NOTICE OF VIOLATION

Duke Energy Corporation
Oconee Nuclear Station
Units 1, 2, and 3

Docket Nos. 50-269, 270, and 287
License Nos. DPR-38, 47, and 55
EA 98-268

During an NRC inspection conducted between April 22 and May 20, 1998, a violation of NRC requirements was identified. In accordance with the "General Statement of Policy and Procedures for NRC Enforcement Actions," NUREG-1600, the violation is listed below:

> 10 CFR 50 Appendix B, Criterion III, Design Control, requires in part, that measures shall be established to ensure that the design basis is correctly translated into specifications, drawings, procedures, and instructions.
>
> Final Safety Analysis Report (FSAR) Section 6.3 states, in part, that the emergency core cooling system is designed to operate by injection of borated water from the borated water storage tank (BWST) by the high pressure injection (HPI) and low pressure injection (LPI) systems and provide long-term cooling by recirculation of injection water from the reactor building emergency sump (RBES) by the LPI pumps. FSAR Section 6.2 states, in part, that the reactor building spray (BS) pump suction is transferred to the RBES when LPI is placed in the recirculation mode.
>
> Technical Specification (TS) 3.3.4 requires, in part, that the BWST have two level instrument channels operable.
>
> Contrary to the above, from installation of BWST level instruments in 1989 until February 1998 and from installation of the reactor building (RB) level instruments in December 1986 until February 1998, the licensee failed to ensure that the design bases for the HPI, LPI and BS systems in the three Oconee Units were correctly translated into drawings and procedures. Specifically, the licensee failed to appropriately account for: (1) the as-built height configuration of the BWST level instrument taps in system design drawings and calibration procedures, thereby failing to maintain two BWST level instrument channels operable as required by TS 3.3.4; and (2) RB level instrument uncertainties in emergency operating procedures. These two design control errors would have significantly affected reactor operators' ability in certain design basis accident scenarios to follow emergency operating procedure (EOP) steps to swap the HPI, LPI and BS system suction from the BWST to the RBES to prevent air entrainment and potential damage in the HPI, LPI and BS pumps (due to BWST vortexing) resulting in an inability to perform their intended safety function. (01012)

This is a Severity Level II violation (Supplement I).

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The NRC has concluded that information regarding the reason for the violation, the corrective actions taken and planned to correct the violation and prevent recurrence and the date when full compliance was achieved is already adequately addressed on the docket in NRC inspection reports, the Licensee Event Report (LER) which you have submitted on this issue, and the materials you presented at the conference. However, you are required to submit a written statement or explanation pursuant to 10 CFR 2.201 if the description therein does not accurately reflect your corrective actions or your position. In that case, or if you choose to respond, clearly mark your response as a "Reply to a Notice of Violation," and send it to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555 with a copy to the Regional Administrator, Region II, and a copy to the NRC Resident Inspector at Oconee, within 30 days of the date of the letter transmitting this Notice of Violation (Notice). Your response may reference or include previously docketed correspondence, if the correspondence adequately addresses the response.

If you contest this enforcement action, you should also provide a copy of your response to the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001.

Under the authority of Section 182 of the Act, 42 U.S.C. 2232, this response shall be submitted under oath or affirmation.

Because your response will be placed in the NRC Public Document Room (PDR), to the extent possible, it should not include any personal privacy, proprietary, or safeguards information so that it can be placed in the PDR without redaction. If personal privacy or proprietary information is necessary to provide an acceptable response, then please provide a bracketed copy of your response that identifies the information that should be protected and a redacted copy of your response that deletes such information. If you request withholding of such material, you <u>must</u> specifically identify the portions of your response that you seek to have withheld and provide in detail the bases for your claim of withholding (e.g., explain why the disclosure of information will create an unwarranted invasion of personal privacy or provide the information required by 10 CFR 2.790(b) to support a request for withholding confidential commercial or financial information). If safeguards information is necessary to provide an acceptable response, please provide the level of protection described in 10 CFR 73.21.

Dated at Atlanta, Georgia
this 5th day of August 1998

Anne or Linda
1 page

NAME	[illegible]	ABoland	CEvans	JJohnson Pages
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NAME	BWestreich	JLieberman			
DATE	7/31 /98	7/3 /98	8/ 4 /98		
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>
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Enclosure 1

DEC 8

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OFFICE	OE	OE:D	DEDE		
Signature		JL			
NAME	BWestreich	JLieberman			
DATE	7/ /98	7/ 31 /98	7/ /98		
COPY?	YES NO	YES NO	YES NO		

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L Watson or
Anne

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LIST OF CONFERENCE ATTENDEES

Nuclear Regulatory Commission

J. Johnson, Deputy Regional Administrator, RII
C. Evans, Regional Counsel, RII
C. Casto, Deputy Director, Division of Reactor Projects (DRP), RII
B. Mallett, Director, Division of Reactor Safety (DRS), RII
A. Boland, Director, Enforcement and Investigations Coordination Staff, RII
K. Landis, Chief, Engineering Branch, DRS, RII
C. Ogle, Chief, Branch 1, DRP, RII
R. Carroll, Project Engineer, Branch 1, DRP, RII
D. Billings, Resident Inspector - Oconee, Branch 1, DRP, RII
M. Thomas, Reactor Inspector, DRS, RII
R. Bernhard, Senior Reactor Analyst, DRS, RII
L. Watson, Enforcement Specialist, EICS, RII
*H. Berkow, Director, Project Directorate II-2,(DRPE) Office of Nuclear Reactor Regulation (NRR)
*D. LaBarge, Project Manager, DRPE, NRR
*S. Malur, DRPE, NRR
*B. Westreich, Enforcement Specialist, Office of Enforcement
*C. Jackson, Reactor System Branch, NRR
*L. Smith, Office of the Executive Director for Operations
*S. Athavale, Division of Reactor Controls and Human Factors, NRR

Duke Energy Corporation (DEC)

M. Tuckman, Executive Vice President, DEC
B. McCollum, Vice President, Oconee Nuclear Station (ONS)
J. Forbes, Station Manager, ONS
M. Nazar, Engineering Manager, ONS
G. Davenport, Operations Manager, ONS
E. Burchfield, Regulatory Compliance Manager, ONS
B. Foster, Safety Assessment, ONS
P. Newton, Chief Council - Nuclear, DEC
M. Barrett, Manager, Probabilistic Risk Assessment Staff, DEC
R. Sweigart, Operating Experience Manager, DEC
G. Swindlehurst, Safety Analysis Director, DEC

*Participated by video-conference

Enclosure 2

PREDECISIONAL ENFORCEMENT CONFERENCE AGENDA
OCONEE NUCLEAR STATION

JUNE 22, 1998, 10:30 A.M.
NRC REGION II OFFICE, ATLANTA, GEORGIA

I. OPENING REMARKS AND INTRODUCTIONS
   J. Johnson, Deputy Regional Administrator

II. NRC ENFORCEMENT POLICY
   A. Boland, Director, Enforcement and Investigation Coordination Staff

III. SUMMARY OF THE ISSUES
   J. Johnson, Deputy Regional Administrator

IV. STATEMENTS OF CONCERNS / APPARENT VIOLATIONS
   C. Casto, Deputy Director, Division of Reactor Projects

V. LICENSEE PRESENTATION

VI. BREAK / NRC CAUCUS

VII. NRC FOLLOWUP QUESTIONS

VIII. CLOSING REMARKS
   J. Johnson, Deputy Regional Administrator

Enclosure 3

A. (EEI 98-12-01)

10 CFR 50.46, delineates acceptance criteria for emergency core cooling systems (ECCS) for light water nuclear power reactors. It states in part that the ECCS must be designed to be capable of long-term cooling of the reactor core.

Technical Specification 3.3.4 requires in part that the BWST have operable two level instrument channels, contain a minimum level of 46 feet of water having a minimum concentration of boron within the limit specified in the Core Operating Limits Report at a minimum temperature of 50 degrees F.

Technical Specification 3.3.1 for the high pressure injection (HPI) system states in part that: (1) two independent trains, each comprised of an HPI pump and flow path capable of taking a suction from the borated water storage tank (BWST) and discharging to the reactor coolant system (RCS) automatically upon engineered safeguards protective system (ESPS) actuation shall be operable; (2) Two independent flowpaths allowing the HPI system to take suction from the discharge of the LPI system by manual-local operator action shall be operable; and (3) the remaining HPI pump and valves HP-409 and HP-410 shall be operable and valves HP-99 and HP-100 shall be open.

Technical Specification 3.3.2 for low pressure injection (LPI) system states in part that two independent LPI trains, each comprised of an LPI pump and a flowpath capable of taking suction from the BWST and discharging into the RCS automatically upon ESPS actuation, together with two LPI coolers and two reactor building emergency sump isolation valves shall be operable.

Technical Specification 3.3.6 requires in part, both reactor building spray (BS) trains, each comprised of a BS pump and a flowpath capable of taking suction from the LPI system and discharging through the spray nozzle header automatically upon ESPS actuation, shall be operable.

Note: The apparent violations discussed in this enforcement conference are subject to further review and are subject to change prior to any resulting enforcement action.

