NRC may elect to develop or acquire additional data from independent or shared facilities. The staff intends to pursue with AECL the potential sources of applicable critical benchmark data from past measurements in the Deuterium Critical Assembly in Japan and from other heavy-water critical experiment facilities in the United States, the United Kingdom, Italy, and elsewhere.

The Task 1 Letter Report, "The Use of Sensitivity Analysis with Experimental Benchmarks to Quantify Bias and Uncertainty in the Calculation of ACR-700 Coolant Void Reactivity," issued in May 2004, reports in detail on the staff's evolving plans to adapt and use advanced S/U methods to evaluate how well the set of existing and planned experimental benchmarks represents the competing neutronic phenomena that determine CVR in the ACR and to quantify CVR prediction bias and uncertainty from the set of benchmark results. The S/U methods in SCALE can be adapted to address the CVR uncertainties caused by cross-section data uncertainties in the calculation of instantaneous values of k(cooled) and k(voided). The assessment of CVR bias and uncertainty will also call for the benchmark evaluation of predicted fuel isotopics against applicable PIE data from the radiochemical assay of irradiated fuel. These will be used to evaluate the CVR uncertainties resulting from the computed burnup-dependent nuclide inventories. The relative concentrations of major actinides affecting CVR are expected to have correlations that amplify their impact of concentration uncertainty on CVR.

8.4 Potential Policy Issues

The staff has identified no issues concerning ACR-700 CVR that would clearly necessitate involving NRC upper management or the Commission in the development and approval of policy guidance. However, this could change if AECL cannot show that the evolving nuclear design of the ACR-700 clearly eliminates any potential for substantially positive void reactivity in LOCAs.

8.5 Conclusions

The staff's preapplication review and analysis activities for AECL FT9 revealed three issues regarding ACR-700 CVR. First, the reviewed preliminary design of the ACR-700 does not meet AECL's stated design goal of negative void reactivity in large-break LOCAs. Second, nuclear design changes would be needed to achieve a void reactivity that is nominally negative and potentially no more than slightly positive after consideration of uncertainties. Third, ACR void reactivity uncertainties are potentially large in relation to AECL's nominal targeted values, and AECL's planned ZED-2 experimental program for the ACR-700 is key to assessing and minimizing such uncertainties.

Pursuant to insights emerging from the NRC's ongoing PIRT process for the ACR-700, the staff performed initial best-estimate neutronic calculations that predict the CVR to be substantially positive in large-break LOCAs. Such a positive reactivity response would cause reactor power to inherently rise during the seconds before shutdown rods can be inserted. This is in notable contrast to PWRs and BWRs, where large-break LOCAs create negative void reactivity that quickly terminates the fission chain reaction before scram.

The positive void reactivity response in the preliminary ACR-700 design arises during the initial checkerboard voiding of alternate coolant channels in the horizontal pressure-tube core. Such voiding patterns appear in all CANDU cores during the first second or two of a large LOCA, persist for many seconds, and are expected to occur over a wide range of postulated break sizes and locations. In this checkerboard voiding pattern, as the name suggests, the liquid coolant in every other core channel has steamed almost completely (e.g., void fraction greater than 98 percent) while the coolant in the remaining channels typically stays at nearly full density (e.g., void fraction less than 5 percent).

The staff's analysis of checkerboard void reactivity in the ACR-700 preliminary design highlights a nonlinear neutronic behavior with voiding that presents particular challenges to the nuclear codes used in reactor transient analysis and to the experimental programs needed for code validation (i.e., evaluating calculation bias and uncertainty). The validation issues concerning void reactivity are heightened because direct in-reactor confirmatory measurements of CVR are inherently difficult during prestartup, startup, shutdown, or power operations, have never been performed in existing CANDU reactors, and are not presently planned by AECL for the ACR-700.

The staff's activities to independently predict and evaluate CVR in the ACR-700 design have included not only independent best-estimate neutronic calculations of the net CVR values, but also independent analysis of the neutronic phenomena that contribute to the net CVR. This analysis shows that a combination of large positive and negative contributors, resulting from the voiding-induced neutron spectral shifts and spatial flux redistributions, govern AECL's targeted small values of CVR. These contributors to CVR appear to be sensitive to fuel design parameters (enrichment level, Dy content), to burnup and uncertainties in the burnup-dependent nuclide concentrations in a given fuel design, and to uncertainties in the cross sections of key nuclides, which include Dysprosium isotopes, as well as U-235, U-238, and Pu-239.

In view of the uncertainty issues, the staff has also initiated the efforts needed to review experimental techniques and to adapt and use recently developed S/U methods to (1) evaluate the adequacy and applicability of AECL's planned set of CVR validation benchmark experiments, (2) identify and pursue additional sources of integral and differential data for use in validating CVR predictions, and (3) systematically apply emerging experimental benchmark results to the quantification of CVR bias and uncertainty for the ACR-700. The staff also intends to engage AECL in the consideration of novel ideas for measuring CVR effects in prestartup, startup, or operational tests of the first ACR-700 core.

In making the transition to the anticipated design certification review phase, the staff will continue and expand its review efforts with AECL on (1) analyzing void reactivity effects in AECL's nuclear design of the ACR-700 submitted for design certification, (2) assessing the adequacy of AECL's planned set of void reactivity experimental benchmarks from ZED-2 and other facilities, (3) identifying and addressing needs for additional benchmark experiments and other measured data, (4) using the emerging set of benchmark results to quantify bias and uncertainty in nuclear code calculations of void reactivity and other safety-related nuclear characteristics of the ACR-700 design, (5) developing specific NRC review guidance, and (6) further developing and using the staff's independent nuclear analysis tools and capabilities as needed for simulating ACR-700 operating states, transients, and accidents.

The staff is still in the process of understanding the relative sensitivity of ACR safety performance (e.g., peak fuel temperatures in accidents) to postulated negative or positive values of CVR. The necessary understanding will accrue from CVR parametric sensitivity analysis to be provided by AECL and verified where necessary by NRC audit calculations. Resulting insights will allow the staff to consider the overall safety impacts of making CVR more or less negative or positive and will also be important for assessing the effects of CVR uncertainties in a given ACR reference design.

The NRC staff has made significant progress in understanding the competing neutronic phenomena in ACR that contribute to the net reactivity changes caused by coolant voiding. The nonlinear CVR phenomenology in the ACR core is characterized by the summation of large positive and large negative contributors that are sensitive to fuel design choices (uranium enrichment, Dy content), fuel burnup levels, core operating conditions, geometry, and the transient distributions of core voiding during accidents.

The NRC staff has begun to evaluate the adequacy of AECL's planned set of ZED-2 critical experiments for use in quantifying bias and uncertainty in the computed predictions of ACR void reactivity. The staff's evaluation efforts will address measurement techniques and uncertainties in the ZED-2 experiments and will use advanced S/U methods to evaluate how well the set of experiment configurations represents the phenomena that govern CVR effects in ACR.

9. ACR PROBABILISTIC RISK ASSESSMENT METHODOLOGY (FT11)

Focus Topic 11 (FT11) involved an examination of the ACR PRA methodology. The outcome desired by AECL for the ACR PRA methodology examination is that, "the NRC staff accepts AECL PRA methodology as sufficient for the purpose of assessing the ACR for licensing in the U.S." (see Langman letter dated July 30, 2003; Langman letter, 108US-01321-021-001, "Further to the ACR Pre-Application Plan—Detailed Deliverables and Schedule for Focus Topics," dated December 18, 2002).

In order to complete an adequate risk assessment of the ACR design, it is necessary to know about the progression and characteristics of potential severe accidents. During the preapplication review, AECL established FT4, which concerns the definition of severe accidents in the ACR and the adequacy of supporting R&D activities. Therefore, the NRC staff limited the scope of its review concerning severe accidents during FT11 to confirming that AECL PRA methodology (1) was based upon the same definition of severe accidents used elsewhere by the applicant, (2) had the capability to reasonably represent the risks associated with severe accidents in the ACR, and (3) had the flexibility to incorporate the results of ongoing and future ACR severe accident research, when such results become available.

The following sections describe the staff's preapplication review for FT11 by discussing the review scope; identifying technical, regulatory, and potential policy issues; and providing staff assessments about the acceptability of the proposed ACR PRA methodology.

9.1 Review Scope

This section identifies the documents the applicant submitted relevant to the ACR PRA methodology, which the staff examined. It also identifies the applicable regulations, regulatory guidance, and standards the staff consulted during its review.

9.1.1 Documents That Were Reviewed

During its examination of the ACR PRA methodology, the staff reviewed the following documents which the applicant submitted:

- generic CANDU probabilistic safety assessment (PSA) methodology (see AECL Report No. 91-03660-AR-001, "Analysis Report: Generic CANDU Probabilistic Safety Assessment—Methodology," Revision 0,July 2002)

- generic CANDU PSA reference analysis (see AECL Report No. 91-03660-AR-002, "Analysis Report: Generic CANDU Probabilistic Safety Assessment—Reference Analysis," Revision 0, July 2002)

- the analysis basis for the ACR PSA (see AECL Report No. 108-03660-AB-001, "Analysis Basis: Probabilistic Safety Assessment—ACR," Revision 1, July 2003)

- the assessment document (see AECL Report No. 108-03660-ASD-008, "Design Assist Role of ACR PSA," Revision 0, February 2004)

- the analysis report on PSA Level 1 (see AECL Report No. 10810-03660-AR-001, "Preliminary Design Assist PSA Level 1—Selected Full Power Event Trees," Revision 1, January 2004)

The applicant supplemented these documents in (1) responses to staff RAIs, (2) a teleconference concerning PRA quality conducted January 22, 2004, and (3) a Category 1 public meeting held February 5–6, 2004, in Rockville, Maryland (see Letter from V. Langman (AECL) to the NRC, "Response to NRC's Requests for Additional Information (RAIs) #3 on PRA Quality," 108US-01321-021-001, February 12, 2004; Letter from Langman (AECL) to the NRC, "Response to NRC's Requests for Additional Information (RAIs) #4 on PRA Analysis Basis," 108US-01321-021-001, April 15, 2004; Letter from Langman (AECL) to the NRC, "Responses to NRC's RAIs 68, 71, and 83 on PRA Analysis Basis, "108US-01321-021-001, April 30, 2004; Letter from Langman (AECL) to the NRC, "Responses to NRC's RAI 37 on PRA Analysis Basis," 108US-01321-021-001, June 17, 2004).

It should be noted that the applicant did not submit the ACR PRA for examination during the preapplication review. The staff was able to identify some general weaknesses in AECL PRA methodology by comparing the generic CANDU PRA reference analysis (see AECL Report No. 91-03660-AR-002, "Analysis Report: Generic CANDU Probabilistic Safety Assessment—Reference Analysis," Revision 0, July 2002) to the generic CANDU PSA methodology (see AECL Report No. 91-03660-AR-001, "Analysis Report: Generic CANDU Probabilistic Safety Assessment—Methodology," Revision 0, July 2002). The staff observes that AECL still needs to properly implement an acceptable methodology to prevent producing a PRA with a less than acceptable scope, level of detail, or technical adequacy.

