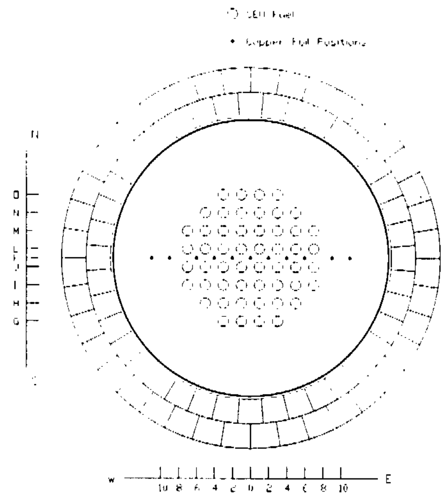
ZED-2 Measurements using Full-Core CANFLEX 0.95% SEU (Scheduled in 2004)

- Flux-maps and buckling measurements in hexagonal lattice
  - existing ZED-2 aluminum PT/CT assemblies with 3 coolants - H_2O, D_2O and air coolant, at 3 or more lattice spacings
- Flux-maps and buckling measurements in square-lattice
  - repeat of above using square lattice
  - copper activation foils will be positioned across the lattice and into the heavy water-reflector to measure the thermal flux peak in reflector on voiding
- Substitution measurements
  - 0.95% SEU into various ref. lattices (validation of substitution method)
  - 7-rod substitutions in ACR-type PT/CT assemblies with H_2O, D_2O and air coolant into ref lattice of 0.95% SEU (in existing ZED-2 aluminum PT/CT)
    - CANFLEX LVRF (1% SEU, Dy with NU in central element)
    - MOX (simulating irradiated ACR fuel)
- Temperature Reactivity Effects of Fuel, Coolant, and Moderator for various configurations

Flux-Map Set Up For Square Lattice



Summary

- Coolant Void Reactivity in CANDU lattice is caused by spatial and spectral changes of neutron flux upon voiding
- The physics design of ACR manipulates these changes to achieve a slightly negative coolant void reactivity with H_2O coolant, by:
  - Using a tighter lattice pitch and a larger gap between pressure tube and calandria tube than the existing designs
  - Adding burnable poison (Dy) in the central fuel pin
  - Using SEU fuel
- Confirmation of Negative CVR by:
  - Comparisons of AECL's physics codes with international physics codes
  - Verification of negative CVR in ZED-2 reactor experiments at CRL

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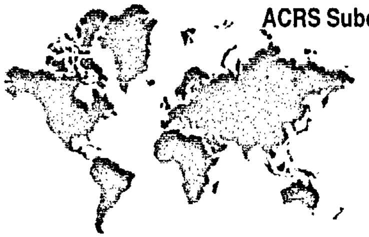
ACR Fuel



Peter G. Boczar
Director, Reactor Core Technology Division
ACRS Subcommittee on Future Plant Designs
Washington D.C.
January 13, 2004

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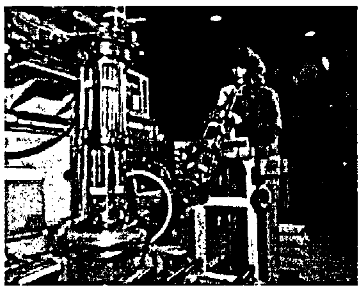
Outline

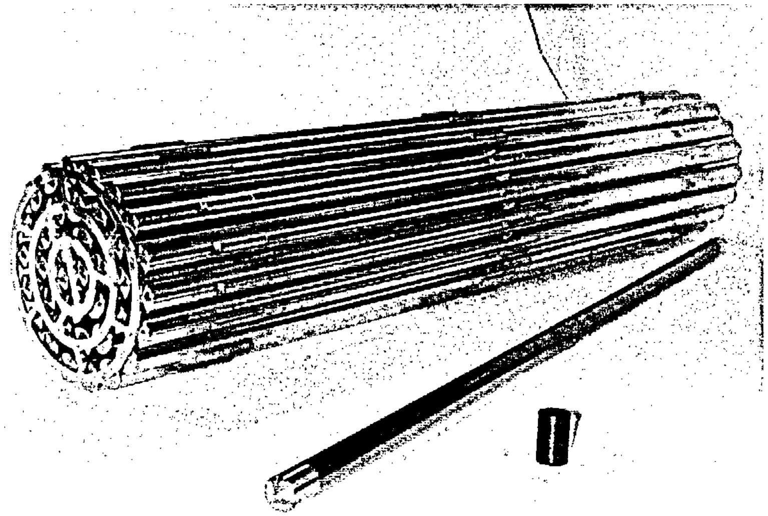
- Introduction to CANDU fuel
- ACR fuel design
- Experience relevant to ACR fuel
  - CANFLEX
  - extended burnup experience
  - low void reactivity fuel
- ACR fuel qualification

Characteristics of CANDU Fuel



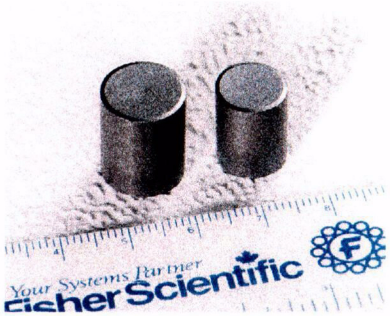
- Small, simple, light-weight
  - 20" length, 4" dia, 50 lb / bundle
  - CANFLEX has only 8 components
- Inexpensive
  - low fuel cycle costs (dollars/unit energy)
- Efficient
  - good use of uranium
- Excellent performance
  - ~ 2 million bundles fabricated; ~ 2 clad defects per million elements
  - on-power defect detection, location and removal
- Easy to manufacture and localize
  - CANDU fuel is manufactured domestically in 7 countries
  - CANDU (and its fuel) licensed in many different regulatory jurisdictions

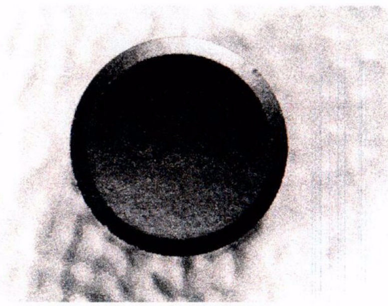
CANDU 6 37-element Fuel



UO_2 Pellets

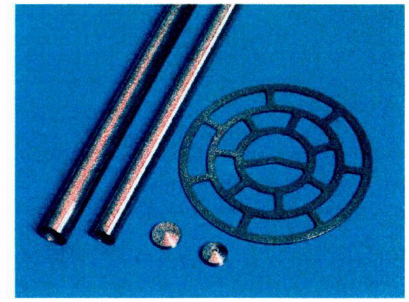
- UO_2, high density (for dimensional stability)
- Chamfers and end-dishes (reduce inter-pellet stresses on clad, volume for fission gas)