(EEI 98-12-01 cont'd)

From initial construction until correction in February 1998, the BWST level instrumentation in all three Oconee Units did not have a height difference calculation included in their instrument calibration; thereby making them inoperable due to indicating as much as 1.5 feet higher than actual BWST level. Consequently, air entrainment (due to BWST vortexing) would have occurred under certain design basis accident scenarios, rendering ECCS (LPI and HPI) and BS pumps inoperable because there was not reasonable assurance that they could fulfill their intended safety functions and assure long-term core cooling in all accident scenarios.

Note: The apparent violations discussed in this enforcement conference are subject to further review and are subject to change prior to any resulting enforcement action.

B. (EEI 98-12-02)

10 CFR 50 Appendix B, Criterion III, Design Control, requires in part, that measures shall be established to ensure that the design basis is correctly translated into specifications, drawings, procedures, and instructions.

Failures to appropriately account for the as-built height configuration of the borated water storage tank (BWST) level instrument taps (in system design drawings and calibration procedures from initial construction until February 1998) and the reactor building level instrument uncertainties (in emergency operating procedures since instrument installation in December 1986 until February 1998) did not meet the design control requirements of 10 CFR 50 Appendix B, Criterion III. Consequently, air entrainment (due to BWST vortexing) would have occurred under certain design basis accident scenarios, rendering ECCS (LPI and HPI) and BS pumps inoperable because there was not reasonable assurance that they could fulfill their intended safety functions and assure long-term core cooling in all accident scenarios.

Note: The apparent violations discussed in this enforcement conference are subject to further review and are subject to change prior to any resulting enforcement action.

Oconee Nuclear Station

Predecisional Enforcement Conference

June 22, 1998

Enclosure 4

Agenda

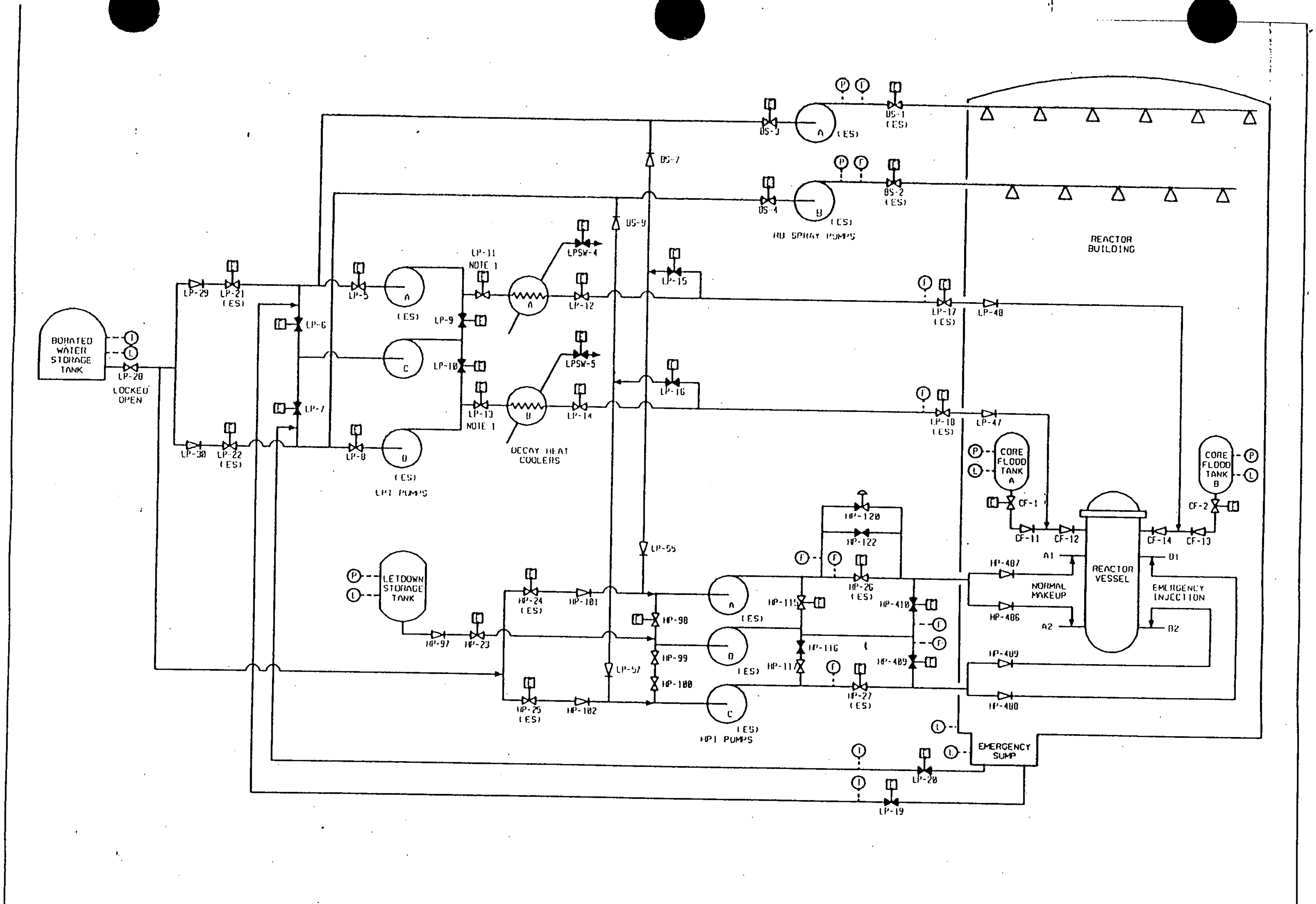
- Apparent Violations
- Overview of Swapover
- Sequence of Events
- Root Cause
- Corrective Actions
- Safety Significance
- Closing Remarks

Perspective

- BWST issue found as part of Recovery Plan initiative
- HPI/LPI SITA comprehensive
  - » Self-initiated
  - » Critical
- BWST and sump level issues self-identified through SITA
- Management response and corrective actions prompt and comprehensive
- Safety perspective
  - » No impact on public health and safety
  - » No significant impact on core damage frequency
  - » The ECCS system might not have performed its intended function under certain conditions
- Recovery Plan will continue efforts to find problems

Apparent Violations

- Restatement of apparent violations:
  - » Violation of Technical Specifications 3.3.1 (HPI), 3.3.2 (LPI), 3.3.4 (BWST), and 3.3.6 (RBS) in that incorrect calibration of BWST level transmitters rendered safety systems inoperable
  - » 10 CFR 50 Appendix B, Criterion III, in that licensee failed to incorporate design information into BWST setpoint calibration procedures and the EOP



TYPICAL FOR UNITS 1, 2, & 3.

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

SUMMARY FLOW DIAGRAM OF EMERGENCY CORE COOLING AND

Overview of Swapover

- Oconee design is a manual swapover to sump recirculation
  - » Three redundant indications of BWST level and two redundant indications of Wide Range Reactor Building Water Level
- Large Break LOCA
  - » When BWST level ≥ 6 feet AND Wide Range Reactor Building Water Level ≥ 4 feet:
    - – Open sump suction valves (LP-19 and LP-20)
  - » When BWST level ≤ 6 feet and > 2 feet:
    - – Close BWST outlet valves (LP-21 and LP-22)

Overview of Swapover

- Small Break LOCA
  - » When BWST level ~ 10 feet:
    - – Establish conditions with one LPI pump in operation
    - – Open LPI discharge cross-connect valves (LP-9 and LP-10)
    - – Open LPI to HPI valves (LP-15 and LP-16)
  - » When BWST level ≥ 6 feet AND Wide Range Reactor Building Water Level ≥ 4 feet:
    - – Open sump suction valves (LP-19 and LP-20)
    - – Close HPI BWST suction valves (HP-24 and HP-25)
  - » When BWST level ≤ 6 feet and > 2 feet:
    - – Close BWST outlet valves (LP-21 and LP-22)

Review of BWST Level Calibration Issue

- Self-Initiated Technical Audit (SITA) of HPI and LPI Systems conducted in November and December of 1997
- PIP initiated on 1/12/98 to address potential BWST level errors
- System engineer evaluated PIP and schedule was established based on experience and expected outcome
- Requested instrument surveys based on apparent drawing discrepancies
- System engineer received initial BWST transmitter elevation surveys on 2/11/98