The staff's review of the ACR PRA methodology has identified some preliminary qualitative risk insights concerning the ACR design. However, the staff has not reached any conclusions on the acceptability of the ACR design from a risk perspective as a result of the FT11 preapplication review effort because the applicant did not submit the actual ACR PRA for review.

9.1.2 Applicable Regulations, Regulatory Guidance, and Standards

In accordance with 10 CFR 52.47(a)(1)(v), an application for design certification must contain a design-specific PRA. Section 5.3.3 of NUREG-1226, "Development and Utilization of the NRC Policy Statement on the Regulation of Advanced Nuclear Power Plants," dated June 1988 indicates that PRAs performed for advanced reactors should accomplish the following:

- address internal and external events
- address various plant operating states (full power, low power, refueling, etc.)
- confirm the bases for component and system selections
- confirm the adequacy of overall plant design
- identify and correct any areas of high risk
- confirm the adequacy of plant responses to severe accidents and mitigation measures

The first two items define the overall expected scope for an advanced reactor PRA. The remaining items describe how AECL should use PRA results in an application for design

certification. In order to obtain more detailed information to guide its examination, the staff consulted the following information sources:

- RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 1, November 2002, provides guidance on the use of PRA in risk-informed decisions on changes to the licensing basis of existing nuclear power plants. The staff believes this guidance is relevant to the ACR since it provides five key principles that are generally applicable to risk-informed decisionmaking, as well as high-level staff expectations concerning PRA quality.

- Chapter 19 of the SRP provides, in part, general guidance on how the staff can combine information from a PRA with other pertinent information in the process of making a regulatory decision.

- Section 19.1 of the SRP addresses PRA technical adequacy and "concerns any licensee request submitted for NRC review and approval for which PRA can play an effective role in the decisionmaking process."

- RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," issued for trial use in February 2004, describes one acceptable approach for determining that the quality of a PRA is sufficient to provide confidence in its results, such that the staff can use the PRA in regulatory decisionmaking for LWRs. The staff believes this guidance is relevant to the ACR because it is of a general nature, and the ACR shares certain characteristics with current-generation PWRs.

- ASME RA-Sa-2003, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications," Addenda, December 5, 2003, sets forth requirements for the PRA used to support risk-informed decisions for commercial LWRs. Specifically, this standard establishes requirements for a Level 1 PRA analysis of core damage frequency (CDF) and a limited Level 2 PRA analysis sufficient to evaluate large early release frequency (LERF) for internal events while at power. The staff has endorsed this standard with various clarifications and qualifications in RG 1.200.

- ANSI/ANS-58.21-2003, "American National Standard: External Events PRA Methodology," March 3, 2003, sets forth requirements for external event PRAs used to support risk-informed decisionmaking for commercial nuclear power plants. Specifically, this standard addresses certain natural external hazards (earthquakes, high winds, and external flooding) and human-made external events (airplane crashes, explosions at nearby industrial facilities, and impacts from nearby transportation activities). It is limited to external events that might occur while a nuclear power plant is at nominal full power, and addresses the Level 1 PRA analysis of CDF and the Level 2 PRA analysis of LERF. The staff is currently reviewing this standard, but has not yet endorsed it.

9.2 Regulatory Issues

The staff has not identified any regulatory issues pertaining to the ACR PRA, such as rulemaking or exemptions to existing rules, that need resolution in order to complete the preapplication review or the design certification review of the ACR.

9.3 Potential Policy Issues

During work on FT11, the staff identified a potential policy issue with applying the current risk objectives for advanced reactors to the ACR design. The following sections provide relevant background and define the potential policy issue.

9.3.1 Background

This section summarizes the types of core damage accidents that may occur in an ACR, AECL position on regulatory treatment of core damage accidents, and guidance from the Commission concerning risk objectives for advanced reactors.

ACR core damage accidents. Core damage accidents in the ACR (and CANDU plants in general) may be grouped as follows:

- LCDAs, which only affect a fraction of the fuel. Two subgroups of LCDAs include the following:

  (a) accidents with a local power/cooling mismatch at full power, also termed "single channel accidents" or "single channel events," and including accidents such as severe flow blockage in a fuel channel and SFBs

  (b) accidents with a global power/cooling mismatch at decay power, also termed "complete core events" or "dual failures," and including accidents such as LOCAs followed by loss of the ECCS

- SCDAs, which affect all of the fuel in the reactor

LCDAs are a group of accidents unique to the ACR design (and CANDU plants, in general) because of their use of multiple, separated fuel channels, surrounded by a cool, low pressure heavy-water moderator contained within a calandria vessel. This design is substantially different from current generation LWRs, in which the fuel is organized into a core that is contained within a reactor vessel.

AECL position. AECL has stated that LCDAs represent a group of accidents that, in terms of their consequences, exist somewhere between DBAs and SCDAs (see AECL Report No. 108-03600-AB-003, "Analysis Basis: Safety Basis for ACR," July 22, 2003; and Letter from Langman (AECL) to the NRC, "Regulatory Treatment of Limited Core Damage Accidents for the ACR-700," 108US-01321-021-001, May 10, 2004). As a result, AECL argued that LCDAs should not be classified as DBAs because they have a lower probability of occurrence; likewise, LCDAs should not be classified as SCDAs because they have a lower magnitude of fission product release from the fuel when compared with potential severe accidents in U.S. LWRs.

With respect to risk objectives for the ACR, AECL has proposed that the SCDA frequency be less than 10^{-5} per reactor-year and the large release frequency (LRF) be less than 10^{-6} per reactor-year. The staff notes that the CNSC is also considering these risk objectives. For LCDAs, AECL has proposed the following actions:

- Using design-centered assumptions and analyses, demonstrate that the accident sequence for each type of LCDA will not result in fuel melting or propagation beyond the initially affected channels, thereby preventing significant releases. (With respect to AECL proposal for addressing LCDAs, the term "design-centered assumptions and analyses" refers to a best-estimate approach, in contrast to an intentionally conservative approach used in DBA analyses).

- Using design-centered assumptions and analyses, demonstrate that the doses from LCDAs will not exceed the 25 rem TEDE dose limit stipulated in 10 CFR 50.34.

- Demonstrate that the frequency of the event sequence for each type of LCDA is low (i.e., in the range of 10^{-4} to 10^{-6} per reactor-year).

- Apply defense-in-depth to ensure a low frequency of occurrence of LCDAs, including conservative design provisions, qualification requirements, applicable construction codes, and QA. AECL will also apply defense-in-depth to help mitigate LCDAs, including the moderator cooling system, the calandria design, and the containment building.

The staff observes that AECL position does not propose a risk objective related to the total limited core damage frequency (LCDF); specifically, no restriction or limitation on the maximum LCDF has been suggested.

The Commission's risk objectives for advanced reactors. The Commission expressed its overall expectation for the protection of public health and safety from accidents resulting from nuclear power operation in its 1986 Safety Goal Policy Statement (see "Safety Goals for the Operation of Nuclear Power Plants: Policy Statement," Federal Register, August 4, 1986). In addition, the Commission's 1986 Advanced Nuclear Power Plant Policy Statement (see "Regulation of Advanced Nuclear Power Plants: Statement of Policy," Federal Register, July 8, 1986) expressed its expectation that advanced reactor designs will (1) as a minimum, offer the same degree of protection to the public and the environment that is required for current-generation LWRs, (2) provide enhanced safety margins, and (3) comply with the NRC Safety Goal Policy. Over the past 18 years, the staff developed various risk objectives to use as surrogates for the quantitative health objectives in the Safety Goal Policy. SECY-90-016, as modified by the Commission in its associated SRM dated June 26, 1990, states the following risk objectives for evolutionary LWRs:

- CDF less than 10^{-4} per reactor-year
- LRF less than 10^{-6} per reactor-year
- conditional containment failure probability (CCFP) less than 0.1

The staff has utilized the above risk objectives during the design certifications of the U.S. ABWR, System 80+, the AP600, and the AP1000.

9.3.2 Potential Policy Issue

The ACR PRA will estimate the severe core damage frequency (SCDF), LCDF, LRF, and CCFP. In addition, AECL will perform uncertainty analyses for the SCDF, LCDF, and LRF. However, it is not clear how to apply the Commission's risk objectives for advanced reactors, as stated in the SRM to SECY-90-016, to the ACR design. In order to resolve this potential policy issue, the staff will seek Commission direction and guidance in resolving three questions:

- Does the CDF risk objective refer to the total CDF of the ACR (i.e., the sum of the LCDF and the SCDF), or only to the SCDF?

- For the ACR, should there be separate risk objectives for the LCDF and the SCDF?

- For the ACR, should the CCFP apply only to SCDAs, or to all types of core damage accidents?

9.4 Technical Issues

This section provides an overview of the ACR PRA methodology, identifies methodological issues requiring resolution before the application for design certification is submitted, and identifies aspects of the methodology that warrant special attention by the staff during its review of the PRA submitted in support of the design certification application.

9.4.1 Overview of the ACR PRA Methodology

AECL PRA methodology for the ACR, which AECL submitted for staff review during the preapplication process, generally provides high-level guidance rather than a step-by-step process. The ACR PRA consists of a Level 1 analysis (assessment of accidents leading to core damage) and a Level 2 analysis (assessment of accidents leading to a release of radionuclides from the containment). AECL will not perform a Level 3 analysis (assessment of offsite health consequences). The following sections briefly describe the methodology AECL will employ.

Plant operating states. The PRA will examine the scope of plant operating states, including nominal full power and the following three shutdown states:

(1) HTS cold, depressurized, and full (including channel freeze plugging)
(2) HTS cold, depressurized, and drained to the reactor inlet and outlet headers
(3) HTS warmup and cooldown

The PRA presumes that the refueling machine is always "on reactor." The staff understands that this modeling assumption is conservative because on-power refueling operations are only expected to occur about 50 percent of the time during power operation of the ACR. Therefore, the staff accepts the assumption that the refueling is always "on reactor." In addition, the ACR PRA methodology includes the assessment of accidents related to refueling that occur when the refueling machine is "off reactor."

Initiating events. The PRA will address internal initiating events, internal floods (including the dynamic effects of high-energy line breaks), internal fires, and seismic events. The PRA will identify internal initiating events through an engineering analysis, comparison to previous CANDU PRAs, and a review of current-generation CANDU operating experience. AECL will not assess the risks from other types of external events (e.g., aircraft crashes) as part of the application for design certification; rather, AECL intends to address these risks as part of the combined license application process.

Success criteria determination. AECL will determine accident sequence success criteria using T-H calculations conducted with a CANDU-specific version of the Modular Accident Analysis Program (MAAP), termed "MAAP4-CANDU," as well as engineering judgment.

Accident sequence modeling scope. The PRA (both the Level 1 and Level 2 analyses) will address both LCDAs and SCDAs.

Plant damage state definitions. AECL has defined the following 11 plant damage states (PDSs), which it will use to group accident sequences with respect to the accident progression and containment performance:

(1) PDS0—Failure to shutdown

(2) PDS1—Late loss of core structural integrity with high PHTS pressure

(3) PDS2—Late loss of core structural integrity with low PHTS pressure

(4) PDS3—Loss of core cooling with moderator required early as a sustained heat sink (e.g., caused by large LOCA plus loss of ECI)

(5) PDS4—Loss of core cooling with moderator required late as a sustained heat sink (e.g., caused by LOCA plus loss of LTC in ECC mode)

(6) PDS5—Loss of cooling/inadequate cooling in one or more core passes following a LOCA with successful initiation of ECI.

