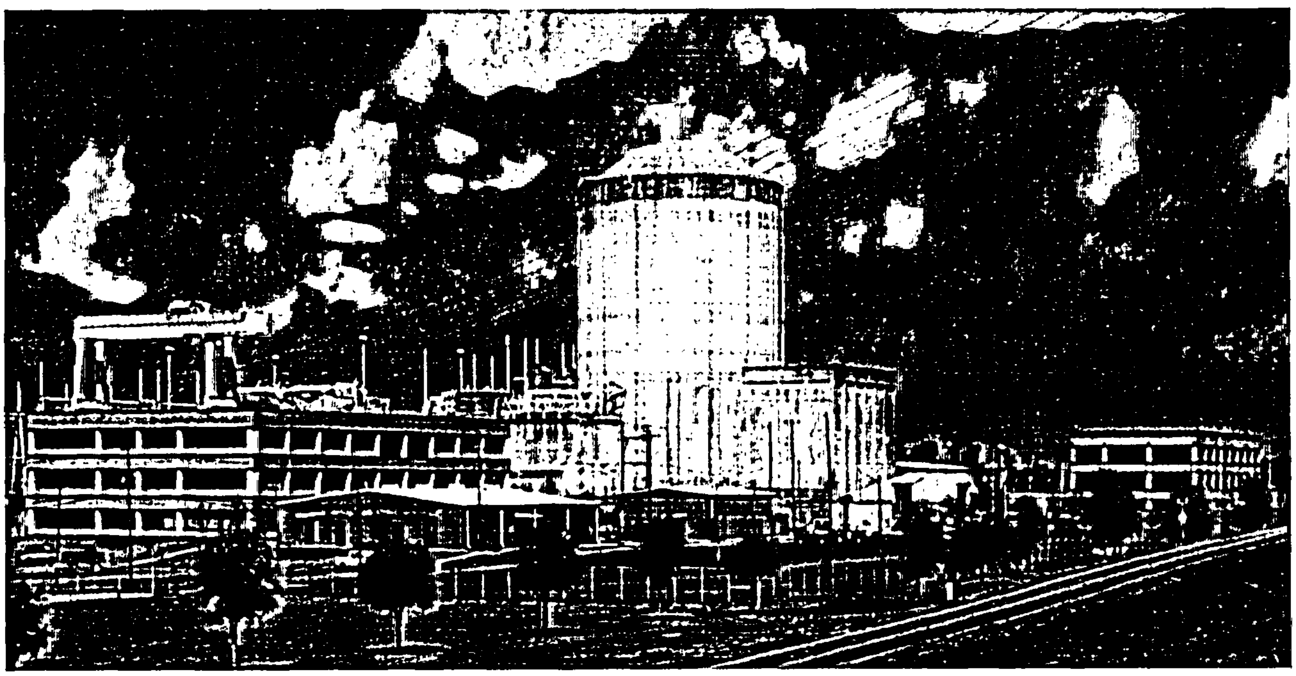
Waterford 3
Extended Power Uprate



ACRS Thermal Hydraulic Phenomena Subcommittee
January 26, 2005

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Waterford 3 Extended Power Uprate Project

ACRS Thermal Hydraulic Phenomena Subcommittee

January 26, 2005

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Tim Mitchell

Engineering Director

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Agenda

- Introduction – Tim Mitchell
- Safety Analysis – Paul Sicard
- Risk Considerations – Jerry Holman
- Engineering Plant Impacts – David Viener
- Operations Impacts
  - Training and Procedures – Gene Wemett
  - Testing – David Constance
- Conclusion – Tim Mitchell

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Introduction

- Project Scope
- Design Basis Improvements
- Oversight & Rigor
- Industry Operating Experience

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Introduction

- Combustion Engineering Nuclear Steam Supply System (NSSS) Pressurized Water Reactor (PWR)
- Entered commercial operation 1985
- 3390 MWt original licensed power
- 3441 MWt Appendix K Margin Recovery
- 3716 MWt Extended Power Uprate (EPU)

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Introduction

- Project Team
  - **Entergy**
  - **Westinghouse (NSSS)**
  - **Enercon (Balance of Plant (BOP))**
  - **Siemens-Westinghouse (Turbine / Generator)**

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Safety Analysis

Paul Sicard

EPU Lead Safety Analysis Engineer

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Scope of Safety Analysis

- Demonstrate Acceptable EPU Impact
  - Fuel
  - ECCS
  - Non-LOCA Events
  - Containment
  - Radiological

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Modification Impact

- Existing safety systems support safety analyses
  - Replace HP turbine steam path
  - Main Generator rewind
  - Replace Main Generator output breakers
  - Main Transformer Improvements
  - Control systems & instrumentation

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Operating Parameters

Parameter	**EPU Value**	**Current Value**
Reactor power	3716 MWt	3441 MWt
Hot Leg temp	601 °F	600.2 °F
Cold Leg temp	541-543 °F	545 °F
RCS pressure	2250 psia	2250 psia
SG pressure	810 psia	831 psia
Steam flow	2301 lbm/sec/SG	2118 lbm/sec/SG
Feedwater temp	449.7 °F	442.7 °F

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Significant Aspects

- Maintain approximate current nominal T_{hot}
- Credit ADVs for secondary pressure control for SBLOCA
- 1999 LBLOCA evaluation model
- CENTS vice CESEC for non-LOCA transients
- AST methodology for dose calculations

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Technical Specification Changes

Technical Specification changes include:

- Added ADV Technical Specification
- Raised minimum BAMT concentration
- Lowered maximum SIT volume
- Lowered SG Pressure – Low PPS setpoint
- Add minimum containment temperature
- 75 gal/day SG primary-secondary operational leakage

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Analysis Changes

Parameter	EPU	Current
RCS Cold Leg Temperature Range (TS 3.2.6)	536 – 549°F	541 – 558°F
T_{cold} Program	541 – 543°F ramp	545°F constant
Minimum Pressurizer Pressure (TS 3.2.8)	2125 psia	2025 psia
# SG Tube Plugging Limit	1000 per SG	700 per SG
Minimum Boric Acid Makeup Tank (BAMT) Boron Concentration (TS 3.1.2.7 and TS Figure 3.1-1) (minor volume changes)	4900 ppm	3950 ppm

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Analysis Changes

Parameter	EPU	Current
SG Low Pressure Setpoint (TS Tables 2-1 and 3.3-4)	666 psia	764 psia
Non-LOCA Transient Analysis Code	CENTS	CESEC
LBLOCA Evaluation Model (EM)	1999 EM	1985 EM
Safety Injection Tank (SIT) Level Maximum Level (TS 3.5.1)	77.8%	83.8%
Post-LOCA Long-Term Cooling (LTC) Approach Changes	lower plenum not credited in mixing volume	lower plenum credited in mixing volume

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Analysis Changes

Parameter	EPU	Current
Fuel Failure for Return to Power Main Steam line Break (MSLB)	Yes	No
Statistical Convolution for Fuel Failure	Yes	Yes for selected events
Reactor Coolant Radioisotopic Concentration	ANSI N18.1	ANSI N237

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Analysis Changes: Dose

Parameter	EPU	Current
Source Term Methodology	RG 1.183 (AST)	RG 1.4 (TID-14844)
Primary-to-Secondary Leak Rate (per SG) (TS 3.4.5.2)	75 gal/day (operational)	720 gal/day
Atmospheric Dispersion Factors	New	Original license
ICRP30 Dose Conversion Factors	Yes	Yes, for selected events
Control Room Doses analyzed	(AST) Yes, including SBLOCA	Only LBLOCA and FHA

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Fuel

- Cycle 14
- Fuel Mechanical Design Unchanged
- Standard 16x16 fuel design
- 18 month fuel cycle
- Erbia burnable poison (since Cycle 9)
- 217 total assemblies
- 100 fresh assemblies (larger batch size)
- Acceptable fuel rod corrosion and duty

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Containment Analysis

- Current LOCA Mass & Energy releases account for EPU
- MSLB Mass & Energy releases generated for EPU
- GOTHIC analyses
- Peak pressures: 35.16 psig LOCA
  41.88 psig MSLB
  (44 psig acceptance limit)

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Transient Analysis Topics

- Use of CENTS vs. CESEC:
  - CENTS to replace CESEC for non-LOCA transient analyses
  - CENTS generically approved for CE plants
- Credit 3 sec time delay for LOOP after trip for SGTR
- Demonstrate compliance with acceptance criteria

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Pressurization Events

- Limiting Anticipated Operational Occurrence: Loss of Condenser Vacuum
  2732 psia
  (2750 psia acceptance criteria)
- Limiting Fault: Feedwater Line Break
  2753 psia
  (3000 psia acceptance criteria)

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ECCS Performance Analysis

LBLOCA:

- Update method to 1999 EM (CENPD-132, Supplement 4-P-A)
- Currently 1985 EM (Supplement 3-P-A)
- Max Peak Clad Temperature (PCT) of 2164°F

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ECCS Performance Analysis

SBLOCA:

- No methods change: CENPD-137-P, Supplement 2-P-A (S2M Evaluation Model)
- Credit automatic operation of ADVs for secondary pressure control
  --ADVs Safety Related
- 1040 psia analysis setpoint
- Charging Pumps no longer credited
- 0.055 ft^2 break: Max PCT 2018°F

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ECCS Performance Analysis

LOCA Long Term Cooling:

- Post-LOCA boric acid precipitation analysis assumes mixing volume of core and part of outlet plenum
- Analysis per CENPD-254 methodology
- Hot leg injection 2-3 hours post-LOCA demonstrates margin to solubility limit

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AST Dose Analyses

- AST needed to address GL 2003-01 Control Room Habitability
  - Tracer gas test conducted April 2004
  - License amendment under staff review
- Bound control room inleakage:
  - Recirculation Mode: 100 CFM (79 CFM measured)
  - Pressurized Mode: 65 CFM (36 CFM measured)

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AST Dose Analyses

- Analyses extended to non-LOCA radiological events and Small Break LOCA
- High Control Room X/Q due to proximity of ADVs to Control Room Air Intakes
- Assume leakage of 0.375 GPM for faulted SG (MSLB, FWLB)
- Assume 150 gal/day for intact SGs (75 gal/day TS limit only for SBLOCA)
- Credit existing operator action to select preferred control room air intake

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AST Results

Results for Limiting Events:

	Fuel Failure	EAB TEDE	LPZ TEDE	MCR TEDE
I.C. MSLB	10%	0.60	0.19	4.89
FWLB / O.C. MSLB	0%	0.23	0.12	3.62
CEA Ejection	15%	1.03	0.65	2.41
SGTR (PIS)	0%	0.99	0.21	4.85
LBLOCA	RG 1.183	5.30	2.37	2.95
SBLOCA	100%	1.96	1.08	3.93
FHA	60 rods	0.55	0.085	0.11

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AST Dose Analyses

Conclusions

- Meet 10CFR50.67 and GDC19 acceptance criteria
- Supports EPU

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Risk Considerations

Jerry Holman

Manager, Nuclear Engineering

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Scope Of Risk Assessment

- Address Impact On
  - Initiating Event frequency
  - Success criteria
  - Equipment failure rates
  - Operator response times and Human Reliability Analysis (HRA)
  - CDF and LERF
  - External events
  - Shutdown

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Risk Assessment Results

- Initiating Event Frequency
  - No new initiators
  - No change in frequencies
- Success Criteria
  - CENTS analyses to confirm success criteria
  - No changes

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Risk Assessment Results

- Equipment Failure Rates
  - Comprehensive reviews of equipment performed
  - Systems operate within allowable limits
  - No impact on PRA failure rates or results
  - Existing monitoring programs and model update will account for any additional system wear

