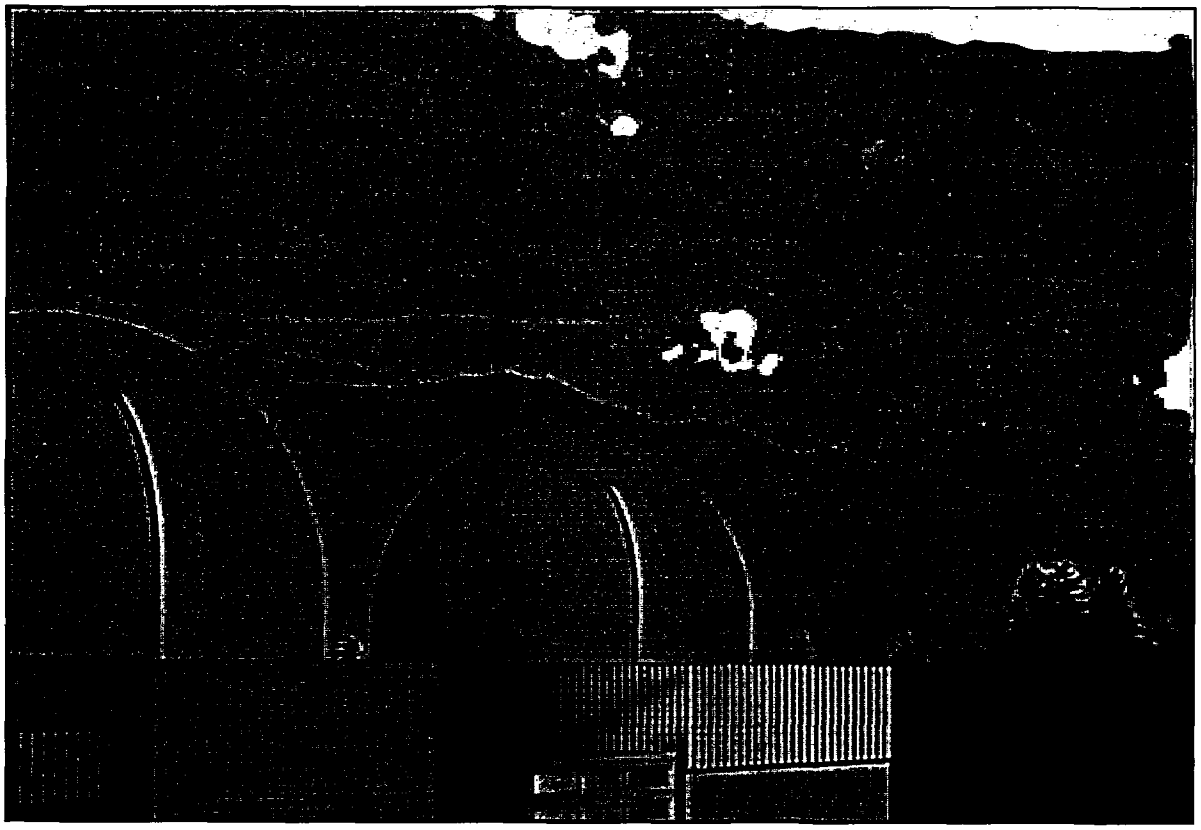
Advanced CANDU Reactor (ACR-700) Design



Meeting with the ACRS
Washington DC
January 13, 2004

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Advanced CANDU Reactor (ACR-700) Design
Pre-Application Review
ACRS Meeting Agenda

January 13, 2004:

8:30 a.m.	Introductory Comments - NRC
8:45 a.m.	Introductory Remarks - AECL
9:00 a.m.	ACR Design Overview - AECL
10:00 a.m.	ACR Pre-Application Scope, Rationale, and Expectations - AECL
10:30 a.m.	Break
10:45 a.m.	Brief Overview of Key and Selected Focus Topics - AECL
10:50 a.m.	Class 1 Pressure Boundary Design - AECL
11:10 a.m.	Computer Codes and Validation Adequacy - AECL
11:20 a.m.	On-Power Fueling - AECL
11:40 p.m.	Confirmation of Negative Void Reactivity - AECL
12:00 p.m.	Lunch
1:00 p.m.	ACR Fuel Design - AECL
1:20 p.m	PRA Methodology - AECL
1:40 p.m.	Break
2:00 p.m.	Planned Pre-Application Review Process of Key Focus Topics - NRC - Class 1 Pressure Boundary Design - NRC - Computer Code Validation: Thermalhydraulics - NRC Physics Code - NRC - On-Power Fueling - NRC - Confirmation of Negative Void Reactivity - NRC - PRA Methodology - NRC
4:30 p.m.	ACR Pre-Application Review Schedule - NRC
4:45 p.m.	Opportunity for Public Comments
5:00 p.m.	Adjourn

ACR Design Overview

Stephen Yu
Program Manager, ACR Product Development
ACRS Subcommittee on Future Plant Designs
Washington D.C.
January 13, 2004

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Outline

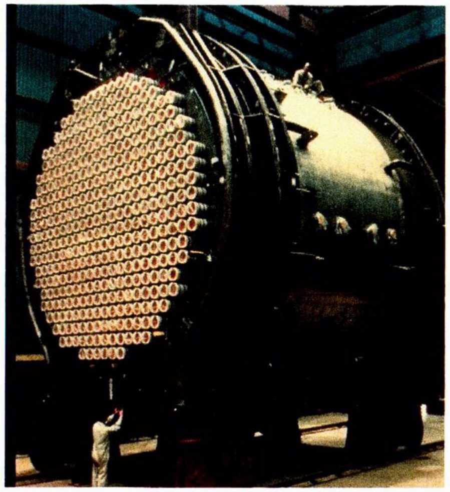
- Introduction
- ACR Design Features
  - Fuel
  - Fuel channels
  - Core design
  - Reactor Coolant System
  - On-power Fueling
- Engineered Safety Features
  - Shutdown System 1
  - Shutdown System 2
  - ECC
  - Containment
- Severe Accident Resistance and Mitigation
- Operational Features

- Introduction
- ACR Design Features
- Engineered Safety Features
- Severe Accident Resistance and Mitigation
- Operational Features

ACR

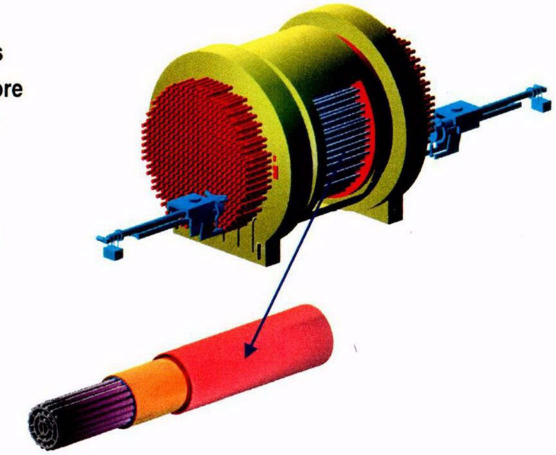
- The ACR is an evolutionary extension of the CANDU 6 plant, which has ten units in operation on four continents, and one unit currently under construction

CANDU 6 Reactor



CANDU Intrinsic Features

- Channel reactor
  - Horizontal channels
  - Pressure tube as core pressure boundary
  - Water cooled
  - Water moderated
- Separate coolant and moderator
- Short fuel bundles replaceable on-line