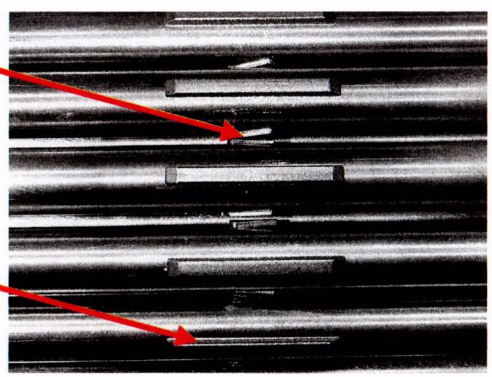
Clad, CANLUB, Endcaps, Endplates

- Clad
  - thin, collapsible (~0.016")
  - excellent heat transfer to coolant
  - low neutron absorption, Zr-4
- CANLUB
  - graphite coating applied to inside of clad provides protection against power ramp failures
- Endcaps
  - seal the fuel element
  - thin to reduce neutron absorption, good heat transfer
  - profiled to interact with fuel channel and fuel handling components
- Endplates
  - thin to minimize neutron absorption
  - flexible to accommodate fuel element differential expansion
  - strong and ductile to provide structural support and element separation



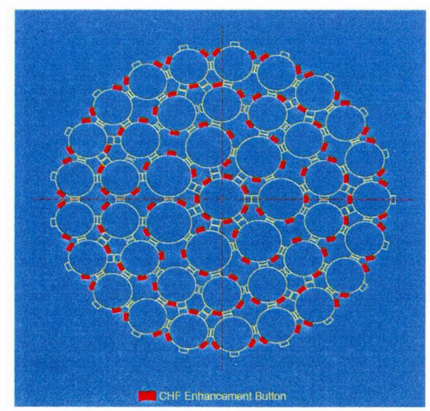
Spacers, Bearing Pads

- Inter-element spacers
  - provide element separation at the bundle midplane
- Bearing pads
  - provide element-to-pressure tube separation



CHF-Enhancing Buttons (CANFLEX)

- Appendages are attached on the 1/4 and 3/4 bundle planes



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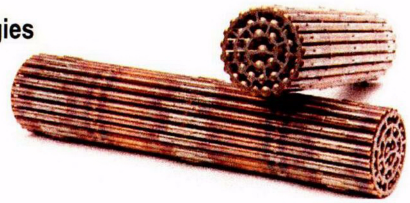
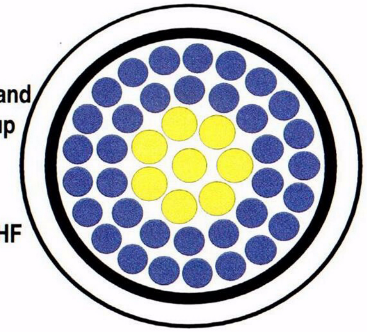
ACR Fuel Design

- Evolutionary extension of current fuel
  - extensive experience base on underlying technologies
- Based on 3 underlying technologies
  - CANFLEX geometry
  - low void reactivity fuel
  - extended burnup
- Key design features
  - 2.1% U^{235} in outer 42 elements
  - 7.5% Dy in nat. UO_2 in central element
  - 21 MWd/kg burnup

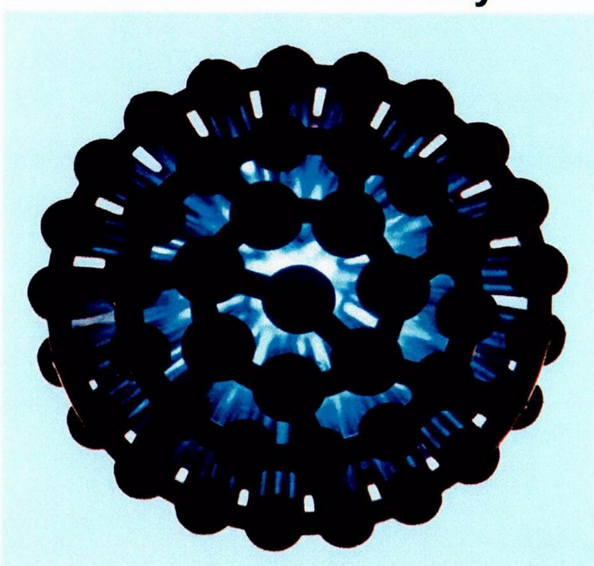


CANFLEX Geometry

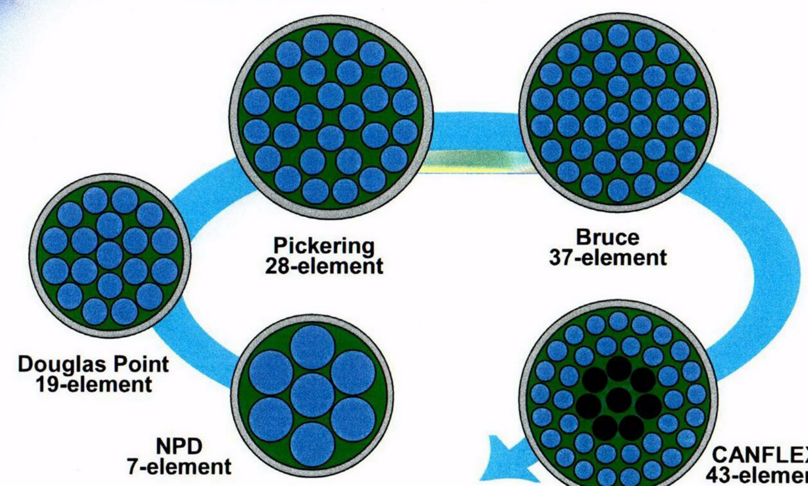
- ACR fuel based on CANFLEX Mk IV geometry
  - 43 elements, 2 element sizes
  - greater "subdivision" reduces ratings and facilitates achievement of higher burnup
  - "buttons" increase CHF
  - qualified for NU fuel
  - higher bearing pads further improve CHF compared to Mk IV