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Risk Assessment Results

- Operator Response Times / HRA
  - CENTS analyses to determine available action times
  - Higher decay heat reduced operator action times
  - Major impact is reduction of recovery time for loss of feedwater

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Risk Assessment Results

Scenario	Pre-EPU Time Available	Post-EPU Time Available
Recover feedwater for early loss of feedwater	82.6 min	68.3 min
Recover feedwater for late loss of feedwater (battery depletion)	5.1 hr	4.1 hr
Recover feedwater for late loss of feedwater (CSP depletion)	12.3 hr	11.3 hr

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Risk Assessment Results

- Internal Events (per year):
  - CDF increase = 3.5E-7
  - LERF increase < 1.0E-7
  - New CDF = 5.9E-6

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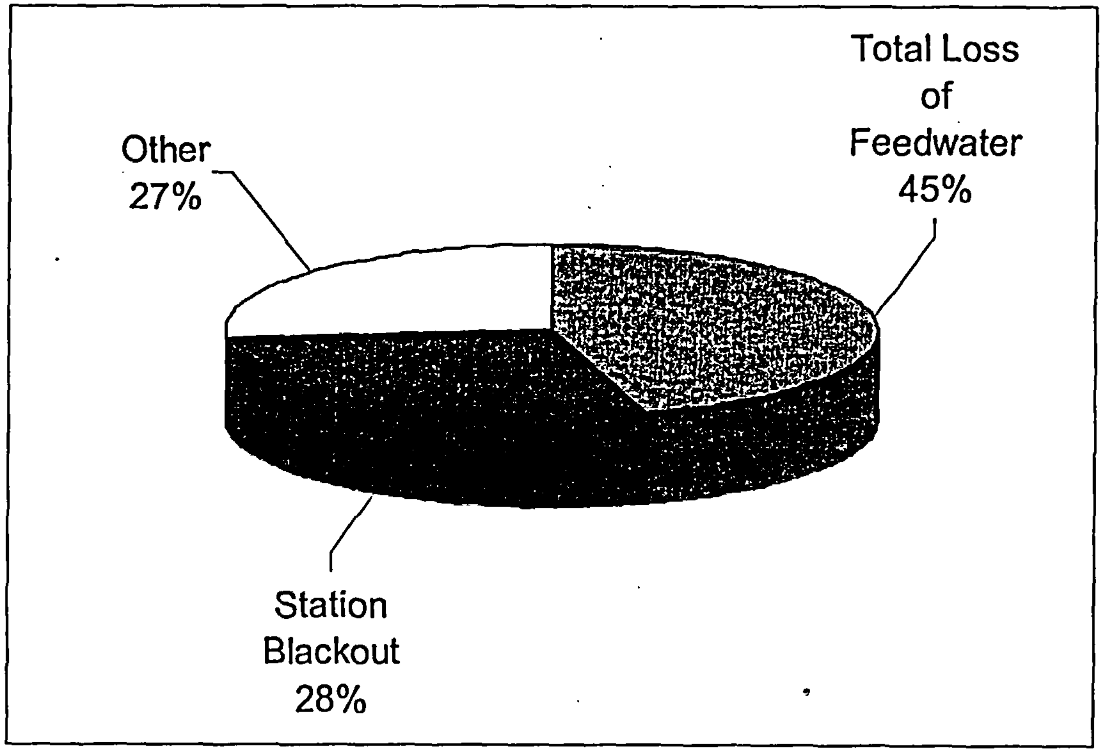
Risk Assessment Results

EPU Sequence Contribution



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Risk Assessment Results

- External Events
  - Slight increase in fire CDF due to operator response time reduction
  - No impact on other external events

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Risk Assessment Results

- Shutdown Risk
  - EPU has no unique or significant impacts
  - No changes to shutdown operations protection plan

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Risk Assessment Results

Conclusions

- All PRA model elements reviewed for impact
- Minor reduction in Operator recovery times
- EPU has a very small impact on risk

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Engineering Plant Impacts

David Viener

EPU Lead Mechanical Engineer

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Significant Modifications

- Replace HP turbine steam path
- Main Generator rewind & alkalizer skid
- Replace Main Generator output breakers
- Replace Main Transformer A
- Increase cooling on Main Transformer B

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Significant Modifications (cont.)

- FW heater drain valve capacity increase
- Condenser tube staking
- Control systems & instrumentation
  - Setpoint, range and scale changes
  - 4 transmitters to be replaced

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Engineering Plant Impacts

- Decay Heat
  - Ultimate Heat Sink
    - System Capable of Dissipating Heat Loads for Normal, Shutdown and Accident Conditions
    - Water Sources are Adequate to Maintain Cooling of Essential Plant Equipment
    - Equipment Operating Times Increased Post-Accident which Impacts Emergency Generator Fuel Oil

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Engineering Plant Impacts

- Decay Heat
  - Emergency Diesel Generator Fuel Oil
    - Raised fuel oil minimum capacity requirement to maintain 7 day supply per current licensing basis.
    - Commitment to add additional storage.

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Engineering Plant Impacts

- Decay Heat (Cont'd)
  - Emergency Feedwater
    - System Flow Capable of Mitigating against Feedwater Demand Events
    - Normal and Backup Condensate Sources are Adequate to Bring Plant to Shutdown Cooling Entry Conditions

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Engineering Plant Impacts

- Decay Heat (Cont'd)
  - Shutdown Cooling
    - Capable of Achieving Cold Shutdown Conditions in accordance with Reactor Systems Branch (RSB) Branch Technical Position (BTP) 5-1
    - Refueling Technical Specification Time Limits to Reduce Shutdown Cooling Flow remain Unchanged

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Engineering Plant Impacts

- Decay Heat (Cont'd)
  - Fuel Pool Cooling
    - Reracking in 1998 assumed an 8.0% Uprate in the Decay Heat Removal Analysis
    - EPU Proposes a 1.5% Increase
    - Decay Heat Removal Analysis Bounds Capacity of Fuel Pool
    - Current Fuel Pool Temperature Limits will be Maintained
    - Bounding Time to Boil Analysis remains Unchanged

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Engineering Plant Impacts

- Containment Overpressure
  - Containment Overpressure not Credited in the ECCS Pumps Net Positive Suction Head Analysis
  - EPU Maintains this Assumption
- PWR Safety Injection Sump
  - Systems Inside Containment will be Unchanged
  - Minimum Containment Water Level remains Unchanged
  - Sump Temperature change is Negligible

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Engineering Plant Impacts

- Vibration
  - Steam Generator
    - Detailed tube bundle evaluation
    - Dryers and Dryer Supports evaluated
    - Palo Verde Dryer Design – Operating at Higher Flow Rates than W3 Proposes.
  - Secondary System
    - Feedwater Heaters, Moisture Separator Reheater, and Condenser Evaluated
    - Condenser Tube Staking Required
  - Vibration Monitoring Program
    - Monitor Secondary Systems pre- and post-EPU based on Industry Operating Experience.

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Engineering Plant Impacts

- Flow Accelerated Corrosion (FAC)
  - Power Uprate effects evaluated using CHECWORKS
  - No component replacements required
  - Outage inspection sampling increased based on EPU conditions
  - Piping systems impacted will continue to be monitored to detect any deviation from predicted wear rates.

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Engineering Plant Impacts

- Alloy 600
  - Reactor Coolant System
    - Nominal Thot increasing by 0.8 °F
    - Nominal Tcold decreasing by 2 °F
    - Impact on crack initiation rate is negligible
  - Steam Generator
    - NEI 97-06 program continues to assure SG tube integrity post EPU

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Engineering Plant Impacts

- Grid Stability
  - Short Circuit, Transient Stability and Offsite Voltage Stability Studies Re-performed
  - Short Circuit Study Determined Generator Output Breakers were marginal
  - Installing larger generator output breakers for EPU

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Engineering Plant Impacts

Conclusion

With the proposed modifications, Waterford 3 plant design can safely operate at the proposed EPU conditions

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Operator Impact/Training

Gene Wemett

Assistant Operations Manager

Entergy

Operator Impact/Training

- Operations oversight
- Review of all modifications and evaluations for impact on operation
- Procedure impact

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Operator Impact/Training

- Training
  - Phase I, EPU seminars on modifications, Technical Specification (TS) changes and procedure changes (complete)
  - Phase II, Crew training on plant modifications (in progress)
  - Phase III, Crew training on procedure changes, setpoint changes, TS changes (begins in March)
    - Crews evaluated on the uprated plant simulator prior to refueling outage
    - Crews evaluated on TS, procedure and setpoint changes

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Operator Impact/Training

- Controls and Displays
  - Changes minimal
    - Change to allow more precise setting of Atmospheric Dump Valve setpoint
    - Turbine will be operated exclusively in single valve
  - Some display ranges will be re-scaled

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Operator Impact/Training

- Technical Specifications (TS)
  - Parameter changes
  - One new Atmospheric Dump Valve TS
- Normal and Off-normal Procedures
  - No new procedures
- Emergency Operating Procedures
  - No change to type and nature of actions
  - No new actions

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Operator Impact/Training

Conclusion

The changes brought about by power uprate to unit's operation are minimal and acceptable to the Operations Department.

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Power Ascension Testing

David Constance

Operations

Entergy

Power Ascension Testing

- Reactor Engineering Tests / Power Verification
- Transient and Steady State Data Record
- Post Modification Testing
- Plant Maneuver Test (100%-90%-95%)
- Post 100% Testing, Data Collection & Surveys
- Vibration Monitoring

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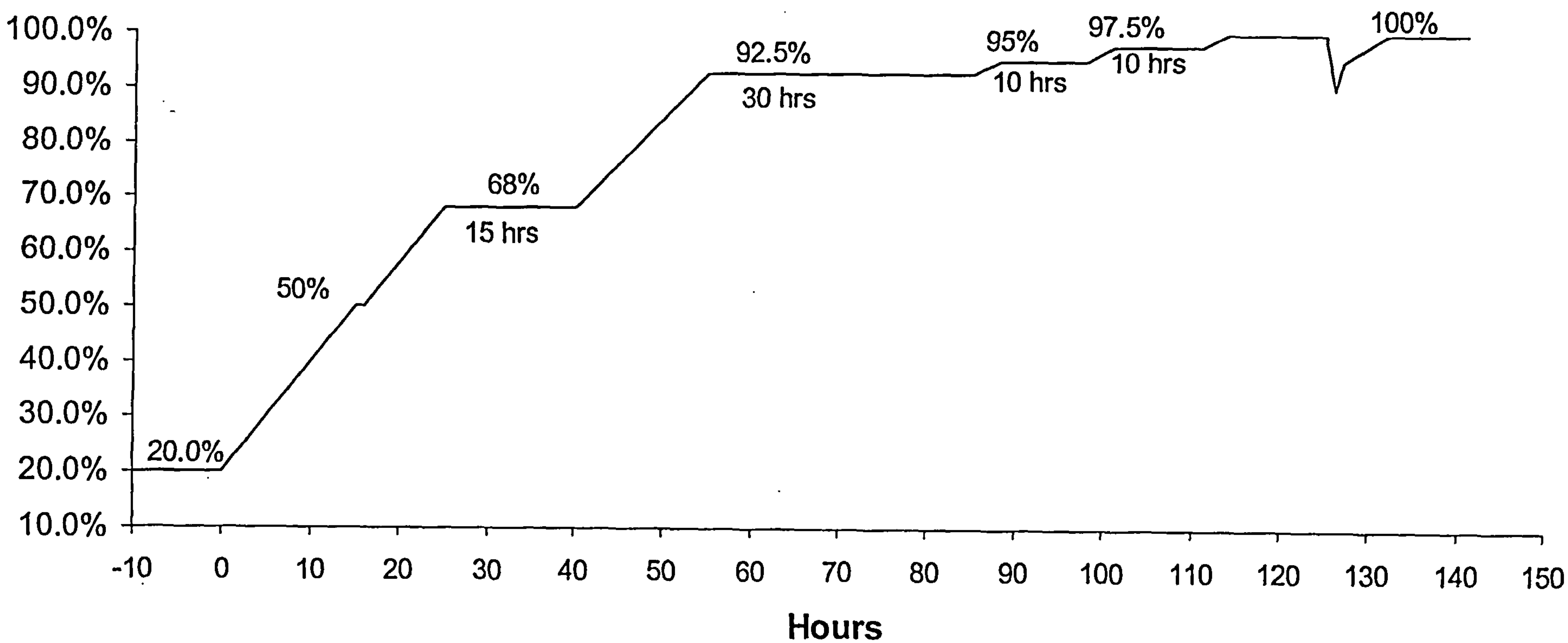
Power Ascension Profile



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Power Ascension Testing

- Low Power Physics Testing (LPPT) remains unchanged for EPU
- Data sets
  - Collected every 10% from 20-100%
  - Collected at 7 different power plateaus
  - Approximately 1000 parameters monitored
  - Data will be automatically collected and processed
  - Data evaluated against predetermined criteria
- Plant Safety Subcommittee reviews results report at each power plateau (>68%), and recommends continued power ascension.