ACR Optimizes the Channel Concept

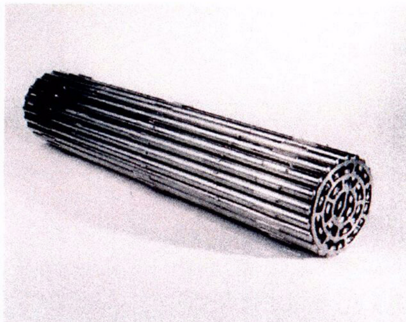
- Current operating CANDU reactors
  - Natural Uranium (NU) fuel
  - Heavy water (D_2O) coolant
  - Heavy water (D_2O) moderator
- ACR – relax constraint of Natural Uranium Fuel and –
  - Use Slightly Enriched Uranium (SEU) fuel
  - Use light water coolant
  - Reduce core size and reduce amount of heavy water moderator
    - Increase reactor coolant system (RCS) pressure
    - Increase thermal efficiency
- Retain intrinsic proven CANDU features

- Introduction
- **ACR Design Features**
- Engineered Safety Features
- Severe Accident Resistance and Mitigation
- Operations

C01

Fuel

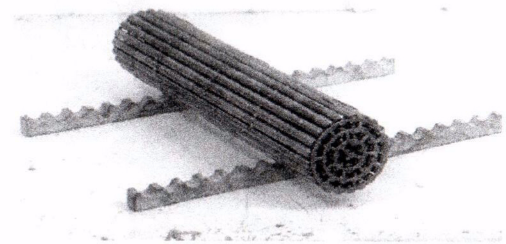
- 0.5m (1.6 foot) long CANFLEX fuel bundle
- On-power refueling
- 43 fuel rods in each bundle
  - 2.1 wt% ^{235}U SEU in 42 rods, in the form of UO_2 pellets
  - NU + 7.5% dysprosium in central rod
- Fuel burn-up 21,000 MWd/MT (U)
  - Higher than NU CANDU average
  - Modest vs. LWRs
- Higher bundle power, lower rod rating than current CANDU



Fuel

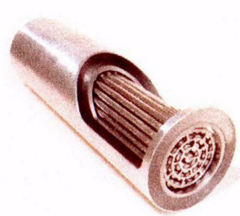
- Short fuel bundles limit radioactivity release from defects or in accidents
  - Defected fuel removable during operation
- Collapsible clad – good fuel/clad heat transfer
- Low internal pressure – small amount of strain relieves internal pressure and reduces likelihood of flow blockage

CANFLEX-NU fuel bundle removed from Point Lepreau



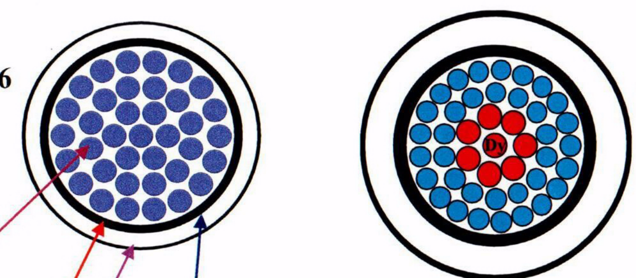
ACR Fuel Channel Design

- Pressure Tube
  - Thicker to reduce stresses during normal operation
  - Chemistry specification optimized to reduce deformation and corrosion
  - Only the outlet end of the pressure tube experiences temperatures greater than in CANDU 6
- Calandria Tube
  - Zr-4 has materials properties equivalent to Zr-2
  - Thicker tube to withstand spontaneous pressure tube failure



Fuel Channel (End View)

CANDU 6



ACR

- 37-rod NU fuel
- Zr – 2.5% Nb pressure tube
- Zr-2 calandria tube
- Insulating gap between pressure tube and calandria tube

- 43-rod SEU fuel
- Thicker Zr – 2.5% Nb pressure tube
- Stronger Zr-4 calandria tube
- Larger gap between pressure tube and calandria tube

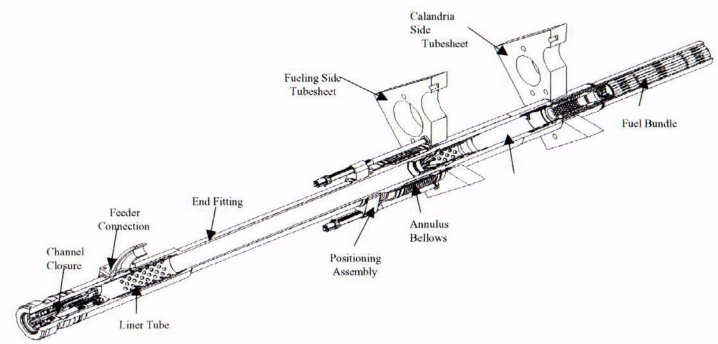
C02

ACR Fuel Channel

Design Feature	ACR	CANDU 6
Pressure Tube	Zr-2.5%Nb	Zr-2.5%Nb
Thickness	6.5 mm	4.2 mm
Fast flux	same	same
Temperature	279-325°C	266-311°C
Pressure	13.2-12 MPa	11.2-10MPa
Calandria Tube	Zr-4	Zr-2
Thickness	2.5 mm	1.4 mm
O.D.	156 mm	132 mm

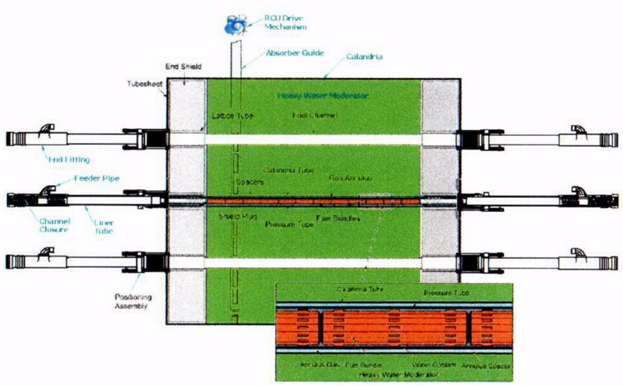
Fuel Channels

- Pressure tube defects will leak allowing time for operator action before rupture. Leakage into annulus is detectable.
- Pressure tube failure contained within calandria tube
- Channel failures will not propagate to other channels, will not fail calandria, nor incapacitate shutdown systems



ACR Reactor

- Horizontal fuel channels surrounded by low temperature, low pressure moderator
- Steel calandria contains moderator and supports fuel channels
- All reactivity devices in moderator
- Shielding water (H_2O) surrounds calandria for thermal and biological shielding



Core Design

- Long prompt neutron lifetime (0.33 ms.) relative to LWRs
  - Due to D_2O moderation (neutron economy)
- Low reactivity hold-up in the control system
  - Due to on-power refueling for long-term reactivity control
  - Typically total of ~9 mk in movable control devices
  - Additional –12mk in Control Absorbers (normally out of core)
  - No need for boron in the coolant to hold down reactivity
  - Limits extent of reactivity accident
- Control rod ejection physically impossible
  - Reactivity control mechanisms penetrate the low-pressure moderator, not the coolant pressure boundary; do not interact with the fuel

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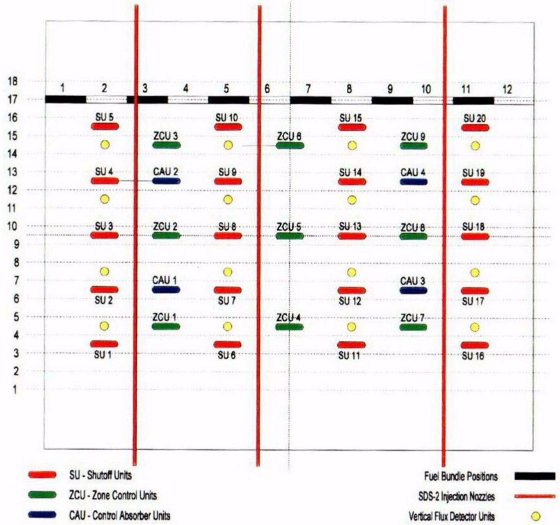
Core Design – cont'd

