IRIS
INTERNATIONAL REACTOR INNOVATIVE AND SECURE

Westinghouse Electric Co.
Pre-Application Initial Meeting

US Nuclear Regulatory Commission
October 3, 2002

PRELIMINARY AGENDA

8:30 AM	Introduction, Scope of Pre-application and Licensing Approach	M. D. Carelli/ C. L. Kling
8:50 AM	IRIS Design Overview	M. D. Carelli/ B. Petrovic
	IRIS Safety	
9:40 AM	The Safety-by-Design IRIS Approach	M. D. Carelli
9:50 AM	Engineered Safety Features	L. E. Conway
10:15 AM	Break	
10:30 AM	Effect of Safety-by-Design - Design Basis Accidents	L. Oriani
11:10 AM	Pre-application Objective A: Testing guidance	M. D. Carelli
11:20 AM	Pre-application Objective B: Risk informed licensing	C. L. Kling
11:30 AM	Q&A	
12:00 Noon	Adjourn	

IRIS CONSORTIUM ATTENDEES

Mario Carelli	Westinghouse Science & Technology	Chief Technologist; Director, IRIS Program
Charles Kling	Westinghouse New Plant Projects	Manager, New Plant Engineering IRIS Licensing
Lawrence Conway	Westinghouse Science & Technology	Principal Engineer IRIS Design
Bojan Petrovic	Westinghouse Science & Technology	Senior Engineer IRIS Core Neutronics
Luca Oriani	Westinghouse Science & Technology	Senior Engineer IRIS Safety Analyses
Charles Brinkman	Westinghouse Nuclear Services	Director, Washington Operations
John Polcyn	Bechtel Power Corporation	Vice President
Daniel Ingersoll	Oak Ridge National Laboratory	ORNL - IRIS Program Manager

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IRIS CONSORTIUM

<u>Industry</u>		
Westinghouse	USA	Overall coordination, core design, licensing
BNFL	UK	Fuel and fuel cycle
Ansaldo Energia	Italy	Steam generators design
Ansaldo Camozzi	Italy	Steam generators fabrication
ENSA	Spain	Vessel and internals
Washington Group EMD	USA	Pumps, CRDMs
NUCLEP	Brazil	Containment, pressurizer design
Bechtel	USA	BOP, AE
<u>Laboratories</u>		
ORNL	USA	I&C, PRA, core analyses, shielding
ININ	Mexico	Neutronics, PRA support
CNEN	Brazil	Transient and safety analyses

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IRIS CONSORTIUM (Cont'd.)

Universities		
Polytechnic of Milan	Italy	Safety analyses, shielding, thermal hydraulics, steam generators design, advanced control system
Tokyo Inst. of Technology	Japan	Advanced cores, PRA
University of Zagreb	Croatia	Neutronics, Safety analyses
University of Pisa	Italy	Containment analyses
Power Producers		
TVA	USA	Maintenance, utility feedback
Eletronuclear	Brazil	Developing country utility feedback
Associated US Universities (NERI programs)		
MIT	USA	Advanced cores, maintenance
U. California Berkeley	USA	Neutronics, advanced cores
U. of Tennessee	USA	Modularization, I&C
Ohio State	USA	In-core power monitor, advanced diagnostics
Iowa State (Ames Lab)	USA	On-line monitoring
U. of Michigan (& Sandia Lab)	USA	Monitoring and control

PRE-APPLICATION RATIONALE

- Background
  - IRIS will rely to a large extent on AP600/AP1000 precedents
  - IRIS licensing activities will hit high gear after AP1000 design certification (~ 2004)
  - Early NRC feedback will be most beneficial in addressing long lead items and new licensing issues
- "Focused" Pre-application - First phase with targeted completion mid 2003
- Possible second phase from mid-2003 to start of formal application (late 2004/early 2005)

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FIRST PHASE PRE-APPLICATION OBJECTIVES

- Objective A: Identify technical issues for NRC study and review of proposed test program for the IRIS design
- Objective B: Define approach to focused application of risk-informed licensing
- Develop scope, schedule and budget for NRC review work
- Develop a schedule for IRIS design certification (current objective 2008)

GUIDANCE FROM NUREG-1226

- Top level regulatory criteria
  - Defense-in-depth
  - Commission safety goals
  - Severe accident policy
  - Applicable industry codes and standards
- Consistency with current regulations
  - Compare IRIS to current SRP
  - Use "Highly risk-informed" approach for IRIS

GUIDANCE FROM NUREG-1226

- **Selection of licensing basis events**
  - Safety-by-design approach emphasizes prevention of accidents by eliminating their occurrence, or reducing their probability, or limiting their consequences to acceptable levels, rather than engineering to cope with consequences
  - Supplemented by PRA guided analysis and design
- **Selection of equipment safety classification**
  - Counteract with passive systems those accidents which must still be considered

LICENSING APPROACH

- **IRIS is based on proven LWR technology, newly engineered**
- **IRIS specifically designed to meet or be significantly within current licensing requirements**
  - Some current licensing requirements can be relaxed and IRIS will still meet top level safety goals
- **Enhanced safety through safety-by-design and simplicity**
  - PRA to guide final design and safety analysis