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Testing Considerations

- The proposed plant modifications either have
  - No significant impact on transient response, or
  - Have been evaluated using a calculation model
- No physical changes to the Nuclear Steam Supply System
- No new interactions that affect system response
- No changes to controller algorithms

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Testing Considerations

- Post Modification Testing demonstrates that the component/systems will perform as designed
- Power ascension data collection confirm acceptable operation
- Maneuvering test provides further confirmation
- Benchmarked calculational model evaluates postulated transient conditions

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Power Ascension Testing

Conclusion

- The planned post modification testing and startup tests confirm that the analyses, modifications and adjustments necessary for EPU have been completed properly
- Adequate safeguards are in place to insure a controlled, closely monitored, conservative approach to the new licensed power level

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Concluding Remarks

Tim Mitchell

Engineering Director

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End of Presentation

ONLY ONE COPY AVAILABLE

Reactor Systems Branch Audit Calculations

L. W. Ward

US Nuclear Regulatory Commission
Division of Systems Safety and Analysis
Reactor Systems Branch

ACRS Meeting

January 26, 2005

Reactor Systems Branch Audit Calculations

Agenda

- o Large Feedwater Line Break

- o Limiting Small Break LOCA

- o Post-LOCA Long Term Cooling
  (Boric Acid Precipitation and Timing for Simultaneous Hot/Cold Side Injection)

REACTOR SYSTEM BRANCH AUDIT CALCULATIONS
Waterford EPU

- Large Feedwater System Pipe Break
  - Alternate Methodology Verified Peak RCS Pressure
  - Conservative Analysis Assumptions (break at the elevation of the tube sheet)

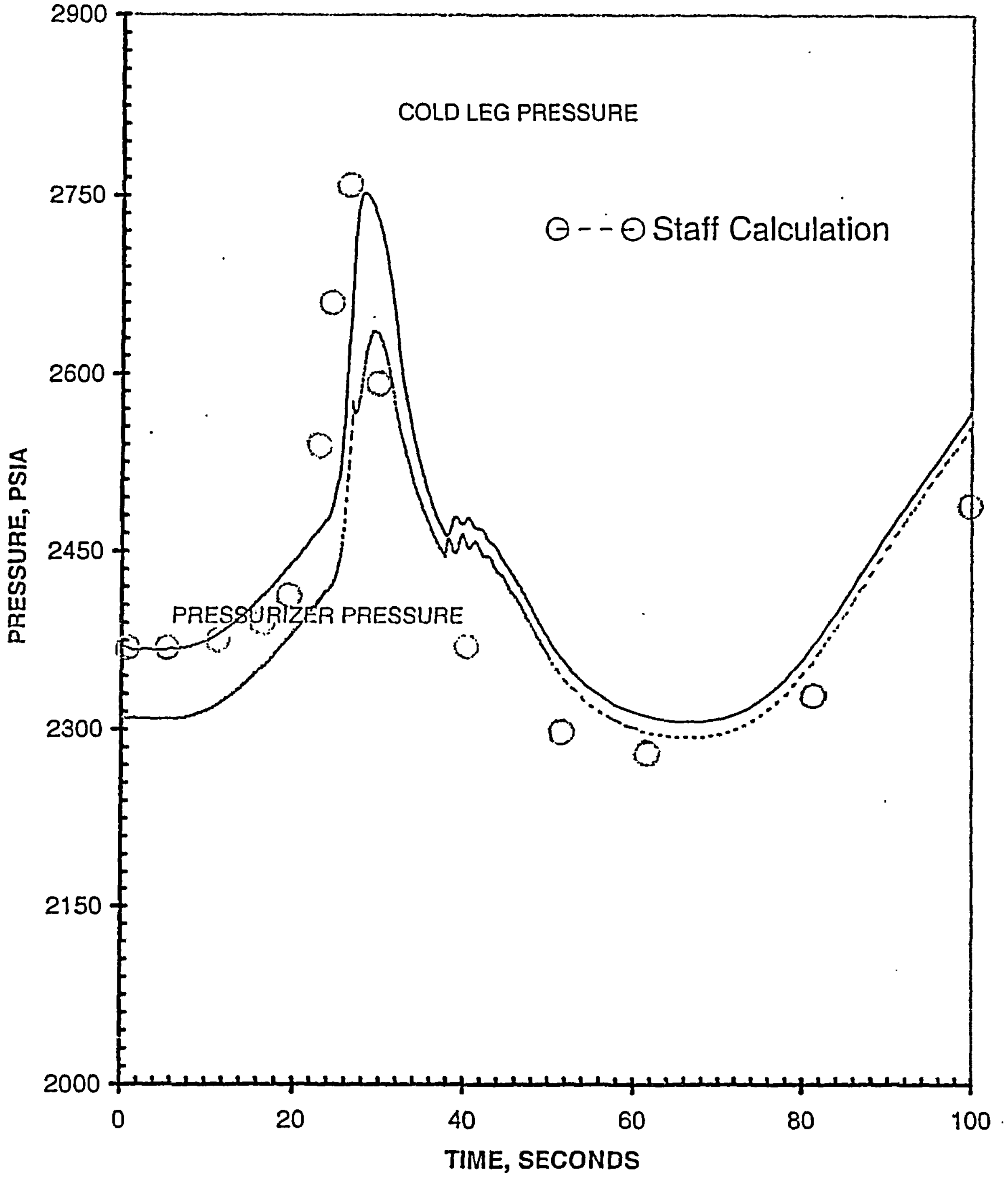


Figure 2.13.2.3.1-2
Feedwater System Pipe Break (Large)
RCS Pressure vs. Time

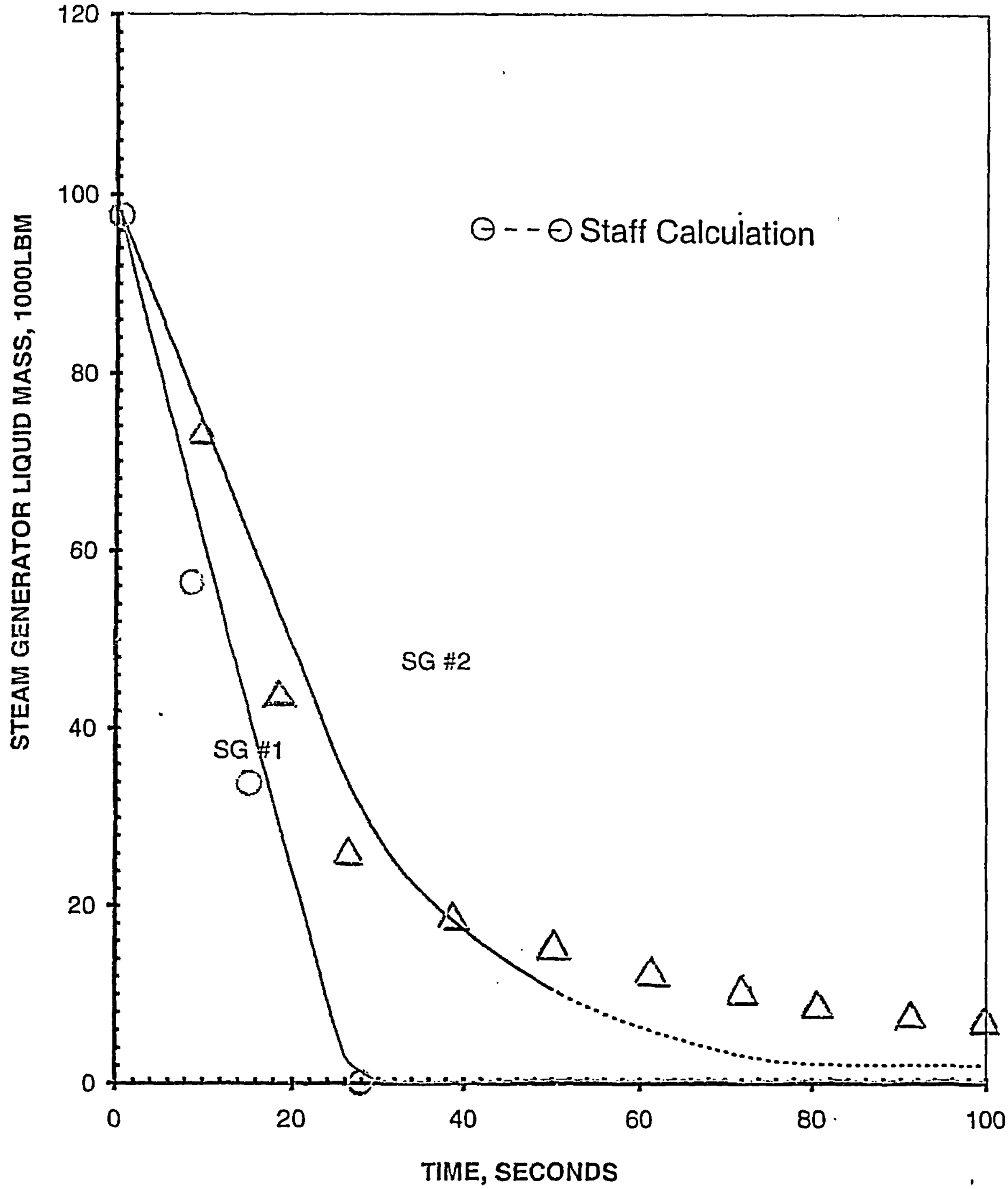


Figure 2.13.2.3.1-9
Feedwater System Pipe Break (Large)
SG Liquid Mass vs. Time

Con't

- o Limiting Small Break LOCA in the Pump Discharge Leg

  - Staff Calculations Reproduced CEFLASH-4AS Core Transient Two-phase Level for the Limiting Small Break(0.055 ft^2 CLB)

  - No Credit for Accumulator Injection

  - Conservative Analysis Assumptions (Top Skewed Axial shape, Diesel Failure, 1.2 Decay.Heat Multiplier)