- Low reactivity worth of a reactivity control device
- Short-term and spatial reactivity control by zone controllers in 9 assemblies
- Each controller assembly worth ~1 mk



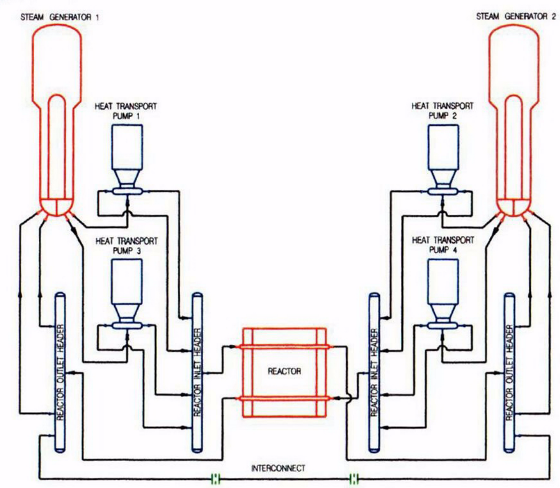
Safety and Control Parameters

Total Delayed Neutron Fraction (ß)	0.0056
Prompt Neutron Lifetime (millisecond)	0.33
Bulk and Spatial Control	Zone Controllers in 9 Assemblies No Adjuster Rods
Fast Power Reduction	4 Control Absorber Units
Shutdown System (SDS1)	20 Shutoff Units
Shutdown System (SDS2)	6 Injection Nozzles (reflector region)

Core Design – cont'd

- Coolant void reactivity coefficient
  - Use of SEU fuel allows flexibility in choice of CVR
  - Chosen small to increase operating margins related to safety
- Total coolant void reactivity is about –7 mk
- Gives negative power coefficient and more stable control
- Inherent reactivity decrease on LOCA, loss of electrical power
- Very small reactivity increase after main steam line break (due to limited boiling in normal operation to about 2% outlet quality)
- Reactivity is reduced should a single channel (pressure tube + calandria tube) fail resulting in H_2O mixing with heavy water moderator

Reactor Coolant System – cont'd

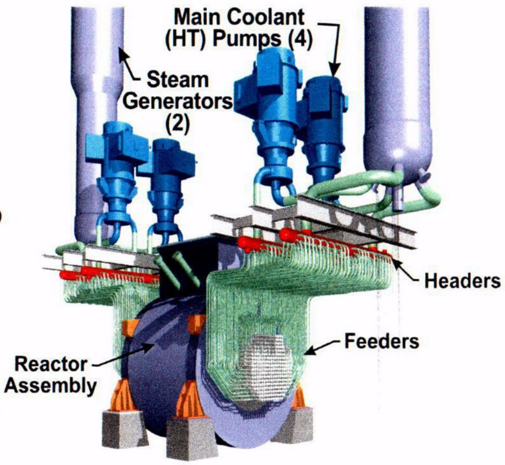


Note "figure-eight" layout

C04

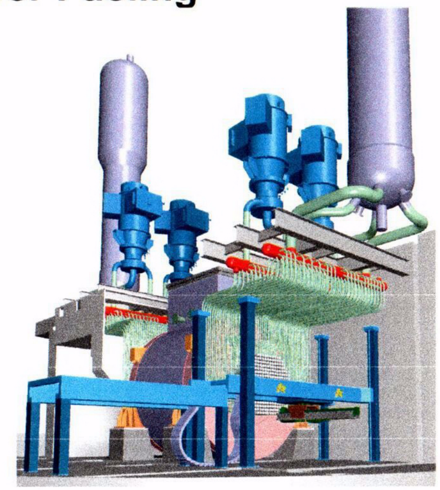
Reactor Coolant System

- Each channel is connected at its inlet and outlet by small diameter (<3.5 inch) feeder pipes to headers above reactor
- No large pipes at or below core level
- Parallel / series pumps – pump seizure mitigation
- Heat sinks above the core - natural circulation, even with some void
  - No preferred flow direction in the channels in the long term

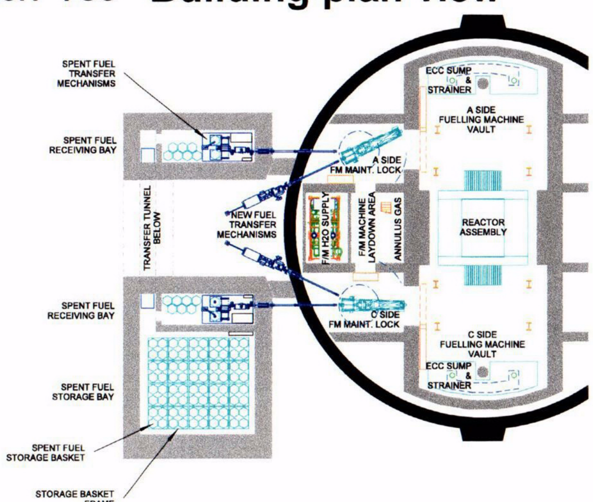


On-Power Fueling

- Each fuel channel contains 12 fuel bundles
- 2 bundles of irradiated fuel are removed and 2 bundles of fresh fuel are inserted using two fueling machines connected to each end of a channel
- The fueling machine has a movable Class 1 pressure vessel that connects to the new and irradiated fuel ports and fuel channels in sequence to move the fuel



ACR 700 Building plan view



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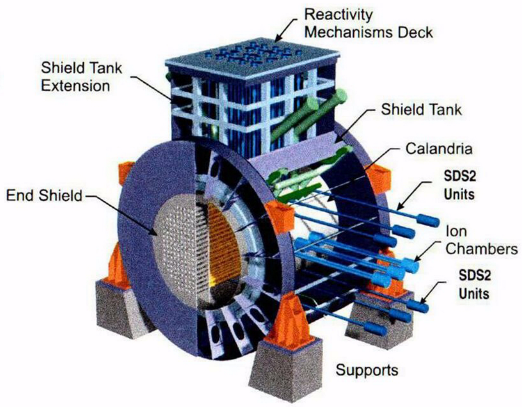
- Operations

C05

Shutdown Systems

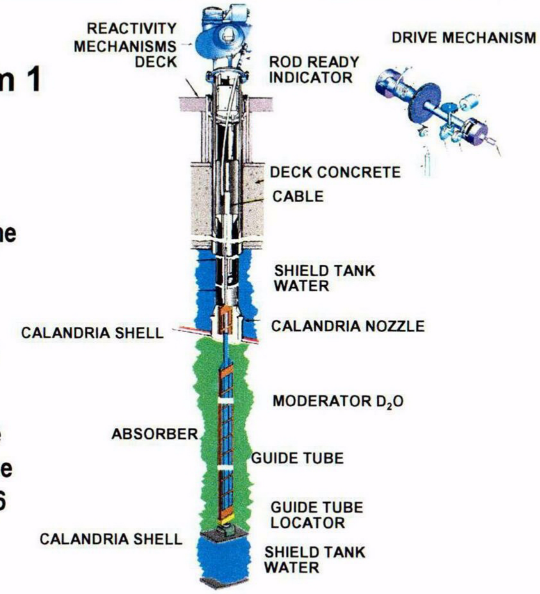
Two fully independent passive shutdown systems:

- SDS1 – rods drop by gravity into moderator
- SDS2 – liquid absorber injected into reflector / moderator by pressurized gas