LICENSING APPROACH

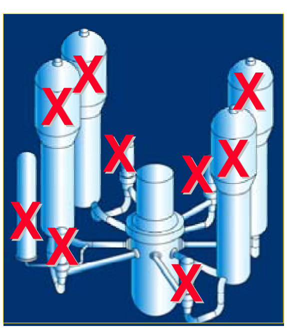
- **Obtain early NRC input on testing and licensing issues**
- **Establish continuing interaction with and feedback from NRC and ACRS as design progresses**

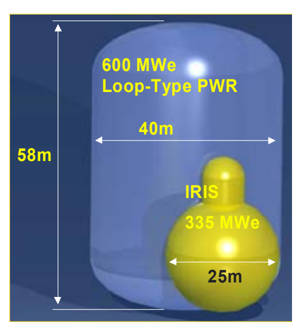
INTERNATIONAL REACTOR INNOVATIVE AND SECURE - DESIGN OVERVIEW -

IRIS DESIGN OVERVIEW OUTLINE

- Configuration
- Snapshots
- Core design
- Integral primary system components (SG, pump, pressurizer)

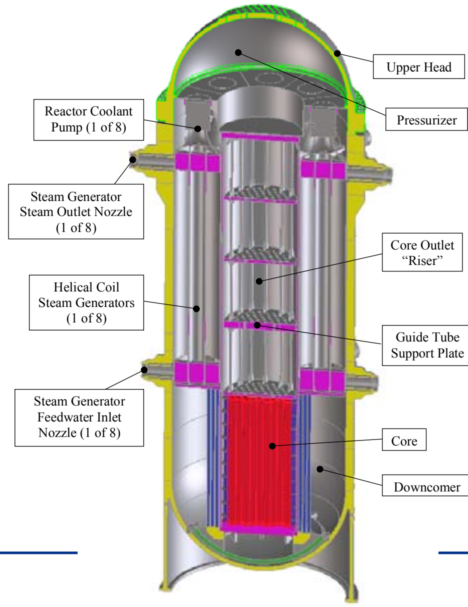
IRIS INTEGRAL PRIMARY SYSTEM CONFIGURATION





Integral vessel configuration eliminates loop piping and external components, thus enabling compact containment and plant size

INTEGRAL REACTOR COOLANT SYSTEM



ENSA Equipos Nucleares, S.A.

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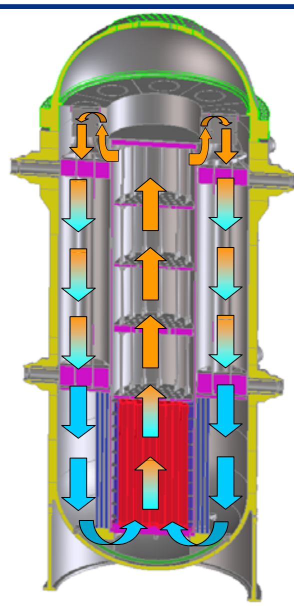
INTEGRAL REACTOR COOLANT SYSTEM MAIN FLOW PATH



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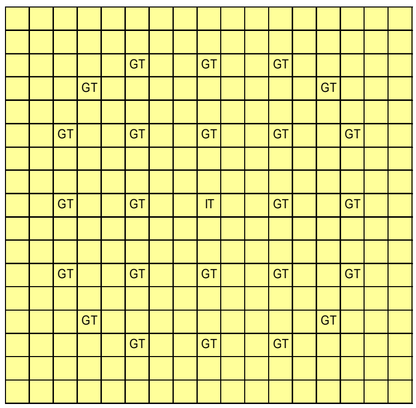
EVOLUTION OF IRIS DESIGN CHARACTERISTICS

Characteristic	Original Concept	Other Options	Current Reference
Plant power rating	$\leq$ 300 MWt		1000 MWt
Refueling scheme	Straight burn	1/3 core batch	1/2 core batch
Core life	Up to 15 yrs	Up to 8 yrs	3-4 yrs
Lead rod burnup	$\geq$ 100,000 MWd/t	Up to 100,000 MWd/t	< 62,000 MWd/t
Fuel enrichment	< 20% fissile	8-10% UO_2, 12% MOX	Up to 4.95% UO_2
Core configuration	Tight lattice, hexagonal	Exotic rod shapes	Standard square lattice
Neutron spectrum	Epithermal		Thermal, enhanced moderation (larger P/D)
Control	No soluble boron		Limited soluble boron
Coolant circulation	Maximized natural circulation through low inlet temp, allow boiling		Very similar to present PWRs

CORE DESIGN REQUIREMENTS

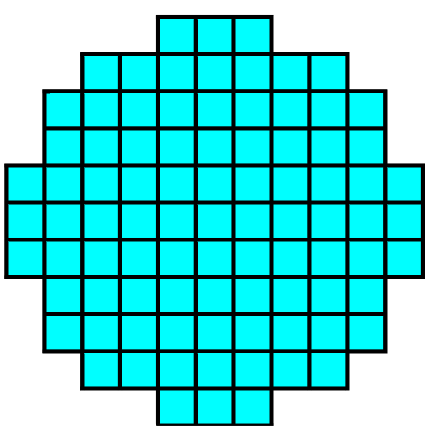
- **Medium reactor power - 1000 MWt**
- **Target: deployable ~2012**
- **No fuel licensing issues**
- **Current fuel technology**
- **Extended fuel cycle (up to 4 years)**