CANFLEX Geometry



Evolution of CANDU Fuel



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Other Design Features

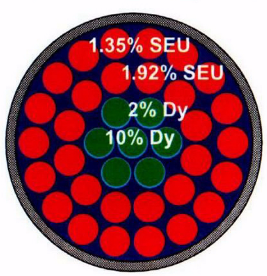
- Optimized pellet design
  - in smaller elements (highest ratings)
    - larger chamfers, deeper dishes, shorter pellets
    - more internal void for accommodating fission gas release
    - reduces inter-pellet clad strain
- Slightly thicker clad
  - to accommodate higher coolant pressures and temperatures

Summary of CANFLEX NU Qualification

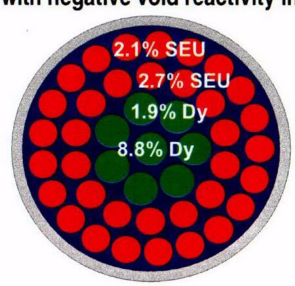
- Design requirements documented in Design Requirements, Design Verification Plan
- Tests and analysis confirmed that CANFLEX met all requirements
  - strength
  - impact and cross-flow
  - fueling machine compatibility, endurance
  - sliding wear
  - fuel performance (NRU irradiations)
  - CHF thermal hydraulic
- Demonstration Irradiation (DI) in Point Lepreau 1998 to 2000
  - 2 channels, 24 bundles
  - irradiation of 24 bundles currently taking place in Wolsong 1
- Design qualification program documented in Fuel Design Manual
- Ready for commercial implementation in CANDU 6 reactor

Experience with Low Void Reactivity Fuel

- ACR fuel is variant of LVRF
- Generic testing done for
  - 37-element LVRF (NU burnup, with negative void reactivity in CANDU 6)
  - CANFLEX LVRF (3x NU burnup, with negative void reactivity in CANDU 6)



37-element LVRF



CANFLEX LVRF

Overview of LVRF Testing

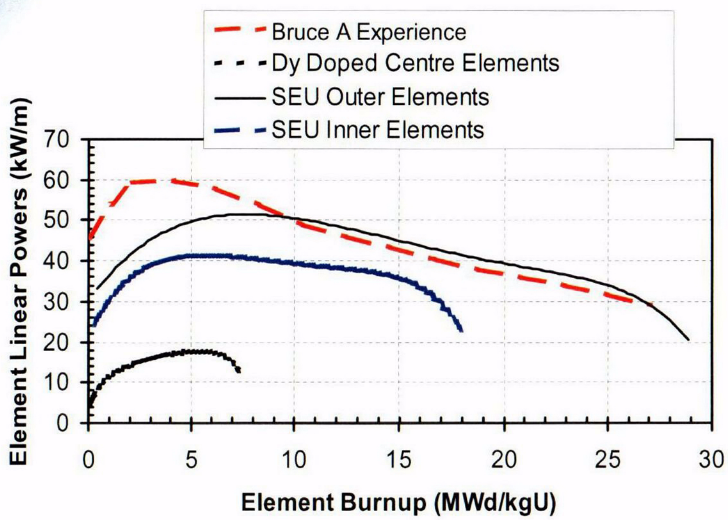
- Dy_2O_3 -UO_2 pellet fabrication
  - measurement of thermal properties
  - corrosion behavior of UO_2
- Bundle fabrication
- Irradiation testing in NRU & PIE
  - Dy-doped demountable elements with Dy levels of 1 to 15%
  - prototype bundles
- Reactor physics
  - ZED-2 measurements
    - void reactivity
    - fine structure
  - WIMS validation
- Thermalhydraulics
  - measurements
  - modeling
- Safety experiments
  - interactions with Zircaloy
  - grain-boundary inventory
- CANFLEX LVRF currently being qualified for Bruce Power implementation
  - enrichment, Dy content tailored to meet station needs
  - synergistic with ACR fuel qualification

Extended Burnup Irradiation Experience

- Power reactor experience
  - >230 37-element bundles achieved burnups > 17 MWd/kg in Bruce A
- Research reactor experience
  - >24 bundle and element irradiations in NRU > 17 MWd/kg
    - 15 irradiations with burnups greater than 21 MWd/kg
  - 10 of 24 irradiations also experienced power ramps
  - several irradiations ongoing
- Qualified irradiated fuel databases
  - 28-element, 37-element and CANFLEX
- Good confidence in ACR fuel performance based on our experience
  - ACR power envelope is below the high power envelope for which we have experience
  - ACR fuel pellet design is optimized for extended burnup, based on our experience base and assessments

NRU Fuel String

ACR Power Envelope vs. Bruce A Experience



ACR Fuel Qualification

- Will ensure ACR has full thermal integrity, structural integrity, and compatibility with interfacing systems
- Comprehensive, integrated set of in-reactor tests, out-reactor tests, and analyses
- Qualified computer facilities, codes, and staff
- US fuel consultants providing guidance

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Approach

- Systematically evaluate impact of all significant operating and damage mechanisms, individually and in combination
- Confirm consequences are within acceptable limits via combination of
  - in-reactor tests
  - out-reactor tests
  - analyses, and
  - engineering judgment
- Envelope all permitted operational and design configurations
- Ensure sufficient margins exist that account for burnup, peak element rating, coolant temperature and flow rate

Summary

- ACR fuel builds on an extensive experience base
  - CANFLEX geometry
  - low void reactivity fuel
  - enriched fuel (extended burnup performance)
- ACR fuel qualification will be facilitated through recent AECL experience in fuel qualification
  - CANFLEX Mk IV fuel with natural uranium
  - current qualification of CANFLEX-LVRF for Bruce Power
- ACR fuel qualification will entail out-reactor tests, in-reactor tests, and analyses
- Numerous background papers on CANDU fuel have been sent to US NRC
- ACR fuel report, summarizing ACR fuel design, experience base, fuel design requirements, and qualification plan will be sent to US NRC shortly

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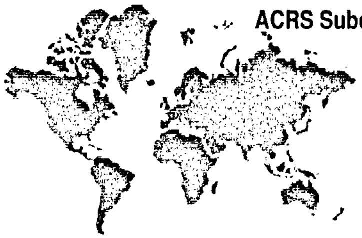
ACR PRA Methodology

Raj Jaitly
Manager, PSA and Safety Design
ACRS Subcommittee on Future Plant Designs
Washington D.C.
January 13, 2004



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Historical Perspective of AECL's PRA Projects