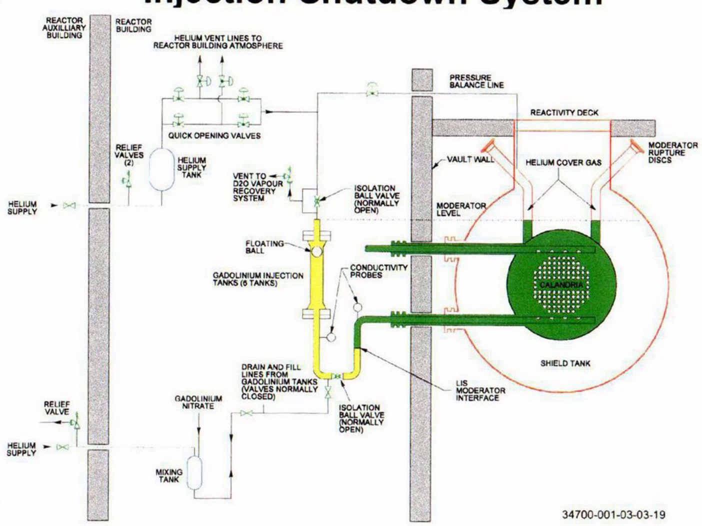


Shutdown System 1 – Mechanical Shutoff Units

- Drive mechanism same as CANDU 6 design
- Passive shutoff unit insertion into core by gravity
- Smaller ACR-700 core achieves approx. same drop time as CANDU 6



Shutdown System 2 – Liquid Injection Shutdown System



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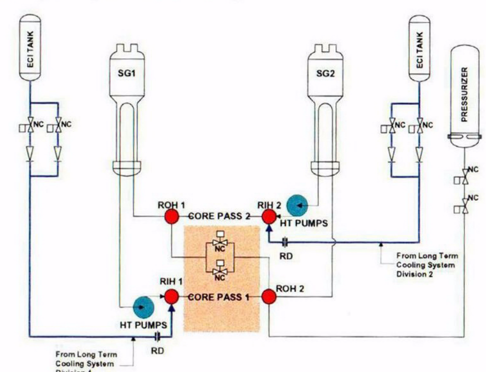
ECC System

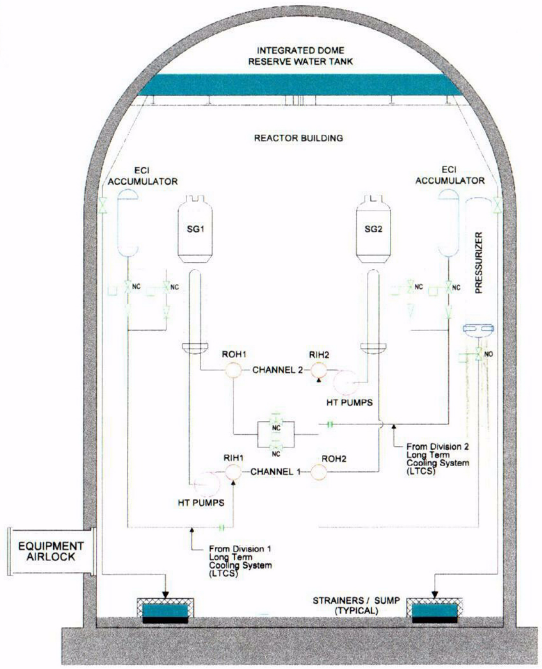
Two stage Emergency Core Cooling System

- Initial injection from pressurized tanks in containment
- Long term pumped recovery



C06

Emergency Coolant Injection (ECI)

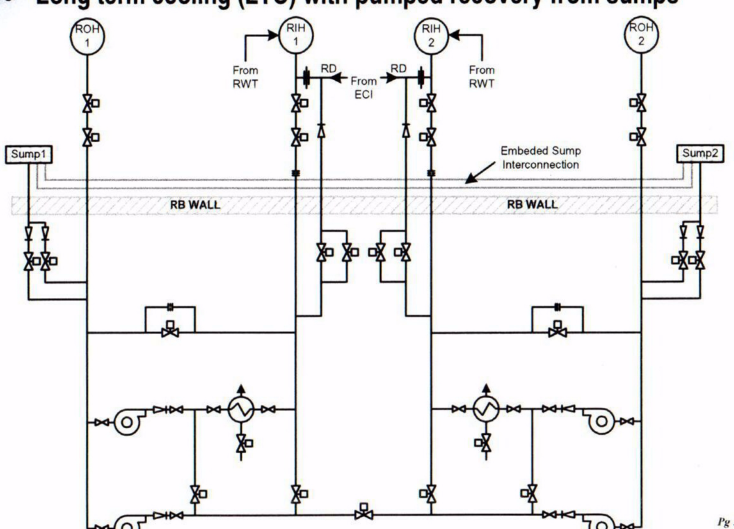


Passive ECI Design Features

- Pressurized Accumulators
  - ECI Accumulators are pressurized by nitrogen gas
- One-Way Rupture Disks
  - One-way rupture disks isolate the ECI system from the RCS
  - Support a large differential pressure in the RCS to ECI direction
  - Burst at a relatively small differential pressure in the ECI to RCS direction
- Floating Ball Shutoffs
  - Seal against a seat at the bottom of the accumulators when the water level becomes low, terminating injection and passively preventing injection of nitrogen gas into the Reactor Coolant System

LTC

- Long term cooling (LTC) with pumped recovery from sumps



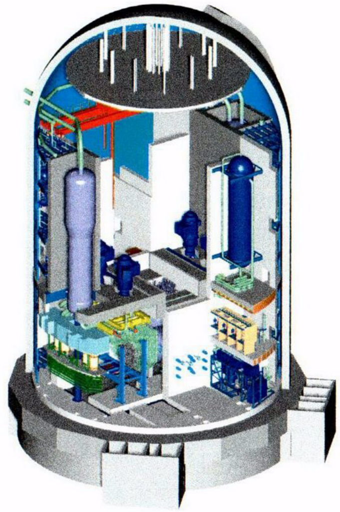
ECC System Parameters

Normal RCS pressure	12 MPa(a)
Normal RCS volume	~120 m^3
ECC Water Tanks	2 (Pressurized with gas)
ECC Water Tank Volume (each)	170 m^3
ECC Working Pressure	5.0 MPa(a)
Injection Lines	2
Major Valves	4
Major Check Valves	2
Rupture Disks	2 (one-way)

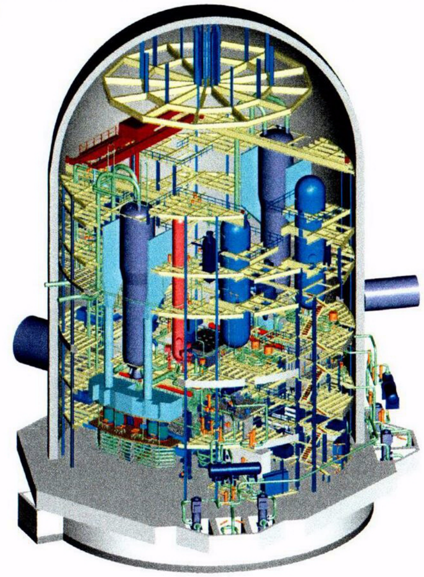
C07

Containment

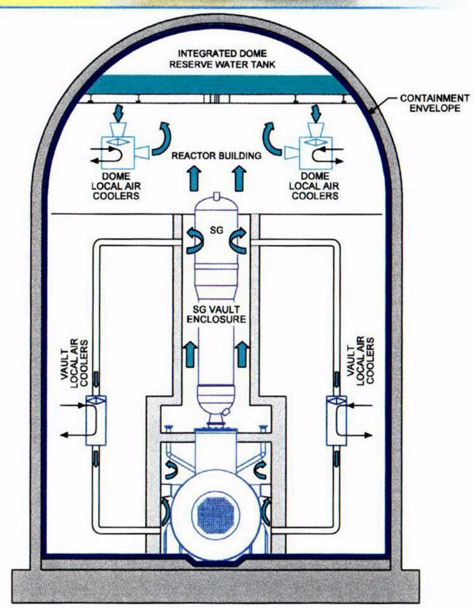
- Steel-lined dry containment similar to conventional PWR
- Elevated Reserve Water Tank for ECC and core damage accident prevention / mitigation
- Air coolers for heat removal
- Passive autocatalytic hydrogen recombiners for core damage accidents