(7) PDS6—Power cooling mismatch with late ECC injection because of multiple channel failure

(8) PDS7—Power cooling mismatch in a single channel with containment overpressure

(9) PDS8—Power cooling mismatch in a single channel with no containment overpressure

(10) PDS9—Tritium release without fuel damage (e.g., deuterium deflagration in the cover gas, fast release of the moderator into the containment, and slow release of the moderator into the containment)

(11) PDS10—Fueling machine failures (when the fueling machine is "off reactor")

Accident sequence modeling approach. AECL will delineate accident sequences using the small event tree-large fault tree approach. It will link support system fault trees to frontline

system fault trees (support state methodology will not be used). In general, the event tree headings directly correspond to the fault tree top events; however, AECL may develop bridging fault tree logic to relate the system-level fault trees to certain event tree headings. Furthermore, AECL may modularize fault trees on a functional basis to reduce the PRA solution time and the time spent reviewing the accident sequence quantification results.

Reliability data. As the ACR has no operating experience, AECL will base its reliability data on generic sources, with an emphasis on utilizing generic data sources that are based on the operating experience of current-generation CANDUs. The staff understands that many ACR systems and components are similar to the systems and components in current-generation CANDUs; therefore, it is reasonable to conclude that they will have similar reliability characteristics. As a result, the staff concludes that use of generic data sources based on the operating experience of current generation CANDUs is an acceptable approach.

Common-cause failures (CCFs). In the initial stages of the PRA, AECL will employ the beta-factor method to model CCF events. Initially, AECL will use a screening value of 0.1 for all beta factors. After initial PRA quantification, AECL will refine the beta factors for significant CCF contributors using the Unified Partials Method (UPM) (see AECL Technology Report No. SROA-R13, "UPM 3.1: A Pragmatic Approach to Dependent Failures Assessment for Standard Systems," April 1996), which provides a structured and auditable approach to beta-factor estimation based on consideration of the following attributes:

- redundancy and diversity
- physical separation
- understanding of the design, including experience with the actual design or similar designs; novelty of the design; complexity of the design; and the potential for misfit (a gap in understanding that can exist where equipment has not been specifically designed to fit its application, but has been taken "off the shelf")
- extent of reliability analysis conducted on the design
- man-machine interface
- safety culture of the plant operating staff
- control of the environment in which the design is installed
- extent of testing conducted for environmental influences that affect CCF, such as shock, vibration, temperature, and humidity

For situations involving three or more levels of redundancy, AECL may replace the beta-factor approach with the alpha-factor approach, which allows the modeling of partial common-cause group failures (e.g., two out of three components fail as a result of a common cause).

Human reliability analysis. The basic framework for conducting the human reliability analysis (HRA) is the Accident Sequence Evaluation Program (ASEP) HRA methodology (see NUREG/CR-4772, "Accident Sequence Evaluation Program Human Reliability Analysis

Procedure," February 1987). AECL will include both preaccident and postaccident human errors. The ASEP HRA method includes methods for considering possible dependencies among human errors. AECL will consider both diagnosis (cognitive) and execution (implementation) errors in its estimation of the probabilities of postaccident human errors.

Quantification. AECL will provide estimates of the frequency of each PDS, in addition to the LCDF, SCDF, LRF, and CCFP. AECL will lower the truncation frequency used to generate the PRA results until convergence of the frequency estimates is demonstrated. The final accident sequence quantification will credit recovery actions, which AECL will append to individual cut sets using a set of recover rules.

Parametric uncertainty analysis. The PRA methodology uses a Monte Carlo approach to propagate parametric uncertainties (e.g., uncertainties in initiating event frequencies, failure rates, CCF parameters, etc.). The applicant will develop uncertainty distributions for the LCDF, SCDF, and LRF, in keeping with the advice of the ACRS (see letter from Powers (ACRS) to the NRC, "Pre-Application Phase of the AP1000 Standard Plant Design—Phase I," September 14, 2000).

Sensitivity studies to explore modeling uncertainties. The PRA methodology generally makes conservative assumptions whenever uncertainty exists about phenomenological issues and human responses to abnormal conditions. AECL has stated that it will conduct selective sensitivity analyses to assess modeling uncertainties. AECL will also calculate and review importance measures (e.g., risk achievement worth and Fussell-Vesely) to obtain insights for optimizing the ACR design.

Seismic risk methodology. AECL will conduct a seismic margins analysis (SMA) of the ACR, not a complete seismic PRA. In SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light Water Reactor (ALWR) Designs," April 2, 1993, the staff accepts this approach for the examination of evolutionary and advanced LWR seismic risk, as approved by the Commission in the associated SRM. In addition, the staff has previously accepted the use of SMA during the design certifications of the ABWR, System 80+, the AP600, and the AP1000.

Fire risk methodology. The methodology for assessing risks caused by internal fires, which is based upon the International Atomic Energy Agency (IAEA) Safety Series No. 50-P-4, "Procedures for Conducting Probabilistic Safety Assessments of Nuclear Power Plants," 1992, uses a successive screening approach to focus attention on the most significant fire scenarios. In SECY-93-087, the staff recommended using simplified methods to evaluate fire risks, as approved by the Commission. In addition, the staff has previously accepted the use of simplified fire risk assessment methods during the design certifications of the ABWR, System 80+, the AP600, and the AP1000.

The PRA methodology consists of the following four major steps:

(1) Divide the plan into fire areas (separated from other areas with a fire barrier rating of 3 hours or more) and fire zones (based on a fire hazards analysis).

(2) Perform qualitative screening analysis, which consists of the following steps:

(a) Identify the plant equipment to be considered in the fire PRA, including equipment that can affect a PRA-credited safety function if actuated spuriously by the hot-short mode of failure and cables.

(b) Identify potential fire propagation pathways, considering the potential for fires to propagate across adjacent fire areas and fire zones, automatic fire suppression capabilities, and the efficacy of fire barriers.

(c) Screen out fire areas and fire zones if a fire within the area or zone does not cause initiating events other than a plant trip, does not include any PRA-credited equipment, or does not propagate to other fire areas or fire zones that contain PRA-credited equipment.

(3) Perform quantitative screening analysis, which consists of the following:

(d) Calculate fire initiating event frequencies using a CANDU-specific fire database developed by AECL.

(e) Identify fire-induced initiating events.

(f) Quantify the fire-induced SCDF.

(g) Screen out fire areas and fire zones if their SCDF is less than 10^{-7} per year.

(4) Perform a detailed analysis by refining the fire-initiating event frequency, considering the possibility that the fire is self-extinguishing, modeling fire growth and propagation using the COMPBRN IIIe computer code, and crediting automatic and manual fire suppression as appropriate.

Internal flooding analysis. The methodology for assessing risks from internal floods uses a successive screening approach to focus attention on the most significant flood scenarios. The methodology consists of the following six major steps:

(1) Determine flood areas based on design flood calculations, general arrangement drawings, consideration of flood barriers, and similar information.

(2) Identify flood area characteristics in terms of potential flooding sources and the location of safety-related and PRA-credited systems and equipment.

(3) Perform qualitative screening analysis to screen out flood areas from further analysis based mainly on the location of safety-related systems and equipment.

(4) Perform quantitative screening analysis to screen out flood areas from further analysis based on a conservative estimation of the flood-induced SCDF.

(5) Refine preliminary quantitative results by eliminating conservatisms.

(6) Perform detailed analysis of potential flood scenarios, crediting local operators' actions, as appropriate.

Level 2 PRA. The ACR PRA methodology for conducting the Level 1 PRA consists of the following six steps:

(1) Define the term “large release” and provide a technical rationale.

(2) Group accidents into categories, called core damage states (CDSs), chosen by expert judgment that broadly represent reactor and containment conditions of similar radioactivity release from the fuel, fuel coolability conditions, and containment integrity challenges. The tentative CDSs are as follows:

(a) CDS1 covers DBAs and most LCDAs. All of these accidents involve limited releases from the fuel, all are directly cooled by the HTS or ECCS, and there are no strong containment integrity challenges. Maintaining the quasi-steady CDS1 state requires few, if any, long-term manual interventions.

(b) CDS2 covers the remaining LCDAs. These accidents do not have direct fuel cooling, so the fuel release is considerable and produces combustible gases that pose a containment integrity challenge. These accidents could potentially result in a large release if the containment function is unavailable (e.g., if a containment bypass occurs). Maintaining the quasi-steady CDS2 state requires few, if any, long-term manual interventions.

(c) CDS3 covers SCDAs that are arrested in process system vessels. These accidents involve significant core geometry changes (corium debris is formed), significant releases from the fuel, and moderate containment integrity challenges (hydrogen production and steam surges caused by debris relocation). These accidents can potentially cause a large release if the containment function is unavailable and could challenge containment integrity in the longer term (depending on the timing of core disassembly). Maintaining the quasi-steady CDS3 state may require some long-term manual interventions.

(d) CDS4 covers the remaining SCDAs, which are not arrested in process system vessels. These accidents have all the characteristics of CDS3 accidents, and a potential for core-concrete interactions with the attendant additional releases and containment integrity challenges resulting from noncondensable gas generation. These accidents would likely cause a large release if the ex-vessel interactions commence and the containment is not sealed.

(3) Determine CDS frequencies. Note that combinations of PDSs and the long-term interventions determine which PDSs end up in which CDSs.

(4) Determine the characteristics and probabilities of open paths between the airborne radioactivity in the plant and the outside environment. Containment events (CEs) cause open paths, which AECL will categorize and document in a containment event tree (CET). AECL will define CEs using fault tree logic, which includes representations of various severe accident phenomena, systems-related issues, and boundary conditions.

(5) Perform deterministic analyses to estimate the radioactivity source terms outside the containment for all combinations of CDSs and CEs. For CDS1, AECL will use design

basis codes, such as GOTHIC, SMART, and SOPHAEROS. For CDS2–CDS4, AECL will use the MAAP4-CANDU code.

(6) Estimate the LRF.

AECL Level 2 PRA methodology is generally consistent with Level 2 PRAs that have been performed for U.S. nuclear power plants. However, AECL description of its approach is very high level and lacks sufficient detail to communicate what it will do. The staff sought additional information to clarify AECL Level 2 PRA methodology, which AECL has not provided. The staff observes that numerous detailed technical aspects govern the results of a Level 2 PRA, which require that AECL elaborate on the ACR PRA methodology documentation, so that risk analysts and reviewers can understand them.

9.4.2 Methodological Issues

The staff has identified various PRA methodological issues that AECL needs to address during the design certification review.

Methodological Issue 1 - Acceptability of MAAP4-CANDU. The ACR PRA methodology relies upon the MAAP4-CANDU T-H code. In order to ensure that the ACR PRA meets the Commission's expectations concerning PRA technical adequacy, as expressed in RG 1.200, the staff needs reasonable assurance that MAAP4-CANDU can adequately be used to support the PRA technical basis.