- AECL brings many insights of their long PRA experience to the ACR PRA:
  - SDMs - 1978-1983: CANDU 6 and Ontario Hydro's NPPs
  - CANDU 600 Probabilistic Safety Study - March 1988
  - Wolsong 2/3/4 PRA - March 1995
  - KEPRI- Wolsong 2/3/4 Level 2 PRA Review - 1997
  - Qinshan CANDU Unit 1 and 2 PRA - May 2001
  - Generic Level 2 PRA for internal and external events 2002
  - Pickering A Return to Service PRA Review - 1999
  - Lepreau Refurbishment Project Level 2 PRA - ongoing
  - Preliminary PRAs for CANDU 3 and CANDU 9 (1994, 1997)

ACR PRA Scope

- Level 2 PRA covers:
  - Internal events, fires/floods
  - PRA based Seismic Margin Assessment (SMA)
  - Shutdown state PRA
- PRA Targets:
  - ACR summed severe core damage frequency is less than 1E-05/yr
  - ACR summed large release frequency target is less than 1E-06/yr
  - Seismic margin target of the plant high confidence of low probability of failure (HCLPF) is 0.5g based on a 0.3g Design Basis Earthquake (DBE)

Level 2 PRA Objectives

- Design assist role – confirm adequacy of redundancy, separation of safety systems (design assist PRA for internal events already completed)
- Estimate severe core damage and large release frequency for comparison with international goals
- Provide a basis for risk informed / risk based regulation
- Provide input to optimize test and maintenance programs
- Identify risk-dominant sequences for development of severe accident management guidelines
- Provide a basis for development of a tool in future to support decisions on plant maintenance activities

Main Elements of Level 1 PRA

- Identification of initiating events (internal and external)
- Event tree analysis
- Fault tree analysis
- Common Cause Failure analysis (CCFs)
- Human Reliability Analysis (HRA)
- Accident Sequence Quantification (ASQ)
- Uncertainty and sensitivity analysis
- Recovery analysis

(CAFTA + FORTE Package of PRA Codes being used)

ACR PRA Models

- Initiating Events
  - Systematic plant review for initiating events identification
  - Frequencies based on CANDU or international NPP operating experience
- Event Trees
  - Small to medium size event trees with post-IE operator explicitly modeled
- Fault Trees
  - Reliability data
    - Components failure data based on CANDU experience
    - Human Reliability Analysis based on ASEP (NUREG 4772)
    - Common Cause Failure Data - UPM (partial beta) model

Human Reliability Analysis

- HRA approach is based primarily on ASEP (NUREG-4772)
- Pre Accident
  - Calibration, test, maintenance errors
  - Dependency effects
- Post Accident Errors
  - Errors of diagnosis + execution
- Risk Dominant Sequences – use THERP (NUREG 1278) Handbook

CCF Analysis – UPM Methodology

- Why UPM:
  - CANDU CCF data has not been collected
  - Extent of generic data applicability and availability for CANDU components and configurations is an issue
  - UPM criteria can fulfill a design audit role, providing designers with an indication of best practices and their quantitative impact
  - AECL has applied this methodology on Generic CANDU 6 and CANDU 9 PRAs; it has also been committed for the Point Lepreau Refurbishment PRA

CCF Analysis – Evaluation Criteria

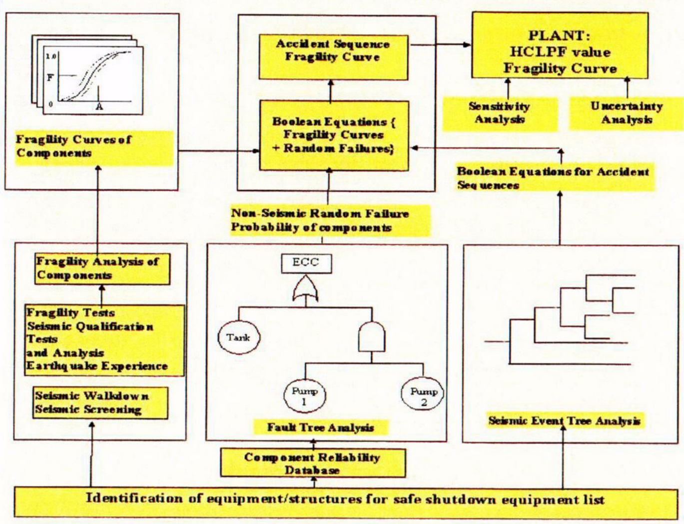
- Eight evaluation criteria:
  - Redundancy and diversity
  - Separation
  - Level of understanding (years of operation, complexity, etc.)
  - Prior analysis of system (fault tree)
  - Man-machine interface
  - Safety culture
  - Control of operating environment
  - Environmental testing

Steps of PRA-Based SMA

- Review internal events PRA model and results
- Select structures / components for seismic capacity analysis
- Perform seismic capacity analysis
- Identify seismically induced initiating events. Develop seismic event trees for these initiating events
- Develop seismic Fault Trees (FTs) (based on internal event FTs)
- Generate minimal cutsets for seismic-induced core damage sequences
- Calculate the HCLPF (High Confidence Low Probability of Failure) value for each seismic core damage sequences

  The plant HCLPF is the lowest sequence HCLPF

PRA Based Seismic Margin Assessment



Fire PRA Approach

- Identify ignition sources: Fire Hazard Analysis for ACR and/or C-6 equipment data base where applicable
- Estimate fire frequency: CANDU fire data base
- Identify PRA-credited equipment: C-6 equipment data base and train/channel based assumption for the cables
- Perform screening analysis to identify potential significant fire areas
- Evaluate fire growth and propagation: COMPBRN IIIe or hand calculation
- Develop fire scenarios including fire detection and suppression probability
- Estimate conditional core damage probability for each fire scenarios
- Estimate Core damage frequency by combining the fire scenario frequency and conditional core damage probability
- Sensitivity analysis and insights for risk management

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Flooding PRA Approach

- Identify flooding sources in each flooding area
- Identify PRA-credited equipment in the areas of concern
- Perform screening analysis to identify Potential significant flooding areas
- Estimate flooding frequencies
- Evaluate flood growth and flood propagation: flood flow rate, floodable volume, flood barrier, etc.
- Develop flood scenarios considering flood protection design features and operator intervention
- Estimate conditional core damage probability for each flood scenarios
- Estimate Core damage frequency by combining the flood scenario frequency and conditional core damage probability
- Sensitivity analysis and insights for risk management

Main Elements of Shutdown State PRA

- Systematically identify low power and planned outage configurations
- In consultations with Operations group, identify/establish maintenance restrictions
- Modify system fault trees to account for system / equipment outage
- Detailed HRA since most mitigation actions need operator action
- Event tree analysis for the postulated events
- Recovery analysis
- Uncertainty and sensitivity analysis