Containment Layout - Detail



Containment Air Flows



- Introduction
- ACR Design Features
- Engineered Safety Features
- **Severe Accident Resistance and Mitigation**
- Operational Features

C08

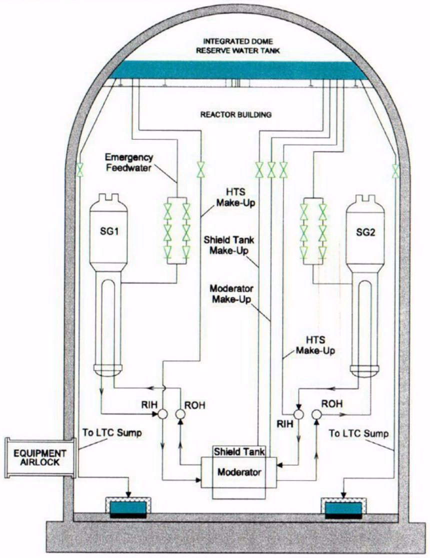
Severe Accident Resistance

Elevated Reserve Water Tank can provide passive makeup by gravity to:

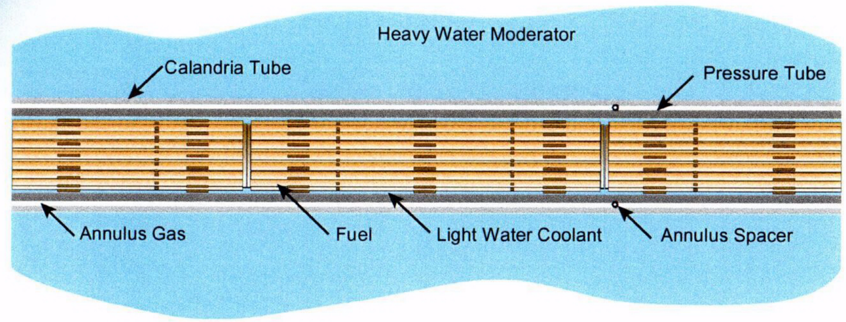
- Reactor coolant system
- Steam generators
- Moderator
- Shield tank

Moderator can remove decay heat from fuel channels without UO_2 melting

Shield tank water can slow down or arrest graceful severe core damage progression



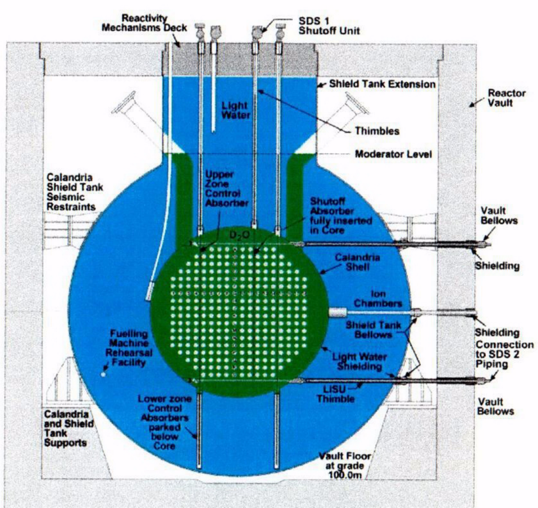
Fuel Channel Surrounded by Water



Fuel is UO_2 clad with Zircaloy-4, in short bundles

Moderator is unpressurized heavy water below 100°C

Moderator Surrounded by Shield Tank

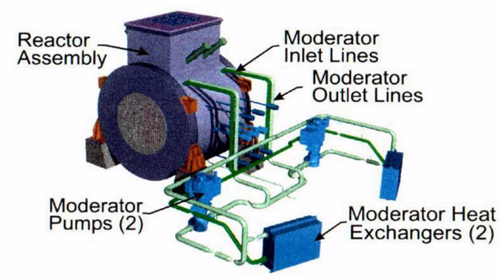


Calandria tank contains 102 m³ of heavy water

Shield tank contains 456 m³ of light water

Moderator System

- The moderator is a separately cooled heavy-water tank surrounding the fuel channels
- In normal operation it removes about 5% of the reactor thermal power which is approximately the same as the decay heat shortly after shutdown...



C09

Limited Core Damage Accidents

- Moderator system removes 5% of thermal power in normal operation
- Can therefore provide an emergency heat sink which maintains core coolability and channel geometry even with no water in the fuel channels (e.g., Loss of Coolant plus Loss of Emergency Core Cooling (ECC))
- Fuel will be damaged but no UO_2 melting
- Design facilitates moderator as emergency heat sink via choice of moderator subcooling
- Active moderator heat removal backed up by passive makeup from Reserve Water Tank

Severe Core Damage Accidents

- Control system and two independent shut-down systems are each capable of safely shutting down the reactor
  - Anticipated Transients Without Scram is not risk-significant
- Severe core damage sequence would result only from multiple failures such as:
  - LOCA *plus* LOECC *plus* Loss of moderator cooling *plus* Loss of Reserve Water Tank makeup to the moderator
- In severe core damage sequence, core geometry is lost but shield tank water can delay progression or contain debris within calandria

Severe Core Damage Delay and Containment

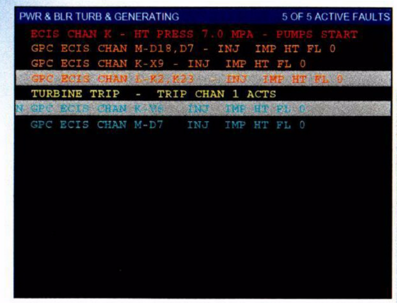
- Severe core damage can be *delayed* for hours due to passive boil-off of moderator and shield tank inventory
- Severe core damage can be *contained* within the calandria vessel if the Reserve Water System is used to make up the shield tank

- Introduction
- ACR Design Features
- Engineered Safety Features
- Severe Accident Resistance and Mitigation
- **Operational Features**

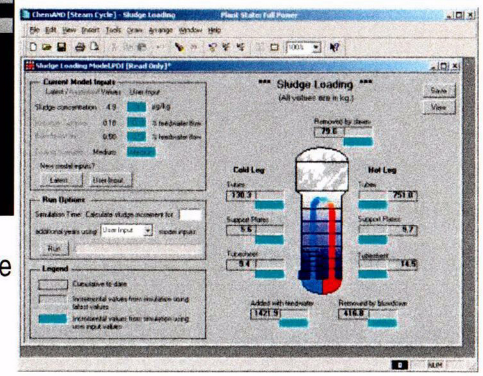
Outage Reduction

- Reducing running failures of essential equipment
  - System / equipment health monitoring: on-line data, readily accessible to all for trend analysis
  - Design and operation connected via Reliability Centered Maintenance (proven from application to existing CANDU reactors)
- Reducing Operator Error
  - Improved Control Room Design
  - Comprehensive plant status available via large-screen display
  - Improved alarm recognition system