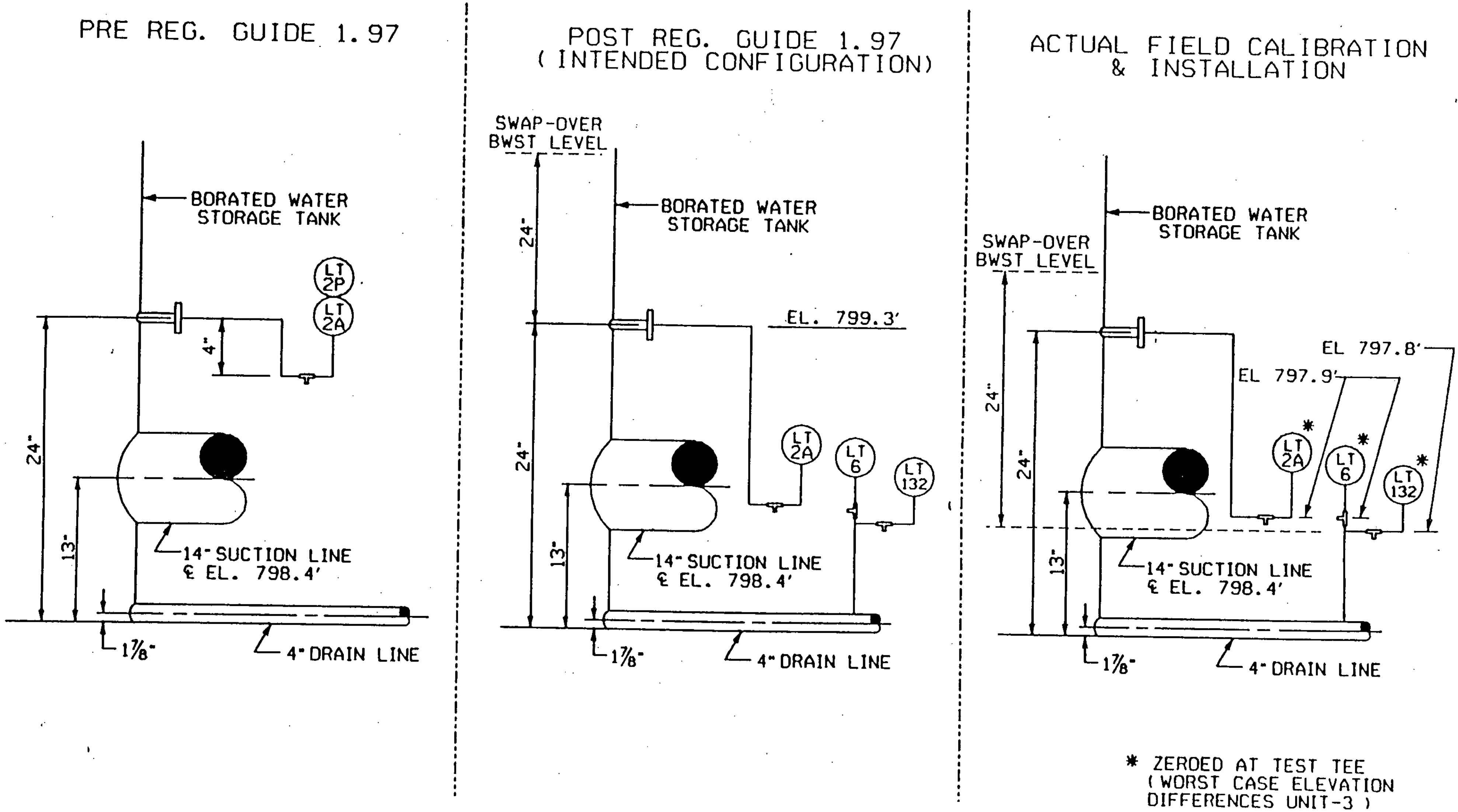
Review of BWST Level Calibration Issue

- Resurvey conducted on 2/12/98
  - » Worst case instrument was calibrated approximately 1.5 feet below lower tap
- BWST level transmitters declared inoperable at 1815 hours on 2/12/98
- One-hour non-emergency notification made at 1830 hours on 2/12/98
- All BWST level transmitters recalibrated by 0431 hours on 2/13/98 and Tech Spec action statements exited
- Site responded promptly to identified problem

Review of BWST Level Calibration Issue

- Two pneumatic level transmitters in 1973
  - » Drawings showed 4" instrument to tap elevation difference
  - » EOP required swapover at 3 feet in BWST
- EOP revised in 1985 to require sump swapover between BWST level of 6 and 2 feet
- Reg Guide 1.97 BWST level modifications implemented in 1989
  - » Three electronic, QA-1 level instruments
  - » Worst case instrument to tap elevation difference of 1.5 feet
- Calibration allowance for elevation differences between tap and instrument implemented in 1998

BWST INSTRUMENTS AND ELEVATION DIFFERENCE BETWEEN TAP AND TRANSMITTER PRE 1989 AND POST 1989



Review of Sump Level Issue

- System Engineer identified apparent EOP conflict on 2/19/98
  - » Conditional requirement to verify $\geq$ 4 feet in the Reactor Building sump may not be met based on revised sump inventory calculations and worst case sump level instrument uncertainties
- PIP was initiated on 2/19/98
- Review concluded on 2/20/98 that EOP guidance did not fully address sump level instrument uncertainties
- Interim guidance provided to the operators at 1700 hours on 2/20/98
- One-hour non emergency notification made at 1710 hours on 2/20/98
- EOP was revised to correct conditional statement prior to 2400 hours on 2/20/98

Review of Sump Level Issue

- Original EOP did not require verification of sump level
- Sump level requirement of 2 feet initially incorporated into EOP in 1985
  - » Requirement added to cover beyond design basis scenarios where inventory may be lost from containment
- Reg Guide 1.97 Wide Range Reactor Building Water Level instruments installed between 1984 and 1986
- EOP incorporated revised sump level of 3.5 feet in 1988
- EOP sump level increased to 4 feet in 1994 to address calculation revisions
- Expected sump inventory changed from 5.3 feet to 4.3 feet in 1997

Prior Opportunities

- Complexity of issue made it difficult to identify error
  - » Normal surveillance and QA activities
  - » BWST level modification in 1989
  - » EOP sump level change in 1994
  - » Reactor Building inventory calculation revision in 1997
  - » Evaluation of Information Notice 91-75

Root Cause of Tech Spec Violation

- Inadequate BWST level calibration procedure that did not contain a head correction for the difference in elevation of the instruments and the associated level tap
  - » Design input requirements for tap location head corrections were not identified for BWST level modification in 1989

Completed Corrective Actions

- Broad approach to corrective actions to look at all potentially affected procedures
- BWST level calibration procedures were revised
- All nine BWST level transmitters were recalibrated to include head corrections
- Identified BWST drawing errors have been corrected

Completed Corrective Actions

- Reviewed head corrections for installed instruments used for surveillances and periodic tests
- Unit 2 narrow range and wide range pressure instruments surveyed and calibration procedures revised

Planned Corrective Actions

- Units 1 and 3 narrow range and wide range RCS pressure calibration procedures will be revised at next available shutdown
  - » Effect evaluated and shown to be negligible
- Implement enhancements to process for controlling static head corrections

Design Control of Instrument Uncertainties

- Special project to upgrade instrument uncertainties initiated in 1996
  - » Upgrades uncertainty calculations
  - » Verifies calculation assumptions agree with calibration procedures
  - » Verifies appropriate application of revised uncertainties in meeting design requirements
  - » Clearly establishes design basis with respect to instrument uncertainties

Design Control of Instrument Uncertainties

- Site Engineering has completed its phase of this project
  - » 80 instrument strings
- General Office
  - » Engineered Safeguards - complete
  - » Reactor Protective System - September 1998
  - » EOP Instruments
    - Detailed review complete
    - Enhance calculations (i.e., update references) by October 1998
    - BWST review had been planned as part of upgrade
- Changes to calculational process warranted based on this event

Root Cause of Design Control Violation

- Process interaction weakness between interrelated design documents
  - » Relationship between BWST level uncertainty calculation, EOP guidance, and calibration procedure regarding head correction
  - » Relationship between EOP guidance, sump level uncertainty calculations, and predicted sump inventory calculations

Completed Corrective Actions

- EOP revised to remove sump level requirement for swapover
- EOP setpoints reviewed for accuracy and completeness
- Benchmarking completed to identify industry best practices regarding the identification, control, and linkage of calculation input assumptions with other documents
  - » 4 sites reviewed
  - » Benchmarking effort completed - 4/98
  - » Results incorporated into planned process changes

Completed Corrective Actions

- Assessment performed to identify improvements to Oconee's calculation process
  - » 12 person week effort completed 4/98
  - » Identified and evaluating 4 major areas of improvement:
    - – Cross disciplinary reviews for linkage of affected documents
    - – Implement SAROS calculation data base
    - – Enhanced process control
    - – Enhanced personnel training
  - » Implementation of recommendations by 12/98

Planned Corrective Actions

- Safety-related, risk significant historical calculations will be enhanced to identify, control, and maintain Oconee-specific calculation inputs
  - » Criteria and scope of calculations to be reviewed - complete by 12/98
  - » Actual calculation review process will begin in 1999 based on identified scope

Planned Corrective Actions

- Perform a risk-informed review of operating experience
  - » Develop systems, equipment, and operator actions that have greatest impact on ONS CDF (complete)
  - » Compile Industry operating experience for above review areas (in progress)
  - » Perform field review of selected OE items (completion during 1999)
  - » Unresolved items addressed through CA program
- Perform an assessment of OE process and implementation by end of August 1998

Safety Significance

- Oconee has analyzed safety significance in four areas:
  - » BWST inventory
  - » Large Break LOCAs
  - » Small Break LOCAs
  - » Reactor Building pressure

Safety Significance

- BWST inventory
  - » Accident analyses credit 40 feet of BWST inventory for sump recirculation
  - » The BWST calibration error did not impact 40 feet of inventory being delivered to the sump
  - » Required Tech Spec level of 46 feet satisfied at all times

Safety Significance

- Key issue: Impact of sump level indicating < 4 feet at the time of swapover
  - » If conditional step not satisfied: Team is forced to make a decision
- Time available dependent on break size

Safety Significance

- Procedural guidance (OMP 1-9)
  - » During off normal plant operations, Emergency and Abnormal procedures are provided to the Operators to respond to these events. Operators are expected to follow procedures when responding to off normal events. However, certain situations may arise where the guidance provided by the emergency or abnormal procedure is deficient or not applicable.
  - » In these cases:
    - – Operators may take reasonable action that deviates from their procedures as necessary to protect the public health and safety.
    - – Such deviations shall be approved by the Operations Shift Manager or, in his absence, the Unit Shift Supervisor or Control Room SRO.

