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U.S. EPR™ Diversity and Defense-in–Depth (D3) Analysis Methodology

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- **Introduction**
- **Meeting Objectives**
  - Review D3 analysis basic assumptions
  - Review analysis approach with specific examples
  - Provide NRC staff opportunity to discuss analysis methods with AREVA analysis personnel

U.S. EPR™ Diversity and Defense-in–Depth (D3) Analysis Methodology

Paul Bergeron
Safety Analysis

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D3 Analysis Methodology Presentation Outline

- Introduction
- Overall Analysis Approach
- Methodology Description
  - Non-LOCA
  - LOCA
  - Core Thermal-Hydraulics
- Specific Analysis Discussion
  - Evaluation (MSLB)
  - Non-LOCA (Increase in Steam Flow)
  - LOCA (Small Break)
- General Discussion

D3 Analysis Methodology Introduction

- **A software common cause failure (SCCF) of the digital Protection System (PS) is postulated**
  - **The SCCF is due to a latent software error in all redundant divisions of the PS that is triggered by an Anticipated Operational Occurrence (AOO) or a Postulated Accident (PA)**
  - **Failure of the PS is due to a common cause software failure; it is <u>not</u> a single failure**
  - **Partial PS failures are considered if activation of a PS function could lead to a more severe condition, e.g., MSIV closure results in isolating the Turbine Bypass System, RCSL**
  - **The Postulated SCCF is a low probability event and is evaluated as a *Beyond Design Basis Event (BDBE)***

Overall Analysis Approach

▶ Used BTP 7-19 as Guidance (Point 2)

- Evaluated each event in Chapter 15 of FSAR
- Relied on Diverse Actuation System (DAS)
- Considered operator action as diversity when time was sufficient
- Performed best estimate (realistic assumptions) analyses to demonstrate diversity
- Adopted BTP 7-19 Acceptance Criteria
  - AOOs: Radiological releases less than 10% of 10CFR100
  - PAs: Radiological releases less than 100% of 10CFR100
  - Integrity of the primary coolant pressure boundary maintained
  - Integrity of the containment maintained

Overall Analysis Approach (continued)

- **Evaluated each initiating event in U. S. EPR FSAR Chapter 15**
  - Assumed DAS functions developed for ATWS
  - Developed engineering arguments for events where BE assumptions and DAS functions bounded by Chapter 15 event response
  - Added DAS functions for D3 for some events to demonstrate acceptance criteria are satisfied
  - Analyzed events where it wasn't clear that the DAS functions and BE assumptions made the Chapter 15 analysis bounding

Overall Analysis Approach (continued)

- **DAS Functions**
  - RT Low SG pressure *
  - RT, Low SG Level *
  - RT, Low-Low RCS Flow (one loop) *
  - RT, Low RCS Flow (two loops) *
  - RT, High neutron Flux (power range) *
  - RT, Low hot leg pressure *
  - RT, High pressurizer pressure *
  - RT, High SG Level
  - Turbine trip on RT *
  - MFW isolation, High SG level (w/RT) affected SG
  - MFW isolation, Low SG pressure affected SG
  - EFW Actuation, Low SG Level *
  - SI Actuation/ SG partial cooldown (TBS) Low pressurizer pressure
  - MSIV Isolation, Low SG pressure *
  - Open H2 mixing dampers
  - Containment isolation, high containment activity

* Identified for ATWS

Overall Analysis Approach (continued)

- **Events analyzed**
  - Single MSIV Closure to evaluate need for High SG pressure trip
  - Increase in Steam Flow to evaluate effectiveness of high neutron flux trip
  - Complete Loss of Flow to confirm DNB margins
  - RCCA withdrawal and RCCA drop without low DNBR trip
  - RCCA ejection to evaluate effectiveness of high neutron flux trip
  - Boron dilution to determine BE response with manual rod control
  - SBLOCA to determine need for RCP trip and partial cooldown function
  - LBLOCA to confirm continuous RCP operation (no RCP trip) has negligible impact
  - Radiological analysis to determine need for automatic control room isolation

Overall Analysis Approach (continued)

- **Events Evaluated (Demonstrated to be acceptable without Specific Analysis)**
  - Decrease in Feedwater Temperature (Bounded by Increase in Steam Flow)
  - Increase in Feedwater Flow (DAS functions adequate)
  - Inadvertent opening of MSRT or MSSV (Bounded by Increase in Steam Flow, post trip bounded by MSLB)
  - MSLB (credit for best estimate neutronics, MTC, scram worth no stuck rod maintains core sub-critical)
  - Turbine trip (DAS functions adequate)
  - Loss of AC (adequate time 1.5 hours to actuate EFW pumps off diesels)
  - FWLB (DAS functions adequate)
  - Partial Loss of Flow (DAS functions adequate)
  - Seized Rotor (DAS functions adequate)

Overall Analysis Approach (continued)

- **Events Evaluated (Demonstrated to be acceptable without Specific Analysis) Continued**
  - Inadvertent Operation of SIS or EBS
  - CVCS Malfunction that Increases Inventory (operator has 24 minutes before pressurizer fills)
  - Inadvertent Opening of PSRV (DAS functions adequate)
  - SGTR
  - Containment Integrity
  - Radiological consequences

Overall Analysis Approach (continued)

- **Analysis Considerations**
  - **Use NRC approved S-RELAP5 code for plant system response**
    - **FSAR models adapted for BE (realistic assumptions) analyses**
    - **Minor enhancements to S-RELAP5 implemented for performance of BE analyses**
  - **Normal operating systems not actuated by PS remain available**
    - Examples
      - Main feedwater control
      - Main steam system, including turbine bypass
      - Other plant support systems continue to operate normally (HVAC, component cooling water, etc.
  - **PS functions fail except where action prevents availability of non-PS function (partial failure)**
    - Example
      - MSIV closure isolates Turbine Bypass System (SBLOCA)

Overall Analysis Approach (continued)

▶ Analysis Considerations (continued)

- **RCSL is assumed to function when the response could lead to more severe consequences (Not credited for mitigation)**
- **Automatic control functions**
  - Main feedwater flow control
  - Pressurizer pressure and level control
  - Main steam generator level and turbine load (pressure control) control
  - Main steam pressure control (Turbine Bypass)
  - EFW Flow control (limits flow to a depressurized SG)
- **Best Estimate flow characteristics for Mitigation Systems**
  - MHSI/LHSI
  - EFW

Overall Analysis Approach (continued)

- **Analysis Considerations (continued)**
  - **Plant Initial Conditions**
    - **Plant is operating at nominal conditions; rated reactor power, temperature and pressure (4590 MWt, Tavg 594 °F, 2250 psia)**
    - **All control rods withdrawn**
      - **Control rod withdrawal and rod ejection events evaluated from PDIL (Power Dependent Insertion Limit) to enable event**
    - **BE neutronics parameters and power distributions**
      - **Includes BE core decay heat**
      - **Equilibrium cycle basis**

Overall Analysis Approach (continued)