Figure 2.12-45

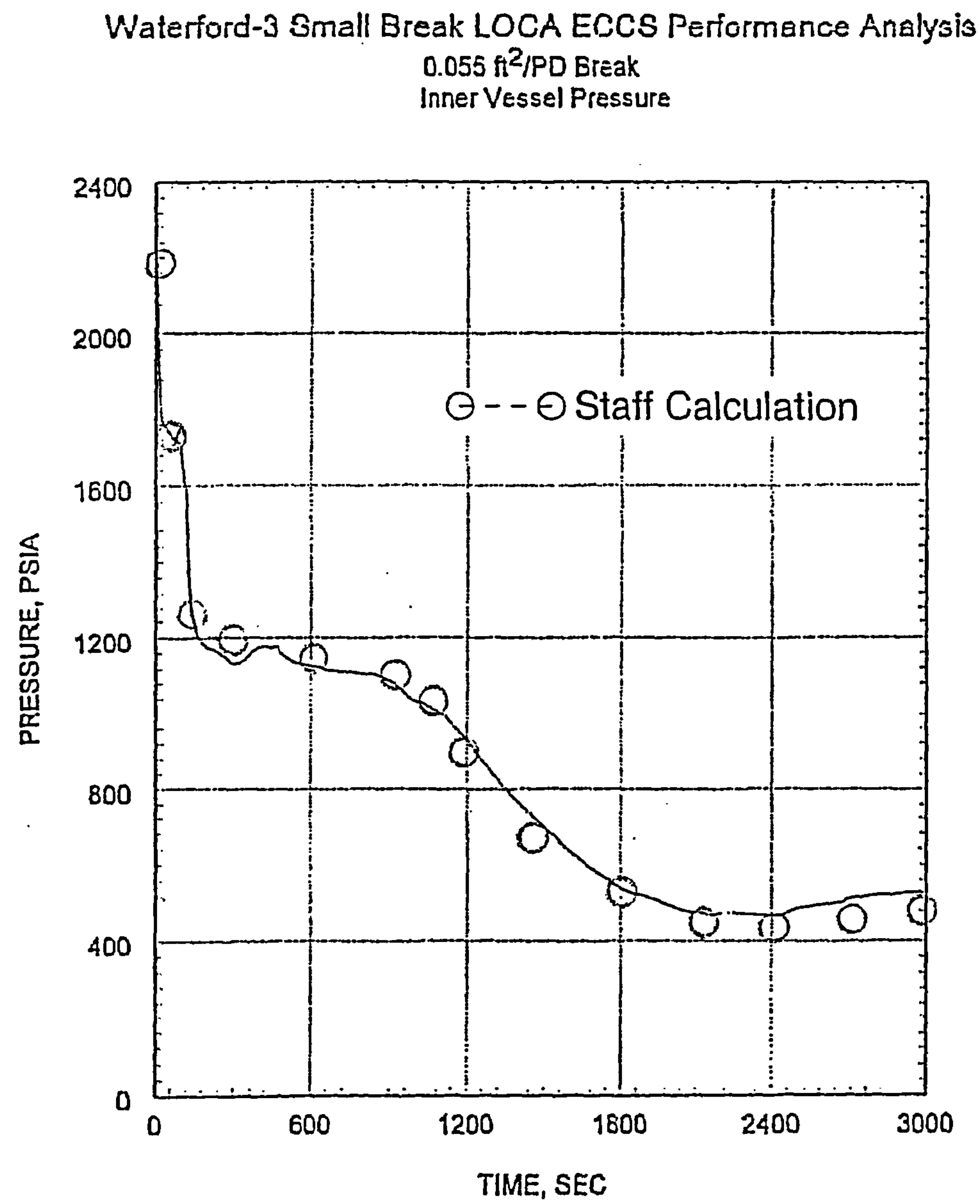
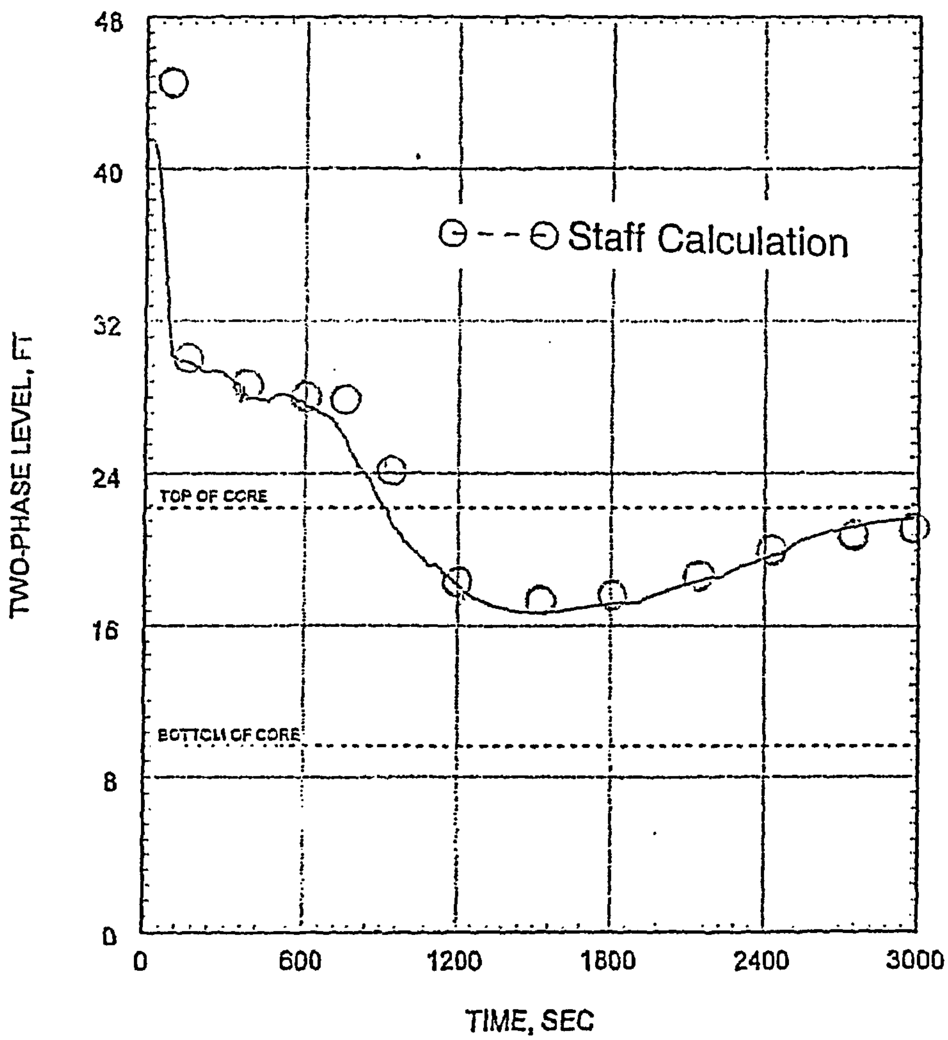


Figure 2.12-48

Waterford-3 Small Break LOCA ECCS Performance Analysis
0.055 ft^2/PD Break
Inner Vessel Two-Phase Mixture Level



Con't

- o Post-LOCA Long Term Cooling (Prevention of Boric Acid Precipitation)

  - Staff Calculations Revealed Error in Mixing Volume (assumed void fraction of 0% in mixing volume following LB LOCAs)

  - Error Produces Precipitation at One Hour vs Four Hours

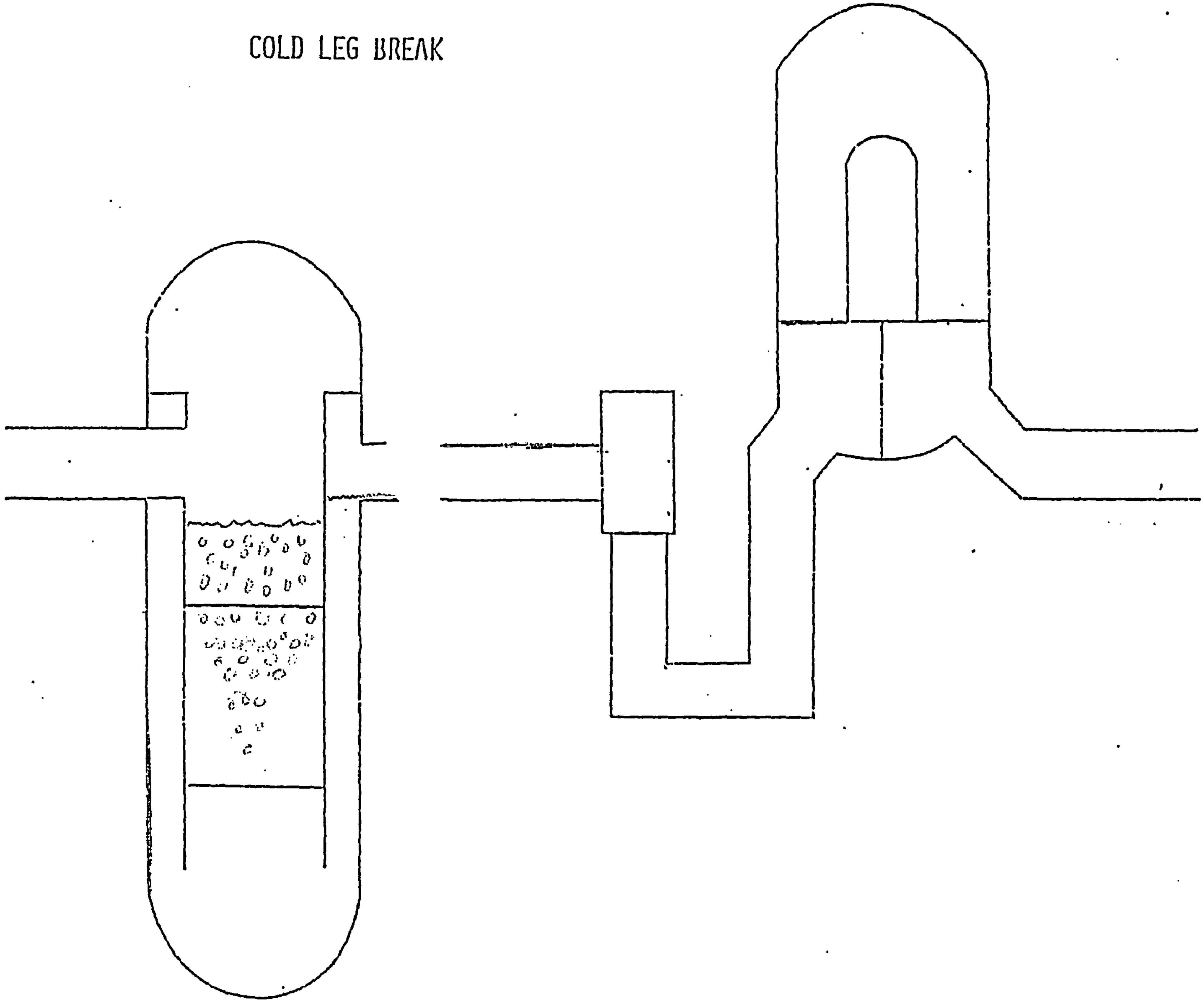
  - Westinghouse has Corrected Error and Modified Licensing Methodology