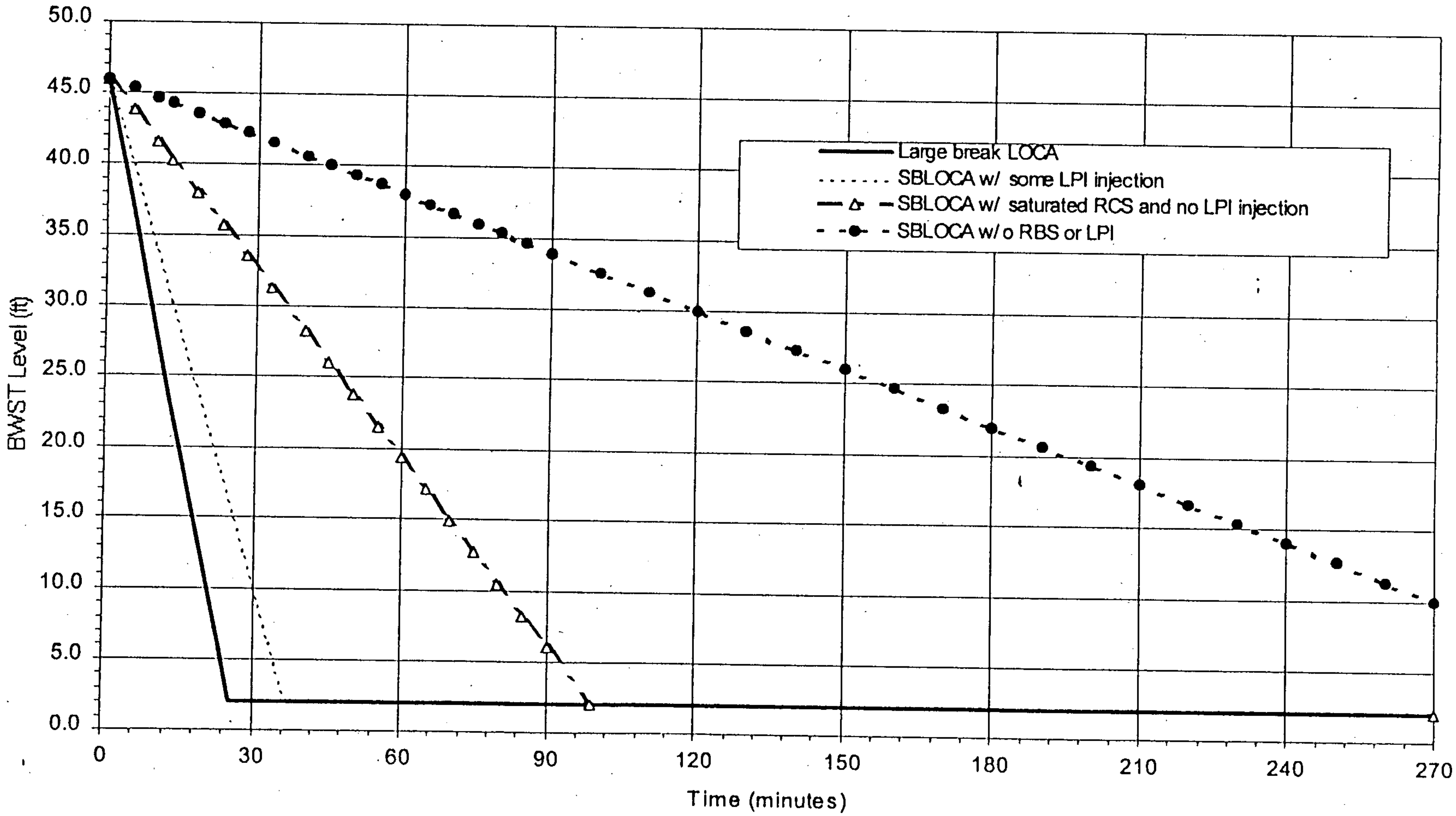
Safety Significance

- Diagnostic tools available to team to assist in decision:
  - » BWST level
  - » Wide Range Reactor Building level
  - » Reactor Building Emergency Sump level
  - » Reactor Building pressure
  - » Subcooled margin monitors
  - » Core exit thermocouples
  - » Radiation alarms
  - » Auxiliary Building waste tank levels
  - » Reports from plant personnel (RP, operators, etc.)
- Earlier step in EOP to verify Reactor Building water level
  - » Operators trained as "commit to memory" on sump swapover initiation

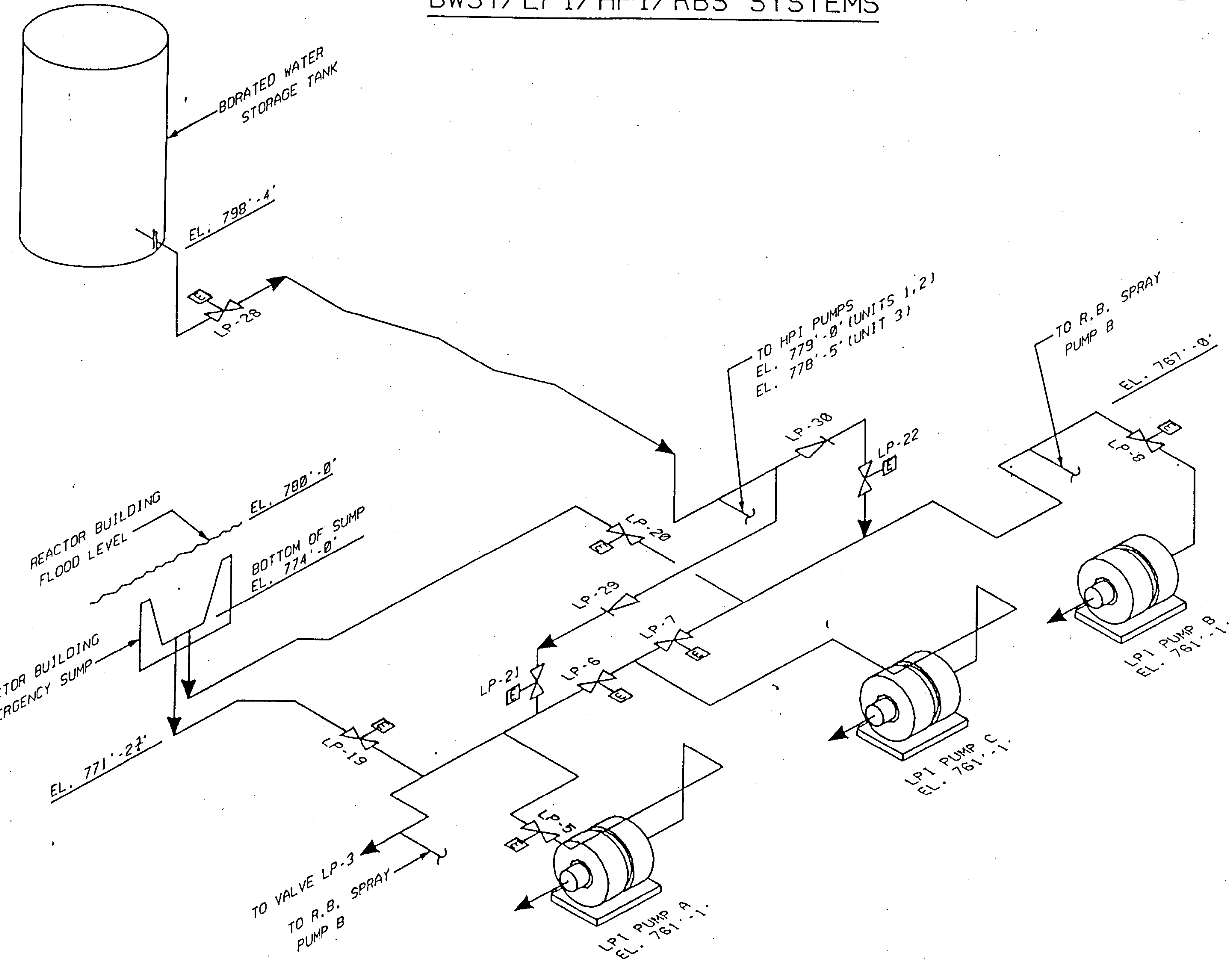
Safety Significance

- Team will decide to either:
  - » Initiate transfer at an indicated level > 2 feet in the BWST or
  - » Stop running HPI, LPI, RBS pumps before an indicated level of 2 feet in the BWST and
    - – Initiate transfer of suction source
    - – Restart pumps
  - » Complexity of either action is very low