- **Analysis Considerations (continued)**

Parameter	Best Estimate (Equilibrium Cycle)	Chapter 15
MTC (pcm/°F) BOC, HFP EOC, HFP	-11.38 -39.4	0 -50
DTC (pcm/°F) BOC, HFP EOC, HFP	-1.40 -1.63	-1.17 -1.85
Scram (pcm) BOC, HFP EOC, HFP	9449 10349	6161 7353
Initial Core Power (MWt)	4590	4612
Tavg (°F)	594	594 ± 4
Pressure (psia)	2250	2250 ± 50
Reactor coolant System Flow Per loop (gpm)	124,741	119,692
Decay Heat	ORIGEN based [1]	ANS 1973
Fq[2] BOC EOC	1.695 (2.1 SBLOCA) 1.613	2.6
FΔH[2] BOC EOC	1.476 (1.557 SBLOCA) 1.425	1.70

1. 5% enrichment, 40 GWD/MTU including actinides
2. Limiting for all Cycles

Overall Analysis Approach (continued)

Analysis Considerations (continued)

	Cycle 1	Equilibrium Cycle
MTC (pcm/°F) BOC EOC	 -7.58 -32.83	 -11.38 -39.40
DTC (pcm/°F) BOC EOC	 -1.34 -1.61	 -1.40 -1.63
SCRAM (pcm) BOC EOC	 11,159 12,153	 9449 10,349

Overall Analysis Approach (continued)

- **Analysis Considerations (continued)**
  - Basic Analysis Assumptions
    - No concurrent Loss of Offsite Power (LOOP)
    - No single failures postulated
    - No concurrent preventive maintenance
    - No stuck control rods
    - Nominal control and trip setpoints
    - No manual operator actions targeted before 30 minutes
  - Plant End State
    - Post-trip mitigation actions completed
    - Plant in stable controlled condition (e.g., hot standby for non-LOCA events)

Overall Analysis Approach (continued)

- **Analysis Considerations (continued)**
  - **Manual Functions Credited**
    - RT
    - Diesel generator loading
    - EFW actuation
    - Operation of EFW for long-term level control
    - SI switchover to hot leg injection
    - MSIV closure
    - Feedwater Isolation (MFW and EFW)
    - Initiation of MHSI
    - Control of MHSI
    - Extended partial cooldown
    - Actuation of EBS
    - Control Room HVAC reconfiguration
    - CVCS isolation (24 minutes initiation time)
    - MSRT

Overall Analysis Approach (continued)

- AREVA U.S. EPR™ D3 Acceptance Criteria Targets
  - AOOs: Conservative target
    - Show that DNB (Departure from Nucleate Boiling) will not occur; therefore, FSAR radiological dose analysis remains bounding
  - PAs: Conservative target
    - Show that fuel failures in FSAR remain bounding; therefore, FSAR radiological dose analysis remains bounding
  - Maintain peak RCS pressures below 120% of design pressure (consistent with ATWS)
  - Maintain containment pressure below structural integrity limit for ultimate pressure capacity (consistent with SRP 3.8.1)
  - For LOCA, maintain PCT below 2200°F

Methodology Description

- S-RELAP5 Code Enhancements
- LOCA
- Non-LOCA
- Core Thermal Hydraulics

U.S. EPR™ D3 S-RELAP5 Code Enhancements

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LOCA Analysis

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S-RELAP5 Code Enhancements

Heat Transfer Modifications

- "S-RELAP5 computes heat transfer by using a nodal model to represent the fluids and heat structures. The current modeling computes heat transfer between the surface of a heat structure and the fluid node by using the fluid temperature at the exit of each node, Tj. That temperature approximation is revised for a best estimate version of S-RELAP5 (ujan08) by using a fluid temperature that is the average of the inlet and exit fluid temperatures, (Tj+Tj-1)/2."
- The Inayatov multiplier formulation is taken from RELAP5/MOD3.3. It offers a more realistic BE computation of the heat transfer coefficient on the shell side of tube arrays. It is new to S-RELAP5 (ujan08) and it is applied to single-phase and two-phase (Chen macro) forced convection heat transfer
- Add ability to multiply LIQHTC

S-RELAP5 Code Enhancements (continued)

- **The following enhancements do not impact the computation of S-RELAP5. These have been added to facilitate input and evaluation of output.**
  - **Reactor Kinetics**
    - Addition of boron reactivity feedback to the SEPARABL reactivity feedback option
    - Addition of the capability to enter reactivity values in units of percent milli (pcm)
    - Addition of the capability to enter a second set of reactivity tables (i.e. for moderator density, fuel temperature, and boron density), that replace the original set of tables when a specified trip occurs
    - Addition of reactivity feedback variables for moderator (rkmodfbk), Doppler (rkdopfbk), boron (rkborfbk), and scram (rkscrfbk)
  - **Additional Enhancements**
    - Addition of the capability to enter time-dependent junction input in volumetric flowrate units
    - Addition of the capability to enter maximum opening and closing rates for servo valves.

S-RELAP5 Code Enhancements (continued)

Verification & Validation

All changes were verified

- Heat Transfer Validation – Model validated with a simple steam generator which has a theoretical and a nodal solution. Verification provided for both shell & tube sides and for forward and reverse flow.
- Validation of Inayatov enhancement – The enhanced correlations (single-phase and two-phase forced convection) in S-RELAP5 were compared with calculated correlations in an Excel spreadsheet.
- Verification of Addition of Boron Reactivity Feedback to SEPARABL option;
  - HBR SBLOCA Model
  - Modified Edward's Pipe

S-RELAP5 Code Enhancements (continued)

Verification & Validation (continued)

Verification of:

- Reactivity Table Input Units (pcm=1.0E-5 beta)
  [convert to $ = (σ (pcm) .10^{-5})/β]
- Second Set of Reactivity Feedback Tables Activated By Trip
- Time Dependent Junction Input Units (gpm or cmph)
- Additional Minor Edit Variables (moderator, Doppler,boron & scram reactivity feedback)
  - Use HBR SBLOCA Model

Verification of Servo Valve Rate Limits

- Simple S-RELAP5 model

U.S. EPR™ D3 LOCA Analysis Methodology

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LOCA Methodology Description

- **Large Break**
  - No changes from FSAR
- **Small Break**
  - Nodalization Changes
  - Modeling and Input Changes

SBLOCA Methodology Description

- **Nodalization Changes**
  - Additional detail has been added to the reactor vessel upper head and upper plenum nodalization
  - Additional detail has been added to the pressurizer dome nodalization
  - Steam generator swirl-vane separators and dryers component nodalization has been modified

SBLOCA Methodology Description

Modeling and Input Changes

- RCPs operate at rated speed with no SG tube plugging
- SG Blowdown is represented (drawn from the axial economizer hot side)
- No automatic pump trip
- Only DAS Functions available for Protection
- Charging System available for make-up
- Initial Conditions Nominal Full Power
- Fq = 2.1, FΔH = 1.557
- Decay Heat based on ORIGEN versus ANS1973 plus 20%
- Normal operating plant systems not actuated by PS remain available
  - Main Feedwater control
  - Turbine Bypass (MSIVs close on containment 1st stage pressure)
  - Component Cooling
- All Four trains of SI (MHSI & LHSI) available at BE flow
- Long-term operator action to open MSRTs