FUEL ASSEMBLY



- UO_2 fuel
- Enrichment <5%
- Standard fuel rod
- Enhanced moderation
- 17x17 fuel assembly
- Incorporates standard <u>W</u> design features
- Long plenum eliminates potential rod internal pressure issues

CORE CONFIGURATION



1,000 MWt CORE

89 FUEL ASSEMBLIES

Core barrel ID/OD = 275/285 cm

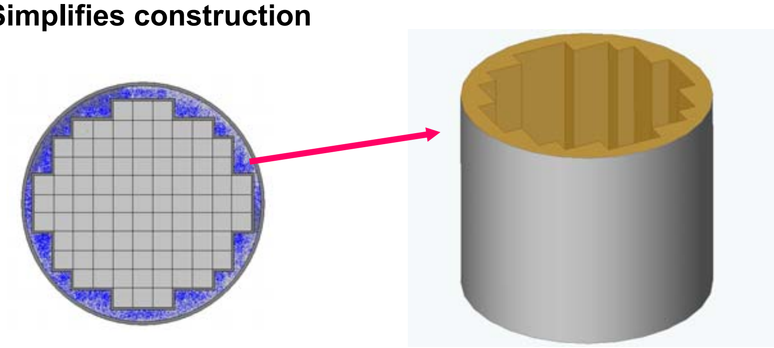
Active Core Height = 14 ft

NOTE: Flexibility in plenum length (fuel assembly length) - no impact on integral reactor vessel

NEUTRON REFLECTOR

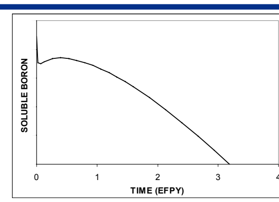
STAINLESS STEEL NEUTRON REFLECTOR
(instead of baffle plates)

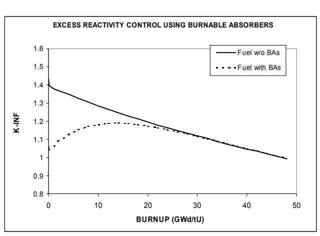
- **Improves neutron economy**
- **Simplifies construction**



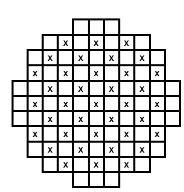
REACTIVITY CONTROL

SOLUBLE BORON
- **Standard PWR approach**
- **Additionally, reduced maximum concentration**





BURNABLE ABSORBERS
- **Standard <u>W</u> design**
  - **Zirconium Diboride IFBA**
  - **Erbium IBA**


Typical RCCA pattern

CONTROL RODS (RCCAs)
- **Standard <u>W</u> design**
- **Black CRs and Gray CRs**

REFUELING OPTIONS (4.95% UO_2)

	Not part of this application	REFERENCE OPTION	Not part of this application
	Single Batch (Straight Burn)	Two-Batch (Partial Reload)	Three-Batch (Partial Reload)
FAs @ Nominal Enr. 4.95 w/o	69	~44	28-36
FAs @ Low Enr. 2.6 w/o	20	--	--
Cycle Length (EFPY)	4.0	3.0-3.5	2.0-3.0
Avg. BU for 4.95 w/o FAs	38-40,000	46-53,000	<62,000
Peak Rod BU (MWd/tU) (est)	<50,000	<62,000	<75,000

NO FUEL LICENSING ISSUES

STANDARD W FUEL/ASSEMBLY FEATURES

- UO_2 fuel
- Licensed ^{235}U enrichment (up to 4.95 w/o)
- Standard rod size (0.374" O.D.)
- Standard assembly lattice (17x17)
- Standard burnable absorbers (IFBA, erbium)
- Standard active fuel length (14 ft., XL)
- Standard reloading strategy (partial reload)

➡ CURRENT (LICENSEABLE) FUEL TECHNOLOGY

IMPROVED CORE PERFORMANCE

- Enhanced moderation and fuel utilization
- Longer cycle (3-4 years)

WHILE MAINTAINING

- Standard design limits
- Discharge burnup within currently approved limits

NUCLEAR SHIP OTTO HAHN WAS POWERED BY AN INTEGRAL REACTOR

German nuclear-powered freighter & research facility:

- Launched in 1964 and commissioned in 1968

- Sailed 650,000 miles in 10 years without any technical problems

IRIS HELICAL COIL STEAM GENERATORS

- Allows thermal expansion, good heat transfer characteristics
- LMFBR SG operating experience
- Fabricated and tested for LWR
- Test confirmed performance (thermal, pressure losses, vibration, stability)
- IRIS - 8 SG, 8.5 meters long, same bundle diameter as Ansaldo test
- Once through with superheat