Advanced Control and Monitoring Capabilities

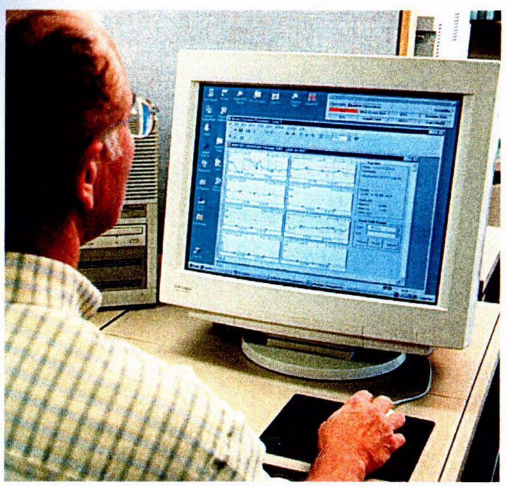


Advanced control room technology and alarm monitoring capability



Advanced plant performance monitoring and diagnostic capability

ChemAND System Health Monitor

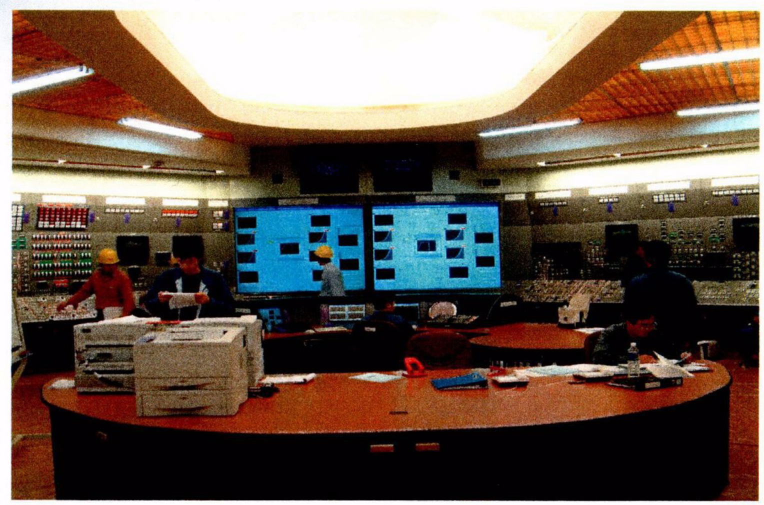


Installed at some CANDU Plants

<u>Features</u>

- Real time monitoring of all chemically controlled systems
- Alarms
- Diagnostics
- Predictions

Qinshan Main Control Room



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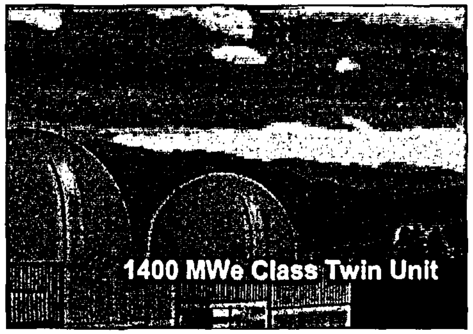
The ACR

- Evolutionary product from >50 years of R&D
- Design optimized
- Enhanced safety including passive features
- Enhanced operability



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ACR Pre-Application Scope, Rationale and Expectations

Vince Langman
ACR Design Certification Program Manager
ACRS Subcommittee on Future Plant Designs
Washington D.C.
January 13, 2004

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ACR Pre-Application Review

SCOPE

- Phase 1: June 2002 to August 2003
  - Establish CANDU-specific focus topics
  - Extensive technical familiarization meetings
  - Reports submitted for NRC review in support of focus topics, including understanding of ACR technology base
  - Respond to NRC staff RAIs

ACR Pre-Application Review

SCOPE

- Phase 2: September 2003 to September 2004
  - Further technical meetings on focus topics
  - Participate in NRC PIRT meetings for ACR
  - Additional reports being submitted for NRC review
  - Respond to NRC staff RAIs

ACR Pre-Application Review

RATIONALE

- Focus the pre-application review effort on the CANDU-specific aspects of the ACR design that are not easily addressed by the current NRC regulations
- Deal with focus topics that:
  - Are inherent to the ACR design (i.e., design aspects that cannot and should not be changed)
  - Have prohibitively large monetary and/or schedule impacts if proposed AECL positions not accepted by the NRC staff

Focus Topics for Pre-Application Review

1. Class 1 pressure boundary design (key)
2. Design basis accidents and acceptance criteria
3. Computer codes and validation adequacy (key)
4. Severe accident definition and adequacy of supporting R&D
5. Design philosophy and safety-related systems
6. Canadian design codes and standards
7. Distributed control systems and safety critical software

Focus Topics for Pre-Application Review (continued)

8. On-power fueling (including fuel design) (key)
9. Confirmation of negative void reactivity (key)
10. Preparation for Standard Design Certification Docketing
11. ACR PRA Methodology
12. ACR Technology Base

NRC Familiarization Complete

ACR Familiarization Meetings

- Design and Technology Base (September 25, 26, 2002)
- Physics (December 4-5, 2003)
- Fuel Channels (December 4-5, 2003)
- Quality Assurance (December 4-5, 2003)
- Thermal Hydraulics (February 5-6, 2003)
- Constructability (February 24-25, 2003)
- Safety Design Philosophy (March 27, 2003)

NRC Familiarization Complete

ACR Familiarization Meetings (continued)

- Severe Accidents and PRA (May 6-8, 2003)
- Analysis Methodology and Computer Codes (May 15-16, 2003)
- Details of RD-14M Results (June 4-5, 2003)
- On-Power Fueling (September 3, 2003)
- CANFLEX Fuel Design (September 4, 2003)

Information Submitted During Pre-Application

- ACR Technical Basis Document
- ACR Technical Description
- PRA Methodology for ACR
  - Generic PRA Methodology
  - PRA Methodology for ACR
- Safety Analysis Basis
- Initial Conditions and Standard Assumptions
- ACR Anticipatory R&D
- Technology of fuel channels
- Technology of on-power fueling

Information Submitted During Pre-Application

- Safety Computer Code Validation
  - Manuals and validation reports for CATHENA (TH) and physics codes
  - CATHENA code, input deck and description
  - Physics code suite
  - Safety Analysis Computer Code Qualification – Status & Plan
- Severe accidents
  - MFMI test program
  - Severe accident progression in ACR
  - Severe accident R&D program

Information Submitted During Pre-Application

- ACR Safety Analysis
  - Safety analysis methods (trip coverage, fuel and fuel channel, containment and thermal hydraulics, including physics)
- System classification (i.e., safety related systems, definition and design requirements)
  - ACR approach to safety related systems
- Comparison of 10CFR50 Appendix B and ASME NQA-1-1994 Requirements versus CSA N286 Series of Standards
- Identification and Initial Assessment of USNRC GSIs Applicable to ACR

ACR Pre-Application Review

EXPECTATIONS

- NRC staff will identify whether there are any impediments to licensing the ACR in the US
- Success paths identified for any unresolved pre-application focus topics
- Assessment of the completeness of AECL's R&D programs that exist, or are proposed/planned, in support of the ACR
- Provide estimates of cost and schedule for the NRC's Design Certification review of the ACR

ACR Pre-Application Review

EXPECTATIONS

- Parallel ongoing licensing reviews of the ACR offer an excellent opportunity for regulatory synergy between the CNSC and the NRC
  - Common major documents for review
  - Similar time frames for reviews
  - Co-operation between internationally-respected, well-established, mature regulatory bodies (i.e., avoid overlap of effort)
- Opportunity to integrate the extensive licensing experience of the CNSC and the NRC to develop a common North American technical basis for licensing the ACR-700 in both Canada and the US

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Overview of Key Focus Topics

Vince Langman
ACR Design Certification Program Manager
ACRS Subcommittee on Future Plant Designs
Washington D.C.
January 13, 2004

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Focus Topic #1 (Key)
Class 1 Pressure Boundary Design

Desired outcome:

The NRC staff accepts the principal design features of the ACR RCS pressure boundary (i.e., the use of Zr-2.5wt%Nb pressures tubes, rolled joints, closure plugs, 403SS end fittings, and fueling machines as components of a Class 1 pressure boundary)

Of note:

- CSA standards, fitness for service guidelines, PT inspection
- Extensive technology base
- Extensive operating experience

Focus Topic #3 (Key)
Computer Codes and Validation Adequacy

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

Of note:

- Similar approach to code validation in Canada and the US (PIRT, validation matrices, etc.)
- Extensive work on formal code validation in recent years in Canada

Focus Topic #8 (Key)
On-Power Fueling
(Including Fuel Design)

Desired outcome:

The NRC staff accepts the ACR CANFLEX fuel design and the process of on-power refueling

Of note:

- Extensive successful experience with various CANDU fuel designs
- ACR CANFLEX based on proven technologies
- Extensive successful experience with on-power fueling

Focus Topic #9 (Key) Confirmation of Negative Void Reactivity

Desired outcome:

The NRC staff accepts that the ACR has a negative void reactivity

Of note:

- R&D program underway to provide ACR-specific validation data for physics code suite to be used for ACR analysis

Focus Topic #11 ACR PRA Methodology

Desired outcome:

The NRC staff accepts AECL's PRA methodology as sufficient for the purpose of assessing the ACR for licensing in the US

Of note:

- ACR approach consistent with international and US practice
- Severe core damage accident and large release frequency goals are the same as for other new reactor designs in the US

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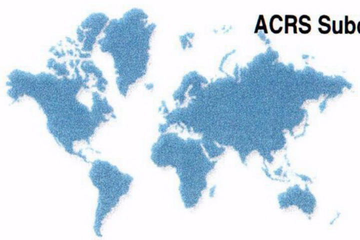
The ACR Class 1 Pressure Boundary

Marc Léger
Director, Materials Engineering
ACRS Subcommittee on Future Plant Designs
Washington D.C.
January 13, 2004



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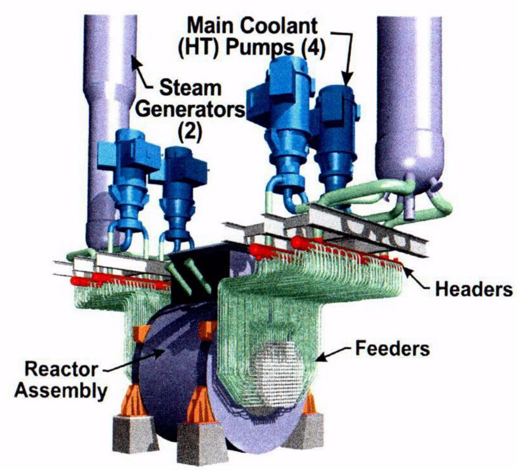
Outline

- Pressure Boundary Features of the ACR
- CANDU Experience
- Pressure Tube Leak-before-Break
- Fuel Channel Standards
  - Pressure Tubes
  - End Fittings
  - Channel Closures for on-power refueling
  - Inspection
    - Material Surveillance
- Summary

Pressure Boundary Features of the ACR

- Piping, valves, pressure vessels – designed to ASME
- Feeder pipes – multiple, small diameter, pipes from headers to fuel channels – also designed to ASME

Reactor Coolant System



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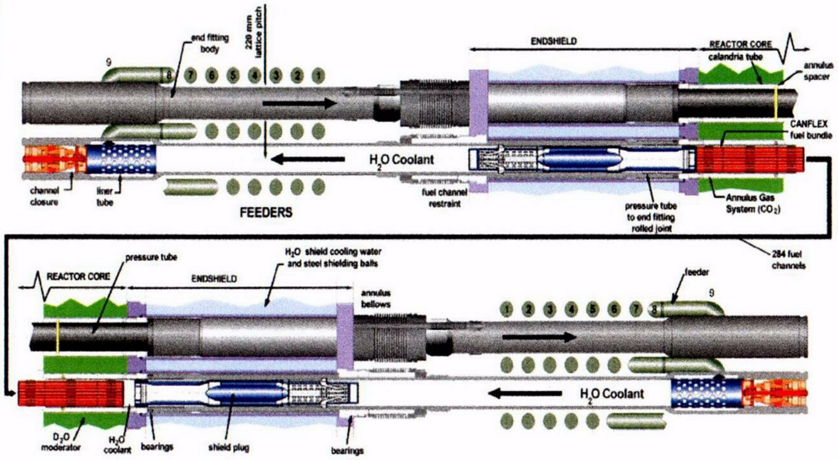
Pressure Boundary Features of the ACR

- Fuel Channel – designed to Canadian Standards
  - Designed to meet intent of ASME with accommodation for pressure tube and refueling requirements
    - Material exceptions
      - Zr-2.5%Nb pressure tube
      - Modified 403 SS end fitting
    - Design differences
      - Rolled joint between pressure tube and end fitting
      - Channel closure for refueling

ACR Fuel Channel



CANDU Fuel Channel Experience

- Power-reactor experience with pressure tube reactors in CANDU community began 41 years ago
- Approximately 400 reactor-years of operation of large CANDU's worldwide starting in 1971
- Longest-operating, Zr-2.5%Nb pressure tubes currently have ~ 150,000 hours of operation

CANDU Experience

- Pressure tubes change dimensions over their lifetime
  - Maximum 4.5% diametral expansion and ~7% wall thinning expected in ACR during a 30 year pressure tube lifetime
    - Due to irradiation creep and growth in anisotropic material
- Dimensional changes are accommodated by design
  - Zr-2.5% Nb deformation performance is well-understood and predictable within acceptable bounds
    - Experience and R&D programs cover the range of ACR conditions
    - Elongation accounted for in feeder clearances and stresses
    - Impact of diametral expansion on fuel cooling is taken into account

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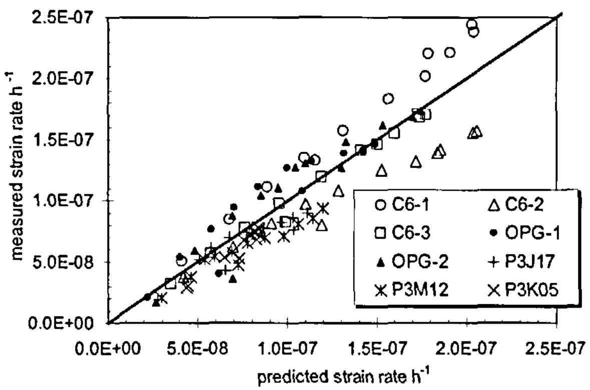
Diametral Strain Rates



CANDU Experience

- Pressure tube creep ductility limits are large
  - Material deformation under irradiation is occurring with stress exponent close to 1, i.e. strain rate is almost proportional to stress
  - Low stress exponents correspond to very high strains to failure (superplastic behavior)
  - Tests of pressure tube materials indicate high failure strains
  - Material surveillance of tubes removed from service has not identified any microstructural changes indicative of a potential creep ductility limit