BWST Depletion vs. Time



BWST/LPI/HPI/RBS SYSTEMS



Safety Significance

Event	RB Pressure at time of swapover	RBS actuated?	HPI piggyback operation?	Time available prior to swapover initiation	Key Actions	Comments
Hot Leg LBLOCA	< 10 psig	Yes	No	23 min	Open sump suction valves (LP-19 and LP-20) before BWST level < 2 feet	Finite volume of air entrained through LPI and RBS pumps
Cold Leg LBLOCA	> 10 psig	Yes	No	23 min	Open sump suction valves (LP-19 and LP-20) before BWST level < 2 feet	Finite volume of air entrained through LPI and RBS pumps (no air entrainment if LP-19 and LP-20 opened at > 4 feet BWST level)
SBLOCA > 0.025 ft^2	> 10 psig	Yes	No	$\geq$ 33 min	Open sump suction valves (LP-19 and LP-20) before BWST level < 2 feet	Finite volume of air entrained through LPI and RBS pumps (no air entrainment if LP-19 and LP-20 opened at > 4 feet BWST level)
SBLOCA between 0.005 and 0.025 ft^2	**> 10 psig**	**Yes**	**Yes**	**1.5 hours**	**Open sump suction valves (LP-19 and LP-20) before BWST level < 3.5 feet**	**No air entrainment in HPI pumps**
SBLOCA < 0.005 ft^2	< 10 psig	No	Maybe	4.5 hours	Open sump suction valves (LP-19 and LP-20) and close BWST isolation valves (LP-21 and LP-22) before BWST level < 3.5 feet	Depressurization and transition to LPI likely Ample time to complete actions if piggyback is required

Safety Significance

- Defense in depth
  - » Indications available to the operators on loss of inventory outside the Reactor Building
  - » Earlier step in EOP to verify Reactor Building water level
  - » STA will be monitoring inventory
  - » Operating team decision vs. individual
  - » TSC will be activated for SBLOCA prior to swapover
  - » "C" LPI pump available
  - » If loss of HPI in SBLOCA, EOP guidance to depressurize to LPI

Safety Significance

- Reactor Building Pressure
  - » Containment pressure well below design limit at time of swapover
  - » Reactor Building Cooling Units provide heat removal capability post-swapover to maintain pressure less than design limit
  - » Opening sump suction valves (LP-19, LP-20) assures continued operability of Reactor Building Spray

Safety Significance Summary

» BWST inventory requirements satisfied at all times

» Core cooling assured for large break LOCAs

  – Duration of air entrainment limited

  – LPI and RBS pumps would have performed intended functions

» Low likelihood of HPI System failure during certain small break LOCAs

  – Small breaks > 0.025 ft^2 and < 0.005 ft^2: confident will succeed

  – Small breaks between 0.005 ft^2 and 0.025 ft^2:

    - Success of HPI possible but not assured
    - EOP will likely take plant to condition allowing LPI to provide cooling

» Reactor Building integrity not compromised

Risk Significance

- Duke performed a detailed precursor analysis
  - » Frequency of LOCA break sizes of concern estimated
  - » Recovery actions using LPI Pump C modeled
  - » Human error due to BWST and sump level issues modeled

Risk Significance

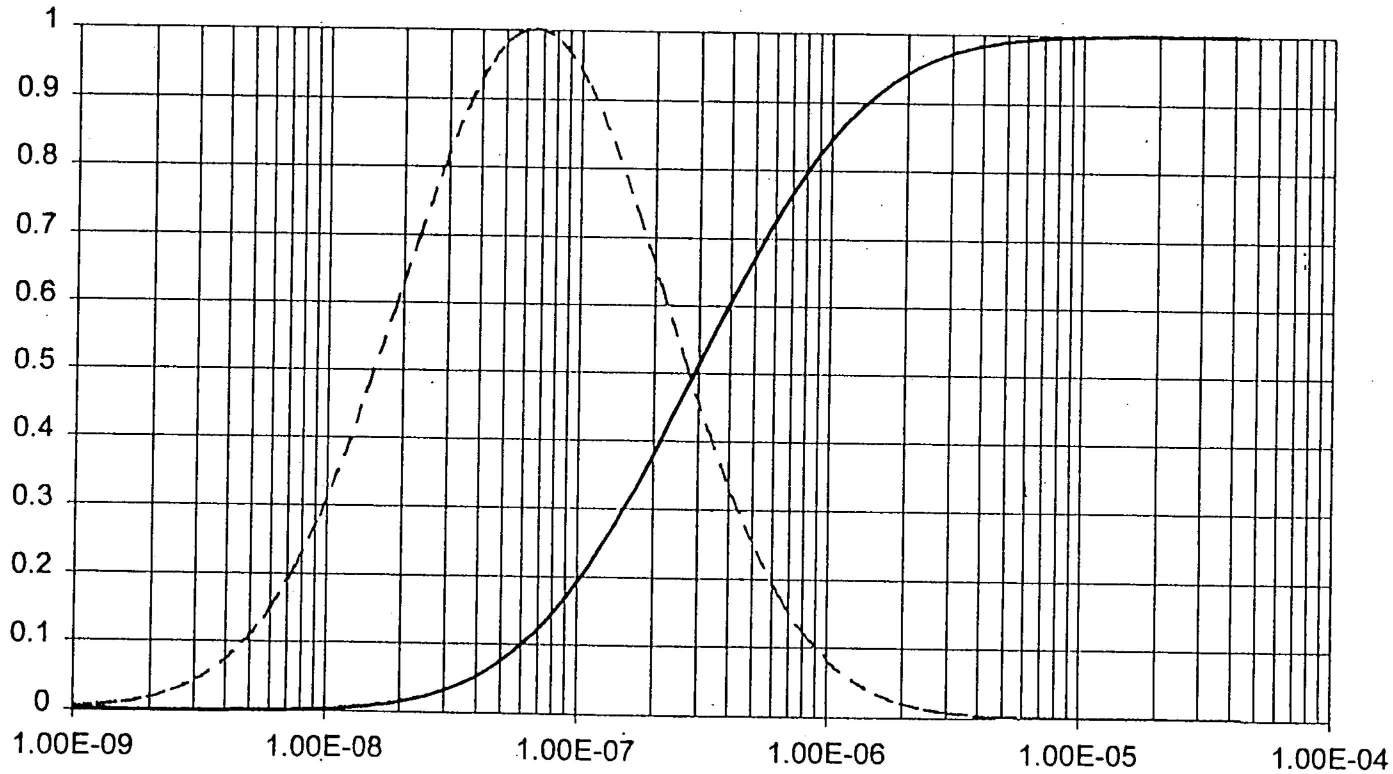
- Event has a conditional core damage frequency on the order of 1E-6
  - » Best estimate slightly below the precursor threshold of 1E-6
- Potential use of the manually operated LPI pump C is a factor in keeping the probability low
- Without credit for LPI pump C, the estimated CDF increases to approximately 1E-05

Risk Significance

- Consideration of uncertainty is important in understanding the potential risk significance
  - » Point estimate 7.0E-07
  - » 5^{th} percentile 4.3E-08
  - » Median 2.9E-07
  - » Mean 7.2E-07
  - » 95^{th} percentile 2.4E-06

Risk Significance

Log-Normal Distribution of CDF Results



Management Perspective

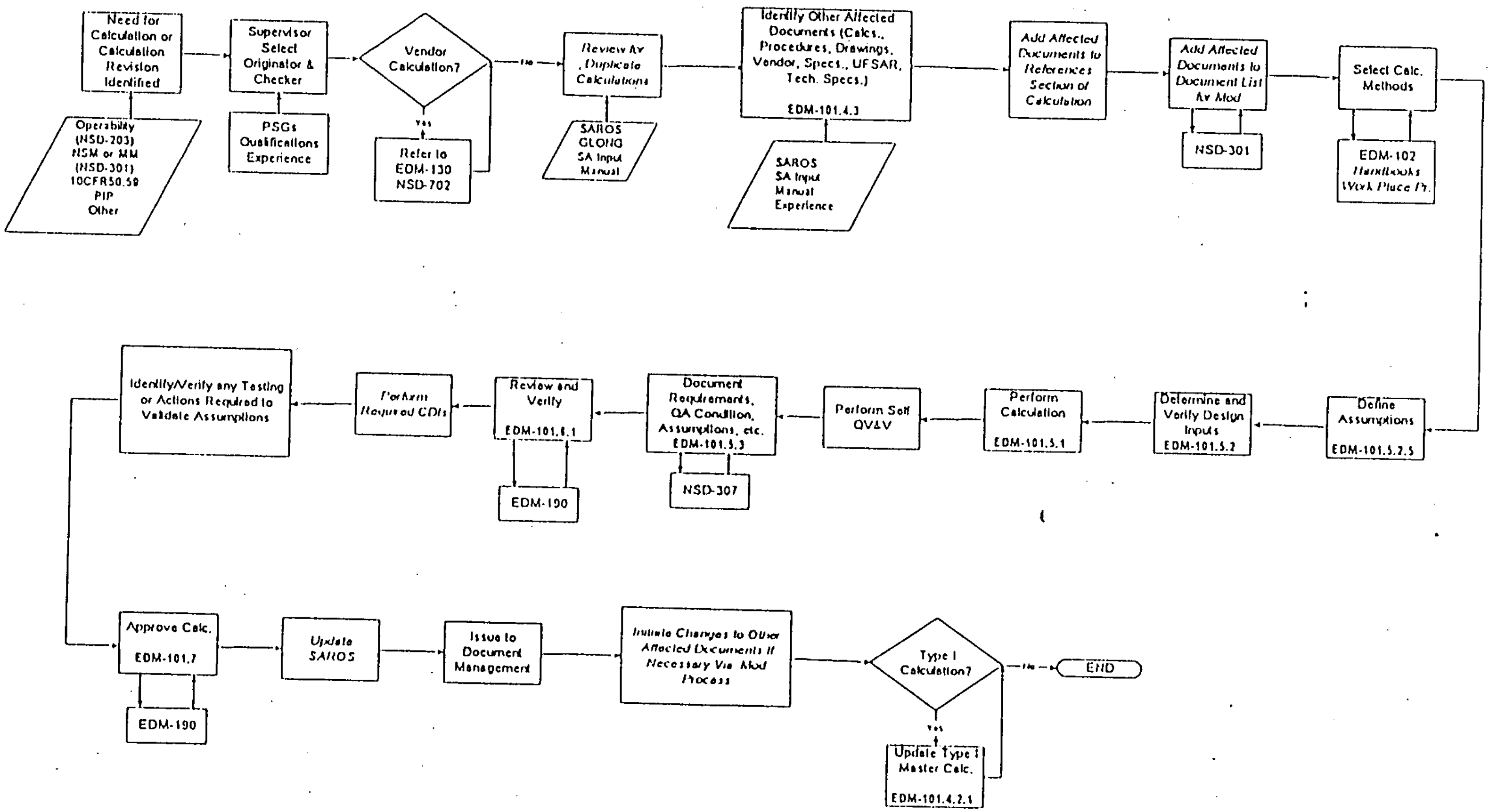
- Both apparent violations involve old design issues
- Both apparent violations self-identified through a voluntary initiative
- Immediate corrective actions taken, and long-term comprehensive corrective actions underway, should preclude recurrence
- Based on length of time situation existed and extent of effort needed to fully understand the condition, problem was not likely to be identified by routine efforts
- Safety significance is well understood
  - » The ECCS system might not have performed its intended function under certain conditions
  - » Impact to core damage frequency is reduced by ONS design features
- Enforcement discretion appears warranted based on Section VII.B.3 of the enforcement policy