ANSALDO NUCLEARE
Division of Ansaldo Energia SpA

ANSALDO

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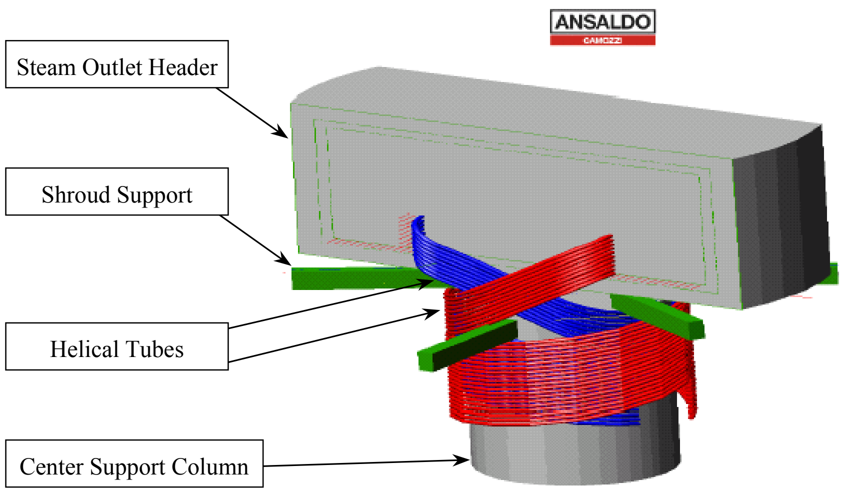
HELICAL COIL STEAM GENERATORS



HELICAL COIL STEAM GENERATORS

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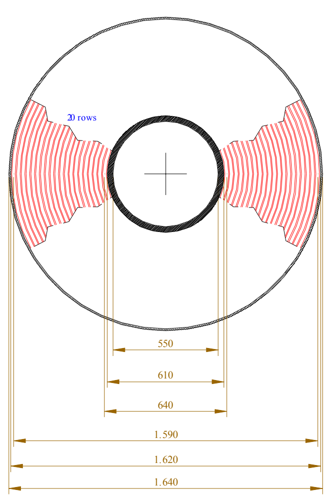
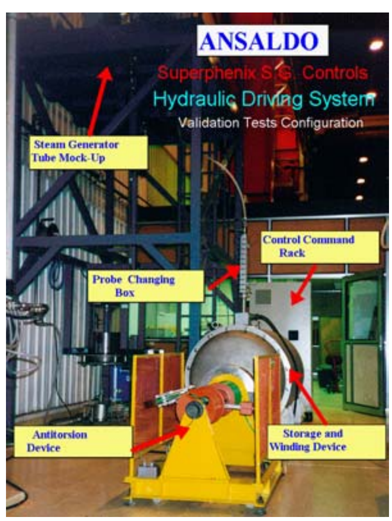
IRIS STEAM GENERATOR
RADIAL ARRANGEMENT



STEAM GENERATOR MAINTENANCE

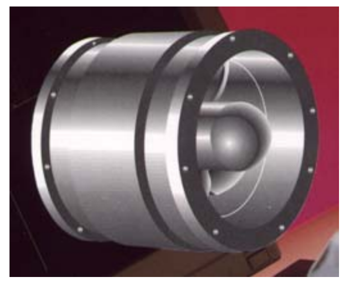
- **Prior Experience exists for helical steam generators (SuperPhenix)**
  - **Signal acquisition system. Ultrasound and visual inspection probes (Framatome)**
  - **Inside tube cleaning and gauging system (Ansaldo)**
  - **Hydraulic driving system (Ansaldo)**
- **Applicability to IRIS verified**
- **Flanged access to headers for ISI and maintenance. No need to remove vessel head.**
- **Bolted steam generators for easy removal**

ANSALDO MOCKUP OF STEAM GENERATOR INSPECTION SYSTEM



PRIMARY COOLANT SPOOL PUMPS

- Developed by Washington Group for marine and chemical applications, requiring large flowrate and low developed head
- Completely immersed, no vessel penetration except electrical cable
- High temperature motor, water lubricated bearings
- Virtually no maintenance
- Reduced vibration
- Operating experience
- Tested up to 500°C
- Must be qualified for nuclear applications



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PRIMARY COOLANT SPOOL PUMPS

Parameter	Design Value (preliminary Pump design)
Fluid Temperature, °F	623
System Pressure, psia	2250
Volumetric Flow Rate per Pump, gpm	14,000
Pump Head, ft	60-70
Brake Horsepower, hp	~300
Coastdown characteristic	Comparable to AP600/AP1000

Washington EMD

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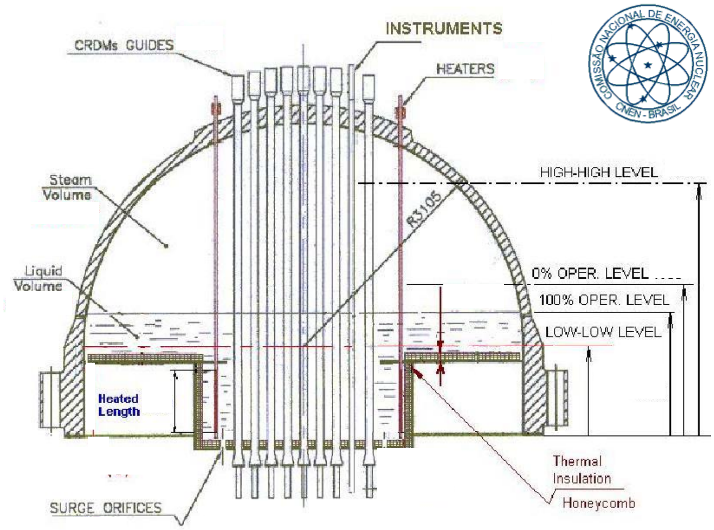
PRESSURIZER