U.S. EPR™ D3 Non-LOCA Analysis Methodology

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Non-LOCA Analysis

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Non-LOCA Analysis Methodology

- **Nodalization changes between FSAR and D3 analysis models**
  - Core model nodalization modified for more realistic representation of assymetric events
  - Reactor vessel upper head nodalization modified to allow for more realistic evaluation of events that may result in upper head voiding
  - Steam generator swirl-vane separators and dryers component modeling modified to be more realistic for the evaluation of events that may result in steam generator overfill

Non-LOCA Analysis Methodology (continued)

Model Changes / Additions

- **Steam Generator Level Indication**
  - Best estimate SG level (narrow range and wide range) based on differential pressures between the lower and upper tap locations
  - Chapter 15 SG level based on collapsed liquid level
- **Pressurizer pressure and level control systems always active**
- **Steam Generator Blowdown System Added**
- **Emergency Feedwater Steam Generator Level Control System Added**
- **Turbine Bypass System Added**
- **Chemical Volume and Control System functions**
  - Makeup and letdown functions added
  - Auxiliary sprays added
- **Main Feedwater System Details Added**
  - FLCV, FLIC, LLCV, LLIV, VLLCV, and MIV individually modeled and controlled

U.S. EPR™ D3 Core Methodology

Michael Bradbury
Core T-H Analysis

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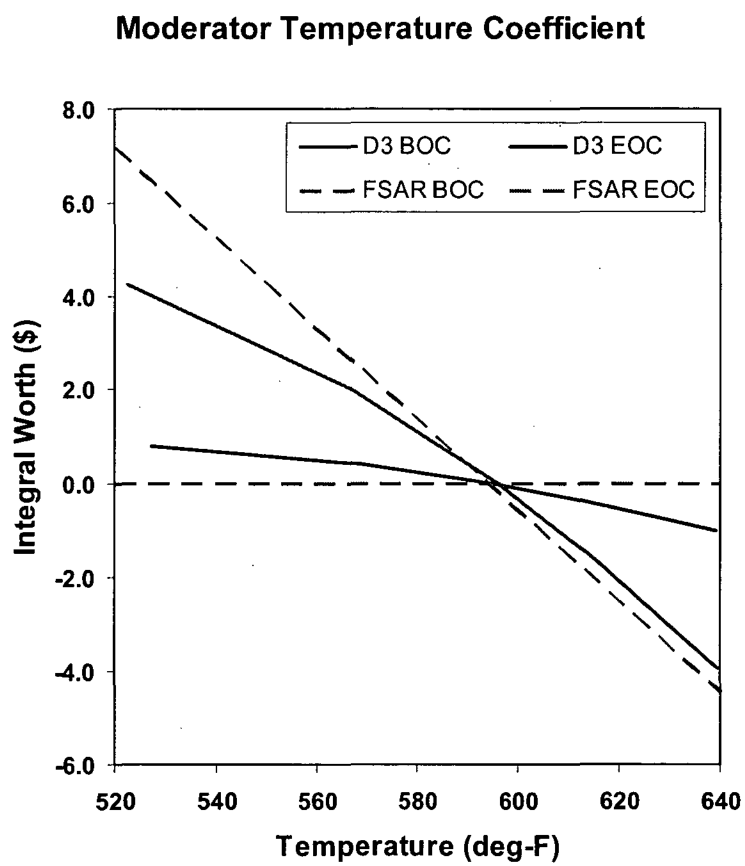
Core Methodology
Neutronics Parameters

- Moderator Temperature Coefficient
- Doppler Temperature Coefficient
- SCRAM reactivity curves

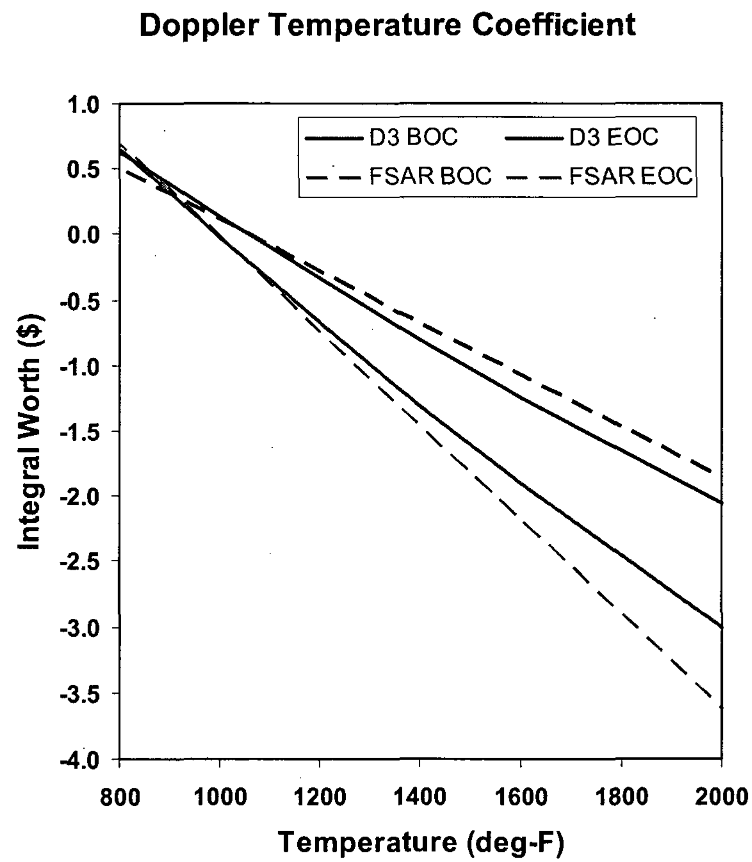
Core Methodology
Moderator Temperature Coefficients

- **D3**
  - **no biasing**
  - **table of $ vs. moderator density**
- **FSAR**
  - **biased by 2 pcm/°F**
  - **BOC – most positive bounding value**
  - **EOC – most negative bounding value**
  - **bounding constant pcm/°F**
    - BOC → 0 pcm /°F
    - EOC → -50 pcm /°F