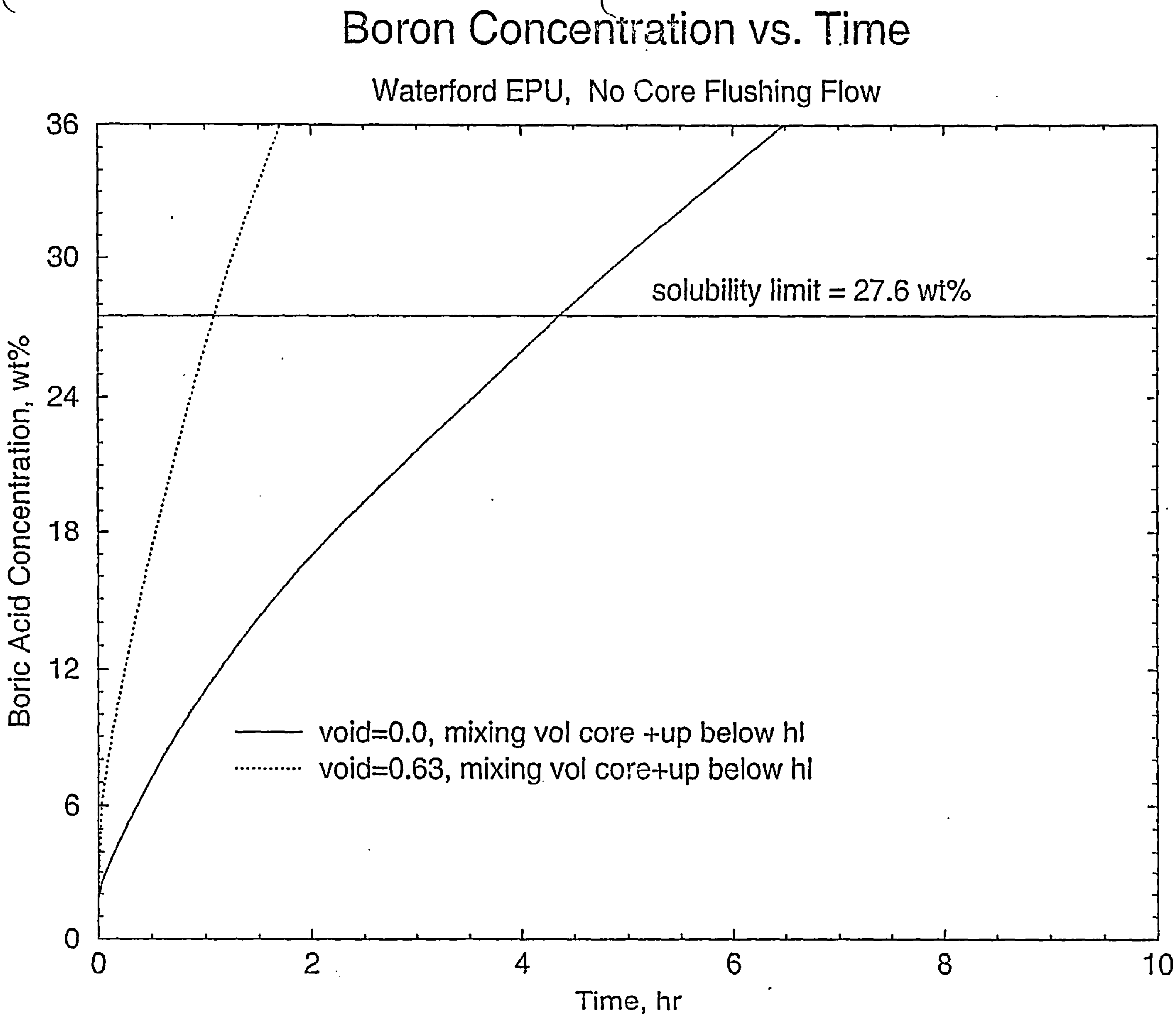
    Mixing Volume Reflects Liquid in Core and Upper Plenum to Hot Leg Top EL (vs mixing vol to hot leg bottom elevation)

    Minimum Containment Pressure Raised to 20 psia (vs 14.7 psia)

    Performed Min. Cont. Pressure Calculation using NRC Approved Methodology (GOTHIC)

COLD LEG BREAK

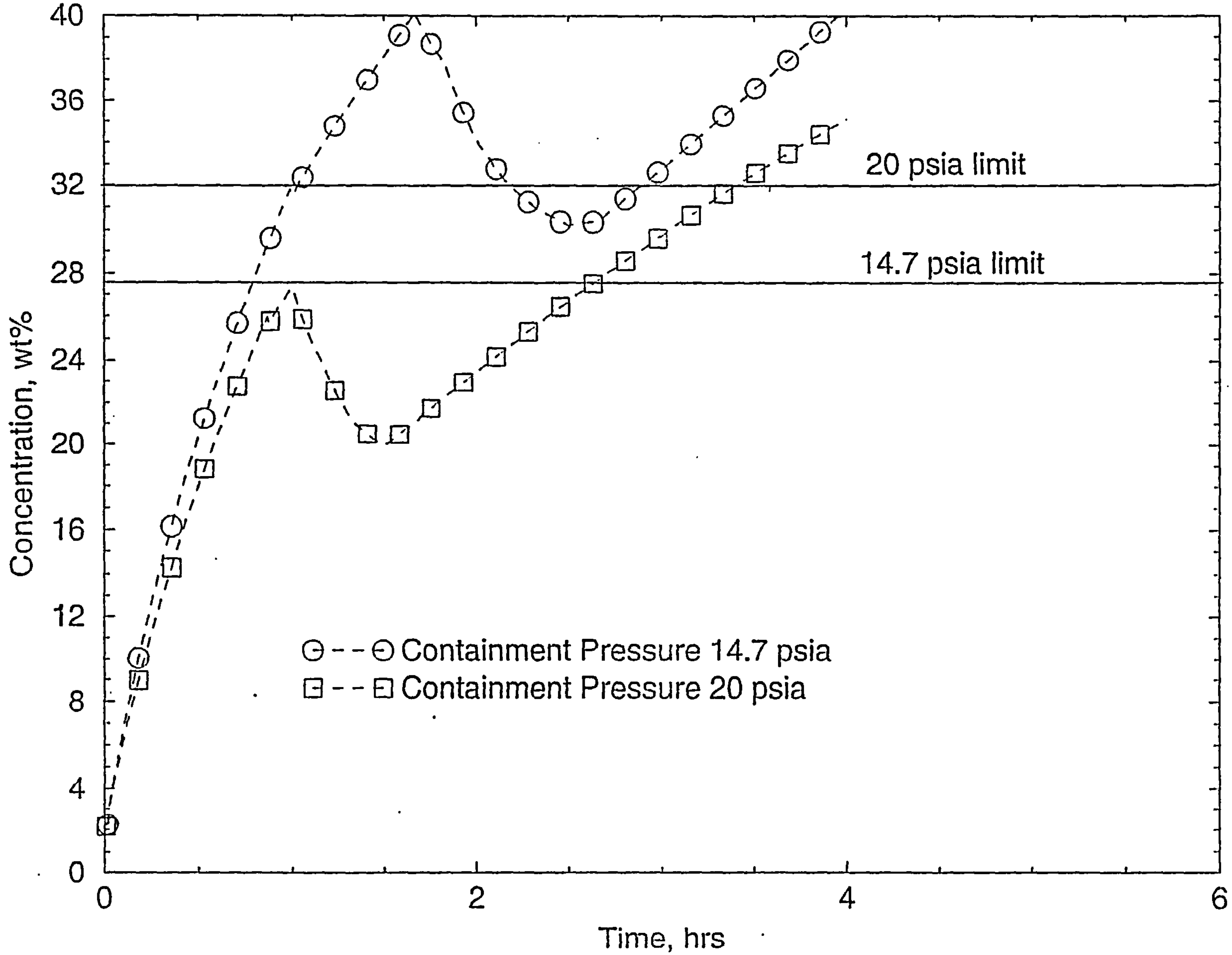




Boron Concentration vs. Time
Waterford EPU, No Core Flushing Flow
Boric Acid Concentration, wt%
Time, hr
solubility limit = 27.6 wt%
void=0.0, mixing vol core +up below hl
void=0.63, mixing vol core+up below hl
0
6
12
18
24
30
36
0
2
4
6
8
10

Boric Acid Concentration vs. Time

Waterford EPU, Effect of Containment Pressure



Con't

- Staff Believes Adequate Margin Remains to Support Power Uprate

  No Credit for Liquid Entrainment (also no removal of boric acid by vapor)

  No Mixing in hot Legs

  Boric Acid Make-Up Tanks, BAMTs, Discharge (6187 ppm)

  Upper Plenum Pressure Higher than Cont. by Loop Pressure Drop

- Westinghouse will Document Changes to Methodology and Revised Analyses

WATERFORD STEAM ELECTRIC STATION, UNIT 3

EXTENDED POWER UPRATE (8.0%)

ACRS THERMAL-HYDRAULIC PHENOMENA
SUBCOMMITTEE MEETING
JANUARY 26, 2005

N. KALYANAM, PROJECT MANAGER
PROJECT DIRECTORATE IV, SECTION 1
DIVISION OF LICENSING PROJECT MANAGEMENT

Waterford 3 EPU

Background

- Originally licensed in 1985 for operation at a reactor core power (CP) not to exceed 3390 Mwt.
- Measurement Uncertainty recapture uprate granted in 2002 to operate at a CP level not to exceed 3441 MWt (a 1.5% increase)
- The extended power uprate (EPU) requests for an increase of 8%, CP level not to exceed 3716 Mwt
- Largest pressurized water reactor (PWR) power uprate to date

Waterford 3 EPU

Major Plant Modifications

- Upgrade the high pressure turbine
- Rewind main generator (MG) / provide associated auxiliaries
- Install higher capacity MG output circuit breakers, disconnect switches, and bus work
- Main transformers modifications
- Replace/upgrade control valves for the heater drain system, reheat system safety valves
- Stake the condenser tubes

Waterford 3 EPU

Time-Table for EPU Implementation

- Entergy plans to implement the Waterford 3 EPU in one increment.
- Completion of plant modifications necessary to implement the EPU is planned prior to the end of refueling outage 13 in the spring of 2005.
- With the approval of this license amendment request, the plant will be operated at 3716 MWt starting in Cycle 14.

Waterford 3 EPU

Comparison of Operating Parameters

	Curr. Value	EPU Value
Reactor Power, MWt	3441	3716
Hot Leg Temperature F	600.2	601.0
Cold Leg Temperature F	545	543
RCS Pressure, psia	2250	2250
RCS Flow, lbm/sec	44,522	45,808
SG Pressure, psia	831	810
Steam Flow, lbm/sec/SG	2118	2301
Final Feedwater Temp. F	442.7	449.7

Waterford 3 EPU

Staff Review Approach

- The first PWR EPU to follow RS-001
- Utilized Standard Review Plan (SRP)
- Used Acceptable Codes and Methodologies
- Requests for Additional Information (RAIs)
- Total of 30 supplements received
- Audits/Independent Calculations in Selected Areas

Waterford 3 EPU

Principle Areas of Review

- Vessel & Internals - Sections 2.1 to 2.4 (Matrix 1 of RS-001)
- Piping Integrity & Non Destructive Examination - Sections 2.5 to 2.6 (Matrix 1 of RS-001)
- SG Integrity & Chem. Eng. - Sections 2.7 to 2.12 (Matrix 1 of RS-001)
- Evaluation of SSCs - Section 2.2 (Matrix 2 of RS-001)
- Electrical - Section 2.3 (Matrix 3 of RS-001)

Waterford 3 EPU

Principle Areas of Review (Contd.)

- Instrumentation & Controls - Section 2.4 (Matrix 4 of RS-001)
- Balance-of-Plant (BOP) Systems & Related Evaluation - Section 2.5 (Matrix 5 of RS-001)
- Containment Review - Section 2.6 (Matrix 6 of RS-001)
- Habitability, Filtration and Ventilation - Section 2.7 (Matrix 7 of RS-001)
- Nuclear Steam Supply System (NSSS), Accident Analysis, and Other Design Basis Evaluations - Section 2.8 (Matrix 8 of RS-001)

Waterford 3 EPU

Principle Areas of Review (Contd.)