INTERNATIONAL REACTOR INNOVATIVE AND SECURE
- SAFETY -

SAFETY APPROACH

- **Safety By Design**
- **Active, non-safety systems have passive, safety-related back-up to perform nuclear safety functions**
  - » Safety functions automatically actuated, no reliance on operator action
  - » Passive features actuated by stored energy (batteries, compressed air)
  - » Once actuated, their continued operation relies only on natural forces (gravity, natural circulation) with no motors, fans, diesels, etc.
- **Heat sink designed to provide cooling for 7 days without operator action or off-site assistance for replenishing**
- **Additional diverse systems to minimize probability of Core Damage/Radioactivity Release**

IRIS SAFETY BY DESIGN APPROACH

Exploit to the fullest what is offered by IRIS design characteristics (chiefly integral configuration) to:

- Physically eliminate possibility for some accident(s) to occur
- Decrease probability of occurrence of most remaining accident scenarios
- Lessen consequences if an accident occurs

IMPLEMENTATION OF SAFETY BY DESIGN

IRIS Design Characteristic	Safety Implication	Accidents Affected
Integral Layout	No large primary piping	- LOCAs
Large, Tall Vessel	Increased Water Inventory Increased Natural Circulation Can accommodate internal CRDMs *	- LOCAs - Decrease in Heat Removal - Various Events - RCCA ejection, eliminate head penetrations
Heat Removal from inside the vessel	Depressurizes primary system by condensation and not by loss of mass Effective heat removal by SG/EHRS	- LOCAs - LOCAs - All events for which effective cooldown is required - ATWS
Reduced size, higher design pressure containment	Reduced driving force through primary break	- LOCAs
Multiple coolant Pumps	Decreased importance of single pump failure	- Locked Rotor, Shaft Seizure/Break
High design pressure steam generator system	No SG safety valves Primary system cannot over-pressure secondary system Feed/Steam System Piping designed for full RCS pressure reduces piping failure probability	 - Steam Generator Tube Rupture - Steam Line Break - Feed Line Break
Once Through steam generator	Limited Water Inventory	- Steam Line Break - {Feed Line Break}
Integral Pressurizer	Large pressurizer volume/reactor power	- Decrease in Heat Removal, including Feed Line Break - ATWS

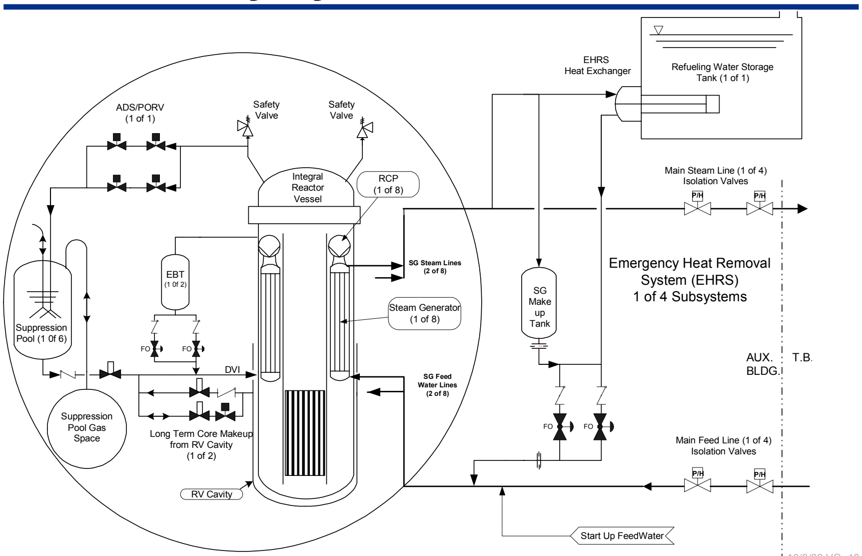
* even though some integral design feature internal CRDMs, their development might not be mature enough for IRIS projected deployment

TYPICAL PWR CLASS IV ACCIDENTS AND THEIR RESOLUTION IN IRIS DESIGN

Design Basis Condition IV Events		Effect of IRIS Safety-by-Design
1	Large Break LOCA	- Eliminated by design (no large piping)
2	Steam Generator Tube Rupture	- Reduced consequences, simplified mitigation
3	Steam System Piping Failure	- Reduced probability, reduced (limited containment effect, limited cooldown) or eliminated (no potential for return to power) consequences
4	Feedwater System Pipe Break	- Reduced probability, reduced consequences (no high pressure relief from reactor coolant system)
5	Reactor Coolant Pump Shaft Break	- Reduced consequences
6	Reactor Coolant Pump Shaft Seizure	
7	Spectrum of RCCA ejection accidents	- [Eliminated by design, requires development of internal CRDMs]
8	Design Basis Fuel Handling Accidents	- No impact