In 1992, Brookhaven National Laboratory (BNL) conducted a high-level review of MAAP Version 3.0B [Reference 30] "...to provide assurance that individual plant examinations (IPEs) will be analyzed with a methodology which [sic] adequately treats significant phenomena and reflects the uncertainty surrounding issues for which confirmatory research is planned or ongoing." In general, BNL concluded that MAAP Version 3.0B is adequate for predicting T-H behavior prior to clad damage. The BNL review cautioned that MAAP users should provide justification for using MAAP if certain T-H conditions are encountered:

- Applicable to both BWRs and PWRs: The break location gives rise to a quasi-steady state two-phase flow condition.

- Applicable only to BWRs: The reactor pressure vessel (RPV) water level and vessel flow conditions may expose the fuel to departure from nuclear boiling (DNB) conditions while MAAP continues to predict adequate core cooling.

- Applicable to both BWRs and PWRs: The reactor has not scrammed (fuel stored energy will not be released).

- Applicable to both BWRs and PWRs: Clad temperature is above 1200 K (1700 F).

BNL also indicated that MAAP Version 3.0B should not be used for determining success criteria after clad damage (e.g., to determine whether or not a core can be successfully reflooded after extensive fuel melting has occurred). After the onset of clad damage, BNL stated that MAAP should be used to provide an appreciation of the uncertainties surrounding containment

performance during a severe accident, and recommended performing a relatively small number of sensitivity studies as follows:

- In-vessel hydrogen generation
- In-vessel core relocation
- In-vessel natural circulation at high pressure
- Hydrogen combustion
- Core/concrete/water interaction
- Anticipated transient without scram (ATWS) scenarios

MAAP Version 3.0B was used to support the majority of the IPEs. The MAAP code has undergone many revisions (e.g., improved models for accident recovery) since the IPEs and the BNL high-level review were completed. The staff has considered the results of the MAAP Version 4.0 in support of PRAs for current generation LWRs. However, the improved MAAP 4.0 models have never been assessed by the staff, in part because the severe accident behavior of current generation LWRs has been scrutinized by using other T-H codes and extensive experimental research.

MAAP4-CANDU is based on MAAP Version 4.0, with modifications to address the horizontal CANDU-type fuel channels and CANDU-specific systems. The staff acknowledges that AECL has conducted and continues to conduct in-depth investigations of the ACR's severe accident behavior, and notes that the analytical tools and experimental results presented by AECL have been partially examined by the staff as part of the preapplication review. Further, the staff understands that AECL is in the process of developing a version of the code that will be specific to the ACR design (designated as “MAAP4-ACR700”), which is expected to be ready in late 2004.

The staff will conduct a high-level assessment of the MAAP4-CANDU code that will address four general areas:

- <u>Assessment of MAAP version used as a basis for the CANDU and ACR versions</u>. Any model improvements and/or additions should be identified, including existing benchmarking and/or performed validation/verification. This activity should be a limited, high-level effort designed to develop a list of additions/improvements in the successive MAAP versions, i.e., inter-relationships of MAAP Version 3.0B, MAAP Version 4.0, MAAP4-CANDU and MAAP4-ACR700 codes.

- <u>Review of the ACR-700 specific models</u>. The review should include the list of CANDU-specific and ACR-specific features and phenomena. Existing benchmarking and/or performed validation/verification should be identified.

- <u>Review of applicability to Level 1 PRA</u>. The focus should be on models affecting success criteria, such as flow solution and water level calculation, in particular if natural processes (passive safety systems) are involved. The review should include relevant conclusions from the ACR-700 PIRT activity. The depth of review of this item will be determined after AECL final decision regarding the use of analytical tools in support of the Level 1 PRA.

- Core degradation and SA phenomena. The review should identify CANDU-specific and ACR-specific degraded core behavior and any restrictions/limitations of the implemented models, such as horizontal core quenching, the potential for fuel relocation, and the potential for fuel-concrete interactions (FCI). The review should include relevant conclusions from the ACR-700 PIRT activity.

The staff notes that assessment of any of the above areas could potentially lead to a full-scope MAAP4-ACR700 review, which may be necessitated after completion of the high-level assessment. The staff will rely on AECL to provide certain information and calculations. Specifically, AECL should:

- Identify those MAAP models and code input assumptions unique to the ACR and/or CANDU design, provide supporting documentation of these models and supporting justification, and provide their assessment of which of these models have greatest impact on success criteria determinations, accident progression, and fission product release characteristics.

- Perform T-H benchmark calculations for frequency-dominant accident sequences using both MAAP and a pedigreed DBA-type thermal-hydraulic code to confirm the adequacy of MAAP predictions related to PRA success criteria.

- Perform severe accident progression calculations using MAAP for accident sequences selected by NRC for purposes of comparisons with NRC-sponsored calculations using MELCOR.

In addition, the staff notes that the following three modeling areas deserve special attention during the high-level assessment because they strongly influence the results of the Level 2 PRA:

- in-vessel hydrogen generation, including cladding surfaces that participate in oxidation process

- in-vessel core relocation, including any blockage model or assumptions, and models on in-core molten pool formation

- models associated with the behavior of molten core debris in the lower head of the RPV/calandria

Methodological issue 2—response to outstanding RAI questions. AECL response to RAI #4 (see Langman letter dated April 15, 2004) did not provide answers to all questions raised by the staff, as listed below.

Question	Topic	Response Date Stated by AECL
86	Definition of large release	5/30/2004
89	Screening criteria for LRF	5/30/2004
90	Determination of LRF	5/30/2004

The staff notes that resolution of these outstanding questions is essential to determine the acceptability of the Level 2 PRA methodology. The current description of the Level 2 PRA methodology is generally correct, although very high level. Numerous detailed technical aspects govern the results of a Level 2 PRA, which require that AECL elaborate on its ACR PRA methodology documentation, so that risk analysts and reviewers can understand them.

The staff will defer resolution of the questions listed above concerning the ACR PRA methodology until it receives the application for design certification.

Methodological issue 3—digital system software reliability. The ACR design incorporates state-of-the-art digital I&C systems. AECL has stated that the PRA will not address software reliability; rather, the software QA program will address this in terms of design, implementation, and exhaustive testing (see Langman letter dated April 15, 2004, response to Question 71).

The staff notes that PRAs submitted with previous design certification applications (e.g., ABWR, System 80+, the AP600, and the AP1000) included software-related errors in their logic models so that the staff could study the sensitivity of risk metrics (such as CDF and LERF) to the probability of software-related errors. Based on the results of these sensitivity studies, the staff believes that software-related errors may be significant risk contributors in the ACR. Should AECL decide not to incorporate software-related errors in the ACR PRA, then the staff expects AECL to provide adequate justification for the omission. The staff directs AECL's attention to the guidance in RG 1.174, Section 2.2.5.5, which concerns completeness uncertainty in PRAs.

The staff also notes that the omission of software-related errors in the PRA may delay or complicate portions of the design certification review that are not directly to the PRA. For example, Appendix 7.0-A to Chapter 7 of the SRP provides an overview of the process for reviewing the unique aspects of digital instrumentation and control (I&C) systems. The second paragraph of Item C.1 states:

> Probabilistic risk assessments (PRAs), such as those conducted under the Individual Plant Evaluation program (see Generic Letter 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities") or required as part of applications under 10 CFR 52, provide information that may prove helpful in determining the appropriate level of review.

The staff observes that it is not possible to use PRA results to establish the review scope for digital I&C system software unless the PRA explicitly models software-related errors.

AECL should propose an approach for considering the impact of software-related errors in the PRA. The staff observes that AECL may find it beneficial to review previous design certification applications (e.g., ABWR, System 80+, the AP600, and the AP1000), staff RAIs, and the staff's safety evaluations to gain insight into the staff's expectations.

9.4.3 Aspects That Warrant Special Staff Attention

There are aspects of the ACR PRA methodology that warrant special attention by the staff during its review of the PRA submitted by AECL in support of its design certification application, as discussed below.

PRA quality. It is essential that the ACR PRA be of sufficient scope, level of detail, and technical adequacy to support risk-informed decisionmaking during the ACR design certification process. AECL has performed a self-assessment (see Langman letter dated June 17, 2004) of the ACR PRA against the ASME PRA Standard (see ASME RA-Sa-2003). Since AECL did not provide the ACR PRA during the preapplication review, it was not possible for the staff to accept or reject AECL self-assessment. As part of the design certification process, the staff will review the ACR PRA against RG 1.200, considering the results of AECL self-assessment and any peer reviews. AECL should adequately document the ACR PRA so that the staff can reach a reasonable assessment of the PRA's quality.

Assumptions about design details. AECL has stated that certain details of the ACR design will not be finalized when it conducts the ACR PRA. Therefore, AECL intends to make reasonable assumptions about such design details in order to proceed with the PRA effort. Those assumptions include the following:

- In the SMA, AECL will assume that ACR-specific structures and components (e.g., the calandria) have a high confidence of low probability of failure value of 0.5 g (see Langman letter dated April 15, 2004, response to Question 78).

- In the fire PRA, AECL will utilize the plant layout plan for the major cable raceways, which shows how the cable trays are located on a division/train basis. However, supplemental assumptions concerning cable routing may be required (see Langman letter dated April 15, 2004, response to Question 79).

AECL should provide a comprehensive list of assumptions it made in the PRA concerning as-yet-to-be-determined design details, including but not limited to, interface requirements between the proposed certified design and the remainder of the plant. The staff expects that AECL will confirm that the final design agrees with the assumptions made in the PRA during the combined license application process.

Implementation of the stated methodology. The staff expects that the ACR PRA will follow the stated PRA methodology of the preapplication review effort that it reviewed during FT11. AECL should capture changes to or extensions of the ACR PRA methodology in a revision to the ACR PRA analysis basis (see AECL Report No. 108-03660-AB-001). In addition, AECL should fully capture all implementation details in the ACR PRA documentation to enable an in-depth staff review. During the preapplication review, the staff noted several instances in which the generic CANDU PRA reference analysis was internally inconsistent (e.g., multiple values were presented for failure rates, calculational errors, etc.). The staff expects that AECL QA process is sufficient to identify and resolve discrepancies, editorial errors, and typographical errors within the ACR PRA.

Common-cause failure analysis. The UPM approach to estimating CCF beta factors may be a viable approach for conducting a design-stage PRA. However, the staff has little experience with this approach and is unable to comment on its acceptability without further review. Because the UPM approach is subjective and only loosely anchored to observed CCF statistics, AECL should confirm the reasonableness of the CCF beta factors produced by the UPM method by comparing them to the statistically based estimates in NUREG/CR-5497, "Common-Cause Failure Parameter Estimations," October 1998. Since CCF is typically a significant contributor to risk, the staff expects that AECL will conduct sensitivity analyses to explore modeling uncertainties involving the CCF analysis.

Human reliability analysis. The staff must confirm that AECL assumptions concerning the procedural basis for operator actions are reasonable. AECL has stated that it will develop symptom-based EOPs for generic actions on power reductions, heat sinks, and containment integrity, and that it will prepare event-based EOPs when entrance conditions are clear (see Langman letter dated April 15, 2004, response to Question 72). Because human actions are typically significant contributors to risk, the staff expects that AECL will conduct sensitivity analyses to explore modeling uncertainties involving the HRA.