- Source Terms & Radiological Analyses - Section 2.9 (Matrix 9 of RS-001)
- Human Performance - Section 2.11 (Matrix 11 of RS-001)
- Power Ascension and Testing - Section 2.12 (Matrix 12 of RS-001)
- Risk Assessment of Power Uprate - Section 2.13 (Matrix 13 of RS-001)

Waterford 3 EPU

Order of NRR Presentation

- Materials & Chemical Engineering Review
- Mechanical & Civil Engineering Review
- Plant Systems Review
- Reactor Systems Review
- Radiological Assessment
- Quality and Maintenance Review
- Risk Assessment of Power Uprate

Waterford 3 EPU

Jim Medoff, Robert Davis, and John Tsao
Materials and Chemical Engineering (EMCB)
Division of Engineering
Office of Nuclear Reactor Regulation

Waterford 3 EPU

Impact on EPU on EMCB Reviews within the scope of Sections 2.1.1 – 2.1.11 of Matrix 1 to NRC Review Standard NRR-RS-001

Section 2.1.1 – Impact on Reactor Vessel (RV) Material Surveillance Program Withdrawal Schedule (10 CFR Part 50, Appendix H)

Section 2.1.2 – Impact on RV Pressure-Temperature (P-T) Limit Curves and Upper Shelf Energy (USE) Assessments (10 CFR Part 50, Appendix G)

Waterford 3 EPU

Impact on EPU on EMCB Reviews (Contd.)

Section 2.1.3 – Impact on Pressurized Thermal Shock (PTS) Assessment for the Waterford 3 RV (10 CFR 50.61)

Section 2.1.4 – Impact on Structural Integrity of the Waterford 3 RV Internal Components and Assessment of the Need for Augmented Inspection Programs

Section 2.1.5 – Impact on Structural Integrity of non-RV/RV-Internal Reactor Coolant Pressure Boundary (RCPB) Components

Waterford 3 EPU

Impact on EPU on EMCB Reviews (Contd.)

Section 2.1.6 – Impact on Leak-Before-Break Analysis (LBB) [10 CFR Part 50, Appendix A, General Design Criterion (GDC) 4]

Section 2.1.7 – Impact on Protective Coating Integrity Assessments (10 CFR Part 50, Appendix B)

Section 2.1.8 – Impact on Flow Accelerated Corrosion (FAC) Programs

Waterford 3 EPU

Impact on EPU on EMCB Reviews (Contd.)

Section 2.1.9 – Impact on Steam Generator Tube Inservice Inspections (TS Requirements)

Section 2.1.10 – Impact on Steam Generator Blowdown System (10 CFR Part 50, Appendix A, GDC 14)

Section 2.1.11 – Impact on Boration Requirements for the Chemical and Volume Control System (CVCS) [TS Requirements]

Waterford 3 EPU

Summary of Results for RV and RV Internals

RV Surveillance Program: The staff confirmed that the changes to the withdrawal schedule satisfied the requirements of 10 CFR Part 50, Appendix H, and conformed to the withdrawal schedule criteria of ASTM E185-82.

USE (Limited by Shell Plate —1003-3): The staff calculated a USE value of 71 ft-lb under the uprated conditions. This plate material satisfies the acceptance criterion of 50 ft-lb at the end of the licensed operating term, as evaluated for the uprated conditions.

Waterford 3 EPU

Summary of Results for RV and Internals (Contd.)

P-T Limits: The staff confirmed that the 32 effective full power year (EFPY) P-T limit curves approved in 2004 were based on the uprated neutron fluence values reported in the licensee's EPU safety evaluation report (SER).

PTS (Limited by Shell Plate —1004-2): The staff calculated a RTpts value of 49 F under the uprated conditions. This plate material satisfies the PTS screening criterion of 270 F for plate materials, as evaluated for the uprated conditions.

Waterford 3 EPU

Summary of Results for RV and Internals (Contd.)

RV Internals: To address potential for aging effects to occur in the RV internals, licensee committed to participate in EPRI-MRP initiatives on PWR RV internals degradation and implement the recommendations resulting from the studies. The specific details of the context of the licensee's commitment on the RV internals will be resolved prior to issuance of the uprated operating license

Waterford 3 EPU

Summary of Results for RCS Piping Integrity and NDE

RCPB Materials: The Waterford 3 EPU results in only a minimal increase in the nominal RCPB hot-leg temperature (+0.8 F) and a slight decrease (-2.0 F) in the RCPB cold-leg temperature. The staff concluded that this will have only a minimal impact on crack initiation and growth rates for the RCPB materials.

Waterford 3 EPU

Summary of Results for RCS Piping Integrity and NDE (Contd.)

- Leak-Before-Break (LBB) Assessment: The operating conditions under the uprated conditions will not alter the conclusions of the previous LBB analysis for the Waterford 3 primary coolant loop piping. Therefore, the staff concludes that the licensee's ability to detect a leak in the Waterford 3 primary coolant loops prior to a limiting loss of coolant accident (LOCA) remains justified.

Waterford 3 EPU

Steam Generator, Protective Coatings, and Water Chemistry

Protective Coating Systems: Changes in pressure, temperature, radiation, and chemistry are bounded by the current design basis. Therefore, the protective coatings remain qualified under the uprated conditions.

Flow-Accelerated Corrosion (FAC): The EPU will cause the wear rates for ferritic pipes in the FAC program to increase. The prediction method in the FAC program has been updated to include the uprated conditions

Waterford 3 EPU

Steam Generator, Protective Coatings, and Water Chemistry (Contd.)

Steam Generator (SG) Tube Inservice Inspection: The EPU may increase SG tube wear at anti-vibration tube support locations. However, the licensee follows inspection guidance in NEI 97-06 and the plant TS to inspect tube wear. The licensee has implemented a conservative primary-to-secondary leakage limit of 75 gallons per day per SG in TS.

Waterford 3 EPU

Steam Generator, Protective Coatings, and Water Chemistry (Contd.)

Steam Generator Blowdown System: The current design of the steam generator blowdown system remains adequate to manage the increase in the feedwater flow rate as a result of the uprated conditions.

CVCS: The boric acid makeup tank volume and concentration will be increased. The boron concentration requirements in the TS have been changed to reflect the uprated conditions.

Waterford 3 EPU

Kamal Manoly

Mechanical and Civil Engineering Branch

Division of Engineering

Office of Nuclear Reactor Regulation

Waterford 3 EPU

Component Evaluation

- Reactor Vessel, Internals, Nozzles, Supports and Control Element Drive Mechanism
- Steam Generator, Reactor Coolant Pump, Pressurizer and Supports
- Nuclear Steam Supply System (NSSS) and Balance-of-Plant (BOP) Piping and Supports
- Safety Related Valves

Waterford 3 EPU

Scope of Review

- Methodology, Loads
- Stresses and Cumulative Fatigue Usage Factors
- Acceptance Criteria, Codes, and Addenda
- Functionality and Impact of EPU on GL 89-10 for MOVs, GL 95-07 for Pressure Locking and Thermal Binding, GL 96-06 for Over-pressurization of Piping Segments Penetrating Containment

Waterford 3 EPU

NSSS and BOP Piping and Supports

EPU evaluation incorporates approved LBB criterion that allows elimination of primary loop pipe breaks postulated in the original design basis. Limiting breaks are in the largest piping branch lines (i.e., MSL, FW, SL, SI, and SDC)

Finite element analysis performed for revised design loads.

Calculated stresses are compared to ASME Code Section III limits.

Waterford 3 EPU

NSSS and BOP Piping and Supports (Contd.)

CUFs for Class 1 piping, calculated based on 40 years and and compared to ASME limit of 1.0.

As a result of EPU evaluation, licensee identified that CCW Shutdown cooling (SDC) heat exchanger outlet piping is currently experiencing higher temperature than design-basis of 175 degree F. Corrective actions involves maintaining piping temperature below 225 degree F via operating procedures. SDC piping stresses meet code limits at this temperature.

Waterford 3 EPU

Flow Induced Vibration

- MSL and FW piping are instrumented at critical locations to monitor vibration levels at current rated power and during EPU power ascension up to the full authorized power level. The vibration monitoring and collected data will be evaluated according to ASME OM3.
- FIV effect on steam dryer is expected to increase at EPU. However, judged to be acceptable based on a comparison to similar plant with same steam dryer design, higher steam flow, higher power level, and higher dynamic pressure. No failure record was identified in the data base for this type of dryer.

Waterford 3 EPU

Flow Induced Vibration due to EPU (contd.)

- Slight increase in FIV on the U-bend tubing, but remains within allowable limits (i.e., maximum stability ratio below 0.8 which is less than the limit of 1.0 and peak stresses are less than material endurance limit).

Waterford 3 EPU

NSSS and BOP Piping Systems and Supports

- Finite element analysis performed for revised design loads
- Calculated stresses compared to ASME Code Section III limits
- Cumulative usage factors (CUFs) for Class 1 piping calculated based on 40 years and compared to ASME limit of 1.0

Waterford 3 EPU

NSSS and BOP Piping Systems and Supports (Contd.)

- As a result of EPU evaluation, licensee discovered that CCW Shutdown cooling (SDC) heat exchanger outlet piping is currently experiencing higher temperature than design-basis of 175°F. Corrective actions involves maintaining piping temperature below 225°F via operating procedures. SDC piping stresses meet code limits at this temperature.

Waterford 3 EPU

Angelo Stubbs

Plant Systems Branch (SPLB)

Division of Systems Safety and Analysis

Office of Nuclear Reactor Regulation

Waterford 3 EPU

Scope of Review

Scope of Review (RS-001, Matrix 5)

Secondary Plant Systems (Steam, Feedwater, Condensate, Circulating Water)

Ultimate Heat Sink and Cooling Water Systems

Main Turbine

Protection from pipe failure, floods, and internally generated missiles

Waterford 3 EPU

Scope of Review (Contd.)

Spent Fuel Pool (SFP) Cooling and Cleanup System

Emergency Feedwater (EFW) System

Fission Product Control and Waste Management Systems

Emergency Diesel Generator (EDG) Fuel Oil Storage and Transfer System

Waterford 3 EPU

Changes which could impact BOP Equipment or Operation

- Increase Decay Heat for EPU Operation
- Modification to HP Turbine
- Changes in system operating parameters (temperature, pressure, and flow)

Waterford 3 EPU

Review Area of Emphasis - Main Turbine

Modification being made to high pressure turbine steam path includes installation of a new high pressure turbine rotor with all reaction blading, a new inner cylinder with stationary blading, a new inlet flow guide, and steam sealing components.

EPU evaluations confirmed that the maximum rotor speed following a reactor trip will remain less than 120% of rated speed and therefore continue to provide adequate overspeed protection.

Waterford 3 EPU

Review Area of Emphasis - Spent Fuel Pool Cooling

EPU increases the decay heat associated with fuel offloaded to the SPF.

Administrative controls will be used for offloading the core to the SPF to ensure that the pool temperature and time to boil will continue to satisfy licensing-basis considerations.

Inputs and methods that will be used by the licensee to determine the core offload limits were reviewed by the staff and found to be acceptable.

Waterford 3 EPU

Review Area of Emphasis - Ultimate Heat Sink

EPU results in an increase in the long-term heat removal requirements following a LOCA.