Core Methodology
Doppler Temperature Coefficients

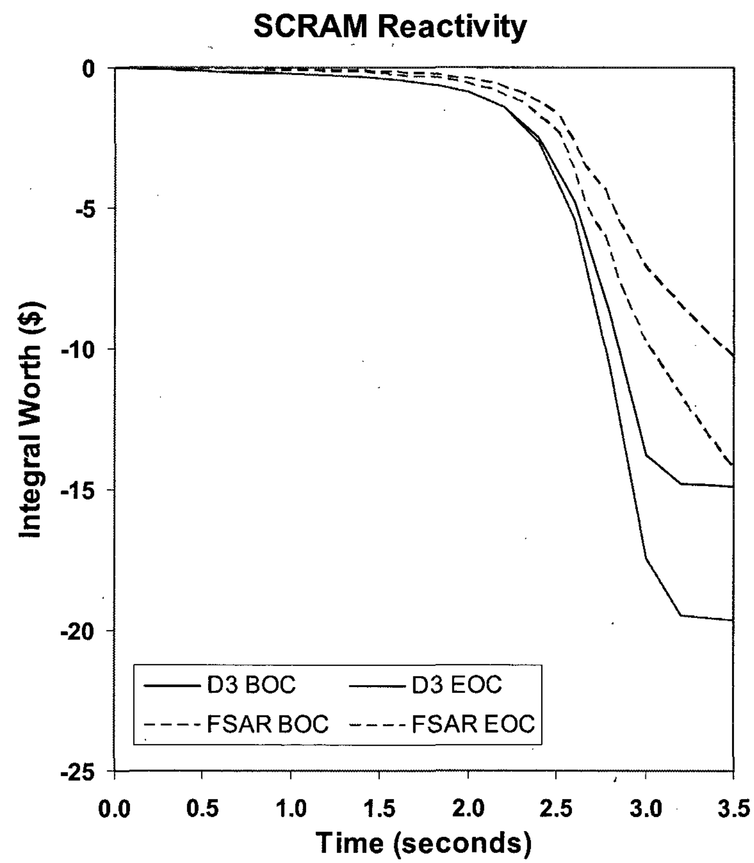
- **D3**
  - **no biasing**
  - **table of $ vs. fuel temperature**
- **FSAR**
  - **biased by 10%**
  - **bounding constant pcm/℉**
    - BOC → -1.17 pcm /℉
    - EOC → -1.85 pcm /℉



Core Methodology
SCRAM Reactivity

TOTAL WORTH

- **D3**
  - BOC: 9449 pcm ($14.87)
  - EOC: 10349 pcm ($19.63)
  - Start drop from bite position
  - All rods are dropped
  - Worth taken from EQ cycle
- **FSAR**
  - BOC: 6161 pcm ($10.35)
  - EOC: 7353 pcm ($14.28)
  - Start drop from PDIL
  - MRR is withdrawn (stuck)
  - Worth bounds all cycles (minimum)



Core Methodology
Thermal-Hydraulic Parameters

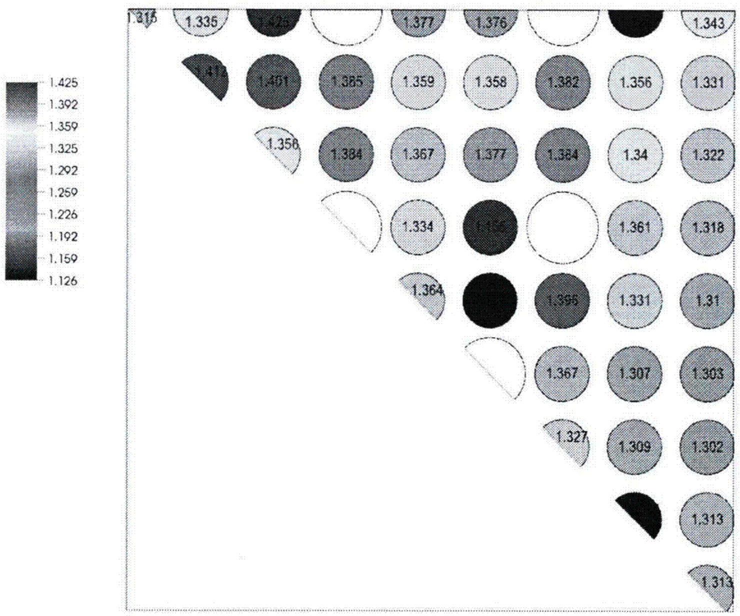
- Power Distributions
- Core Related Uncertainties

Core Methodology Power Distributions

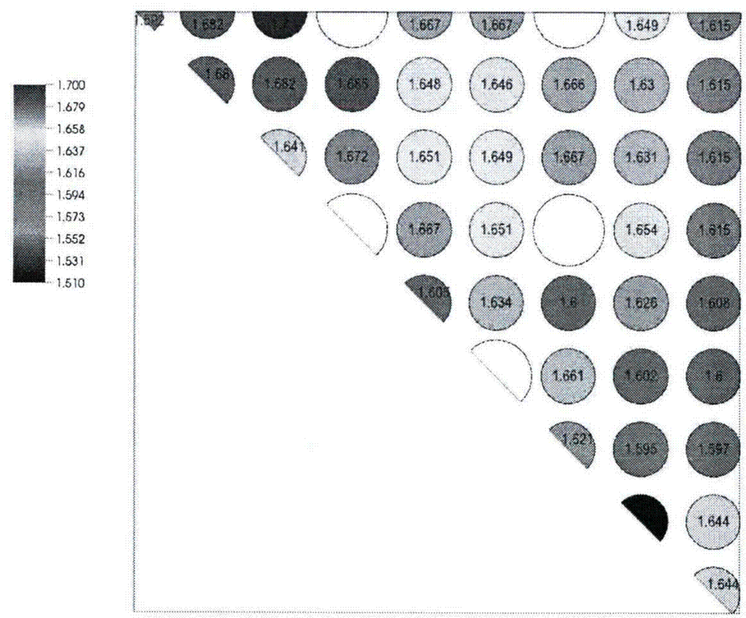
- **D3**
  - **Single axial and radial power distribution for BOC & EOC**
  - **Select power distribution from depletion (no xenon oscillations)**
  - **Select limiting assembly (assembly with greatest $F_{\Delta H}$) from PRISM**
    - Both 18 & 24-month designs
    - All cycles considered
  - **Retrieve pin-to-pin $F_{\Delta H}$ distribution from CASMO for limiting assembly**
  - **Select core average axial flux shape from PRISM (consistent with $F_{\Delta H}$)**
- **FSAR**
  - **Most analyses credited the Low DNBR Channel PS function**
  - **>2000 power distributions considered (including xenon oscillations, single RCCA withdrawal, RCCA misalignments, etc.)**
  - **When the Low DNBR Channel PS function was not credited a DNBR limiting axial and radial power distribution was utilized**