External event analysis. AECL should provide a technical basis for all numerical values it used in the study (e.g., cable damage and ignition temperatures, times to cause fire-induced damage, multipliers on human error probabilities to account for additional psychological stress during external events, fire suppression failure probabilities, flood-induced failure probabilities for doors, etc.). Suitable technical bases should include, but are not limited to, references to reports issued by and prepared for regulatory authorities, references to refereed conference and technical journal papers, references to risk-informed licensee submissions made to the NRC, and the results of AECL-sponsored research. If AECL bases the numerical values used in the external events analyses on expert opinion, then it should thoroughly explain and document the process used to elicit such opinions. AECL numerical values that are founded on assumptions lacking adequate technical bases should be realistically conservative and marked as such; the staff expects that AECL will explore the uncertainty induced from using such values with sensitivity analyses.

Level 2 PRA. As previously discussed, AECL description of its approach is very high level and lacks sufficient detail to communicate what it will do. Therefore, the staff should pay particular attention to all aspects of the ACR Level 2 PRA to ensure the following:

- The interface between the Level 1 and Level 2 PRAs is appropriate.
- The definition of large release generally agrees with previous NRC definitions (e.g., NUREG/CR-6595, "An Approach for Estimating the Frequencies of Various Containment Failure Modes and Bypass Events," Revision 1, October 1998).
- The grouping of accidents into CDSs is appropriate and consistent.
- The CET structure adequately represents the expected progression of severe accidents in the ACR, as determined by T-H simulations performed using acceptable tools and experiments.
- The estimation of LRF uses traceable and acceptable methods.

9.5 Conclusions

Based on its examination of the ACR PRA methodology, the staff has not identified any issues that would preclude certification of the ACR-700 design. The staff's summary of its review of the ACR PRA methodology is as follows:

- The ACR PRA methodology is generally consistent with the methodologies used to develop PRAs of current generation LWRs licensed in the United States.

- The ACR PRA methodology provides high-level guidance to analysts; it is not a step-by-step procedure. Therefore, the staff was unable to examine various implementation details that may have a significant impact on the acceptability of the ACR PRA results.

- Once AECL has resolved the methodological issues identified above to the staff's satisfaction, AECL PRA methodology will be sufficient for the purpose of assessing the ACR for licensing in the U.S.

- While resolution of the methodological issues identified above will require AECL to expend resources, the staff believes that AECL can effectively address them. That is, the staff does not believe that AECL will need to develop any new analytical risk assessment methods to resolve the outstanding methodological issues associated with the ACR PRA.

10. CANFLEX FUEL DESIGN (FT13)

10.1 Review Scope

The staff completed a preapplication review on AECL Report No. 108-37000-LS-001, "ACR Fuel." This report describes the ACR fuel design and verification, as well as its performance during normal operations. This report focuses on the fuel bundle design and steady-state fuel performance code ELESTRES. It does not address fuel safety-related computer programs, such as CATHENA and ELOCA, which analyze fuel performance under accident conditions.

Section 4.2 of the SRP, "Fuel System Design," provides regulatory guidance for reviewing the 43-element CANFLEX fuel bundle design and its adherence to applicable GDC. In addition, the staff incorporated knowledge and insight from recent NRC safety evaluation reports on LWR nuclear fuel designs into the preapplication review.

In accordance with Section 4.2 of the SRP, the main objectives of the fuel system safety review are to assure the following:

- The fuel system is not damaged as a result of normal operation and AOOs.
- Fuel system damage is never so severe as to prevent control rod insertion when required.
- The number of fuel rod failures is not underestimated for postulated accidents.
- Coolability is always maintained.

During the preapplication review, the staff determined the applicability of postulated fuel damage mechanisms and fuel failure mechanisms to the 43-element CANFLEX fuel bundle design. The staff identified deviations resulting from the unique characteristics of the CANFLEX bundle design, the ACR-700 plant design, and/or the ACR-700 unit operation. In addition, the staff highlighted regulatory and policy issues resulting from these unique characteristics.

The staff based its preapplication review of the fuel design and fuel performance computer models, as well as the experience database used to validate their predictions, on recent SERs on LWR fuel design and fuel performance computer codes. This portion of the review was limited for the CANFLEX fuel validation database, because, while the fuel performance validation was discussed in concept, AECL did not present it in detail.

The staff did not address fuel design aspects associated with transportation and long-term storage as part of this review.

10.2 Regulatory Issues

This section documents the following regulatory issues which may necessitate rulemaking or exemptions to 10 CFR Part 50:

- Title 10, Section 50.46(b)(1) and (2), of the Code of Federal Regulations (10 CFR 50.46(b)(1) and (2)) limit peak fuel clad temperature to 2200 EF and clad oxidation to 17 percent ECR. For the ACR-700 design, the postulated stagnation feeder tube break event results in fuel melt and clad temperatures beyond 2200 EF within the affected channel. This event falls within the definition of a small-break LOCA and exceeds the 10 CFR 50.46 criteria.

- Title 10, Section 50.46(b)(3), of the Code of Federal Regulations (10 CFR 50.46(b)(3)) limits the amount of hydrogen generation resulting from a chemical reaction experienced by the cladding. For the ACR-700 design, the PTs are manufactured from a zirconium-niobium alloy and are capable of reacting to produce hydrogen at the elevated temperatures during a LOCA. AECL will need to account for this potential source of hydrogen production (although expected to be small) in its analysis of performance.

- Section 10.3 of the ACR fuel report states, "Qualified facilities, qualified computer codes, and qualified staff will be used in these assessments. This will achieve full compliance with ISO 9001 and CSA 286.7." AECL will need to address differences between these standards and Appendix B to 10 CFR Part 50 as discussed in Section 5.7 of this report.

10.3 Potential Policy Issues

Building upon many years of CANDU experience, AECL will continue its verification and validation program for the ACR fuel design and supporting thermal-mechanical fuel performance models. In addition to recent and ongoing irradiation programs at the NRU test reactor, AECL plans on a final validation of the 43-element CANFLEX fuel design at the Halden facilities in Norway. Because of the duration of these irradiations and the subsequent hot-cell PIEs, AECL will not complete its final validation program in time to support the proposed ACR-700 design certification schedule. As a result, the staff's review and approval of the 43-element CANFLEX fuel bundle design and supporting thermal-mechanical fuel performance model (ELESTRES) will not be completed during the design certification phase.

The staff identified this as a potential policy issue requiring Commission guidance. Note that Section 10.4 of this report details several issues requiring NRC management awareness and/or guidance.

10.4 Technical Issues

This section documents technical issues related to the 43-element CANFLEX fuel bundle design, the associated fuel thermal-mechanical computer models, and the in-reactor experience and empirical database the staff employed to validate the design and the models.

The CANFLEX fuel elements are designed to promote clad collapse onto the fuel column at BOL under normal coolant pressure. To promote this design feature, the thickness of the CANFLEX fuel clad (sheath) is up to 30 percent thinner than PWR fuel clad, and the initial fuel-to-clad gap size is minimal. This behavior differs from LWR fuel designs, and the staff will consider the following actions to ensure a thorough evaluation:

- Evaluate the degree of clad deformation to ensure it is within the elastic region.

- Evaluate the ovality of the clad upon collapse and throughout burnup, especially as rod internal pressure approaches system pressure.

- Determine the potential for longitudinal ridging and evaluate any adverse impact.

- Determine the potential for clad collapse into any axial spaces, especially at the higher coolant pressure of the ACR (relative to past CANDUs), because, in accommodating the higher fission gas release (associated with higher burnup), the ACR fuel element design will include more internal void volume.

- Describe the supporting experience and empirical database, specifically (1) pool-side measurements, (2) hot-cell PIEs, and (3) unirradiated autoclave tests.

- Determine the impact of irradiation hardening on the clad's initial elastic/plastic deformation and ovality, and on the ability of the clad to regain its initial symmetric shape.

- Because ELESTRES does not appear to model the initial collapse of the clad onto the fuel column, the staff will review the impact of this omission on fuel thermal-mechanical models including clad stress/strain calculations.

- Because ELESTRES does not appear to model changes in ovality of the clad or asymmetric diametral gap, the staff will review the impact of this omission on fuel thermal-mechanical models including fuel temperature and clad stress/strain calculations.

AECL fuel design requirements credit available clad ductility in the clad strain evaluation in contrast to a prescriptive limit of 1.0 percent strain. Predicted clad ductility would vary as a function of several parameters, including temperature, burnup, and corrosion (oxide and hydride levels). The staff will review the following actions in order to resolve certain technical issues during the design certification review:

- Identify elastic and plastic strain components as a function of burnup.

- Determine the impact of irradiation hardening on strain predictions.

- Determine the impact of clad corrosion on strain predictions.

- Describe the supporting experience and empirical database, specifically (1) measured versus predicted strain for separate effects, (2) measured versus predicted strain for combined (synergistic) effects, and (3) whether the database bounds ACR fuel design limits.

- Determine the potential limit on allowable steady-state strain based upon the (1) channel flow blockage, (2) allowances within CHF testing or CHF uncertainties, and (3) core physics uncertainties.

- Determine the impact of uniform versus nonuniform stress/strain components, because ELESTRES does not appear to model nonuniform clad stress/strain.

- Determine stress/strain in the end regions of each element, because ELESTRES appears to only model the center region of the fuel element.

The 43 elements within the ACR CANFLEX design are welded at each end to an endplate. The rigidity of the CANFLEX bundle design is required to ensure dimensional stability and compatibility with refueling equipment. The staff will need to resolve the following items during the design certification review:

- Radial power gradients across the fuel bundle promote differential axial thermal expansion which may bend/skew the endplates and apply loads to the endcap welds. This may also result in rod and bundle bowing. What is the negative impact of this potential effect?

- Radial power gradients across the fuel bundle promote differential axial irradiation growth which may bend/skew the endplates and apply loads to the endcap welds. This may also result in rod and bundle bowing. What is the negative impact of this potential effect?

- Are pin-to-bundle power limits required to limit this power gradient?

- What are the design limits on endplate deformation?

- How are these dimensional changes accounted for in core physics predictions?

- How are these dimensional changes accounted for in core T-H predictions and the supporting CHF test program?

The PT confines 12 fuel bundles within a given channel. During its lifetime, PT dimensions change because of irradiation growth and creep. Furthermore, refueling maneuvers may result in scratches on the PT along bearing pad tracks. The staff will need to review the following items during the design certification review:

- What are the design limits on PT dimensional changes and scratches?

- How are these dimensional changes accounted for in core physics predictions?

- How are these dimensional changes accounted for in core T-H predictions and the supporting CHF test program?