Management Perspective

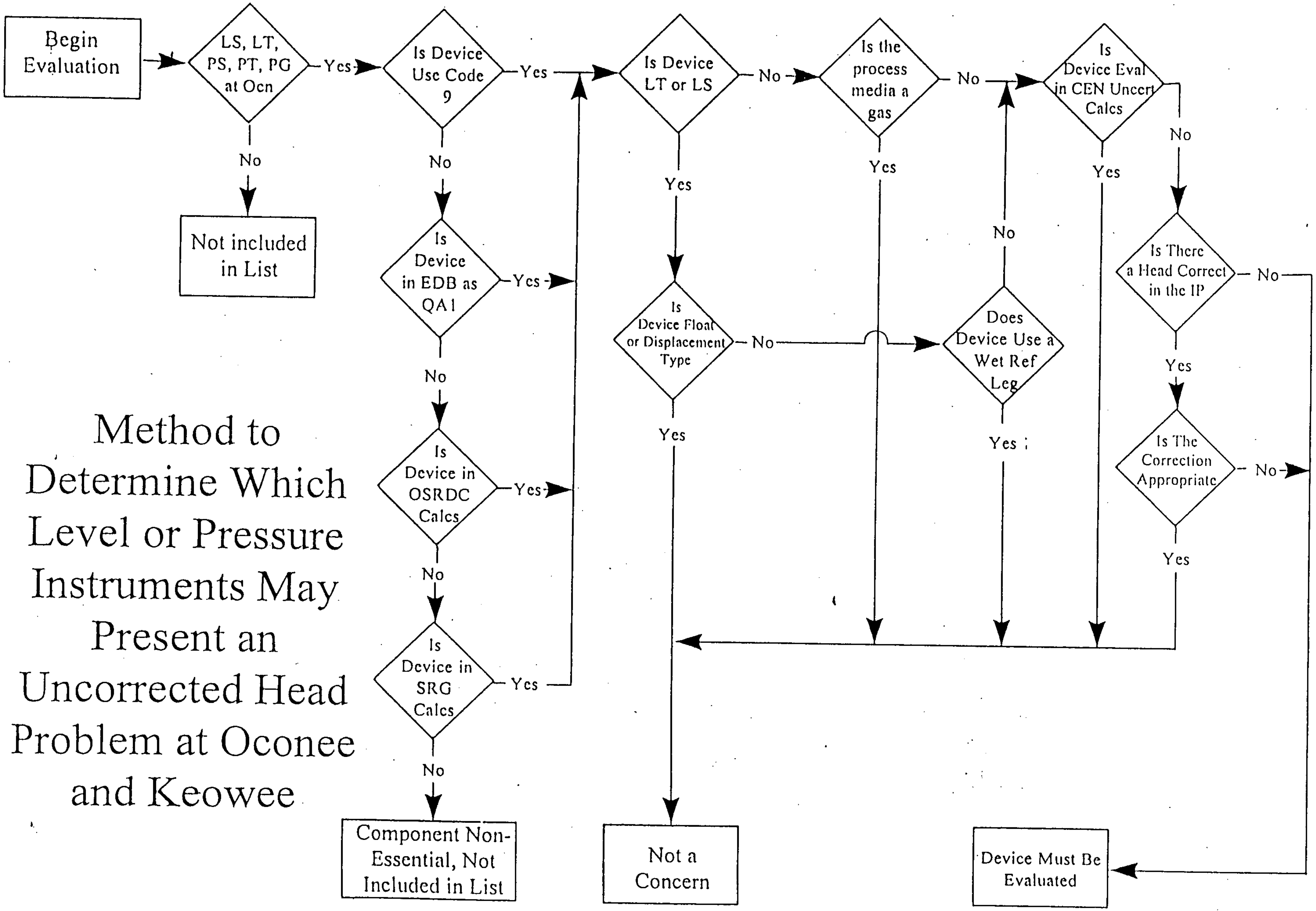
- ONS SITAs clearly demonstrate the value of continued assessments of the design basis
- Significant resources continue to be applied to Oconee design basis and UFSAR initiatives
- Continue to review systems in risk-informed manner per Recovery Plan
- Committed to comprehensive self assessments
- Low threshold for the identification of issues
- Applying significant resources to thoroughly resolve issues

Closing Remarks

Planned Process Changes



Method to Determine Which Level or Pressure Instruments May Present an Uncorrected Head Problem at Oconee and Keowee



Oconee Nuclear Station

BWST/RB Sump Level Instrument Error

Supplemental Report On Core Damage Significance

May 1998

Purpose

The purpose of this report is to present a more complete evaluation of the conditional core damage probability associated with the BWST/RB Sump level instrument calibration error, and also to provide a perspective on the impact of the key parameters on the results and conclusions.

Overview

For significant operational events, Duke routinely performs an assessment of the core damage significance of the event as part of the safety significance consideration. These calculations are done using the PRA Group guidelines on precursor calculations. The level of detail and level conservatism in the calculation are dependent on the safety significance of the event. A simplistic, conservative approach can be used when the assessed conditional core damage probability is seen not to be significant. When this not the case, a more detailed analysis is performed to obtain a realistic or best estimate value.

For the Oconee BWST/RB Sump Level event, the simplest and most conservative estimate of the conditional core damage probability is obtained by assuming that the ECCS fails for all LOCA events. In this case the conditional core damage probability would be simply the annual frequency of the LOCA event (2E-3).

A less conservative, yet still simplistic, treatment of the event would be to give reasonable credit for the operator recovery of the ECCS pumps upon indications abnormal performance. Using an operator recovery failure probability of 0.01, the conditional core damage probability is seen to be 2E-5 (2E-3 x 1.0 x 0.01) with this approach. (This value seems to be in the same range of other conservative estimates reported for a similar event in the industry).

For the Oconee event, a more detailed analysis is considered appropriate to take into account the use of the LPI-C pump and to obtain a more realistic conditional core damage probability.

Analysis

Some of the specific factors considered in the evaluation include:

- plant response to LOCAs of various sizes and the influence of RB Sprays,
- contribution from transient initiators,
- magnitude of errors in the BWST and RB sump level instruments,
- a realistic assessment of the BWST level at which the LPI pumps might be expected to fail,
- operator action to prevent damage to the operating pumps,
- operator action to establish sump recirculation with the 'C' LPI pump, which is normally not needed for LOCA mitigation.

Initiating Frequencies

LOCAs of various sizes are possible and may occur with different frequencies. This analysis considers three size ranges for LOCAs. The small LOCA range includes breaks that are small enough that actuation of reactor building spray is not expected. The RB Spray activation shortens the time available prior to the need for recirculation from the sump, the critical time available for operator recovery action. The small LOCA (3/8 inch to 1.5 inches diameter) frequency from the Oconee PRA revision 2 is 1.44E-03/yr, however, breaks in the upper end of this range (> 1.0 inch) are expected to cause RB spray actuation. For this analysis the frequency of LOCAs small enough that RB spray initiation does not occur is assumed to be ~2/3 of the PRA frequency, 1E-03/yr.