Core Methodology
Radial Power Distribution

D3



FSAR

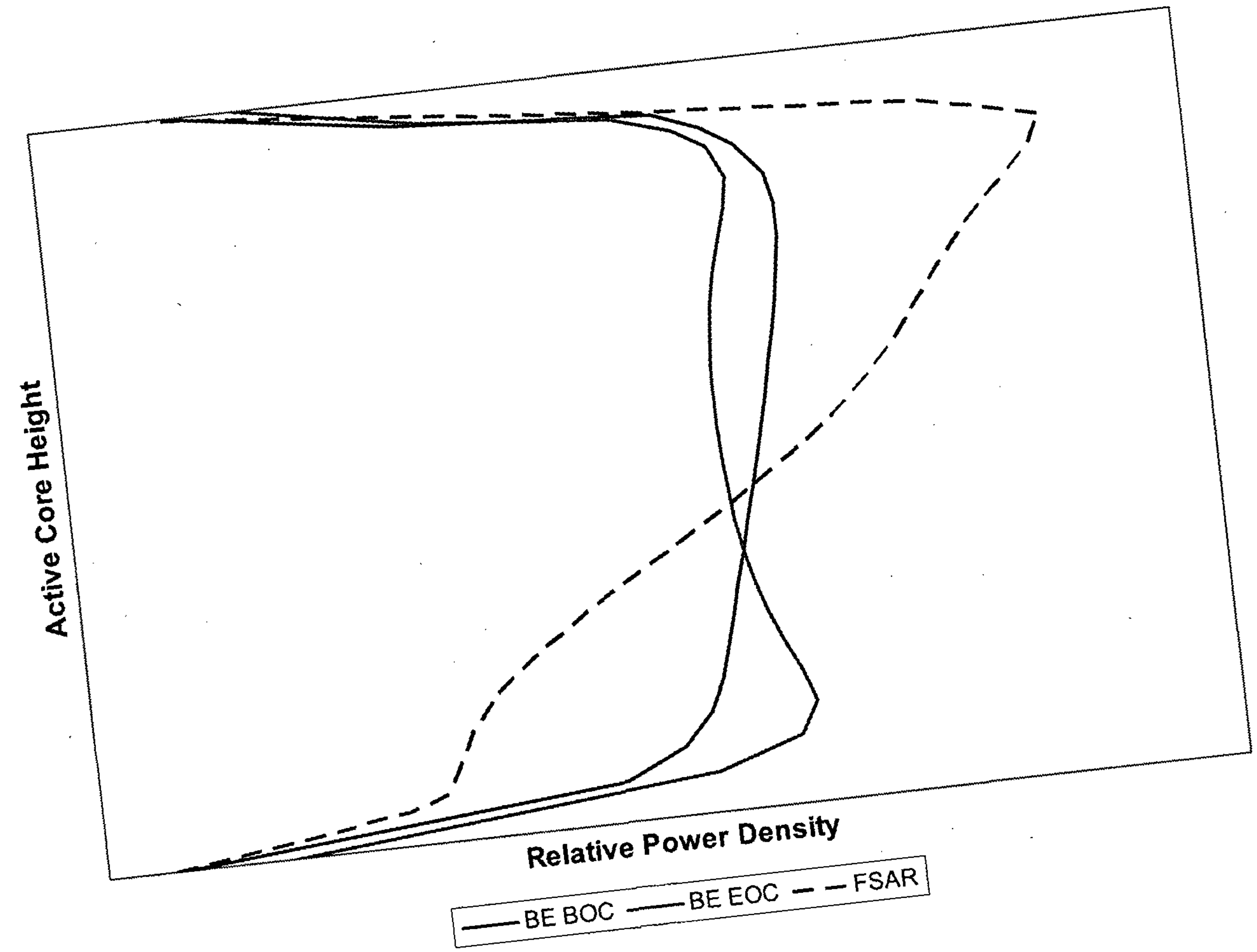


Power Limiting Assemblies (1/8 assembly)

Peak $F_{\Delta H}$ = 1.425
Peak-to-Ave$_{assm}$ = 1.072

Peak $F_{\Delta H}$ = 1.700
Peak-to-Ave$_{assm}$ = 1.038

Core Methodology
Axial Power Distribution



Core Methodology
Core Related Uncertainties

	D3	FSAR
Cold leg temperature measurement		✓
Pressurizer pressure measurement		✓
Inlet flow measurement		✓
SPND measurement		✓
CHF correlation	✓	✓
Low DNBR channel algorithm		✓
Power calorimetric		✓
Reconstructed power distribution		✓
Analytical uncertainties		✓
SPND calibration frequency		✓
Manufacturing deviations		✓
Assembly bow		✓
Rod bow		✓

U.S. EPR™ D3 Specific Analysis Discussion

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Specific Analysis Discussion

- **Event Assessment**
  - **Non-LOCA**
    - Evaluation (MSLB)
    - Analysis (Increase in steam flow)
  - **LOCA**
    - Analysis (Small Break)

MSLB Evaluation Example

FSAR

- **Includes a spectrum (break size, initial power, for both pre and post RT conditions)**
- **Pre-RT**
  - FSAR break sizes considered 10%, 50%, and 100% of steam line area
  - RT occurs on High Core Power Level, Low DNBR, or High SG Pressure Drop
  - Timely RT occurs and acceptance limits are met
- **Post-RT**
  - FSAR considers spectrum breaks and initial power levels including HFP and HZP
  - Limiting case is HZP with a return to power with minimal fuel damage
  - Offsite doses are within acceptable limits

MSLB Evaluation Example (continued)

- **D3**
  - **DAS RTs available on high neutron flux and low SG pressure**
  - **Pre-RT**
    - Smaller breaks that trip on neutron flux are covered by increase in steam flow event
    - Neutron flux is de-calibrated due cooldown and results in peak core power
    - Larger breaks quickly lead to a RT and MSIV closure on low SG pressure
  - **Post-RT**
    - MSIV closure and MFW isolation occur on low SG pressure (DAS)
    - EFW actuates on low SG level and terminated by operator in 30 minutes
    - MTC is dominant parameter in determining return to power and potential fuel damage
    - BE neutronics (MTC, scram worth, no stuck rod) results in significant SDM
    - ARI with BE neutronics core needs to cool to 105 °F before return to critical, well below Tsat at atmospheric pressure
    - No return to power, no fuel damage and no offsite consequences
  - **Following SG dry-out (MFW and EFW isolation) long-term heat removal occurs via unaffected SGs (using MFW or EFW venting steam through MSRTs)**

U.S. EPR™ D3 Non-LOCA Analysis Example

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Core T-H Analysis

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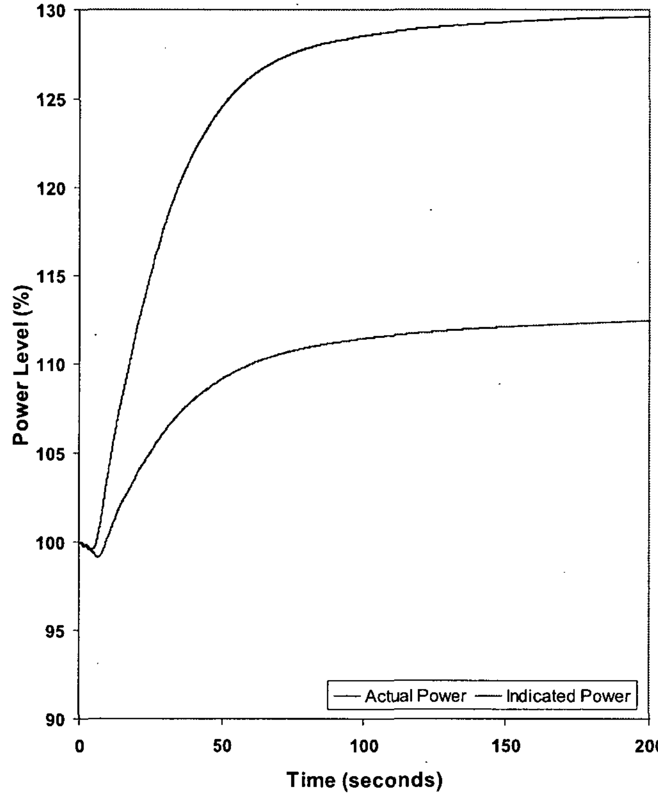
Non-LOCA Analysis Example
Increase in Steam Flow