Level 2 PRA

- Studies for severe accident progression and consequential challenges to containment
- Core damage states to be analyzed include:
  - Moderator ultimate heat sink + existing impairment of containment functions
  - Fuel debris (corium) <u>in</u> vessel + containment failure assessment
  - Fuel debris (corium) <u>ex</u> vessel + containment failure assessment
- Analysis to be performed by MAAP4 CANDU that is part of the Industry Standard Toolset
- Containment reliability assessment by containment event tree

Containment Reliability

- The following containment functions (dormant and mission) are modeled:
  - Airlocks
  - Containment isolation
  - Hydrogen control
  - Reactor building cooling

Summary

- The ACR PRA methodology has been established and submitted to the NRC
- The methodology is generally consistent with international PRA practices. Already applied on Generic CANDU PRA as well as currently being applied for the Point Lepreau Refurbishment Project
- PRA scope covers internal events, internal fires and floods, and the shutdown state
- Seismic risk will be evaluated by PRA based Seismic Margin Assessment

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ACR-700 Pre-Application Review

January 13, 2004

ACRS Future Plant Subcommittee Briefing

Belkys Sosa, Project Manager
Office of Nuclear Reactor Regulation

Planned Pre-Application Review Process

- The approach and criteria to be applied in the review of the ACR-700 are in some cases different from those applied to conventional LWRs because of the unique features and design characteristics of the ACR-700

- The review will identify where new staff positions, regulations and regulatory guidance is needed to address the unique characteristics of the design; such as:
  - Pressure tubes and fueling machine as Reactor Coolant System (RCS) components
  - On-power fueling

- In the application of existing regulations and guidelines, the staff may need to interpret the guidance developed for LWRs for application to non-LWR concepts and issues under review
  - The approach is directed toward ensuring an equivalent level of safety as that of current-generation LWRs

Pre-Application Review Scope
Focus Topics (FT)

- **<u>Class 1 pressure boundary design</u>**
- **Design basis accidents and acceptance criteria**
- **<u>Computer codes and validation adequacy</u>**
- **Severe accident definition and adequacy of supporting R&D**
- **Design philosophy and safety-related systems**
- **Canadian design codes and standards**
- **Distributed control systems and safety critical software**
- **<u>On-power fueling</u>**
- **<u>Confirmation of negative void reactivity</u>**
- **Preparation for Standard Design Certification Docketing**
- **ACR PRA Methodology**
- **ACR Technology Base**
- **<u>Fuel design</u>**

Pre-Application Review Status

- Phase 1 completed - July 31, 2003
  - Staff participated in a series of familiarization meetings and tours of AECL facilities designed to provide a general overview of the ACR design
- Phase 2 on-going – September 30, 2004
  - Detail Meetings on each key focus topic to discuss technical issues
  - PIRT Panels on Thermal hydraulics, Severe Accidents, and Neutronics
  - Staff review of technical information provided by AECL
  - Staff is requesting clarification and additional information to resolve issues
- Schedule for Safety Assessment Report (SAR)
  - NRC Issue SAR: September 2004
  - Staff plans to forward draft SAR to ACRS in July 2004 to support September 2004 ACRS Full Committee meeting

ACR Pre-Application

Phase 2 - Discussions

- Pressure tubes and fueling machine as Reactor Coolant System (RCS) components
  - Design codes and standards
- Definition of design basis accidents and acceptance criteria
- Severe accidents definition for ACR
- Safety analysis computer codes
- On-power fueling
- PRA Methodology
- Quality Assurance (QA)

ACR Pre-Application Review Product

Safety Assessment Report (SAR)

- **Review Scope**
  - Discuss what was reviewed and what guidance it was reviewed against, to the extent that the guidance exists.
- **Technical Issues**
  - Discuss technical issues identified that will require further data, tests, inspections, analyses, or codes.
- **Regulatory Issues**
  - Discuss regulatory issues, such as rules, rulemaking, or exemptions that will need to be resolved.
- **Policy Issues**
  - Discuss policy issues that will need upper management or Commission guidance for resolution.
- **Conclusion**
  - Discuss the feasibility of successfully completing the review.
- **Schedule and Resources**
  - Provide an estimate of the resources required and schedule for completing the review of the specific focus topic area.

NRC STAFF REVIEW OF CLASS 1 PRESSURE BOUNDARY DESIGN (PBD)
AECL Focus Topic #1

January 13, 2004

ACRS Future Plant Subcommittee Briefing

Edmund Sullivan
NRR/DE/EMCB

Pre-Application Review

- Review of Class 1 PBD being performed by Materials and Chemical Engineering and Mechanical and Civil Engineering Branches in NRR
- With assistance from the Materials Engineering Branch in RES

Documents submitted for review include

- Technology of CANDU Fuel Channels - AECL
- Procedures for In-Service Evaluation of Zirconium Alloy Pressure tubes in CANDU Reactors - CANDU Owners Group
- Canadian Standards Association (CSA) standards applicable to CANDU nuclear components
- Published Technical Papers on Fuel Channel Behavior
- Technology of On Power Fueling

Methodology for pre-application review

- Acquire familiarity with ACR-700 design - Phase 1 of pre-application review
- Develop understanding of differences between ACR-700 and plants already operating or reviewed
- Identify where there are existing regulations that may not be met by the ACR-700
- Identify where new regulations may be needed to ensure adequate protection provided by the ACR-700 design

Pre-application Review

- Scope primarily fuel channel design. Will extend into other areas of Class 1 PB, as resources and available information permit

- Thrust of focus topic review is to identify significant challenges to reviewing actual application

- Approach is to identify concerns not to try to resolve issues

- Technical interactions planned with the Canadian Nuclear Safety Commission

Pre-application Review

Review to the depth necessary to identify

- documentation needed by staff to complete pre-application review,
- regulatory requirements that may not be satisfied by ACR-700,
- need for new regulatory requirements,
- safety issues or technical approaches that the staff may have difficulty finding acceptable, and policy issues.