The medium LOCA range for this analysis (1.0 inch to 4.0 inches) includes breaks large enough to cause RB spray actuation but small enough that HPI is needed in the injection phase. For this analysis this includes the remainder of the PRA small LOCA frequency plus the contribution from the 1.5 to 4.0 inch diameter breaks (the medium LOCA range from the Oconee PRA). LOCAs of equivalent diameters greater than 1.5 inches have never occurred. The frequency of such large breaks is estimated from the following.

Using the PWR operating data for the years 1980 through 1996 (783 PWR reactor-years) the not-small LOCA frequency is estimated, from a chi-square distribution with 1 degree of freedom, as .455/(2*783) or 2.9E-04/RY. On a calendar year basis and assuming a conservative capacity factor of 0.9, the frequency is 2.6E-04/yr. This is assumed to be equally split between medium and large breaks.

The medium LOCA frequency is estimated as

$$0.5 * 2.6E\text{-}04 + 4.4E\text{-}04 = 5.7E\text{-}04/yr.$$

The large LOCA frequency is estimated as

$$0.5 * 2.6E\text{-}04 = 1.3E\text{-}04/yr.$$

Some plant transients may, as a result of various equipment failures, require that decay heat be removed by a primary system feed and bleed. The initiating frequency for this category is identified from the Oconee PRA revision 2. The non-LOCA cutsets of interest are those where recirculation is required. Since no transients result in a sufficiently low RCS pressure such that an initial transfer to low pressure recirculation is assumed, the cutsets of interest are those that require operators to establish high pressure recirculation. A review of the Oconee PRA cutsets reveals that the frequency of plant transients requiring a transfer to high pressure recirculation is approximately 8.3E-05/yr. In these sequences the RBCUs are not available, therefore, containment spray actuation would be expected to occur. The steam generators are not available.

RB Sump Level Indication

The actual sump level at the time the BWST level is approaching 2 feet should be approximately 4.5 feet, however, the combination of instrument loop bias and random errors mat result in an indication less than this level. In addition to random error, a bias towards a lower indication is present due to current leakage in the instrument loop. This bias results in a probability of approximately 0.5 of the indication reading less than the desired 4 feet on the worst case transmitter.

LPI Pump NPSH Consideration

For cases where air entrainment could occur and the ECCS pumps begin to indicate unstable operation as the BWST level decreases towards empty, the possibility exists that the air core of the vortex could reach the pump. This would momentarily degrade pump performance, reducing flow, and collapse the vortex. The LPI pump vendor has expressed an engineering opinion that, over a short period of time such as that required to complete the swapover, no pump damage would occur. The momentary erratic performance of the pump would be apparent to the operators and it is expected that they would stop the effected pumps prior to damage. An engineering evaluation has concluded that the pumps would be capable of restarting once the suction source was realigned to the sump.

Operator Actions

The normal operator actions as directed by the EOP are as follows.

For large break LOCAs the RCS pressure rapidly attains the LPI conditions. When the BWST level approaches 6 feet, the operator begins the switchover to the RB sump. Switchover is actually accomplished at the EOP step when BWST level is ≥ 6 feet and RB sump level is ≥ 4 feet. Subsequently, the LPI suction from the BWST is isolated when the BWST level reaches 2 feet. The effect of the BWST instrumentation level calibration error in this event was that the actual level in the BWST could be as much as 18 inches lower than in the calibration which derived the EOP setpoints to manually perform the swapover from the BWST to the RB sump. Thus, the actual BWST level could be as low as 4.5 feet when the indicated level approaches the EOP setpoint of 6 feet. When the 2 feet indication is reached, the actual level could be a slow as 6 inches. Since the LPI pumps are considered capable of operating under these conditions, the time available for successful operator action is unchanged.

For smaller breaks when the RCS pressure stays above the LPI pump shutoff head pressure, the EOP directs the operators to cool and depressurize the RCS to the LPI operating conditions. If the RCS pressure is still at the HPI operating condition and not the LPI operating condition, the HPI pumps suction is swapped from the BWST to the LPI discharge header when the BWST level reaches 10 feet.

For this event, two specific human actions are considered in the analysis. The most significant of these is the operators recognizing that the Reactor Building sump level is not responding in a manner consistent with their expectations given the low BWST level. The operators would have to realize that strict adherence to the EOP is not going to be effective and action to swap the suction source to the sump or terminate pump operation prior to damage is needed. The reliability of this action considers a number of factors. First, the sump level is indicating significant inventory in the sump. The level should indicate 3 feet or more even considering the instrument error. Thus, the operators can see that the water is collecting in the RB and is not being lost to some other location. Second, the inventory in the BWST is being depleted and the operators understand that the BWST is not viable as a long term suction source. Continuing to draw suction from the tank as the level drops to low values, with the symptoms of abnormal pump operation, simply is not an option. Another source is required and the sump is the only alternative. Finally, actions taken by the operating crew to maintain LPI flow in the situation of interest should not be considered as a violation of procedure. The operators are expected to follow the procedure to the point where it becomes obvious that the procedure is not working. At this time they are expected to apply their knowledge of plant systems and equipment to maintain an important plant function. Maintaining injection to the RCS is clearly the intent of the procedure and the crew is aware that this is the case. It is expected that in this situation they would recognize that following the letter of the procedure clearly can not accomplish the intent of the procedure. The human error modeled here is very similar to the Oconee PRA event "Operators Fail To Initiate Low Pressure Recirculation". However, the typical cue for the action, coincident low level in the BWST and RB sump level above setpoint, is not met. For the small and medium LOCA cases where the accident is progressing more slowly a value of 0.01 is assumed. This value is 10 times the probability for the event in the PRA. For large

LOCAs an additional order of magnitude increase in the failure probability is assumed, resulting in a failure probability of 0.1.

Another significant operator action is to use the 'C' LPI pump to provide sump recirculation. This recovery is needed when the operating crew has failed to prevent damage to the 'A' and 'B' pumps by accomplishing the action previously described. It is reasonable to expect that they would recognize that the operating pumps failed do to a loss of NPSH and that the BWST is no longer a viable suction source. Therefore, recovery using the LPI-C implies that the pump is aligned to the RB sump.

The time available for this recovery has been reevaluated using the MAAP code. A range of break sizes is investigated: 1 inch in diameter, 4 inches in diameter, and 2 square feet in area. The operating LPI pumps are assumed to fail when the BWST level reaches 2 feet. Any significant core heat-up is delayed until the boiled up level in the core region falls below the top of the core. Initially, the inventory in the vessel above the core must be boiled away (this process was estimated in the LER to take approximately 7 minutes). A two phase mixture continues to cover the core for several more minutes and then the steam flow cools the upper nodes of the core. Significant core damage does not occur until the level drops near the mid-plane of the core and steam cooling of the upper nodes is no longer effective. The worst case break (the large break) results indicate that there is approximately 45 minutes before a significant core temperature rise occurs. There is an additional 20 minutes or more before the core temperature rises to the point where significant core damage begins. The plots of hottest core node temperature versus time are included as Figures 1 through 3. The analysis indicates that the time available (tens of minutes) is much greater than the time required (approximately 5 minutes). This action is not time critical and its probability of success is dominated by the procedural considerations, the opportunity for self-checking, and independent evaluation by an STA. This recovery also includes the consideration that the RCS may need to be depressurized below the LPI pump shutoff head for some sequences. Therefore, the failure to recover probability is not the same in all sequences and is higher when the additional action for depressurization or operation of an HPI pump in piggy-back mode is expected to be required.

For large LOCA sequences, no depressurization is required and the recovery for this case represents the base value (0.05). This value includes consideration for the human action as well as hardware failures and maintenance unavailabilities. The recovery values for the small and medium LOCAs and the transients are assumed to be higher (0.1) as a result of the potential need to depressurize. Depressurization is only needed if no HPI pumps are available. If all HPI pumps are initially running, one pump is expected to be shut down assuring its availability at the time the LPI-C is started. A smaller failure to recover probability could be given to the medium LOCA recovery; for breaks at the larger end of this range the RCS pressure will fall below the LPI shutoff head and no operator initiated depressurization or HPI is needed. This represents a conservatism in the final result.

For the small breaks, the containment spray pumps are not expected to actuate and at least 5 hours are available before switchover to recirculation is needed. The long time period before sump recirculation is required creates the opportunity for a plant cool down to be completed prior to depletion of the BWST. The failure to cool down is estimated at 0.1.

Sequences

The CDF impact is evaluated by considering the combined results for the sequences of interest. An event tree for consideration of the various accident sequences is included as Figure 4. Core damage occurs when the down branch is followed at each decision point. Success at a branch means that the sequence does not contribute to an increase in CDF as a result of the BWST calibration error. The contribution from each initiator is calculated as the frequency of each initiator times the down branch probability at event tree branch.

The small LOCA sequence proceeds to core damage with the following probability,

$$1E\text{-}3 \times 0.1 \times 0.5 \times 0.01 \times 0.1 = 5.0E\text{-}08/yr.$$

The medium LOCA sequence proceeds to core damage with the following probability,

$$5.7E\text{-}4 \times 1.0 \times 0.5 \times 0.01 \times 0.1 = 2.9E\text{-}07/yr.$$

The large LOCA sequence proceeds to core damage with the following probability,

$$1.3E\text{-}4 \times 1.0 \times 0.5 \times 0.1 \times 0.05 = 3.3E\text{-}07/yr.$$

The transient sequence proceeds to core damage with the following probability,

$$8.3E\text{-}5 \times 1.0 \times 0.5 \times 0.01 \times 0.1 = 4.2E\text{-}08/yr.$$

These sequences have been input to a cutset file as shown in Table 1. The cutset file is used for generation of importance measures and consideration of the uncertainty as described in the following sections.