Engineered Safety Features of IRIS

Safety Systems and Function



Safety Systems and Functions

- **Emergency Heat Removal System (EHRS), 4 trains**
  - Safety grade decay heat removal following SLB, FLB, LOHS events (2 trains actuated)
  - Reactor and containment depressurization/cooling following LOCA events (4 trains actuated)
- **Main Feed and Steam Isolation Valves (MFIV/MSIV), redundant and fast closing**
  - Provides isolation following Steam Generator Tube Rupture event - terminating leak
  - Part of EHRS actuation
- **Automatic Depressurization System (ADS), 1 stage**
  - Assists EHRS to equalize RV and containment pressures following LOCAs at low RV locations
- **Emergency Boration Tank (EBT), 2 tanks**
  - Borates primary system to maintain reactor subcritical at low temperatures
  - Provides diverse means of shutdown for ATWS events
  - Provides a limited amount of water makeup following LOCA and cooldown event
- **Long Term Core Makeup System (LTCMS), 2 trains**
  - Provide passive, long term, water makeup to RV from reactor cavity and suppression pools

Safety Systems and Functions (cont.)

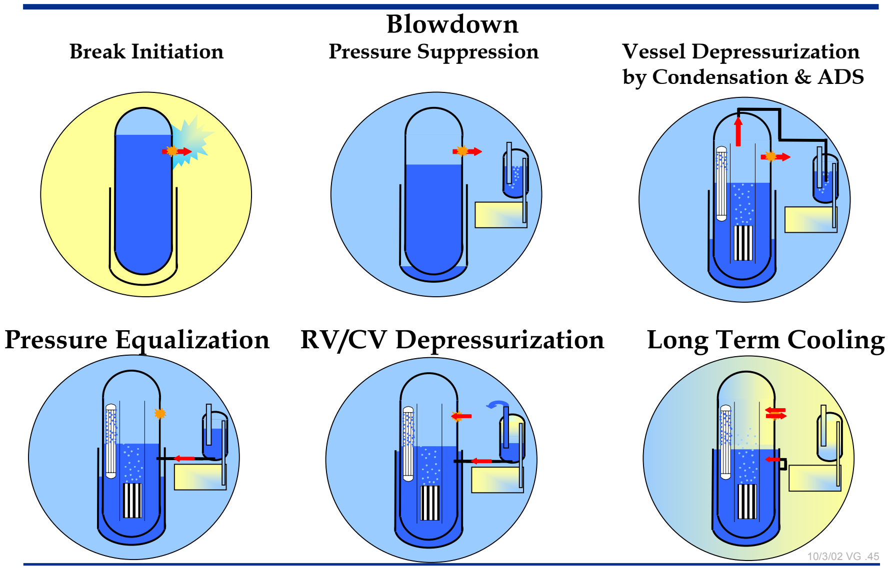
- **Containment System**
  - **High Design Pressure (12 bar_g/175 psig) Steel Shell**
    - » Pressurization following LOCA reduces break flow
    - » Penetrations isolated automatically on high-high pressure
    - » Low leakage - limits offsite dose
  - **Suppression Pool**
    - » Limits containment pressurization to 8 bar_g, following worst DBA
    - » Floods RV cavity after pressure suppression function completed
    - » Source of gravity-driven, borated, makeup water to RV
  - **Reactor Vessel Cavity**
    - » Assures bottom 1/3 of vessel externally flooded following LOCAs
    - » Source of gravity-driven, borated, makeup water to RV for unlimited time
- **Refueling Water Storage Tank**
  - Provides out-of-containment heat sink for EHRS HXs
  - Source of borated water for refueling and heat sink for shutdown accidents

Effect of Safety by Design
- Design Basis Accidents -

Decrease in Reactor Coolant Inventory

- Large LOCAs
  - **Safety by Design**: Integral RV has no loop piping, PZR surge line
  - **Effect**: No postulated large LOCAs
- Small Break LOCAs - (Small (<4-in.) pipes/penetrations in RV)
  - **Mitigation**: (See Next Slide)
  - **Safety by Design**: (1) IRIS large coolant inventory; (2) compact design that allows the design of a small diameter containment with a high design pressure; (3) a depressurization system not based on mass release but on steam condensation inside the RPV
  - **Effect**: IRIS prevents core uncovery for the complete spectrum of anticipated LOCAs

Overview of IRIS SBLOCA Sequence



Decrease in Reactor Coolant Inventory (cont.)

- SBLOCA Sequence of Events and Mitigation
  - **Blowdown:** Reactor Vessel depressurizes and loses mass to the containment. Containment pressure rises.

    Mitigation sequence is initiated with Rx trip and RCP trip. EBT actuated to provide boration. EHRS actuated to depressurize primary system by condensing steam on the Steam Generators (depressurization without loss of mass). ADS actuated to assist the EHRS in depressurizing the system.

    Containment pressure limited by Pressure Suppression System and EHRS.
  - **Pressure Equalization:** RV and CV pressure become equal (CV pressure peak <8 bar_g), break flow stops, gravity makeup of borated water from suppression pool becomes available.
  - **Containment and RV depressurization:** the coupled CV-RV system is depressurized by the EHRS (steam condensation inside the RV exceeds decay heat boiloff), break flow reverses reducing containment pressure, portion of suppression pool water pushed out and assist in flooding the containment cavity.
  - **Long Term Cooling:** RV and CV pressure reduced to <2 bar_g in <12 hours, gravity makeup of borated water from both suppression pool and RV cavity available as required. Long term break flow limited to containment heat loss.

Decrease in Reactor Coolant Inventory (cont.)