Safety Assessment for Pre-application Review

- Review scope,
- Safety/technical issues,
- Regulatory issues,
- Policy issues,
- Conclusions regarding feasibility of successfully completing review, and
- Schedule and resource estimate required for completing review of focus topic

Potential Issues

- Basis for fatigue design curves,
- Basis for governing creep equations,
- Sagging of pressure tubes and hydride blister formation,
- Effect of large number of bent pipes >> erosion corrosion, SCC,
- Effect of irradiation damage, aging and embrittlement,
- Effect of dissimilar metal contacts in typical ACR-700 environment,
- Design of rolled joints,
- Canadian design and inspection codes,
- Code classification of components,

Potential Issues (Cont'd)

- Inspectability of components,
- Scope, methods and frequency of inspection,
- Testability of components,
- Scope, methods and frequency of testing,
- Leak-before-break approach and adequacy of leak detection capability,
- On power fueling as an extension of the Class 1 pressure boundary,
- Design of transport mechanisms in Class 1 component support structure, and
- Component material behavior under severe accident conditions.

NRC STAFF REVIEW OF ACR-700 PIRT

January 13, 2004
ACRS Future Plant Subcommittee Briefing

Jack Rosenthal, Chief
Safety Margins and Systems Analysis Branch
Office of Nuclear Regulatory Research

ACR-700 PIRT

- Objective: Develop initial PIRTs for neutronics, severe accidents and thermal hydraulics
- Purpose: Guide requirements for code modeling and help determine experimental data requirements
- This is a research program to develop infrastructure to support the forthcoming design certification effort

Panel Members

Neutronics	Thermal Hydraulics	Severe Accidents
David Diamond	Samim Anghaie	Michael Corridini
Thomas Downar	Sanjoy Banerjee	Robert Henry
Ron Ellis	Peter Griffith	Salomon Levy
Farzad Rahnema	Yassin Hassan	Dana Powers
Paul Turinsky	Pradip Saha	Karen Vierow
	Novak Zuber	

PIRT Operations

- BNL is the contractor with support from Brent Boyack and Gary Wilson
- CNSC also participating in the PIRT effort
- PIRT process benefiting by extensive support by AECL
- AECL has provided large number of documents, as well as presentations on ACR-700 design, and staff support to answer questions

Thermal Hydraulics

- Specified scenario is a "critical break," defined as the break size leading to early flow stagnation in the core
- This break is ~25% located the a feed header
- Figure of merit is fuel time-temperature history
- Event is divided into two phases: blowdown and reflood
- Plant is decomposed into: systems/components
- Each component is ranked in importance
- Phenomena within each component are identified and ranked by: importance and by state of knowledge, using a scale of high, medium, or low

Neutronics PIRT

- Specified scenario is the large break of an inlet or outlet header, voiding all fuel channels within 1 to 3 seconds
- Figure of merit is Coolant Void Reactivity: CVR = k(voided) – k(cooled)
- Initial PIRT considers only the equilibrium core because initial and transitional cores have yet to be designed
- PIRT tables are organized according to the three main elements of CVR calculation (operating conditions, lattice physics, core simulation) to address fundamental physics as well as safety analysis methods
- "Phenomena" (i.e., parameters and models as well as nuclear reactions, etc) are identified and ranked by importance on a scale of High, Medium, or Low
- The knowledge level of each phenomenon's impact on CVR is assessed as Known, Partially known, or Unknown

Severe Accidents

- Specified scenarios are: 1) single channel event, either critical break in a single feeder pipe or a flow blockage; or 2) whole core event initiated by LOCA or station blackout
- Figure of merit for single channel is potential for damage progression to lower neighboring channels
- Figure of merit for whole core event is debris coolability and containment integrity

Status

- Effort began in September 2003
- Two PIRT meeting held, on October 30-31 and on December 11-12
- Third PIRT meetings to be held in January, 2004 for neutronics and in February, 2004 for thermal hydraulics and severe accidents
- PIRT report due in May 2004

NRC STAFF REVIEW OF COMPUTER CODES AND VALIDATION ADEQUACY
AECL Focus Topic #3

January 13, 2004

ACRS Future Plant Subcommittee Briefing

Walton Jensen
NRR/DSSA/SRXB

Review Objectives

AECL Desired Outcome - The NRC staff accepts the computer codes used in the ACR safety analysis and the adequacy of their validation as sufficient for the purpose of providing a safety analysis for the ACR in the US.

NRC Staff Objectives -

- Scoping review to determine code strengths and weaknesses including areas where additional work or experimental verification is needed.
- Identify additional information requirements.
- Identify any "show stoppers" that would prevent the codes from being used for ACR-700 safety analysis.
- Identify any regulatory or policy issues that will need to be resolved.
- Develop independent capability to audit ACR-700 safety analyses when they are submitted with the DCD.

Scope of Staff Review
ACR-700 Thermal-Hydraulic

Regulatory Standards

- Draft Regulatory Guide DG-1120 "Transient and Accident Analysis Methods"
- Draft Standard Review Plan Section 15.0.2 "Review of Analytical Computer Codes"

Documents to be reviewed

- CATHENA Theoretical Manual
- CATHENA Thermal/Hydraulic Validation Manual
- CATHENA Fuel and Fuel Channel Thermal/Mechanical Validation Manual
- ACR-700 CATHENA Input and Calculational Notes

Resources
ACR-700 Thermal-Hydraulic

- Technical manuals and presentations by AECL

- CATHENA code with preliminary ACR-700 input operational at NRC
  - Currently evaluating a large break in an inlet header

- Preliminary RELAP5 model for comparison with CATHENA results
  - almost complete

- Insights from the PIRT panels
  - phenomena which are important but may be difficult to model

- Input from RES on the adequacy of AECL experimental facilities for code validation
  - Spring 2004

ACR-700 Neutronics Scope

Anthony Attard (NRR/DSSA/SRXB)

Regulatory Standards

- Draft Regulatory Guide DG-1120 "Transient and Accident Analysis Methods"
- Draft Standard Review Plan Section 15.0.2 "Review of Analytical Computer Codes"

Documents to be reviewed

- Code theory manuals for:
  - RFSP - neutron diffusion code for 3D power distribution and burnup
  - WIMS - 2D lattice physics code to generate fuel neutron cross sections for RFSP
  - DRAGON - 3D lattice code to generate cross section data of control devices for RFSP
- Neutronics code validation manuals and data for: RFSP; WIMS and DRAGON
- Neutronics code user manuals for: RFSP; WIMS and DRAGON
- Neutronics code Input and Calculational Notes

Resources
ACR-700 Neutronics

- Technical manuals and presentations by AECL
- ACR-700 Neutronics codes operational at NRC for sensitivity evaluations
- Contractor assistance in place to review theory of codes and available data base
  - Brookhaven
- Insights from the PIRT panels
  - Significant phenomena which may not be modeled correctly
- Input from RES on the adequacy of AECL experimental facilities for code confirmation and validation
  - Spring 2004