The EPU evaluation shows that the wet and dry cooling towers have sufficient capacity to accommodate post-LOCA heat loads, and that sufficient water volume will continue to be available in one basin to meet the 30 day heat removal criterion. (The staff conclusion is pending resolution of how the licensee accounts for measurement uncertainty)

Waterford 3 EPU

Review Area of Emphasis - Emergency Feedwater System

EPU increases the decay heat generated and thus the EFW heat removal requirements.

Initial water source of EFS pumps is the condensate storage pond, the backup source is wet cooling tower basin.

Evaluation shows that the increased demand for emergency feedwater for plant cooldown will continue to be met.

Waterford 3 EPU

Review Area of Emphasis - EDG Fuel Oil Storage and Transfer (FOST) System

- EPU increases decay heat and therefore the operating duration for components used to remove decay heat increases.
- The licensee proposed a change to the TSs to increase the minimum required usable volume in the FOST to assure enough fuel oil to meet the 7 day operating criterion for the EDGs.
- The staff found the proposed change to the TS requirements to be acceptable.

Waterford 3 EPU

Richard Lobel

Probabilistic Safety Assessment Branch (SPSB) / Containment System

Division of Systems Safety and Analysis (DSSA)

Waterford 3 EPU

Containment topics

Mass and Energy Release from LOCA to Containment

Containment LOCA Analysis

Subcompartment Analysis

Mass and Energy Release from Main Steam Line Break to Containment

Containment Main Steam Line Break Analysis

Containment Environmental Qualification Analysis

Waterford 3 EPU

Mass and Energy Release from LOCA to Containment

Mass and Energy calculations done utilizing NRC-approved Westinghouse methods:

Ceflash-4a
Flood 3
Contrans

Calculations done for a previous license amendment (Amendment No. 165)

Waterford 3 EPU

Subcompartment Analysis

Subcompartment: Any fully or partially enclosed volume within the primary containment that houses high energy piping and would limit the flow of fluid to the main containment volume

Reactor cavity pressurization limiting

Significant margin

Waterford 3 EPU

Mass and Energy Release from MS Line Break to Containment

- Westinghouse NRC-approved SGNIII Computer Code used to calculate mass and energy release from ruptured steam line into the containment

Waterford 3 EPU

Containment Main Steam Line Break Analysis

- Containment pressure and temperature calculated with GOTHIC 7

- Calculations consistent with NRC staff approval of GOTHIC 7.0 on another docket

Waterford 3 EPU

Containment Environmental Qualification Analysis

- Peak containment pressure and temperature for EPU conditions bounded by existing EQ plant accident profile
- Time at elevated temperatures at EPU conditions slightly longer
- Licensee analysis confirmed that electrical equipment still qualified
- Containment flood level unchanged

Waterford 3 EPU

Calculation Results-LOCA

	EPU	Acceptance Limit
Peak Pressure (psig)	35.16	44
Peak Temperature (°F)	254.4	269.3
Pressure at 24 hours	15.94	16.62

Waterford 3 EPU

Calculation Results-Main Steam Line Break

	EPU	Acceptance Limit
Peak Pressure (psig)	41.83	44
Peak Temperature (°F)	394.4	413.5

Waterford Steam Electric Station, Unit 3

EXTENDED POWER UPRATE (8%)

Sam Miranda

Reactor Systems Branch (SRXB)

Division of Systems Safety and Analysis (DSSA)

Office of Nuclear Reactor Regulation

Waterford 3 EPU

Review Areas

PUR

2.3.5	Station Blackout
2.6.7	Reactor Coolant System
2.6.8	Safety Injection System
2.6.4.4	Shutdown Cooling System
2.6.9	NSSS Design Transients

Waterford 3 EPU

Review Areas (Contd.)

- 2.6.1 Fuel System Design
- 2.6.2 Nuclear Design
- 2.6.3 Thermal and Hydraulic Design
- 2.6.4.1 Functional Design of CED System
- 2.6.4.2 Overpressure Protection during Power Operation

Waterford 3 EPU

Review Areas (Contd.)

2.6.4.3	Overpressure Protection during Low Temperature Operation
2.12.3	Large Break LOCA
2.12.4	Small Break LOCA
2.12.5	Post LOCA Long Term Cooling
2.13.xx	Non-LOCA Events (including ATWS)

Waterford 3 EPU

Audit of selected Westinghouse analyses of events that are

- Sensitive to the plant's uprated conditions, and/or
- Analyzed with new methods (e.g., the CENTS code)

Waterford 3 EPU

Analyses that were selected for detailed review, including some independent calculations

- Loss of feedwater analysis
- Feedline break analysis
- Steamline break analysis
- Small break LOCA
- Long-term cooling, hot-leg injection switchover time, and Boron precipitation

Waterford 3 EPU

Results

- Steam System Piping Failure, Post-Trip Analysis
  - Fuel clad failure < 2%; none due to centerline melting
- Feedwater System "Large" Pipe Break (0.12 sq ft with LOOP)
  - Max RCS Pressure = 2753 psia [3000 psia] and NRC staff calculation = 2780 psia

Waterford 3 EPU

Results (Contd.)

Loss of Condenser Vacuum

Max RCS Pressure = 2732 psia [2750 psia] and Max SG Pressure = 1186 psia [1210 psia]

Small break LOCA (0.055 sq ft CLB)

Max core uncovery: application vs staff calc < ½ foot difference

Waterford 3 EPU

Review Areas

PUR	SER	Event Description	ANSI	
2.13.1.1.1	2.8.5.1.1	Decrease in Feedwater Temperature	II	Bounded by 2.13.1.1.3
2.13.1.1.2	2.8.5.1.1	Increase in Feedwater Flow	II	Bounded by 2.13.1.1.3
2.13.1.1.3	2.8.5.1.1	Increased Main Steam Flow	II	Analyzed & meets Class II criteria
2.13.1.1.4	2.8.5.1.1	Inadvertent Opening of a Steam Generator ADV (IOSGADV)	II	Analyzed & meets Class II criteria
2.13.1.2.1	2.8.5.1.1	Decrease in Feedwater Temperature with SAF	III	Bounded by 2.13.1.2.3
2.13.1.2.2	2.8.5.1.1	Increase in Feedwater Flow with single active failure (SAF)	III	Bounded by 2.13.1.2.3
2.13.1.2.3	2.8.5.1.1	Increased Main Steam Flow with SAF	III	Analyzed & meets Class III criteria

Waterford 3 EPU

Review Areas (Contd.)

PUR	SER	Event Description	ANSI	
2.13.1.2.4	2.8.5.1.1	IOSGADV with LOOP	III	Analyzed & meets Class III criteria
2.13.1.3.1	2.8.5.1.2	Steam System Piping Failure, Post-Trip Analysis	IV	Analyzed & meets Class IV criteria
2.13.1.3.2	2.8.5.1.2	Mode 3 and 4 All Rods In (ARI) RTP SLB	IV	Bounded by 15.1.3.2
2.13.1.3.3	2.8.5.1.2	Steam System Piping Failure, Pre-Trip Power Excursion	IV	Analyzed & meets Class IV criteria
2.13.2.1.1	2.8.5.2.1	Loss of External Load	II	Bounded by 2.13.2.1.3

Waterford 3 EPU

Review Areas (Contd.)

PUR	SER	Event Description	ANSI	
2.13.2.1.2	2.8.5.2.1	Turbine Trip	II	Bounded by 2.13.2.1.3
2.13.2.1.3	2.8.5.2.1	Loss of Condenser Vacuum (LOCV)	II	Analyzed & meets Class II criteria
2.13.2.1.4	2.8.5.2.2	Loss-of-Normal AC Power	II	Bounded by 2.13.2.1.3 and 2.13.3.2.1
2.13.2.1.5	2.8.5.2.1	Steam Pressure Regulator Failure	II	Bounded by 2.13.2.1.3
2.13.2.2.1	2.8.5.2.1	Loss of External Load with SAF	III	Bounded by 2.13.2.2.3
2.13.2.2.2	2.8.5.2.1	Turbine Trip with SAF	III	Bounded by 2.13.2.2.3

Waterford 3 EPU

Review Areas (Contd.)

PUR	SER	Event Description	ANSI	
2.13.2.2.3	2.8.5.2.1	Loss of Condenser Vacuum with SAF	III	Bounded by 2.13.2.1.3
2.13.2.2.4	2.8.5.2.2	Loss-of-Normal AC Power with SAF	III	Bounded by 2.13.3.2.1
2.13.2.2.5	2.8.5.2.3	Loss-of-Normal Feedwater Flow	III	Analyzed & meets Class II criteria
2.13.2.3.1	2.8.5.2.4	Feedwater System Pipe Breaks	IV	Analyzed & meets Class IV criteria
2.13.2.3.2	2.8.5.2.3	Loss of Normal Feedwater Flow with SAF	IV	Analyzed & meets Class II criteria
2.13.3.1.1	2.8.5.3.1	Partial Loss of Forced Reactor Coolant Flow	II	Bounded by 2.13.3.2.1

Waterford 3 EPU

Review Areas (Contd.)

PUR	SER	Event Description	ANSI	
2.13.3.2.1	2.8.5.3.1	Total Loss of Forced Reactor Coolant Flow	III	Analyzed & meets Class III criteria
2.13.3.2.2	2.8.5.3.1	Partial Loss of Forced Reactor Coolant Flow with Single Active Failure (SAF)	III	Bounded by 2.13.3.2.1
2.13.3.3.1	2.8.5.3.2	Single Reactor Coolant Pump (RCP) Shaft Seizure/Sheared Shaft	IV	Analyzed & meets Class IV criteria
2.13.4.1.1	2.8.5.4.1	Uncontrolled CEA Withdrawal from Subcritical	II	Analyzed & meets Class II criteria

Waterford 3 EPU

Review Areas (Contd.)

PUR	SER	Event Description	ANSI	
2.13.4.1.2	2.8.5.4.1	Uncontrolled CEAW from Low Power	II	Analyzed & meets Class II criteria
2.13.4.1.3	2.8.5.4.2	Uncontrolled CEAW at Power	II	Analyzed & meets Class II criteria
2.13.4.1.4	2.8.5.4.3	CEA Misoperation	II	Reload analyses will meet Class II criteria
2.13.4.1.5	2.8.5.4.5	Inadvertent Boron Dilution	II	Bounded by FSAR 15.4.1.5
2.13.4.1.6	2.8.5.4.4	Startup of an Inactive Reactor Coolant Pump	II	Bounded by FSAR 15.4.1.6
2.13.4.1.7	2.8.5.4.1	CEAW Modes 3, 4 and 5 ARI	II	Bounded by FSAR 15.4.1.7

Waterford 3 EPU

Review Areas (Contd.)

PUR	SER	Event Description	ANSI	
2.13.4.3.1	2.8.5.4.7	Inadvertent Loading of a Fuel Assembly into an Improper Position	IV	Analyzed & meets Class II criteria
2.13.4.3.2	2.8.5.4.6	Control Element Assembly Ejection	IV	Analyzed & meets Class IV criteria
2.13.5.1.1	2.8.5.5.5	Chemical & Volume Control System (CVCS) Malfunction	II	Bounded by FSAR 15.5.1.1
2.13.5.1.2	2.8.5.5.5	Inadvertent Emergency Core Cooling System (ECCS)	II	Bounded by FSAR 15.5.1.2
2.13.5.2.1	2.8.5.5.5	CVCS Malfunction with Single Active Failure (SAF)	III	Bounded by FSAR 15.5.2.1

Waterford 3 EPU

Review Areas (Contd.)