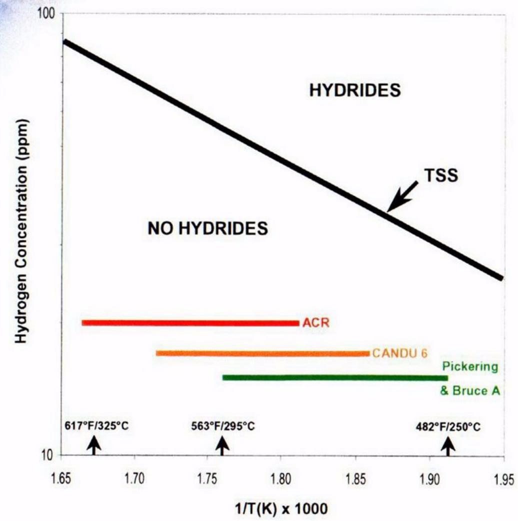
CANDU Experience

- Pressure Tube Integrity
  - No pressure tube leaks due to design / material performance since 1986
    - Early leaks due to high residual stresses near rolled joints and delayed hydride cracking (DHC)
    - Rupture of Zircaloy-2 pressure tube at power due to contact with calandria tube and hydride blistering
    - One rupture at cold conditions from long manufacturing flaw
  - Design, manufacturing and assembly issues that led to early failures have been solved
    - Development of low-stress rolled joint eliminated high residual stresses in pressure tubes near the joints and prevents crack initiation
    - New channel spacer design prevents contact between pressure tube and calandria tube and thereby prevents potential hydride blister formation in the pressure tubes
    - Improved manufacturing practices and better inspection reduces the possibility of long manufacturing flaws

CANDU Experience

- DHC mechanism has been carefully studied and is well understood
  - Cracking not possible at operating temperature given that hydrogen content of tube remains below the hydrogen solubility limit at temperature
  - Crack initiation avoided by low-residual-stress joint technique and clean system preventing debris flaw formation
  - Cracking at lower temperature avoided by pressure reduction

Source and Consequences of Hydrogen Ingress

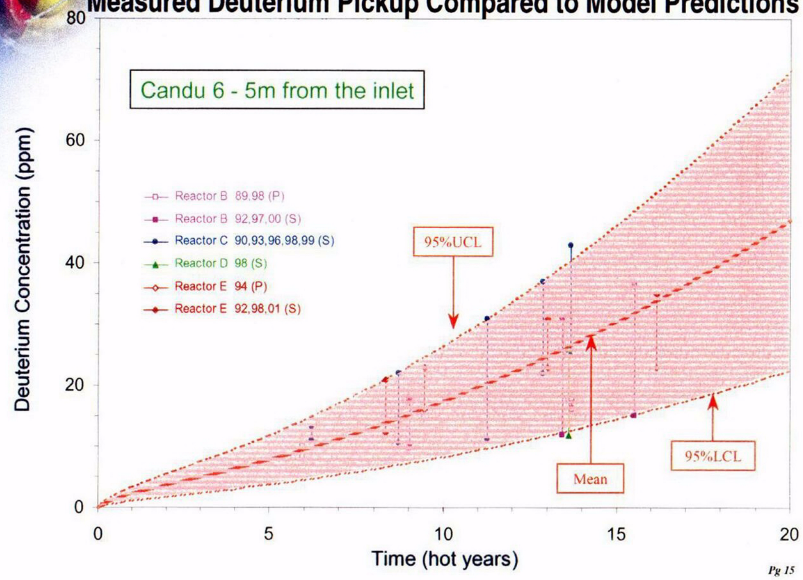


- Corrosion Reaction:
  $Zr + 2H_2O \rightarrow ZrO_2 + 4H$
- fraction of hydrogen absorbed by base metal
- hydrides present when Terminal Solid Solubility (TSS) exceeded
- hydrides can potentially lead to fracture issues
- ensure hydrides are not present during reactor operation at power

CANDU Experience

- **Pressure tubes corrode to produce hydrogen, some of which enters the tube**
  - **In current CANDU reactors, maximum waterside oxide thickness is 20 to 30 μm after 20 years operation**
  - **Maximum hydrogen pickup is approximately 20 ppm H (as D) after 20 years of operation except near rolled joints which exhibit higher hydrogen pickup**
  - **Empirical models of corrosion and hydrogen pickup, based on experimental programs, produce predictions consistent with observations from surveillance tubes**
    - **Models include rolled joint region and main body of the tube**

Measured Deuterium Pickup Compared to Model Predictions



Pressure Tube Leak-before-Break

- **Defense-in-depth for normal operation**
- **Gas circulated through the annuli between pressure tubes and calandria tubes is continuously monitored for moisture content**
- **Fracture toughness and crack growth rates of the pressure tube material remain at a level at which any tube, that could potentially develop a crack, would leak allowing time for leak detection, response and safe shutdown before the crack becomes unstable**
- **LBB is demonstrated by a sequence-of-events analysis using conservative assumptions**

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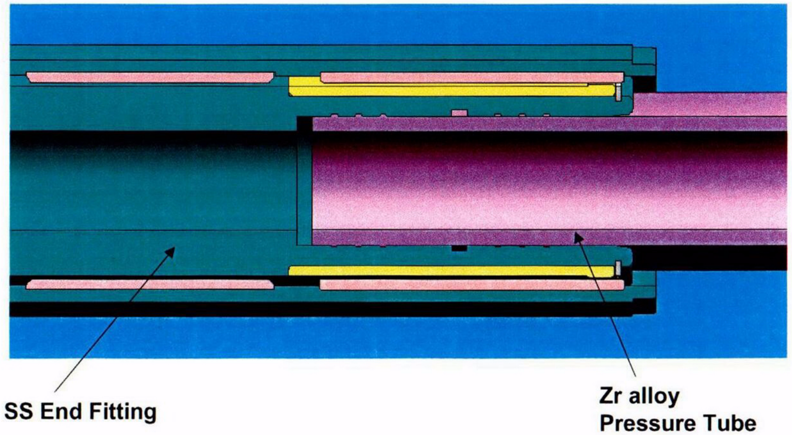
Fuel Channel Standards

- Pressure tube
  - Pressure tubes are designed to CAN/CSA N285.2 Standard
  - Tubes meet CAN/CSA N285.6 Standard and additional AECL Technical Specifications for material
  - Zr-2.5%Nb is an ASTM Standard B353 (UNS R60901) material
  - ASME – type criteria apply for allowable design stress levels
  - Tubes are a consistent, high quality product
  - Current production tubes have improved properties compared to earlier production achieved by improved material specifications and production methods – especially with respect to fracture toughness properties after irradiation – a result of R&D programs in fracture area

Fuel Channel Standards

- Rolled joint
  - Rolled joints will meet CAN/CSA N285.2
  - Designed to ASME Section III NB-3200 – "Design by Analysis"
  - Reliable, strong, mechanical joint suitable for zirconium alloy to stainless steel connection – able to withstand 3x design-condition axial load including pressure
  - Qualification is carried out on production-grade joints
  - Each reactor joint checked for designed pressure tube wall reduction and leak rate on installation

Pressure Tube to End Fitting Joints



Fuel Channel Standards

- End Fitting
  - Designed to CAN/CSA N285.2 and ASME Section III NB-3200
  - Material specified by CAN/CSA N285.6
    - Each end fitting made from a single forging
    - Modified 403 martensitic SS
      - High strength and corrosion resistance with acceptable fracture toughness
  - Excellent operating experience in CANDU – no identified issues from operation

C14

Fuel Channel Standards

- Channel Closures
  - Channel closures are removable pressure boundary components at the outboard end of an end fitting required to permit on-power fueling
  - Satisfy ASME Class 1 design rules
  - Satisfy the following requirements specified by CAN/CSA N285.2:
    - Closure shall be locked in place to prevent inadvertent removal
    - Closures shall be leak tested each time they are installed prior to removal of the fuelling machine
  - CAN/CSA N285.2 requires interlocks to prevent fueling machine from disengaging before closure is in place

Fuel Channel Standards

- Periodic Inspection Programs for CANDU fuel channels are designed to monitor for any generic degradation and are defined by CAN/CSA N285.4
- Single channels removed for material surveillance purposes from "lead" reactor units
  - Source of fracture toughness, corrosion, hydrogen isotope pick up and DHC growth rate data after irradiation

Summary

- CANDU fuel channels are a proven technology licensed in five jurisdictions including Korea
- An extensive technology base supports the design

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