One of the primary objectives of SRP Section 4.2 is to ensure that the number of fuel rod failures during a postulated accident is not underestimated. AECL intends to employ a detailed mechanistic fuel element model (ELOCA) to predict fuel element performance under severe local conditions, and to quantify clad integrity and ultimately radiological source term. Allowing severely degraded clad-to-coolant heat transfer without a presumption of clad failure, and the resulting release of gas gap activity, is a significant departure from current radiological source

term guidance. The design certification review will include an assessment and verification of the validity of the supporting empirical database, specifically the following items:

- clad integrity as a function of time at elevated temperatures
- measured versus predicted clad strain for separate effects
- measured versus predicted strain for combined (synergistic) effects
- clad corrosion rates at elevated temperatures

The design certification review will verify the fuel integrity acceptance criteria against predicted clad temperature, clad strain, and oxidation rates. AECL should verify the fuel thermal-mechanical models under dynamic accident conditions against data, especially the following models:

- fuel temperature model
- fuel pellet thermal expansion and relocation models
- fission gas release model
- clad creep model

AECL should quantify the uncertainties associated with model predictions and the empirical database, and should determine how to apply this as a source term penalty.

- Figure 5-1 of the ACR fuel report depicts fuel element ratings for the different bundle positions as a function of burnup for the 37-element bundle design. While the 43-element CANFLEX bundle is expected to exhibit lower element ratings, fuel power levels are still predicted to greatly exceed those in U.S. LWR designs. Thus, AECL should consider the following actions:

  – Determine the impact of prolonged, high element ratings on fuel pellet stability, pellet cracking, and fission gas release .

  – Determine the impact of the continued approach to melting temperatures on fuel pellet morphology caused by end flux peaking.

  – Determine the impact of the approach to fuel melting temperatures during power ramps (e.g., refueling or power maneuvers).

- To avoid past fuel reliability issues, AECL has developed power ramp limitations and introduced CANLUB graphite coating. Thus, AECL should consider the following actions:

  – Provide the basis for the power ramp limitations including past in-reactor experience and empirical testing.

  – Make the following two determinations, because past experience indicates an increasing degradation of the CANLUB coating with burnup:

    (1) the impact of degradation of the CANLUB coating as a function of burnup and operating power for the CANFLEX design

(2) the impact of the loss of function of the CANLUB coating with its burnup degradation, and how this loss of function limits power ramps at end of life or dictates loading/refueling strategy

- The unique characteristics and potential consequences of the LCDAs will require the following actions to provide significant technical validation:

  – Confirm that the violation of fuel melting criteria during any postulated feeder line break is limited to a single fuel channel.

  – Confirm that the violation of fuel melting criteria during any credible flow blockage event is limited to a single fuel channel.

  – Confirm that the molten mass escaping from the ruptured pressure/calandria tubes will not propagate into a more severe transient. Specifically, verify that the LCDAs do not result in the following two conditions:

    (1) a pressure pulse which challenges the integrity of the calandria vessel

    (2) failure of the calandria tube in neighboring fuel channels

- Unlike typical PWR fuel assembly axial power distributions, fuel elements in the CANFLEX bundle experience end flux peaking. AECL should determine the impact of potential end flux peaking on the following:

  – endcap weld integrity
  – ELESTRES fuel performance model predictions
  – core physics predictions and the buildup of fission products
  – core T-H calculations and the supporting CHF test program

- AECL will need to develop licensed limits on the 43-element CANFLEX fuel design for the following parameters:

  – peak element, bundle, and channel power ratings
  – peak fuel element and peak bundle burnup limits (e.g., 28 GWd/MTU)
  – U-235 enrichment limit on SEU fuel (e.g., 2.1 wt%)
  – Dy limit for center element (e.g., 7.5 wt%)
  – clad oxide and hydride limits (e.g. 100 microns and 600 ppm)

AECL will base its V&V of the ACR fuel design and supporting thermal-mechanical fuel performance models upon prototypical test bundles assembled in machine shops at CRL. AECL has not yet identified a commercial fuel fabrication facility (along with its fabrication process and quality control). While the staff will review the design specifications of the 43-element CANFLEX fuel bundle and its expected in-reactor performance, future validation of the commercial fuel fabrication facility to manufacture in strict compliance to these design specifications will be required.

AECL needs to define and defend specified acceptable fuel design limits (SAFDLs), in accordance with GDC 10, “Reactor Design,” for the ACR fuel design. With the exception of the

centerline fuel melt, AECL-proposed fuel design limits do not coincide with LWR requirements. Because 10 CFR Part 50 does not define numerical values for SAFDLs, differences between LWR and ACR SAFDLs will not necessitate exemptions. However, the unique characteristics of the ACR fuel design and the proposed fuel integrity design limits may require policy consideration.

The CANFLEX fuel elements are designed to promote clad collapse onto the fuel column at BOL under normal coolant pressure. To promote this design feature, the thickness of the CANFLEX fuel clad (sheath) is up to 30 percent thinner than PWR fuel clad. Furthermore, the fuel-to-clad gap size is minimal, and axial gaps along the pellet stack are limited.

For the ACR-700 design, the postulated single channel flow blockage event results in significant fuel melt within the affected channel. The molten fuel causes PT and then calandria tube failure, which initiates a LOCA. This combined event and its consequences (which could be initiated by a simple lapse in foreign material control) are unique to the ACR design and do not fall within current regulations.

Predicted cladding strain during normal operation may exceed the 1.0 percent design limit currently imposed on LWR fuel. AECL design criteria allow credit for available ductility as a function of burnup, temperature, and corrosion.

The CHF corresponds not to DNB, but to incipient channel dryout. Allowing degraded clad-to-coolant heat transfer (e.g., DNB) or channel voiding during normal operation or AOOs is a significant departure from current design criteria.

Exceeding the DNB SAFDL is permitted under GDC 10, “Reactor Design,” during less probable limiting fault events. However, allowing severely degraded clad-to-coolant heat transfer without a presumption of clad failure is a significant departure from current radiological source term guidance. AECL plans to address this as follows:

- AECL intends to employ a detailed mechanistic fuel element model (ELOCA) to predict fuel performance under these severe, dynamic local conditions. Validation of a transient fuel performance model and quantification of the uncertainties associated with these predictions is a significant effort.

- AECL has proposed fuel integrity acceptance criteria. Limits on clad temperature, clad strain, and clad oxidation would demonstrate clad integrity well beyond DNB and channel dryout.

In the event of an end plug or shield plug malfunction, it may not be possible to refuel a single channel. Current CANDU operating licenses allow a 365-day duration for fuel bundles to remain within the defective channel beyond the intended refueling schedule. AECL believes that failure of channel components which prevent refueling a single channel should be classified as an AOO. The staff, however, believes that fuel bundle design and regulatory criteria should continue to apply to fuel bundles remaining within these faulted channels.

During plant outages and lower modes of operation, the configuration of various systems undergoes several planned evolutions. The probability of unplanned incidents or failures (e.g., loss of shutdown cooling) are sufficiently likely that AECL must account for them. AECL has

not defined or evaluated the consequences of these incidents. For example, CANDUs do not currently off-load fuel during maintenance outages. AECL will need to address whether these plants are allowed to use the polar crane with fuel present in the core below, or whether these plants have emergency procedures (time to boil, time to uncovery, containment evacuation and isolation, etc.) for these outage configurations.

During on-power fueling activities, AECL will postulate all credible incidents during each stage of the fuel maneuvers. AECL has not defined or evaluated the consequences of these incidents. On-power fueling creates many more scenarios than the postulated LWR single fuel assembly drop incident (in the spent fuel pool (SFP) or refueling pool). Accidents range from ejecting 12 full-power bundles onto the containment floor, to potentially melting a single bundle in the fuel transfer tube outside the containment, to dropping a basket of 36 bundles in the SFP. Each of these scenarios has different source terms and different release paths.

RGs 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors, issued in 1972, and 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," issued in 2000, provide radiological source term guidelines for calculating dose consequences. Estimated fission gas release for the ACR fuel design exceeds these regulatory guidelines for non-LOCA events. Figure 4-3 of the ACR fuel report (which illustrates measured fission gas release as a function of element power rating and burnup) estimates that the outer fuel elements will experience a 15–20 percent fission gas release. This elevated level of fission gas release is attributed to the high element rating of approximately 17 KW/ft of the ACR (relative to a typical PWR fuel rod power of 8 KW/ft).

10.5 Conclusions

The AECL ACR fuel report provides a reasonable description of the 43-element CANFLEX bundle design basis and validation plan. In its conclusions, AECL states the following:

> The qualification plan for ACR fuel is designed to confirm that the fuel has full thermal integrity, full structural integrity, and full compatibility with interfacing systems. This will be achieved by confirming that all appropriate parts of ACR fuel will meet SAFDLs. Towards that end, AECL will conduct a comprehensive, integrated set of in-reactor tests, out-of-reactor tests, analyses and engineering judgements. Qualified facilities, qualified computer codes, and qualified staff will be used in these assessments.

This statement represents a commitment to demonstrate acceptable ACR fuel performance. While the 43-element CANFLEX fuel design and the ACR-700 represent a departure from past CANDU designs, the staff believes AECL will be able to design an acceptable CANFLEX fuel bundle, building upon past experience coupled with a comprehensive validation plan.

AECL has the unique ability to qualify fuel designs and validate computer models at state-of-the-art facilities, including in-reactor testing at CRL. In addition to the past CANDU database and more recent NRU irradiations, AECL is planning a final demonstration program at Halden.

Based upon the staff's review, the most significant issues in the ACR-700 fuel design review are expected to be as follows:

- verifying fuel element integrity under severe, dynamic transient conditions, including degraded clad-to-coolant heat transfer regimes (e.g., beyond DNB and dryout) and licensing of the fuel integrity acceptance criteria

- validating the transient thermal-mechanical fuel performance models and quantifying the uncertainty associated with these predictions

- validating the steady-state clad strain model and associated fuel design criteria

- not completing AECL final fuel design validation program in time to support the ACR-700 design certification schedule

- ensuring that fuel damage is limited to a single fuel channel, given the unique characteristics, consequences, and regulatory approach to the LCDAs.

11. REFERENCES

References that include an NRC ADAMS Accession Number are available in the NRC's Agencywide Document Access and Management System (ADAMS), which is accessible from the Public Electronic Reading Room at http://www.nrc.gov/reading-rm.html. Select "Web-Based Access" and search for the Accession Number.

American National Standards Institute/American Nuclear Society (ANSI/ANS)

ANSI/ANS-58.21-2003, "American National Standard: External Events PRA Methodology," March 3, 2003.

American Society of Mechanical Engineers (ASME)

ASME RA-Sa-2003, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications," Addenda, December 5, 2003.

Atomic Energy of Canada, Ltd. (AECL)

— — — — —, "A Plan for the Pre-Application Review of the Advanced CANDU Reactor (ACR)," September 26, 2002. (ADAMS Accession No. ML022810658)

— — — — —, "ACR Workshop—Core Design and Reactor Physics" (ADAMS Accession No. ML022810624), presented September 25, 2002.