- **FSAR**
  - **Spectrum of cases**
    - HFP, 25%, and HZP
    - BOC and EOC
    - Increase in steam flow 10% to 60 % (6 turbine bypass valves)
    - With and without LOOP
    - With and without automatic rod control
    - Considers feedwater heater train out of service
  - **Trips**
    - Small increases (<20%) → High Core Power Level or new steady state
    - Intermediate increases (20% - 50%) → Low DNBR, large increases (60%) → SG Pressure Drop or Low SG Pressure
  - **EOC – RT issued from Low DNBR channel PS function at ~7 seconds**
  - **BOC – RT on High SG Pressure Drop PS function (~6 seconds)**
    - 0 MTC results in no power increase due to cooldown
    - ACT is credited because it will withdraw RCCA in an attempt to increase T_{ave}

Non-LOCA Analysis Example
Increase in Steam Flow (cont.)

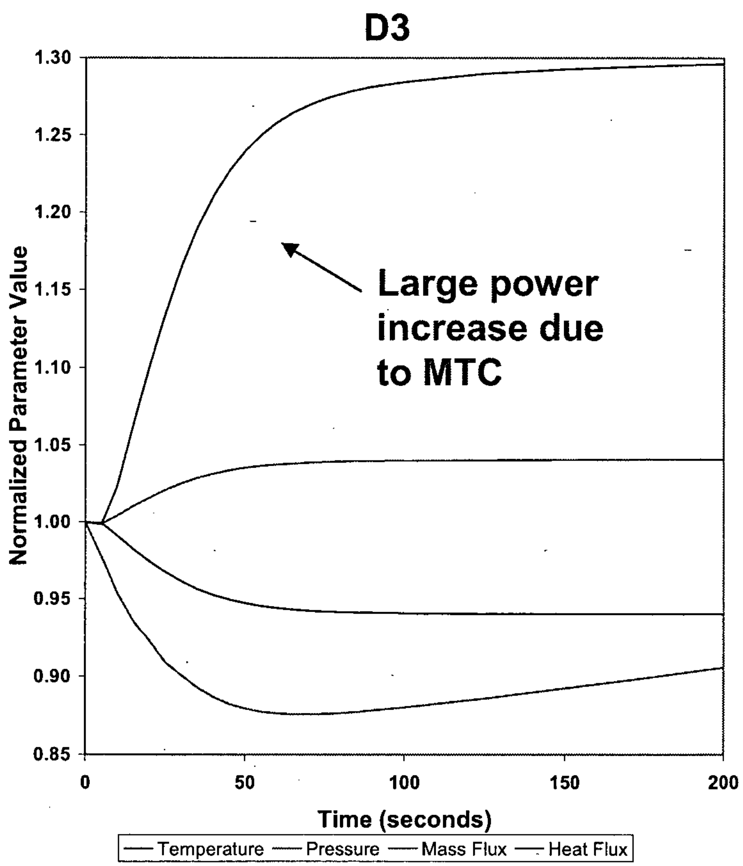
- **D3**
  - **Spectrum of cases at HFP**
    - BOC and EOC
    - No LOOP
    - Automatic rod control available until RT
    - All 6 valves on TBS inadvertently open (results in peak power)
  - **EOC – RT issued from High Neutron Flux function in DAS at ~21 seconds**
  - **BOC – No RT issued**
    - BOC conditions allow slower ascension in power than EOC (less negative MTC)
    - slow ascension in power allows larger cooldown
    - larger cooldown leads to more attenuation in the downcomer
    - more attenuation in the downcomer leads to a lower power reading from the excore detectors
  - **BOC considered more limiting for DNB and LHGR → EOC not analyzed**

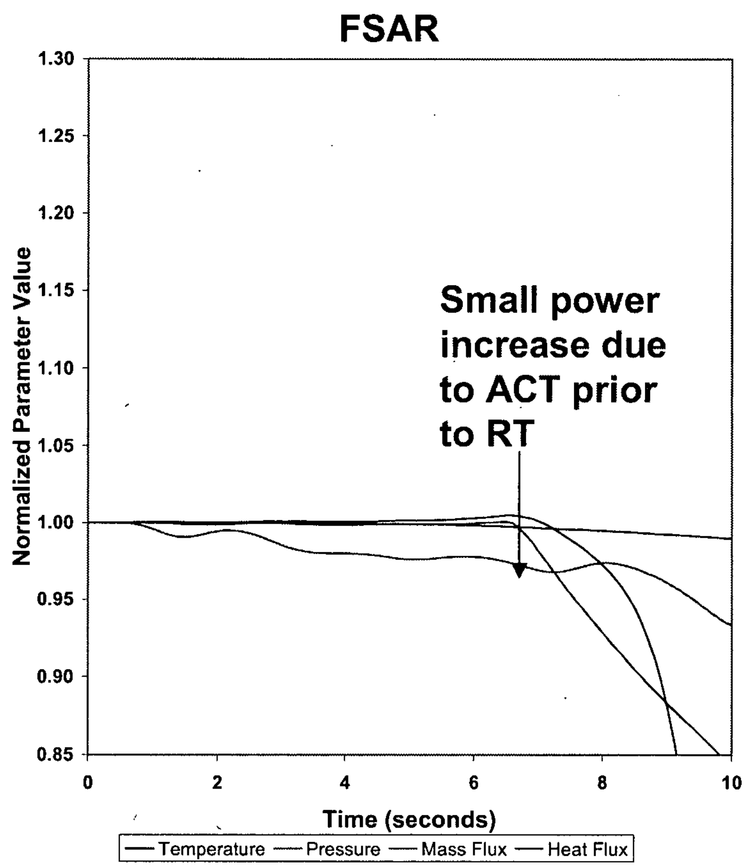
Actual vs. Indicated Power



BOC, 60% Steam Flow Increase Case

Non-LOCA Analysis Example
Increase in Steam Flow (cont.)