PUR	SER	Event Description	ANSI	
2.13.6.3.1	2.8.5.6.3	Small Primary Line Break Outside Containment	III	Analyzed & meets Class III criteria
2.13.6.3.2	2.8.5.6.2	Steam Generator Tube Rupture with LOOP	IV	Analyzed & meets Class III criteria
2.13.6.3.3	2.8.5.6.3	Loss of Coolant Accident (LOCA) Radiological Consequences	IV	Analyzed & meets Class IV criteria
2.13.6.4	2.8.5.5.6	Inadvertent Opening of a Pressurizer Safety Valve	IV	Bounded by SBLOCA, 2.12.4

Waterford 3 EPU

Review Areas (Contd.)

PUR	SER	Event Description	ANSI
2.13.7.3.1		Radioactive Waste Gas System Leak or Failure	RS-001 Section 2.5.5.1, SRP 11.3
2.13.7.3.2		Liquid Waste System Leak or Failure	RS-001 Section 2.5.5.2, SRP 11.2
2.13.7.3.3		Postulated Radioactive Releases Due to Liquid Containing Tank Failures	RS-001 Section 2.5.5.2, SRP 11.2

Waterford 3 EPU

Review Areas (Contd.)

PUR	SER	Event Description	ANSI	
2.13.7.3.4		Radiological Consequences of Fuel Handling Accidents		Bounded by FSAR 15.7.3.4 SRP 15.7.4
2.13.7.3.5		Spent Fuel Cask Drop Accidents		Bounded by FSAR 15.7.3.5 SRP 15.7.5
2.13.8	2.8.5.5.7	Anticipated Transients Without Scram (ATWS)	n/a	Meets the requiremen ts of 10CFR50.6 2

Waterford 3 EPU

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U.S. Nuclear Regulatory Commission

Waterford 3 EPU

Reactor Systems Branch Audit Calculations

- Agenda
  - Large Feedwater Line Break
  - Limiting Small Break LOCA
  - Post-LOCA Long Term Cooling

  (Boric Acid Precipitation and Timing for Simultaneous Hot/Cold Side Injection)

Waterford 3 EPU

Reactor Systems Branch Audit Calculations

- Large Feedwater System Pipe Break
  - Alternate Methodology Verified Peak RCS Pressure
  - Conservative Analysis Assumptions (break at the elevation of the tube sheet)

Waterford 3 EPU

Reactor Systems Branch Audit Calculations

- Limiting Small Break LOCA in the Pump Discharge Leg
  - Staff Calculations Reproduced CEFLASH-4AS Core Transient Two-phase Level for the Limiting Small Break(0.055 ft2 CLB)
  - No Credit for Accumulator Injection
  - Conservative Analysis Assumptions (Top Skewed Axial shape, Diesel Failure, 1.2 Decay Heat Multiplier)

Waterford 3 EPU

Reactor Systems Branch Audit Calculations

- Post-LOCA Long Term Cooling (Prevention of Boric Acid Precipitation)
  - Staff Calculations Revealed Error in Mixing Volume (assumed void fraction of 0% in mixing volume following Large Break LOCAs)
  - Error Produces Precipitation at One Hour vs Four Hours
  - Westinghouse has Corrected Error and Modified Licensing Methodology
    - Mixing Volume Reflects Liquid in Core and Upper Plenum to Hot-Leg Top EL (vs mixing vol to hot-leg bottom elevation)
    - Minimum Containment Pressure Raised to 20 psia (vs 14.7 psia)
    - Performed Min. Cont. Pressure Calculation using NRC Approved Methodology (GOTHIC)

Waterford 3 EPU

Reactor Systems Branch Audit Calculations

- Staff Believes Adequate Margin Remains to Support Power Uprate
  - No Credit for Liquid Entrainment (also no removal of boric acid by vapor)
  - No Mixing in Hot Legs
  - Boric Acid Make-Up Tanks, BAMTs, Discharge (6187 ppm)
  - Upper Plenum Pressure Higher than Cont. by Loop Pressure Drop
- Westinghouse will Document Changes to Methodology and Revised Analyses

Waterford Steam Electric Station, Unit 3

EXTENDED POWER UPRATE (8%)

PROBABILISTIC SAFETY ASSESSMENT BRANCH (SPSB)
Containment and Accident Dose Assessment Section – Design Basis Accident (DBA) Dose Assessment

Michelle Hart

Waterford 3 EPU

Dose Assessment Review

- Regulatory Requirements
  - 10 CFR Part 100
  - GDC-19
- Review Conducted in Accordance with Applicable SRP Sections and RS-001
  - Licensee's analyses followed applicable guidance
    - Any differences were justified by the licensee and found acceptable by the staff
  - Staff performed confirmatory dose analyses

Waterford 3 EPU

Design Basis Accidents Evaluated

- Main Steam Line Break
- RCP Locked Rotor
- Control Rod Ejection
- Small Line Break Outside Containment
- SGTR
- LOCA
- Fuel Handling Accident
- Spent Fuel Cask Drop

Waterford 3 EPU

Draft SE Open Item

- Control Room Habitability Dose Analyses

Waterford 3 EPU

CRH Assessment

- Submitted EPU control room dose analyses
  - Used unverified values for control room unfiltered inleakage
    - Testing planned to take place while the EPU under review and Entergy would modify the EPU submittal as necessary
  - Only for those accidents already evaluated in the FSAR
    - LOCA, FHA

Waterford 3 EPU

CRH Assessment (cont.)

- Tracer gas test results not bounded by EPU dose analysis assumptions
  - Separate full-scope AST submittal
    - Submitted 7/15/04, w/ 4 supplements through 10/19/04
    - Control room dose analyses in EPU supplanted by AST
    - Control room dose analyzed for all DBAs
    - Used control room unfiltered inleakage assumption based on tracer gas test results
  - AST review scheduled for completion by March 10, 2005
    - No apparent technical problems meeting schedule

Waterford 3 EPU

Dose Results

- All exclusion area boundary (EAB) and low population zone (LPZ) doses meet Part 100 and are within SRP dose acceptance criteria for each DBA for the EPU
- Control room doses will be evaluated against GDC 19 and SRP 15.0.1 dose acceptance criteria as part of the review of the Waterford AST amendment request.
  - EPU can not be implemented without AST implementation

WATERFORD, UNIT 3

Extended Power Uprate Test Program

Paul Prescott

Senior Operations Engineer

Quality and Maintenance Section (IPSB)

Division of Inspection Program Management (DIPM)

Waterford 3 EPU

Test Program

- IPSB responsible for review coordination. Secondary review branches responsible for reviewing application to ensure SSCs will perform satisfactorily in service.

- SRP 14.2.1, "Generic Guidelines for Extended Power Uprate Testing Programs," provides guidance for testing programs based on RG 1.68 and plant specific initial test program. Guidance calls for performance of large transient testing (LTT) and considers original Power Ascension Test program and EPU related plant modifications.

Waterford 3 EPU

Test Program

- Guidance acknowledges that licensees may propose alternative approaches to testing. SRP provides supplemental guidance for staff evaluation of alternative approaches. Licensees responsible to justify proposed alternative approaches.

- Staff has previously approved 12 EPUs ranging from 106 -120 % over the licensed thermal power without performing LTTs.

Waterford 3 EPU

Test Program

- Previously accepted justifications for not performing LTT for Pre-RS-001/SRP EPU applications:
  - The licensee's test program will monitor important plant parameters during EPU power ascension.
  - TS surveillance and post-mod testing will confirm the performance capability of the modified components.
  - Operating history and experience at other uprated light-water reactors (LWRs).
  - LTT is not needed for Code analyses benchmarking.

Waterford 3 EPU

Test Program

- Previously accepted justifications for not performing LTT were applicable to the Waterford 3 EPU application:
- Staff perceived need for additional transient testing
  - Resulted in a staff RAI to the applicant (10/26/04)
  - The licensee response, accepted by the staff, was based on:
    - Consideration of previous operating experience,
    - Analytical methods,
    - Analysis of potential unexpected systems interactions,
    - Effects on design margin, and
    - Limited scope of modifications.

Waterford 3 EPU

Test Program

- Regulatory Guide 1.68 testing "Objectives"
  - Operator training and familiarization,
  - Confirmation of design and installation of equipment,
  - Benchmarking of analyses codes and models, and
  - Confirmation of the adequacy of emergency and operating procedures.

Staff basis for requiring performance of LTT should consider the above.

Waterford 3 EPU

Summary

- SRP 14.2.1 allows for justification for not performing EPU Power Ascension Tests.
- Twelve domestic LWRs have implemented staff approved EPUs (up to 120% OLTP) without performance of LTT.
- Conducting LLTs would not provide significant new information regarding transient modeling and component performance.

WATERFORD STEAM ELECTRIC STATION, UNIT 3

EXTENDED POWER UPRATE (8.0%)

Martin A. Stutzke
Probabilistic Safety Assessment Branch
Division of Systems Safety and Analysis
Office of Nuclear Reactor Regulation

Waterford 3 EPU - Risk Evaluation

- Risk Evaluation performed to:
  - Demonstrate that risk are acceptable
  - Determine if "special circumstances" exist, SRP 19, Appendix D, that could rebut the presumption of adequate protection provided by meeting current regulations
- Review conducted per NRR RS-001, Rev. 0, "Review Standard for Extended Power Uprates," Matrix 13, "Risk Evaluation."

Waterford 3 EPU - Risk Evaluation

Review Scope

- Internal Events
- External Events
  - Internal flooding (screening approach)
  - Internal fires (FIVE methodology)
  - Seismic events (seismic margins analysis)
  - HFO external events (NUREG-1407 screening)
- Level 2 PRA (modified NUREG/CR-6595)
- Shutdown events (Qualitative; per SRP 19)
- PRA quality

Waterford 3 EPU - Risk Evaluation

Overall results (1 of 3)

- Internal Events
  - CDF ~ 6E-6/y, ΔCDF ~ 4E-7/y
  - LERF ~ 2E-6/y, ΔLERF ~ 7E-8/y
- Internal flood CDF ~ 2E-6/y, timing of associated operator actions do not depend on reactor power level
- Internal fire
  - CDF ~ 8E-6/y, ΔCDF ~ 7E-10/y
  - ΔLERF ~ 7E-11/y

Waterford 3 - Risk Evaluation

Overall results (2 of 3)

- Seismic risk
  - Waterford 3 classified in NUREG-1407 as a reduced scope plant
  - Increase in power level not expected to affect equipment survivability or response
  - No change in the safe shutdown pathways
- HFO events screened out in IPEEE; increase in power level does not affect HFO event occurrence frequencies

Waterford 3 EPU - Risk Evaluation

Overall results (3 of 3)

- Shutdown risk
  - Very little change to shutdown schedule
  - Shutdown Operations Protection Plan (SOPP) used to ensure
    - Decay heat removal
    - RCS inventory control
    - Vital control power
    - Reactivity control
    - Containment closure
  - Increase in power level not expected to affect
    - Shutdown equipment reliability
    - Availability of equipment or instrumentation used for contingency plans

Waterford 3 EPU - Risk Evaluation

PRA Quality

- IPE submitted 8/8/92, accepted 1/3/97
- IPEEE submitted 7/28/95, accepted 7/27/00
- Owners Group peer review in January 2000
- Several PRA updates, latest was June 2003
- PRA maintained as a quality record
- Staff checked the resolution of IPE, IPEEE, and peer review findings

Waterford 3 EPU - Risk Evaluation

Staff Conclusions

- Licensee has adequately modeled and/or addressed the potential risk impacts
- Risks are acceptable because RG 1.174 risk acceptance guidelines are met
- Proposed EPU does not create "special circumstances" that rebut the presumption of adequate protection provided by meeting current regulations