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U.S. NUCLEAR REGULATORY COMMISSION

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DATE OF MEETING
07/01/2003

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Docket Number(s)	50-269, 50-270, 50-287
Plant/Facility Name	OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3
TAC Number(s) (if available)	MB8086, MB8087, MB8088
Reference Meeting Notice	JUNE 6, 2003
Purpose of Meeting (copy from meeting notice)	TO DISCUSS DEFENSE-IN-DEPTH AND DIVERSITY ANALYSIS ASSOCIATED WITH DIGITAL UPGRADE OF RPS/ESPS

NAME OF PERSON WHO ISSUED MEETING NOTICE	TITLE
L. N. OLSHAN	PROJECT MANAGER

OFFICE
NRR

DIVISION
DLPM

BRANCH
PDII-1

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Oconee Nuclear Station

RPS/ES D^3 Analysis

July 1, 2003

Duke Energy

Agenda

- Introductions
- Licensing interactions
- RPS/ES D^3 analysis
- Manual Operator Response Times
- Leak Detection Capability
- Applicability of LBB for RCS Piping
- 50.46 Single Failure Assumption
- Closing Remarks

Licensing Interactions

- Duke appreciates the unique nature of this submittal
  - Initial meeting with staff on March 7, 2002
  - Working closely with lead reviewer
  - Followed established guidance
  - Analysis submitted on March 20, 2003
- NRC response needed as input into design of RPS/ES digital upgrade
  - Eliminates need for diverse LPI actuation feature

RPS/ES D^3 Analysis

- Summary of Methodology
- Analysis Results
  - Diverse Design Features Credited
  - Operator Actions Credited
- Conclusions

Summary of Methodology

- Duke used new replacement SG T/H analysis methodologies that are currently under NRC review (non-LOCA PCT scope)
  - Extensions of existing NRC-approved methodologies currently in UFSAR
  - Codes: RETRAN-3D, VIPRE-01, RELAP5/MOD2-B&W, FATHOMS, SIMULATE, LOCADOSE
- SBLOCA analyzed by FANP using RELAP5/MOD2/B&W
- LBLOCA addressed by leak detection and low probability (otherwise need diverse actuation of Low Pressure Injection System)

Summary of Methodology (cont.)

Transients and Accidents Considered

- Control rod bank withdrawal at zero power
- Control rod bank withdrawal at full power
- Boron dilution at full power
- Loss of coolant flow
- Locked rotor
- Dropped control rod
- Turbine trip
- Steam generator tube rupture
- Control rod ejection
- Large steam line break
- Small steam line break
- Small break LOCA
- Large break LOCA
- Loss of main feedwater
- Loss of offsite power
- Main feedwater line break

Summary of Methodology (cont.)

Assumptions

- Typical conservative initial conditions
- No loss of offsite power
- No single failures
- Integrated Control System (ICS) in automatic
- Realistic core power distribution (SBLOCA only)
- Realistic core flood tank initial conditions (SBLOCA only)
- Realistic operator actions and times
- Credit for AMSAC (trip turbine and start EFW on loss of main feedwater)
- Credit for existing Diverse Scram System (DSS) at 2450 psig RCS pressure
- Credit for Automatic Feedwater Isolation System (AFIS) on low SG pressure
- Pre-existing SG tube leakage at administrative limit

Summary of Methodology (cont.)

Acceptance Criteria

- Offsite dose limits based on R. G. 1.183
  - Large steam line break 25 rem TEDE (EAB & LPZ)
  - Loss of flow 2.5 rem TEDE (EAB & LPZ)
  - Control Room 5 rem TEDE
- RCS overpressure limit is 3250 psia (ASME Service Level C), same as ATWS acceptance criterion for B&W plants (Note: 3000 psig error in submittal)
- Reactor Building overpressure limit is 125 psi based on 98% of ultimate strength (design pressure is 59 psig)

Summary of Methodology (cont.)

Results Categories

- 1 – RPS and ESPS not actuated / no adverse impact
- 2 – Event terminated by DSS actuation / no adverse impact
- 3 – Event bounded by another event
- 4 – Analysis required and results show acceptance limits are met
- 5 – Acceptance limits not met / fail diversity and defense-in-depth

Analysis Results

Category 1 – RPS and ESPS Not Actuated / No Adverse Impact

- Dropped control rod
- Steam generator tube rupture
- Small steam line break (for RCS pressure response and offsite doses)

- The UFSAR analysis does not credit automatic RPS or ESPS actuation

Analysis Results (cont.)

Category 2 – Event Terminated by DSS Actuation / No Adverse Impact

- Control rod bank withdrawal at zero power
- Turbine trip
- Loss of main feedwater
- Loss of offsite power
- Main feedwater line break

- The DSS mitigates the event when RCS pressure reaches 2450 psig

Analysis Results (cont.)