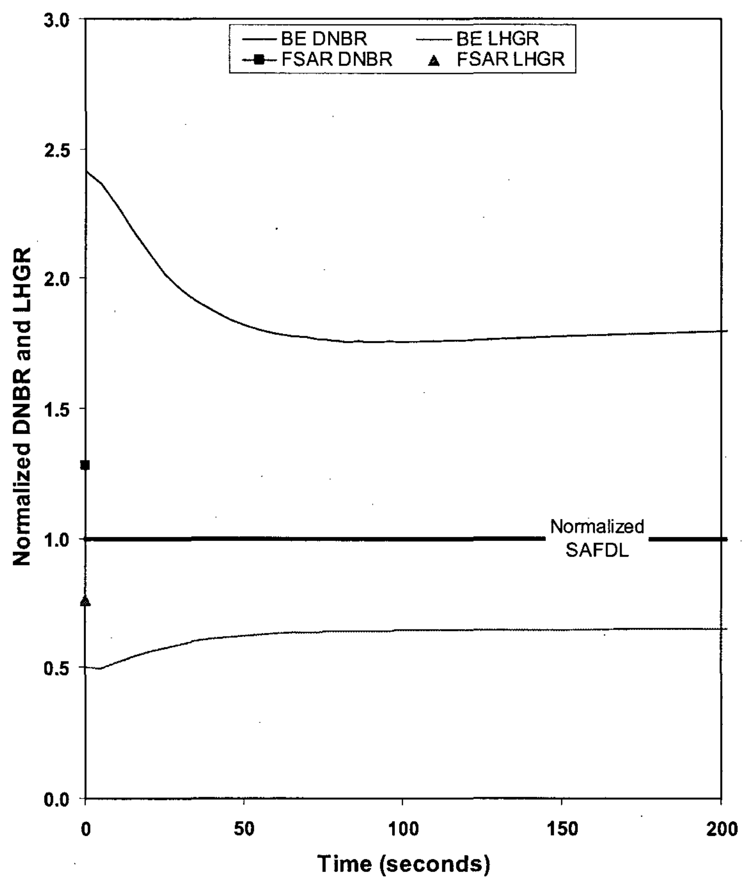




BOC, 60% Steam Flow Increase Case

Non-LOCA Analysis Example
Increase in Steam Flow (cont.)

- Increase in steam flow overcools the primary side
- Decrease in core inlet temperature leads to:
  - increase in power ↓DNB / ↑LHGR
  - decrease in pressure ↓DNB
  - increase in mass flux ↑DNB
- D3
  - Initial margins provide protection for DNB and LHGR
- FSAR
  - Negligible power increase and rapid RT lead to little degradation of DNB or LHGR



BOC, 60% Steam Flow Increase Case

U.S. EPR™ D3 LOCA Analysis Example

Liliane Schor
LOCA Analysis

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SBLOCA Analysis Example

- **PS protection (FSAR)**
  - RT on low RCS pressure
  - SIS actuation on low RCS pressure and partial cooldown (MSRT)
- **DAS protection (D3)**
  - RT on low RCS pressure
    - DAS setpoint selected to allow PS actuation first, while providing adequate protection
  - SIS actuation on low RCS pressure
    - DAS setpoint selected to allow PS actuation first, while providing adequate protection
- **Analysis assumptions**
  - BE decay heat and break flow model (HEM)
  - All four ECCS trains and EFW trains available with BE flows
  - No turbine bypass or MSRT
  - MFW isolated in high containment pressure
  - No automatic pump trip

SBLOCA Analysis Example
(continued)

- **No single failure or preventive maintenance**
- **No Loss of Offsite Power, resulting in continued RCP operation**
- **Post-RT response is controlled**
  - Heat removal by steam generators only through MSSVs
  - RCS makeup by MHSI/LHSI and charging pumps
  - Containment isolation by DAS
- **Sensitivities performed to evaluate effect of RCP trip**
  - Cover full break spectrum of SBLOCAs
  - Cover RCP trip times up to 30 minutes

SBLOCA Analysis Example
(continued)

Parameter	Best Estimate	Chapter 15
Decay Heat	Origen based-40 Gwd/Mtu, 5% enrichment	1971/1973 *1.2
Core Bypass Flow	Heavy Reflector+baffle: 1.04% CRGT and Thimbles: 2.4% Upper Head Spray: 0.36% Total Core Bypass: 3.8%	Heavy Reflector+baffle: 1.24% CRGT and Thimbles: 3.93 % Upper Head Spray: 0.33% Total Core Bypass: 5.5%
MSRT Opening	N/A	1414.9 psia
CVCS Flow	1 pump: 176 gpm 2pumps: 335 gpm	N/A
Scram (pcm)	10349 (EOC)	6161 the minimum between all cycles

SBLOCA Analysis Example
(continued)

DAS Functions

- RT occurs Low Hot Leg Pressure DAS trip
- Turbine Trip (TT) on RT
- SIS actuation on low pressurizer pressure: 4-trains MHIS / LHSI / Accumulators inject into the cold leg when the primary pressure reaches their respective shutoff head
- MSRT are not available for partial cooldown from the DAS.
- Turbine Bypass System (TBS) is not available to perform a partial cooldown due to MSIV closure on high containment pressure (PS partial failure). The analysis assumes MSIV isolation at RT.
- Full MFW isolation occurs on high containment pressure. The analysis assumes MFW isolation at RT.
- EFW actuation is available on low SG level; no EFW isolation function available in DAS.

SBLOCA Analysis Example Results

- **Breaks divided into 3 categories:**
  - **Very small breaks 1.0 – 2.5 inch ID**
    - Rely on SG to remove energy through MSSVs
    - Tripping the RCPs has no impact on PCT due to high inventory
    - Primary pressure remains above the MHSI shutoff head until EFW actuation or loop seal clears
    - EFW actuated on low SG inventory
      - A CONMAS multiplier set to 0.01
      - EFW injected at T_{sat} (558°F)
    - For 1.0 inch breaks, the loop seal may take hours to clear; operator will need to take manual action to perform cooldown through the MSRT
  - **Intermediate breaks 3.0 – 6.0 inch ID**
    - Rely on SG and break to remove energy
    - At the time of loop seal clearing, the primary pressure drops below SG secondary pressur
    - EFW does not initiate since SG level does not drop to the EFW initiation setpoint
  - **Larger breaks 6.5 – 9.7 inch ID**
    - The primary pressure drops below the secondary pressure early in the event
    - The MSSVs do not open
    - More significant core uncovery, however the MHSI, LHSI and accumulators inject, short core uncovery time

SBLOCA Analysis Example Conclusions

- The results of the analysis show that for all SBLOCA cases analyzed the maximum PCT is 1260°F
- The DAS architecture does not require the incorporation of an automatic RCP trip or the incorporation of automatic MSRT partial cooldown in order to mitigate the SBLOCA event

Overall D3 Conclusions

- AREVA D3 analysis methodology is systematic and comprehensive
- Analysis is consistent with BTP 7-19 and DI&C-ISG-02
- Results of analysis methodology provide a basis for design of an effective diverse actuation system (DAS) ANP-10304 Rev 1 Appendix A

UNITED STATES NUCLEAR REGULATORY COMMISSION

U.S. Nuclear Regulatory Commission

MEETING WITH AREVA NP, INC.

Discussion of June 2010 Diversity and Defense-in-Depth Request for Additional Information

Two White Flint, Rockville, Maryland

July 22, 2010

Agenda

- Objectives
- Diverse Actuation System Functionality
- Diverse Actuation System Implementation
- Diverse Actuation System Diversity Guidance
- Diverse Actuation System DAS Performance Guidance
- Addressing I&C Vulnerabilities
- Questions / Comments
- References

Objectives

- Discuss:

  -Diverse Instrumentation and Control Regulations and Guidance

  -Diversity and Defense-in-Depth Analysis Guidance

  -Diversity Best Estimate, Deterministic Analysis, Limits and Goals

Diverse Actuation System Functionality

- **10 CFR part 50, Appendix A, General Design Criterion 22, Protection System Independence**

  The protection system shall be designed to assure that the effects of postulated accident conditions on redundant channels do not result in loss of the protection function. Design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function.