— — — — —, Assessment Document No. 108-03660-ASD-001, "Systematic Review of Plant Design for Identification of Initiating Events," Revision 1, January 2004. (ADAMS Accession No. ML040700097)

— — — — —, Assessment Document No. 108US-01321-ASD-001, "Identification and Initial Assessment of US NRC Generic Safety Issues Applicable to ACR," Revision 0, April 2003. (ADAMS Accession No. ML031340532)

— — — — —, Assessment Document No. 108US-01910-ASD-001, "Comparison of 10 CFR Part 50, Appendix B, and ASME NQA-1-1994 Requirements Versus CSA N286 Series of Standards," February 2003. (ADAMS Accession No. ML030920638)

— — — — —, Design Requirement No. 10810-35000-DR-001, "Fuel Handling and Storage System," Revision 0, March 2003. (ADAMS Accession No. ML040580382)

— — — — —, Design Requirement No. 10810-35300-DR-001, "Spent Fuel Transfer and Storage System," Revision 0, July 2003. (ADAMS Accession No. ML040580380)

— — — — —, Design Requirement No. 10810-63581-DR-001, "Fuel Handling Controls System," Revision 1, April 2003. (ADAMS Accession No. ML040580385)

— — — — —, “Introduction to PIRT Expert Panel,” presented October 2003.

— — — — —, Letter to the NRC, No. 108US-01321-021-001, “Responses to NRC’s RAIs on On-Power Fueling,” May 17, 2004. (ADAMS Accession No. ML041470201)

— — — — —, Letter to the NRC No. 108US-01321-021-001, “Response to NRC’s Requests for Additional Information (RAIs) on Quality Assurance,” May 31, 2004. (ADAMS Accession No. ML041560207)

— — — — —, Licensing Submission Document No. 108-35000-LS-001, “The Technology of On-Power Fueling,” Revision 0, September 2003. (ADAMS Accession No. ML0326809900)

— — — — —, “PIRT Expert Panel Meeting on ACR-700,” presented December 2003.

— — — — —, Report No. 91-03660-AR-001, “Analysis Report: Generic CANDU Probabilistic Safety Assessment—Methodology,” Revision 0, July 2002. (ADAMS Accession No. ML030430072)

— — — — —, Report No. 91-03660-AR-002, “Analysis Report: Generic CANDU Probabilistic Safety Assessment—Reference Analysis,” Revision 0, July 2002. (ADAMS Accession No. ML030430074)

— — — — —, Report No. 108-01200-440-001, “ACR R&D Status Report 2003.” (ADAMS Accession No. ML04280202)

— — — — —, Report No. 108-03500-AB-004, “Fuel and Fuel Channel Safety Analysis Methodology,” Revision 1, September 2003. (ADAMS Accession No. ML032810065)

— — — — —, Report No. 108-03660-AB-001, “Analysis Basis: Probabilistic Safety Assessment—ACR,” Revision 1, July 2003. (ADAMS Accession No. ML032200010)

— — — — —, Report No. 108-03600-AB-003, “Analysis Basis: Safety Basis for ACR,” Revision 0, July 22, 2003. (ADAMS Accession No. ML032200007)

— — — — —, Report No. 10810-01371-TED-001, “ACR-700 Technical Description,” Revision 0, June 2003. (ADAMS Accession No. ML032030391)

— — — — —, Report No. 10810-01371-TED-001, “ACR-700 Technical Description,” Revision 1, March 2003. (ADAMS Accession No. ML041350423)

— — — — —, Report No. 10810-03510-AB-001, “Initial Conditions and Standard Assumptions Safety Analysis Basis,” Section 2, Revision 0, August 2003. (ADAMS Accession No. ML032370079)

— — — — —, Report No. 108-37000-LS-001, “ACR Fuel,” Revision 0, January 2004. (ADAMS Accession No. ML042770001)

— — — — —, Report, “ACR-700 Limited Core Damage Accidents,” presented in a meeting with the NRC, April 5, 2004. (ADAMS Accession No. ML040990675)

— — — — —, Report No. 108-126810-LS-02, “ACR Limited and Severe Core Damage Accidents: Supporting R&D,” Revision 0, November 2003. (ADAMS Accession No. ML033280557)

— — — — —, Report No. ACR 108-126810-LS-001, “Phenomenology for Limited and Severe Core Damage Accidents in an ACR,” September 2003.

— — — — —, Report No.108US-03621-LS-001, “Classification of Safety Related Systems and Structures,” April 30, 2003. (ADAMS Accession No. ML031340516)

— — — — —, Report No. 108US-03621-LS-002, “Canadian Codes and Standards Used in CANDU Plants,” Revision 0, March 2004. (ADAMS Accession No. ML040700139)

— — — — —, “Response to NRC’s Requests for Additional Information (RAIs) #5 on Class 1 Pressure Boundary Design and Materials Review of Fuel Channels and On-Power Fueling,” March 31, 2004. (ADAMS Accession No. ML040990605)

— — — — —, Safety Design Guide No. 108-03650-SDG-001, “Safety Related Systems,” Revision 3, March 2004. (ADAMS Accession No. ML041000172)

— — — — —, Safety Design Guide No. 108-03650-SDG-002, “Seismic Requirements,” Revision 3, March 2004. (ADAMS Accession No. ML041000174)

— — — — —, Safety Design Guide No. 108-03650-SDG-004, “Separation of Systems and Components,” Revision 3, March 2004. (ADAMS Accession No. ML041000177)

— — — — —, “Scenario Description for ACR-700 Inlet Header Critical Break LOCA,” January 21, 2004. (ADAMS Accession No. ML040560464)

Authored Reports

Ingham, P., et al, AECL Report No. 108US-03500-225-003, “Scenario Description for RD-14M Inlet Header Critical Break LOCA Test B9401,” Revision 0, February 2004. (ADAMS Accession No. ML040540037)

Kowalski, et. al., AECL Report No. RC-2702, “CATHENA Mod-3.5c/Rev. 0 Fuel and Fuel-Channel Thermal-Mechanical Effects Validation Manual,” 2001.

Mallory, J., AECL Report No. RC-2240, “Validation Plan for CATHENA Mod-3.5c,” 2000.

Mallory, J.P. and K.A. Arsenault, AECL Report No. RC-2701, “CATHENA Mod-3.5c/Rev 0 Systems Thermal-Hydraulic Validation Manual,” 2001.

McGee, G. et al., AECL Report No. 108US-03532-225-001, “CATHENA Simulation of RD-14M Critical Break LOCA Experiment B9401,” Revision 0, February 2004.

Mills, P.J., et al., AECL Report No. RC-2151-0, “Validation of CATHENA Mod-3.5c/Rev. 0 for Phenomena Relevant to Fuel-Channel Thermal-Mechanical Behavior: Overall Plan,” Revision 0, 1999.

Pascoe, J., et al., AECL Report No. RC-1517, “A Phenomenology-Based Matrix of Tests for Use in Validation of Thermal-Hydraulics Codes Employed in CANDU Safety Analysis,” Revision 1.0, November 1995. (ADAMS Accession No. ML032090358)

Rouben, B., AECL Report No. TTR-370, “RFSP Program Description,” Revision 1, COG-94-580, April 1995.

Sils, H.E., and N. Popov, AECL Report No. 108US-03500-LS-001, “PIRT for Critical Inlet Header Break LOCA in ACR-700,” Revision 0, February 2004. (ADAMS Accession No. ML040540292)

Wright, D., et al., AECL Report No. 108US-03532-225-002, “Scenario Description for ACR-700 Inlet Header Critical Break LOCA,” Revision 0, February 2004. (ADAMS Accession No. ML040540048)

Canadian Standards (CSA)

CSA CAN/CSA-N285.0-95, “General Requirements for Pressure-Retaining Systems and Components in CANDU Nuclear Power Plants.”

CSA International Standard, N285.2-99, “Requirements for Class 1C, 2C, and 3C Pressure-Retaining Components and Supports in CANDU Nuclear Power Plants.”

CSA-N286.0.1-92, “Commentary on the Principles for Quality Assurance Programs of CSA N286 Series Standards.”

CSA-N286.0-92, “Overall Quality Assurance Program Requirements for Nuclear Power Plants.”

CSA-N286.1-00, “Procurement Quality Assurance for Nuclear Power Plants.”

CSA-N286.2-00, “Design Quality Assurance for Nuclear Power Plants.”

CSA-N286.3-99, “Construction Quality Assurance for Nuclear Power Plants.”

CSA-N286.7-99, “Quality Assurance of Analytical, Scientific, and Design Computer Programs for Nuclear Power Plants.”

Correspondence

Kim, J., NRC, Letter to V.J. Langman, AECL, “Requests for Additional Information—CATHENA Code for ACR-700 Application,” May 14, 2004. (ADAMS Accession No. ML041040166)

Kim, J., NRC, Letter to V.J. Langman, AECL, “Requests for Additional Information—ACR-700 Pre-application Class 1 Pressure Boundary Design and Materials Review of Fuel Channels and On-power Fueling,” March 19, 2004. (ADAMS Accession No. ML040760030)

Kim, J., NRC, Letter to V.J. Langman, AECL, “Requests for Additional Information—ACR-700 Pre-Application Quality Assurance Review,” May 5, 2004. (ADAMS Accession No. ML041260008)

Langman, V.J., AECL, Letter to the NRC, 108US-01321-021-001, “Further to the ACR Pre-Application Plan—Detailed Deliverables and Schedule for Focus Topics,” December 18, 2002. (ADAMS Accession No. ML023640013)

Langman, V.J., AECL, Letter to B. Sosa, NRC, 108US-013210-021-001, “Phase 2 of ACR Pre-Application Review,” July 30, 2003. (ADAMS Accession No. ML032180580)

Langman, V.J., AECL, Letter to the NRC, 108US-01321-021-001, “Response to NRC’s Requests for Additional Information (RAIs) #3 on PRA Quality,” February 12, 2004. (ADAMS Accession No. ML040490250)

Langman, V.J., AECL, Letter to B. Sosa, NRC, “Proprietary Responses to NRC’s RAIs 106, 113, and 121 on Class 1 Pressure Boundary Design and Materials Review of Fuel Channels and On-Power Fueling,” April 14, 2004. (ADAMS Accession No. ML041120336)

Langman, V.J., AECL, Letter to the NRC, 108US-01321-021-001, “Response to NRC’s Requests for Additional Information (RAIs) #4 on PRA PRA Analysis Basis,” April 15, 2004. (ADAMS Accession No. ML041130200)

Langman, V.J., AECL, Letter to the NRC, 108US-01321-021-001, “Responses to NRC’s RAIs 68, 71, and 83 on PRA Analysis Basis,” April 30, 2004. (ADAMS Accession No. ML041310073)

Langman, V.J., AECL, Letter to the NRC, 108US-01321-021-001, “Regulatory Treatment of Limited Core Damage Accidents for the ACR-700,” May 10, 2004. (ADAMS Accession No. ML041340669)

Langman, V.J., AECL, to the NRC, “Proposed Accident Source Term for ACR-700,” May 28, 2004. (ADAMS Accession No. ML041560213)

Langman, V.J., AECL, Letter to the NRC, “Response to NRC’s Request for Additional Information (RAIs) on Event Categorization,” June 7, 2004. (ADAMS Accession No. ML041680370)

Langman, V.J., AECL, Letter to the NRC, 108US-01321-021-001, “Responses to NRC’s RAI 37 on PRA Analysis Basis,” June 17, 2004. (ADAMS Accession No. ML041750266)

Powers, D.A., Chairman, Advisory Committee on Reactor Safeguards, Letter to Richard A. Meserve, Chairman, NRC, “Pre-Application Phase of the AP1000 Standard Plant Design—Phase I,” September 14, 2000. (ADAMS Accession No. ML003750183)

Sosa, B., NRC, Letter to V.J. Langman, AECL, “Advanced CANDU Reactor (ACR-700)—Phase 2 of Pre-Application Review,” February 4, 2004. (ADAMS Accession No. ML032750547)

Van Adel, R., President and Chief Executive Officer, AECL, Letter to the NRC, “Pre-Application Review of ACR-700,” 115-01321-021-002, June 19, 2002. (ADAMS Accession No. ML021750473)

International Atomic Energy Agency (IAEA)

IAEA, “Procedures for Conducting Probabilistic Safety Assessments of Nuclear Power Plants, Level 1,” Safety Series 50-P-4, 1992.