Schedule for Thermal/Hydraulics and Neutronics

- RAIs to AECL by March 31, 2004
- Safety Assessment Report July 31, 2004

CONFIRMATORY ANALYSIS OF ACR-700 COOLANT VOID REACTIVITY (AECL Focus Topic #9)

January 13, 2004

ACRS Future Plant Subcommittee Briefing

Donald E. Carlson

RES/DSARE/REAHFB

NRC Confirmatory Analysis of ACR-700 Coolant Void Reactivity

<u>AECL Focus Topic #9</u>
Confirmation of Negative Void Reactivity

AECL Desired Outcome: Staff confirmation that the Coolant Void Reactivity (CVR) is negative over range of operating conditions

- Void reactivity is key to evaluating the design in relation to GDC-11, Reactor Inherent Protection

- Void reactivity effects can significantly impact the progression of analyzed transients and accidents

Confirmatory Analysis of CVR Key Observations

- AECL's nominal value of CVR is:
  - only slightly negative (e.g., k(v) – k(c) = -0.007 = -7 mk)
  - a combination of positive and negative nuclear effects
  - sensitive to core design and operating parameters

- Evaluation of bias and uncertainty in the calculated CVR predictions (i.e., validation) will figure prominently in staff conclusions

  - In-reactor measurements of CVR are difficult and not planned by AECL

  - Validation of computed CVR predictions will be based on ACR-specific benchmark measurements in AECL's ZED-2 critical facility

  - Validation question: When code calculations predict a small negative CVR, how confident are we that the actual CVR will indeed be negative in view of prediction bias and uncertainty?

Confirmatory Analysis of CVR Ongoing and Planned Activities (1 of 3)

Significant Result from Phase 1 Pre-application Activities:

In June 2003, AECL changed the fuel design to make CVR more negative

Pre-application interactions on CVR:

- Technical exchanges on CVR analysis and validation, including facility tours of ZED-2
- First RAIs submitted in March 03; AECL responses and supporting documents provided in June and Nov 2003
- Status report on RES in-house CVR analysis activities provided in Sep 2003
- NRC PIRT activities started in Sep 2003, including presentations by AECL and participation by CNSC staff
- CVR is initial focus of Neutronics PIRT to be completed in March 2004

Confirmatory Analysis of CVR Ongoing and Planned Activities (2 of 3)

Completing Phase 2 Pre-application Activities on CVR:

- RES to provide input on status, initial results, and plans for CVR confirmatory analysis (Focus Topic #9) - in May 2004

- RES to provide initial report on related PIRT results – in April 2004

- RES to provide related input on status, initial insights, and plans for assessing neutronics validation data for CVR, etc (also part of Focus Topic #3) - in May 2004

- RES to provide related input on estimated resources and schedules for CVR confirmatory analysis and validation, including related work to establish core models (PARCS code) for audit analysis of ACR-700 transients and accidents – in June 2004

Confirmatory Analysis of CVR Ongoing and Planned Activities (3 of 3)

CVR Confirmatory Analysis and Related Work for Design Certification Phase:

- **Independent static calculations of nominal CVR values using detailed models with existing state-of-the-art methods (MCNP)**
  - MCNP modeling and analysis with RES in-house cross-checking against MONK
  - MCNP analysis will reflect and supplement phenomenology insights from PIRT
  - Detailed MCNP modeling studies will help qualify the more approximate models and methods to be used by NRC nuclear code suite for reactor transient analysis (SCALE+PARCS)

- **Validation benchmark analysis to evaluate CVR bias and uncertainty**
  - Adapt and apply sensitivity and uncertainty analysis methods to (a) assess applicability and coverage of semi-prototypic ZED-2 benchmarks and (b) derive CVR bias and uncertainty
  - Review and assess ZED-2 measurement techniques for ACR
  - Identify potential needs for additional integral and/or differential data – early emphasis

- **Provide SCALE lattice data and PARCS core models for simulating ACR-700 operations and transients**
  - Adapt and apply SCALE to model ACR-700 fuel lattice and transverse reactivity devices
  - Adapt and apply PARCS to model ACR-700 core with lattice data from SCALE
  - Integrate and test SCALE data and PARCS models and coupling with TRACE T/H
  - Analyze impacts of CVR variations on ACR-700 reactivity transients

NRC STAFF REVIEW OF ON-POWER REFUELING
AECL Focus Topic #8

January 13, 2004

ACRS Future Plant Subcommittee Briefing

Steven Jones

NRR/DSSA/SPLB

OBJECTIVE

On-Power Refueling Not Previously Licensed in the U.S.

- Establish Feasibility of Design Certification
  - **Regulatory Issues - Possible Exemptions from Existing Regulations or Rulemaking**
  - **Policy Issues - New Criteria for Evaluation of Design and New Classes of Design-Basis Events**
  - **Technical Issues - New Methods of Review or Analysis**

- Develop Regulatory and Policy Framework to Support Design Certification

REGULATORY ISSUES

Comparison of Basic Design Against Regulations

- Review of 10 CFR Parts 50 and 52
- Identify Applicable Regulations – For Example:
  - 10 CFR 50.68 Criticality accident requirements
  - 10CFR 50.55a Codes and standards (Division of Engineering)
- Identify Need for Exemptions or Rulemaking to Support Design Certification

POLICY ISSUES

Comparison of Basic Design Against General Design Criteria

- Review of General Design Criteria and Proposed ACR-700 Design Criteria
- Select Applicable Design Criteria for Functional Capability
  - Criticality Prevention
  - Fuel Cooling/Residual Heat Removal
  - Mechanical Handling of Fuel
  - Instrumentation
  - Emergency Cooling
  - Containment
- Identify Policy Issues Involving New Design Criteria or Different Application of Existing Criteria

POLICY ISSUES

Evaluation of Proposed Design Basis Events

- Review of Proposed Design Basis Events Against CANDU Reactor Fuel Handling Operating Experience and Failure Mode Analysis of Basic Design
- Establish Scope of Credible Design Basis Events and Acceptance Criteria for Fuel Handling Accident Analyses
- Identify Policy Issues Involving New Design Basis Events or Different Acceptance Criteria

TECHNICAL ISSUES

Evaluation of Methods of Review and Analysis

- Review of Proposed Methods of Review and Analysis Against Existing NRC Regulatory Guidance for Similar Events
- Identify Technical Issues Involving Different Methods of Review or Analysis for Resolution