- **SRM to SECY-93-087, Item II.Q, Defense Against Common-Mode failures in Digital Instrumentation and Control Systems**

  If a postulated common-mode failure could disable a safety function, then a diverse means, with a documented basis that the diverse means is unlikely to be subject to the same common-mode failure, shall be required to perform either the same function or a different function.

DAS Implementation

- **10 CFR 50.62, Requirements for Reduction of Risk from Anticipated Transients Without Scram (ATWS) Events for Light-water-cooled Nuclear Power Plants**

  - Each pressurized water reactor must have equipment from sensor output to final actuation device, that is diverse from the reactor trip system....

  -This equipment must be designed to perform its function in a reliable manner....

- <u>**Quality Guidance**</u>

  SRM to SECY-93-087, Item II.Q, Defense Against Common-Mode failures in Digital Instrumentation and Control Systems

  The diverse or different function may be performed by a non-safety related system if the system is of sufficient quality to perform the necessary functions under the associated event conditions.

  Generic Letter 85-06, Quality Assurance Guidance for ATWS Equipment that is not Safety-Related

  Explicit quality assurance guidance is provided in its enclosure. The lesser safety significance of diverse equipment (i.e., ATWS equipment) necessarily results in less stringent quality assurance guidance.

DAS Implementation continued

BRANCH TECHNICAL POSITION 7-19, GUIDANCE FOR EVALUATION OF DIVERSITY AND DEFENSE-IN-DEPTH IN DIGITAL COMPUTER-BASED INSTRUMENTATION AND CONTROL SYSTEMS

Design Diversity

- Design diversity
- Equipment diversity
- Functional diversity
- Human diversity
- Signal diversity
- Software diversity

Diversity Performance Guidance

Anticipated Operational Occurrences

For each AOO in the design basis occurring in conjunction with each single postulated CCF, the plant response calculated using best estimate analyses should not result in radiation release exceeding 10 % of the 10CFR100 guideline or violation of the integrity of the primary coolant pressure boundary.

Postulated Accidents

For each postulated accident in the design basis occurring in conjunction with each single postulated CCF, the plant response calculated using best-estimate analyses should not result in radiation release exceeding the 10CFR100 guideline or violation of the integrity of the primary coolant pressure boundary, or violation of the integrity of the containment (i.e., exceeding coolant system or containment design limits).

Diversity Guidance

NUREG/CR-6303 Diversity Attributes

Design Diversity	Equipment Diversity	Functional Diversity	Human Diversity	Signal Diversity	Software Diversity
Different technologies	Different manufacturers of fundamentally different designs	Different underlying mechanism	Different design organization	Different reactor or process parameters sensed by different physical effects	Different algorithms, logic, and program architecture
Different approaches within a technology	Same manufacturer of fundamentally different designs	Different purpose, function, control logic, or actuation means	Different engineering management team within the same company.	Different reactor or process parameters sensed by the same physical effect	Different timing, order of execution
Different architecture	Different manufacturers making the same design	Different response time scale	Different designers, engineers, or programmers.	The same reactor or process parameter sensed by a different redundant set of similar sensors	Different operating system
	Different versions of the same design		Different testers, installers, or certification personnel.		Different computer language

Diverse Actuation System Performance Guidance

- **SRM to SECY-93-087, Item II.Q, Defense Against Common-Mode failures in Digital Instrumentation and Control Systems**

  In performing the assessment, the vendor or applicant shall analyze each postulated common-[cause] failure for each event that is evaluated in the accident analysis section of the safety analysis report (SAR) using best-estimate methods. The vendor or applicant shall demonstrate adequate diversity within the design for each of these events.

Diverse Actuation System Performance Guidance continued

Anticipated Operational Occurrences

For each AOO in the design basis occurring in conjunction with each single postulated CCF, the plant response calculated using best estimate analyses should not result in radiation release exceeding 10 % of the 10CFR100 guideline or violation of the integrity of the primary coolant pressure boundary.

Postulated Accidents

For each postulated accident in the design basis occurring in conjunction with each single postulated CCF, the plant response calculated using best-estimate analyses should not result in radiation release exceeding the 10CFR100 guideline or violation of the integrity of the primary coolant pressure boundary, or violation of the integrity of the containment (i.e., exceeding coolant system or containment design limits).

DAS Performance Guidance continued 10CFR100 Limit Requirements

- **10CFR100.21(c)(1)**

  (c) Site atmospheric dispersion characteristics must be evaluated and dispersion parameters established such that:

  (1) Radiological effluent release limits associated with ***normal operation*** from the type of facility [i.e., stationary power reactor] proposed to be located at the site can be met for any individual located offsite

- **10CFR34(a)(ii)(D)**

  (D) ... Special attention must be directed to plant design features intended to mitigate the radiological consequences of accidents. In performing this assessment, an applicant ***shall assume a fission product release from the core into the containment assuming that the facility is operated at the ultimate power level contemplated.***

- Therefore, an applicant's diversity and defense-in-depth deterministic analysis should ***assume a fission product release from the core into the containment assuming that the facility is operated <u>at the ultimate power level contemplated.</u>***

DAS Performance Guidance continued
10CFR100 Limit Requirements

- **10CFR100.21(c)(2)**

  (2) Radiological dose consequences of postulated accidents shall meet the criteria set forth in 50.34(a)(1) of this chapter for the type of facility proposed to be located at the site;

- **10CFR34(a)(ii)(D)(*1*) and (2)**

  (1) An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, ***would not receive a radiation dose in excess of 25 rem*** total effective dose equivalent (TEDE).

  (2) An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage) ***would not receive a radiation dose in excess of 25 rem*** total effective dose equivalent (TEDE)

Addressing I&C Vulnerabilities

- **Branch Technical Position 7-19, Acceptance Criteria**

  The applicant/licensee should (1) demonstrate that sufficient diversity exists to achieve these goals, (2) identify the vulnerabilities discovered and the corrective actions taken, or (3) identify the vulnerabilities discovered and provide a documented basis that justifies taking no action.

REFERENCES

- 10 CFR Part 50

  [*http://www.nrc.gov/reading-rm/doc-collections/cfr/part050.html*]
- SRM to SECY-93-087, Item II.Q, Defense Against Common-Mode failures in Digital Instrumentation and Control Systems [*ADAMS Accession No. ML003708056*]
- NUREG/CR-6303, Method for Performing Diversity and Defense-in-Depth Analyses of Reactor Protection Systems [*ADAMS Accession No. ML071790509*]
- Generic Letter 85-06, Quality Assurance Guidance for ATWS Equipment that is not Safety-Related [*ADAMS Accession No. ML031140390*]
- Branch Technical Position 7-19, Guidance for Evaluation of Diversity and Defense-in-depth in Digital Computer-based Instrumentation and Control Systems

  [ADAMS Accession No. ML070550072]