Miscellaneous

ACR-700, 10810-03660-ASD-005, “Review of Design Features for Severe Accident Management,” January 2004. (ADAMS Accession No. ML040580378)

ACR USA Report 108US-31100-LS-001, “The Technology of CANDU Fuel Channels,” Revision 0, August 13, 2003. (ADAMS Accession No. ML032300450)

AEA Technology, “UPM 3.1: A Pragmatic Approach to Dependent Failures Assessment for Standard Systems,” SRDA-R13, April 1996.

Altiparmakov, D.V., “Calculation of Coolant Void Reactivity Components in Lattices of NG CANDU Type,” FFC-RRP-398, January 2002. (ADAMS Accession No. ML040050570)

Altiparmikov, D.V., “WIMS-AECL Theory Manual,” COG-00-077, September 2001. (ADAMS Accession No. ML031120598)

Ardron, K.H. and S. Banerjee, “Flooding in an Elbow Between a Vertical and Horizontal or Near-Horizontal Pipe,” *International Journal of Multiphase Flow*, 1986.

Beuthe, T.G. and B.N. Hanna, “CATHENA MOD-3.5c/Rev. 0 Theoretical Manual,” COG-00-008, CANDU Owners Group, Inc., November 2000.

Buell, J.R., et al., “Description of the Large-Scale Header Facility,” COG-95-101, 1999. (ADAMS Accession No. ML033640367)

Casal, J.J., R.J.J. Stamm’ler, E.A. Villario, and A.A. Ferri, “HELIOS: Geometric Capabilities of a New Fuel-Assembly Program,” Proceedings of the International Topical Meeting—Advances in Mathematics, Computations, and Reactor Physics, Vol. 2, pp. 10.2.11-13. Pittsburgh, Pennsylvania, April 28–May 2, 1991.

Chaxton, K.T., J.G. Collier, and A.J. War, “H.T.F.S. Correlation for Two-Phase Pressure Drop and Void Fraction in Tubes,” AERE-R7162, 1972.

Diamond, D., A. Hanson, and J. Saccheri, “ACR-700 CVR Calculations,” Letter report for Task 1, June 24, 2004. (ADAMS Accession No. ML042780640)

Downar, T.J., C.A. Cotton, and D. Lee, “Assessment of PARCS Core Simulator for ACR-700 Application,” Letter report for Task 2, June 14, 2004. (ADAMS Accession No. ML042780641)

Downar, T.J., et al, “PARCS: Purdue Advanced Reactor Core Simulator,” PHYSOR 2002, Seoul, Korea, October 7–10, 2002.

Douglas, S.R., “Void Reactivity Error Assessment for CANDU Reactors,” COG-01-030, June 2001. (ADAMS Accession No. ML040050583)

Douglas, S.R., R.C. Chow, R.E. Donders, and R.T. Jones, “System validation manual for WIMS-IST/DRAGON-IST/RFSP-IST reactor physics code suite,” COG-01-144/FFC-RRP-378, November 2001. (ADAMS Accession No. ML031120565)

Marleau, G.A. and R. Hebert, “A User Guide for DRAGON: Version DRAGON,” Ecole Polytechnique de Montreal Report 000331, Release 3.04, IGE.174, Revision 5, April 2000. (ADAMS Accession No. ML032160626)

“MCNP—A General Monte Carlo N-Particle Transport Code, Version 5—Volume I: Overview and Theory, Volume II: Users Guide,” LA-UR-03-1987 and LA-CP-03-0245, X-5 Monte Carlo Team, Los Alamos National Laboratory, September 2003.

“Plant Performance Specification—ACR-700 Human Factors Engineering Program Plan,” (HFEPP) Revision 1, June 23, 2003. (ADAMS Accession No. ML041000306)

Popov, N.K. and U.S. Rohatgi, “Effect of Interfacial Shear and Entrainment Models on Flooding Predictions,” AIChE Symposium Series, “Heat Transfer,” Seattle, Washington, 1983.

Poston, D.L. and H.R. Trellue, “User’s Manual, Version 2.0 for MonteBurns,” LA-UR-99-4999, Los Alamos National Laboratory, September 1999.

Shin, K.S., “Data Report for Large Scale Header Tests (Tests No. 226 to 237),” CWAPD-472, April 1987.

Shin, K.S., “Data Report for Steady-State Tests Conducted in the Large-Scale Header Test Facility (Test No: 764 to 877),” COG-90-206, June 1991. (ADAMS Accession No. ML041690256)

Smith, N., M. Armishaw, and A. Cooper, “Current Status and Future Direction of the MONK Software Package,” 7th International Conference on Nuclear Criticality Safety (ICNC2003), Tokai-mura, Japan, October 20–24, 2003.

Solbrig, C.W., J.H. McFadden, R.W. Lcyzkowski, and E.D. Hughes, “Heat Transfer and Function Correlations to Describe Steam-Water Behavior in Nuclear Safety Studies,” AIChE Symposium Series 74, “Heat Transfer: Research and Applications.”

Trapp, J.A. and V.H. Ransom, “A Choked-Flow Calculation Criterion for Nonhomogeneous, Nonequilibrium, Two-Phase Flows,” *International Journal of Multiphase Flow*, Vol. 8, No. 6, 1982.

Whitlock, J.J., W.J. Garland, and M.S. Milgram, “Effects Contributing to Positive Coolant Void Reactivity in CANDU,” Transactions of the American Nuclear Society, 72, pp. 329–330, 1995.

Williams, M.L., R.J. Ellis, and J.C. Gehin, “The Use of Sensitivity Analysis with Experimental Benchmarks to Quantify Bias and Uncertainty in the Calculation of ACR-700 Coolant Void Reactivity,” Letter Report for Task 1, May 2004.

Valente, J.U. and J.W. Yang, “MAAP 3.0B Code Evaluation Final Report,” Technical Report FIN L-1499, October 1992. (ADAMS Accession No. ML9403220222)

US Nuclear Regulatory Commission (NRC)

— — — — —, “NRC Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants,” August 8, 1985.

— — — — —, “PIRT report for ACR-700 Reactor Systems Analysis” (to be issued).

— — — — —, “Regulation of Advanced Nuclear Power Plants: Statement of Policy,” Federal Register, Vol. 51, p. 24643, July 8, 1986.

— — — — —, “Safety Goals for the Operations of Nuclear Power Plants: Policy Statement,” Federal Register, Vol. 51, p. 30028, August 4, 1986.

— — — — —, SECY-93-087, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light Water Reactor (ALWR) Designs,” April 2, 1993; Staff Requirements Memorandum (SRM) from Samuel J. Chilk, Secretary, to James M. Taylor, EDO, “SECY-93-087: Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs,” July 21, 1993.

— — — — —, SECY-96-128, “Policy and Key Technical Issues Pertaining to the Westinghouse AP-600 Standardized Passive Reactor Design,” June 12, 1996.

Taylor, James M., Executive Director for Operations, SECY-93-092, Policy Issue Memorandum to the NRC, “Issues Pertaining to the Advanced Reactor (PRISM, MHTGR, AND PIUS) and CANDU 3 Designs and Their Relationship to Current Regulatory Requirements” (ADAMS Accession No. ML040210725), April 8, 1993.

Vietti-Cook, Annette L., Secretary, NRC, Policy Memorandum to William D. Travers, Executive Director for Operations, “Staff Requirements—COMJSM-00-0003—Staff Readiness for New Nuclear Plan Construction and the Pebble Bed Reactor,” February 13, 2001. (ADAMS Accession No. ML010440409)

Code of Federal Regulations (10 CFR)

— — — — —, Title 10, “Energy,” Part 50, “Domestic Licensing of Production and Utilization Facilities.”

— — — — —, Title 10, “Energy,” Part 50, Appendix A, “General Design Criteria.”

— — — — —, Title 10, “Energy,” Part 52, “Early Site Permits, Standard Design Certification, and Combined Licenses for Nuclear Power Plants.”

NUREG Reports

— — — — —, NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants LWR Edition,” June 1996 (ADAMS Accession No. ML033580677).

— — — — —, NUREG-0800, Chapter 7, “Instrumentation and Controls,” Revision 4, June 1997.

— — — — —, NUREG-0800, Chapter 19, “Use of Probabilistic Risk Assessment in Plant-Specific, Risk-Informed Decisionmaking: General Guidance,” Revision 1, November 2002.

— — — — —, NUREG-0800, Chapter 19.1, “Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities,” Draft, November 2002.

— — — — —, Draft Standard Review Plan, Section 15.0.2, “Review of Transient and Accident Analytical Methods,” January 2003.

— — — — —, NUREG-1226, “Development and Utilization of the NRC Policy Statement on the Regulation of Advanced Nuclear Power Plants,” June 1988.

— — — — —, Draft NUREG/CR, “Phenomena Identification and Ranking Technique (PIRT) Applied to the ACR-700 Design,” April 2004.

— — — — —, NUREG/CR-0200, Draft, “SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation,” Revision 7, ORNL/NUREG/CSD 2/R7, Vols. I, II, and III, Oak Ridge National Laboratory, June 2004.

— — — — —, NUREG/CR-5535, “RELAP5/MOD3.3 Code Manual, Volume 1: Code Structure, System Models, and Solution Methods,” Revision 1, June 2004.

Chung, B.D., et. al., NUREG/IA-0189, “Improvements of RELAP5/MOD3.2.2 Models for the CANDU Plant Analysis,” October 2000.

Marshall, F.M., et al., NUREG/CR-5497, “Common-Cause Failure Parameter Estimations,” October 1998. (ADAMS Accession No. 9811170329)

Pratt, W.T. and V. Mubayi, NUREG/CR-6595, “An Approach for Estimating the Frequencies of Various Containment Failure Modes and Bypass Events,” Revision 1, August 2003.

Spore, J.W., et al., Los Alamos National Laboratory, NUREG/CR-6724, “TRAC-M/FORTRAN 90 (Version 3.0) Theory Manual,” July 2001.

Swain, A.D., NUREG/CR-4772, “Accident Sequence Evaluation Program Human Reliability Analysis Procedure,” February 1987. (ADAMS Accession No. ML8706120044)

Regulatory Guides (RG)

— — — — —, DG-1120, “Transient and Accident Analysis Methods,” December 2002.

— — — — —, RG 1.174, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,” Revision 1, November 2002.

— — — — —, RG 1.200, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-informed Activities,” issued for trial use, February 2004.