- Steam Generator Tube Rupture
  - **Mitigation**: SGTR detection, Rx Trip, closure of MSIV/MFIVs and decay heat removal.
  - **Safety by Design**: In IRIS, the OTSGs, steam and feed lines and the EHRS are designed for full primary pressure
  - **Effect**: No release of primary fluid (radioactivity) once the MSIV/MFIVs are closed;

    Primary water will fill the faulted SG and break flow will stop when primary/secondary pressure equalizes. No need for coolant injection in the primary to avoid core uncovery;

    No SG overfill-overpressure-water relief-safety valve failure, resulting in unisolable containment bypass scenario.

    The number of tubes assumed to fail has limited effect on the system response and does not impact the final plant state.

Increase in Heat Removal from the Primary System

- Excessive increase in secondary steam flow; Inadvertent opening of a steam generator relief or safety valve:
  - **Mitigation**: Rx Trip on low steam pressure & safety grade decay heat removal
  - **Safety by Design**: Once-though steam generators; steam generators designed for full RCS pressure
  - **Effect**: Steam flow equals feed water flow, and cannot increase and cause a significant increase in heat removal
- Steam Line Break:
  - **Mitigation**: Rx Trip on low steam pressure & safety grade decay heat removal
  - **Safety by Design**: Once-though steam generators contain limited water mass
  - **Effect**: Consequences eliminated (no cooldown), and reduced (containment pressurization limited)
- Inadvertent operation of the EHRS heat exchanger:
  - **Mitigation**: None required (see below)
  - **Safety by Design**: The EHRS is designed to not operate even if the isolation valve(s) inadvertently opens.
  - **Effect**: No component level failure leads to inadvertent operation.

Decrease in Heat Removal by the Secondary System

- Loss of Offsite Power; Turbine Trip; Loss of Normal Feedwater; Feed System Piping Failure (Feed Line Break)
  - **Mitigation**: Similar to AP600/AP1000 and current PWRs; Rx Trip & safety grade decay heat removal
  - **Safety by Design**: Once-through steam generators have a limited water inventory; however, this is counteracted by the large primary side thermal inertia (IRIS primary inventory is ~1.6 times AP1000, while power is only ~0.3)

    IRIS has a large pressurizer steam volume (steam volume-to-power ratio is ~4.5 times AP1000) to limit pressure increase
  - **Effect**: Reactor trip setpoint(s) is quickly reached (e.g., low feedwater flow); no pressurizer overfill or high pressure relief.

    Feedline break M&E release to containment limited, effect on containment limited

Decrease in Reactor Coolant Flow Rate

- Complete Loss of Flow (CLOFA) Event
  - **Mitigation**: Rx Trip, sufficiently long pump flow coastdown, & large core thermal margin
  - **Safety by Design**: Spool-type RCPs have sufficient coastdown <u>without</u> added flywheel
  - **Effect**: No significant difference from Westinghouse AP600/AP1000 transient response

- Locked Rotor/Shaft Seizure (LRSS)
  - **Mitigation**: Event terminated by Rx Trip
  - **Safety by Design**: Large core thermal margin; 8 reactor coolant pumps versus 4 for AP600/AP1000 (flow reduction due to loss of one pump limited)
  - **Effect**: Reduced consequences (no DNB predicted using AP1000 evaluation model/assumptions)

Increase in Reactor Coolant Inventory

- Inadvertent Actuation of Emergency Boration
  - **Mitigation**: None required (see below)
  - **Safety by Design**: The Emergency Boration Tank (EBT) will not operate even if the isolation valve(s) inadvertently opens, require reactor coolant pumps to be stopped.
    EBTs are sized for boration function. Not required for reactor makeup (no LOCA injection function)
  - **Effect**: EBT recirculation cannot cause reactor overfill or over-pressure

- Inadvertent S-signal
  - **Mitigation**: None required (see below)
  - **Safety by Design**: The EHRS does not inject water, but only removes heat from the reactor via the Steam generators. EBTs can only inject a limited amount of water (EBTs are sized for boration, not injection)
  - **Effect**: Inadvertent S-Signal cannot cause reactor overfill or over-pressure

INTERNATIONAL REACTOR INNOVATIVE AND SECURE
OBJECTIVE A: TESTING GUIDANCE

PRE-APPLICATION SCOPE

- NRC review of IRIS proposed approach to testing (limiting long lead items)

SCOPE

- To develop a complete set of Evaluation Models for IRIS, according to guidelines/approach in DG-1120 (Evaluation Model Development and Assessment Procedure, EMDAP)

1.	Identify and Rank Phenomena (PIRT) Scaling Test Requirements (Define Test Program) Facilities Identification	Pre-Licensing Application Phase I
2.	Testing Campaign Select/Validate Codes Finalize Evaluation Models	Pre-Licensing Application Phase II
3.	Safety Analyses	Design Certification

IDENTIFY AND RANK PHENOMENA (PIRT)

- **A phenomena identification and ranking activity is in progress**
- **Three PIRTs have been developed: SBLOCA, Containment and Transients**
- **AP1000 used as a basis for PIRTs development (Transients PIRT in particular)**
- **PIRT assessment is being used in preliminary selection of codes, assessment of applicability to IRIS of AP1000 evaluation models, testing campaign definition**