NRC STAFF REVIEW OF ON-POWER FUELING
AECL Focus Topic #8

January 13, 2004

ACRS Future Plant Subcommittee Briefing

Patrick Sekerak
NRR/DE/EMEB

AECL Report, "The Technology of On-Power Fueling"

- What it provides
- What it does not provide

CONTENT FOR DESIGN CERTIFICATION (10 CFR 52.47)

- Quality Group Classification of Systems and Components
- Dynamic Analysis & Testing Methods
- Service Loading Combinations
- Service Stress Limits
- Design Transients
- Special Analytical Methods
- Experimental Stress Analysis
- Computer Codes Used
- ITAAC

POLICY ISSUE

- Acceptance Criteria
- CSA Standards as Proposed Alternatives to 10 CFR 50.55a
- Reconciliation of CSA Standards with ASME III, XI, and O/M Codes

NRC STAFF REVIEW OF PROBABILISTIC SAFETY ASSESSMENT
AECL Focus Topic #11

January 13, 2004

ACRS Future Plant Subcommittee Briefing

Martin Stutzke
NRR/DSSA/SPSB

Presentation Outline

- Describe the plan for conducting the pre-application review of the ACR-700 PSA
- Review objectives
- Review guidance
- Review assignments and schedule
- Describe a potential policy issue involving the risk acceptance guideline for core-damage frequency

Review Objectives (1 of 2)

- Determine if the AECL PSA methodology will produce a PSA with adequate scope, level of detail, and technical acceptability to satisfy regulatory needs
- Identify potential issues
- Technical
- Regulatory
- Policy

Review Objectives (2 of 2)

- Develop a schedule and resource estimate for reviewing the PSA submitted with the standard design certification application
- Learn about the ACR-700 design
- Plant layout, construction, systems, etc.
- Accident phenomenology and progression

Review Guidance (1 of 3)

General:

- 51 FR 24643, July 8, 1986, NRC Policy Statement on Regulation of Advanced Nuclear Power Plants
- NUREG-1226, May 1988, Development and Utilization of the NRC Policy Statement on the Regulation of Advanced Nuclear Power Plants
- 10 CFR Part 52.47(a)(v)

Risk Acceptance Guidelines:

- SECY-90-16, June 26, 1990, Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationships to Current Regulatory Requirements
- SECY-93-087, July 21, 1993, Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (LWR) Designs

Review Guidance (2 of 3)

PRA Quality:

- Regulatory Guide 1.174, An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis
- Standard Review Plan, Chapter 19, Use of Probabilistic RISK Assessment in Plant-Specific, Risk-Informed Decision making: General Guidance
- Standard Review Plan, Chapter 19.1, Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities

Review Guidance (3 of 3)

PRA Quality (continued):

- ASME RA-S-2002, Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications
- Regulatory Guide 1.200 (for Trial Use), An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities
- ANSI/ANS-58.21-2003, External-Events PRA Methodology

PRA Methodology

- NUREG-1335, IPE Submittal Guidance
- NUREG-1407, IPEEE Submittal Guidance
- NUREG/CR-3485, PRA Review Manual

Assignments and Schedule (1 of 2)

Who	When	Done	What
SPSB	12/23/03	✔	Issue RAI concerning PSA quality
SPSB	12/29/03	✔	Issue advice on PSA quality expectations
SPSB	12/31/03	✔	Compile review references and matrix
SPSB	1/9/04	✔	Issue RAI concerning PSA methodology
PRAB	3/5/04		Issue draft report on review of: • 91-03660-AR-001, Generic CANDU PSA Methodology • 91-03660-AR-002, Generic CANDU PSA Analysis • 108-03660-AB-001, ACR PSA Methodology

Assignments and Schedule (2 of 2)

Who	When	Done	What
PRAB	4/16/04		Issue final report on review of: • 91-03660-AR-001, Generic CANDU PSA Methodology • 91-03660-AR-002, Generic CANDU PSA Analysis • 108-03660-AB-001, ACR PSA Methodology
SPSB	5/14/04		Complete review of: • 108-03660-AB-001, ACR PSA Methodology • 108-03660-AB-003, Phenomenology for Limited and Severe Core Damage Accidents in the ACR
SPSB	5/28/04		Complete schedule and resource estimate
SPSB	6/25/04		Issue Focus Topic #11 deliverable

Potential Policy Issue (1 of 4)

- The SRM on SECY-90-16 specifies a core-damage frequency goal of 1E-4/year for evolutionary and advanced reactor designs
- For the ACR-700, AECL has defined two types of core-damage accidents:
  - Limited core damage accidents:
    - Accident progression is arrested within the fuel channels
    - No equivalent in LWRs
  - Severe core damage accidents:
    - Corium is formed, which may change its geometry, location, composition and state during an accident
    - Similar phenomenology to severe accidents in LWRs, although the accident progression is different

Potential Policy Issue (2 of 4)

- AECL has defined 10 plant damage states (PDS) that, with one exception, map to either the limited or severe core-damage categories
- The exception, PDS9 pertaining to tritium releases, does not involve any fuel damage
- The staff will ask AECL to determine the frequency of each PDS, including uncertainties

Potential Policy Issue (3 of 4)

- Question #1: How should the staff interpret the core-damage frequency risk acceptance guideline specified in the SRM on SECY-90-16 with respect to the ACR-700?
  - If the guideline applies only to the severe core-damage frequency, should a guideline pertaining to limited core- damage frequency be developed?
  - If the guideline applies to the total (limited and severe) core- damage frequency, should a guideline that limits the severe core-damage frequency to a certain percentage of the total core-damage frequency be developed?

Potential Policy Issue (4 of 4)

- Question #2: Should a guideline pertaining to the frequency of accidents that potentially involve a release but no fuel damage (e.g., tritium release - PDS9) be developed?

ACR-700 Pre-Application Review

January 13, 2004

ACRS Future Plant Subcommittee Briefing

James Kim
NRR/RNRP

ACR-700 Pre-Application Schedule

Phase 1	June 2002 – July 2003
ACR Submittals	December 2002 – March 2004
Requests for Additional Information	May 2003 – March 2004
Phase 2	August 2003 – September 2004
ACRS Information Briefing	January 2004
AECL RAI Responses	June 2003 – April 2004
ACRS Subcommittee Meetings	April - June 2004
Draft SAR to ACRS	July 2004
ACRS Full Committee Meeting	September 2004