Category 3 – Event Bounded by Another Event / No Adverse Impact

- Boron dilution at full power (bounded by control rod bank withdrawal)
- Control rod ejection containment response and dose results (bounded by LOCA)
  - Manual actuation of HPIS at 5 minutes credited
  - Manual actuation of RBCS and RBS at 8 minutes credited
- SBLOCA containment response and doses (bounded by LOCA)
  - Manual actuation of RBCS and RBS at 8 minutes credited

Analysis Results (cont.)

Category 4 – Analysis Required and Acceptance Criteria Met

- Control rod bank withdrawal at full power
  - No cladding failures, so offsite doses are not significant
  - RCS and Reactor Building pressure limits not challenged
- Loss of coolant flow (four-pump coastdown)
  - 26.0% cladding failure and 2.14% fuel melt
  - Radiological doses bounded by two-pump coastdown
  - RCS and Reactor Building pressure limits not challenged

Analysis Results (cont.)

Category 4 – Analysis Required and Acceptance Criteria Met (cont.)

- Loss of coolant flow (two-pump coastdown)
  - 26.6% cladding failure and 2.46% fuel melt
  - RCS and Reactor Building pressure limits not challenged
  - Radiological doses
    - EAB boundary = 2.0 rem TEDE (2.5 rem is limit)
    - LPZ boundary = 0.4 rem TEDE (2.5 rem is limit)
    - Control Room = 1.2 rem TEDE (5 rem is limit)

Analysis Results (cont.)

Category 4 – Analysis Required and Acceptance Criteria Met (cont.)

- Large steam line break
  - 34.0% cladding failure and 4.75% fuel melt
  - RCS pressure limit is not challenged
  - Peak containment pressure is 44 psig
  - Radiological doses
    - EAB boundary = 4.4 rem TEDE (25 rem is limit)
    - LPZ boundary = 0.9 rem TEDE (25 rem is limit)
    - Control Room = 3.4 rem TEDE (5 rem is limit)

Analysis Results (cont.)

Category 4 – Analysis Required and Acceptance Criteria Met (cont.)

- Locked rotor
  - No cladding failures, so offsite doses are not significant
  - RCS and Reactor Building pressure limits not challenged

- Small steam line break
  - Peak containment pressure is 45 psig
  - Manual actuation of RBCS and RBS credited at 8 minutes

Analysis Results (cont.)

Category 4 – Analysis Required and Acceptance Criteria Met (cont.)

- Small-break LOCA
  - Reactor manually tripped by the operator at 2 minutes
  - Reactor coolant pumps manually tripped by the operator at 2 minutes
  - HPI and LPI manually started by the operator at 5 minutes
  - Peak cladding temperature is limited to around 1000°F
  - RCS pressure limit not challenged

Analysis Results (cont.)

Category 5 – Acceptance Limits Not Met

- Large-break LOCA
  - Crediting manual start of HPI and LPI at 5 minutes is not early enough to maintain a coolable geometry
  - LBLOCA does not meet the diversity and defense-in-depth requirements
  - A diverse actuation of LPI is required if LOCA within the scope of the D^3 study
  - LOCA addressed by leak detection and low probability – not required to meet diversity and defense-in-depth requirements.

Conclusions

- Diversity and defense-in-depth demonstrated for all events except large-break LOCA
- Existing diverse plant systems credited for automatic mitigation
  - DSS
  - AMSAC
  - AFIS
  - ICS

Conclusions (cont.)

- New manual operator action times credited
  - Manual reactor trip at 2 minutes (SBLOCA)
  - Manual start of HPI and LPI at 5 minutes (SBLOCA, REA)
  - Manual start of RBCS and RBS at 8 minutes (SBLOCA, REA)
- Acceptance criteria met (except for LBLOCA)
  - Diverse actuation of LPI required for LBLOCA with failure of ESPS
  - Leak detection and low probability justification to preclude LBLOCA

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Operator Response Times

Process Used to Develop Times

- Table Top Discussion Using 3 SROs involved with EOP and familiar with other timing validations
- Scenario analysis to develop operator response times based on previously validated times and operator judgment
- Structured approach for transient mitigation prior to reactor trip (memory)
- Immediate operator actions after reactor trip (memory)
- Symptoms check and parallel actions (manual action response times based on previous validations)

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Leak Detection Capability

- **Oconee's capability and methods are similar to others in nuclear industry**
  - Sump Level Monitoring
  - Radioactive Particulate Monitoring
  - Radioactive Gaseous Monitoring
  - Other means, e.g., RCS leakage calc, RCS makeup flow, LDST level monitoring

Use of LBB for RCS Piping

- Submittal not requesting additional LBB approval of RCS piping
  - Previously approved in SER dated December 12, 1985

- Oconee RCS has Inconel welds

- PWSCC issue is generic, the industry through EPRI is working toward quantifying the effect, if any, of PWSCC on LBB analyses

50.46 Single Failure Assumption

- SWCMF is not a single failure based on NRC endorsed guidelines for licensing digital upgrades.
- NRC RIS 2002-22 endorsed EPRI TR-102348 Rev.1
  - D^3 analysis is considered a beyond design basis concern
  - Recognizes the likelihood of a common case software failure in a high quality digital system is significantly below that of a single active hardware failure

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❖ Closing Remarks