SCALING METHODOLOGY

- The Hierarchical, Two-Tiered Scaling Analysis Methodology (NUREG/CR5809, Zuber, 1991) is adopted to support and confirm the PIRTs

Stage 1 **SYSTEM DECOMPOSITION**	**Stage 2** **SCALE IDENTIFICATION**	**Stage 3** **TOP-DOWN/SYSTEM SCALING ANALYSIS**	**Stage 4** **BOTTOM-UP/PROCESS SCALING ANALYSIS**
PROVIDE: System hierarchy **IDENTIFY:** Characteristic: Concentrations Geometries Processes	**PROVIDE HIERARCHY FOR:** Volumetric concentrations Area concentrations Residence times Process time scales	**PROVIDE:** Conservation equations **DERIVE:** Scaling groups and Characteristics time ratios **ESTABLISH:** Scaling hierarchy **IDENTIFY:** Important process to be addressed in bottom-up/ process scaling analyses	**PERFORM:** Detailed scaling analysis for important local processes **DERIVE AND VALIDATE:** Scaling groups

FOUR CATEGORIES OF TESTS

1. Basic research experiments
   (e.g. 2-phase flow regimes in inclined helical SG tubes)
2. Engineering tests for components design verification
   (e.g. manufacturability of steam generators, pumps)
3. **Separate Effects Tests: verification on scaled models with proper boundary conditions**
4. **Integral Effects Tests: system verification on scaled models**

- **Categories 3 and 4 are required for design certification**
- **Definition of IRIS SETs and IETs will be an outcome of this pre-application**

IRIS - TESTING STRATEGY

- **New testing is required:**
  - **Where data do not exist (new phenomena)**
    - » Quantify safety related performance of new components and systems
    - » Provide data to verify/validate computer codes used in safety analyses
  - **To demonstrate performance/reliability of new components**
- **Rely on existing test data (like AP600) where possible, e.g.**
  - **Emergency Boration Tank (AP600 CMT data)**
  - **EHRS HX heat transfer (W PRHR Hx, GE, and other data)**
  - **Performance of spargers for the suppression pool and ADS (AP600 ADS data)**
  - **Gravity makeup (AP600, OSU and SPES2 integral test data)**

TEST FACILITIES

IRIS Consortium Members Facilities Access

Identified

- **ANSALDO: Steam Generator Mockups**
- **Washington Group EMD: Pumps**
- **SIET (POLIMI): IET and/or SETs**

Under Consideration

- **Westinghouse**
- **ORNL**
- **CNEN**
- **Universities**

Possible Additions

- **PSI, Switzerland (PANDA)**
- **OKBM, Russia (Integral Reactor)**
- **OSU, USA (APEX, MASLWR)**

INTERNATIONAL REACTOR INNOVATIVE AND SECURE

OBJECTIVE B: RISK-INFORMED APPLICATION TO IRIS LICENSING/DESIGN

PRE-APPLICATION SCOPE

- **NRC review of IRIS proposed approach to licensing (combining defense in depth and risk-informed)**

BACKGROUND/BASIS OF RISK-INFORMED LICENSING

- **NRC policies**
  - **Justify changes from current regulations and licensing**
  - **Defense-in-depth remains basis for design and safety reviews**
  - **Documented in NUREG-1226, SECY-97-287, SECY-98-300, SECY-99-264, SECY-01-0188, RG 1.174, RG 1.176 and DG 1110**
- **Industry Implementation and Developments**
  - **South Texas Project (Option 2), CEOG, PBMR and others**
  - **"New regulatory framework" (Risk-informed, Performance based) proposed in NEI 02-02**
  - **"Highly risk-informed" approach outlined by DOE NERI project for new plants**

EXAMPLE APPLICATIONS OF RISK-INFORMED LICENSING

- Emergency Response Planning Zone (Elimination of off-site emergency response is a DOE-Gen IV objective)
- Control room staffing requirements
- Reclassification, elimination, or re-definition of deterministic design basis events
- Severe Accident Mitigation Alternative (SAMA) Systems and/or Analyses

EXAMPLES OF POTENTIAL OF RISK-INFORMED LICENSING (Cont'd)

- Risk-Informed Technical Specifications
  - Integrated Leak Rate Test Interval Extension
  - Allowed Outage Time Extension
- Optimization of In-Service Inspection
- Risk-Informed Physical Security

“HIGHLY RISK-INFORMED” APPROACH PROPOSED FOR IRIS

- Review/determine applicability of current regulations to IRIS Safety-by-Design
- Establish high-level design/safety criteria
- Apply PRA to extent practical, iterate with deterministic design and testing results
- Resolve safety-related defense-in-depth issues with NRC staff

INTERNATIONAL REACTOR INNOVATIVE AND SECURE

CONCLUSIONS

SUMMARY

- IRIS design defined
- Safety "story" quite encouraging; detailed safety analyses in progress
- Early definition of testing necessary to maintain IRIS projected schedule
- Need to define approach to risk informed licensing

NEXT MEETINGS

- Dates to be determined, after Westinghouse submittal of preliminary:
  - Plant Description Document
  - Safety Assessment: study of Chapter 15 accidents, including ATWS
  - PIRT and Scaling Analysis
  - Comparison against current SRP