Results

A point estimate of the increase in CDF associated with the instrument calibration error is made. Probabilities for the various failures, both hardware and human, are selected so that a best estimate calculation is conducted. This analysis results in an estimated increase in CDF of 7.0E-07 for the condition of interest. This is slightly less than the precursor threshold.

Identification of some of the key parameters in the results and the sensitivity to these parameters is presented below.

Key Parameters and Sensitivity Studies

Table 2 is a listing of the events in the cutsets sorted by Risk Achievement Worth (RAW). The Fussel Vesley (FV) and Risk Reduction Worth (RRW) importance measures are also included.

The results are clearly sensitive to the initiating frequencies and large increases or decreases in these frequencies would have a major impact on the results. In particular, the large LOCA initiating frequency can have the largest impact.

The most important human action is LPISTOPRHE. This is a failure to recognize that the procedure is not going to work as intended (non large LOCA sequences) and take action to maintain operability of the A and B LPI pumps. If this action was assumed to always fail, the conditional CDF would rise to 3.8E-05. Conversely, if it was assumed to always succeed the CDF falls to 3.3E-07.

The second most important human action is the failure to recover using the 'C' LPI pump. This is actually four events in the cutset model, however, they are considered as a single action here in order to assess the significance of having the additional pump. If this action was assumed to always fail the conditional CDF would rise to 1E-05.

Uncertainty

The impact of uncertainty is also considered in the analysis. Parameters in the analysis are assumed to have a mean value equal to the point estimate used in the original analysis. Each parameter is assumed to be from a log-normal distribution and is assigned an error factor. The data input into the uncertainty analysis is included in Table 3. A Monte Carlo simulation is used to establish the parameters of the resulting distribution on the increase in CDF. The summary statistics are:

Mean	7.2E-07
5%	4.3E-08
Median	2.9E-07
95%	2.4E-06
Point Estimate	7.0E-07

It is seen that the mean and median values of the distribution, as well as the point estimate, are slightly less than 1E-06. The exact distribution on the CDF is not log-normal, however, for purposes of a graphical representation, a log-normal has been assumed. The median of the log-normal is assumed to be the median of the distribution and an error factor is computed from the 5^{th} and 95^{th} percentile values. The cumulative distribution function and a normalized (maximum value set to 1) probability density function are included as Figure 5.

Conclusions

The point estimate of the conditional core damage frequency of this event is calculated to be 7E-07, slightly below the precursor threshold of 1E-06. Operator recognition that, should the RB sump level indication be in the unfavorable side, the EOP step is not effective to safely mitigate the accident and that recovery action is needed to secure the operating LPI pumps on symptoms of cavitation is an important factor. If correct operator action for LPIP-A/B is conservatively assumed not to occur, the conditional core damage probability would be approximately 3.8E-5. Similarly, the action to use the standby LPI pump (LPI pump C) if timely action to accomplish the previous operator action did not occur is also an important factor affecting the numerical result. If this LPIP-C operator action is conservatively assumed not to occur (or if LPIP-C is assumed not available), the conditional core damage probability would be approximately 1E-05.

Considering realistic ranges in the values of the individual basic events associated with the core damage sequences, and performing a Monte Carlo simulation of the problem, it is seen that the mean value of the conditional core damage probability is 7.2E-07 with a 90% range of 4.3E-08 to 2.4E-06.

#	Inputs	Description	Rate	Exposure	Event Probability	Probability
1	MLOCA	Medium LOCA		5.70E-04	5.70E-04	2.85E-07
	SUMPLVLDEX	Probability That RB Sump Level Indicates Less Than 4 Feet (setpoint for swap)		5.00E-01	5.00E-01	
	LPISTOPRHE	Operators Fail To Recognize LPI Pump Cavitation and Realign to RB Sump		1.00E-02	1.00E-02	
	LLPC0MLRHE	Operators Fail To Recover Using LPI-C for a Medium LOCA		1.00E-01	1.00E-01	
2	LLOCA	Large LOCA		1.30E-04	1.30E-04	3.25E-07
	SUMPLVLDEX	Probability That RB Sump Level Indicates Less Than 4 Feet (setpoint for swap)		5.00E-01	5.00E-01	
	LPISTP2RHE	Operators Fail To Recognize LPI Pump Cavitation and Realign to RB Sump (LL)		1.00E-01	1.00E-01	
	LLPC0LLRHE	Operators Fail To Recover Using LPI-C for a Large LOCA		5.00E-02	5.00E-02	
3	SLOCA	Small Loss of Coolant With No RB Spray Actuation		1.00E-03	1.00E-03	5.00E-08
	SLCOOLDDHE	Operators Fail To Cool To LPI Entry Conditions Prior To BWST Depletion		1.00E-01	1.00E-01	
	SUMPLVLDEX	Probability That RB Sump Level Indicates Less Than 4 Feet (setpoint for swap)		5.00E-01	5.00E-01	
	LLPC0SLRHE	Operators Fail To Recover Using LPI-C for a Small LOCA		1.00E-01	1.00E-01	
	LPISTOPRHE	Operators Fail To Recognize LPI Pump Cavitation and Realign to RB Sump		1.00E-02	1.00E-02	
4	TRANHPR	Transient Requiring High Pressure Recirculation		8.30E-05	8.30E-05	4.15E-08
	SUMPLVLDEX	Probability That RB Sump Level Indicates Less Than 4 Feet (setpoint for swap)		5.00E-01	5.00E-01	
	LPISTOPRHE	Operators Fail To Recognize LPI Pump Cavitation and Realign to RB Sump		1.00E-02	1.00E-02	
	LLPCTRNRHE	Operators Fail To Recover Using LPI-C for a Transient Initiator		1.00E-01	1.00E-01	

Total Probability = 7.0E-07

Table 1 Cutsets

Event Name	Probability	RAW	FV	RRW	Description
LLOCA	1.30E-04	3.56E+03	4.63E-01	1.863	Large LOCA
TRANHPR	8.30E-05	713.70	5.92E-02	1.063	Transient Requiring High Pressure Recirculation
MLOCA	5.70E-04	713.35	4.06E-01	1.684	Medium LOCA
SLOCA	1.00E-03	72.20	7.13E-02	1.077	Small Loss of Coolant With No RB Spray Actuation
LPISTOPRHE	1.00E-02	54.13	5.37E-01	2.158	Operators Fail To Recognize LPI Pump Cavitation and Realign to RB Sump
LLPC0LLRHE	5.00E-02	9.80	4.63E-01	1.863	Operators Fail To Recover Using LPI-C for a Large LOCA
LPISTP2RHE	1.00E-01	5.17	4.63E-01	1.863	Operators Fail To Recognize LPI Pump Cavitation and Realign to RB Sump (LL)
LLPC0MLRHE	1.00E-01	4.66	4.06E-01	1.684	Operators Fail To Recover Using LPI-C for a Medium LOCA
SUMPLVLDEX	5.00E-01	2.00	1.00E+00	0.000	Probability That RB Sump Level Indicates Less Than 4 Feet (setpoint for swap)
LLPC0SLRHE	1.00E-01	1.64	7.13E-02	1.077	Operators Fail To Recover Using LPI-C for a Small LOCA
SLCOOLDDHE	1.00E-01	1.64	7.13E-02	1.077	Operators Fail To Cool To LPI Entry Conditions Prior To BWST Depletion
LLPCTRNRHE	1.00E-01	1.53	5.92E-02	1.063	Operators Fail To Recover Using LPI-C for a Transient Initiator

Table 2 Importance Listing Sorted By Risk Achievement Worth (RAW)

Event Name	Probability	EF	Distribution	Description
LLOCA	1.3E-04	10	L	Large LOCA
LLPC0LLRHE	0.05	5	L	Operators Fail To Recover Using LPI-C for a Large LOCA
LLPC0MLRHE	0.1	2	L	Operators Fail To Recover Using LPI-C for a Medium LOCA
LLPC0SLRHE	0.1	2	L	Operators Fail To Recover Using LPI-C for a Small LOCA
LLPCTRNRHE	0.1	2	L	Operators Fail To Recover Using LPI-C for a Transient Initiator
LPISTOPRHE	0.01	5	L	Operators Fail To Recognize LPI Pump Cavitation and Realign to RB Sump
LPISTP2RHE	0.1	5	L	Operators Fail To Recognize LPI Pump Cavitation and Realign to RB Sump (LL)
MLOCA	5.7E-04	5	L	Medium LOCA
SLCOOLDDHE	0.1	2	L	Operators Fail To Cool To LPI Entry Conditions Prior To BWST Depletion
SLOCA	1E-03	5	L	Small Loss of Coolant With No RB Spray Actuation
SUMPLVLDEX	0.5	1.5	L	Probability That RB Sump Level Indicates Less Than 4 Feet (setpoint for swap)
TRANHPR	8.3E-05	5	L	Transient Requiring High Pressure Recirculation

Table 3 Input Parameters for Uncertainty Analysis

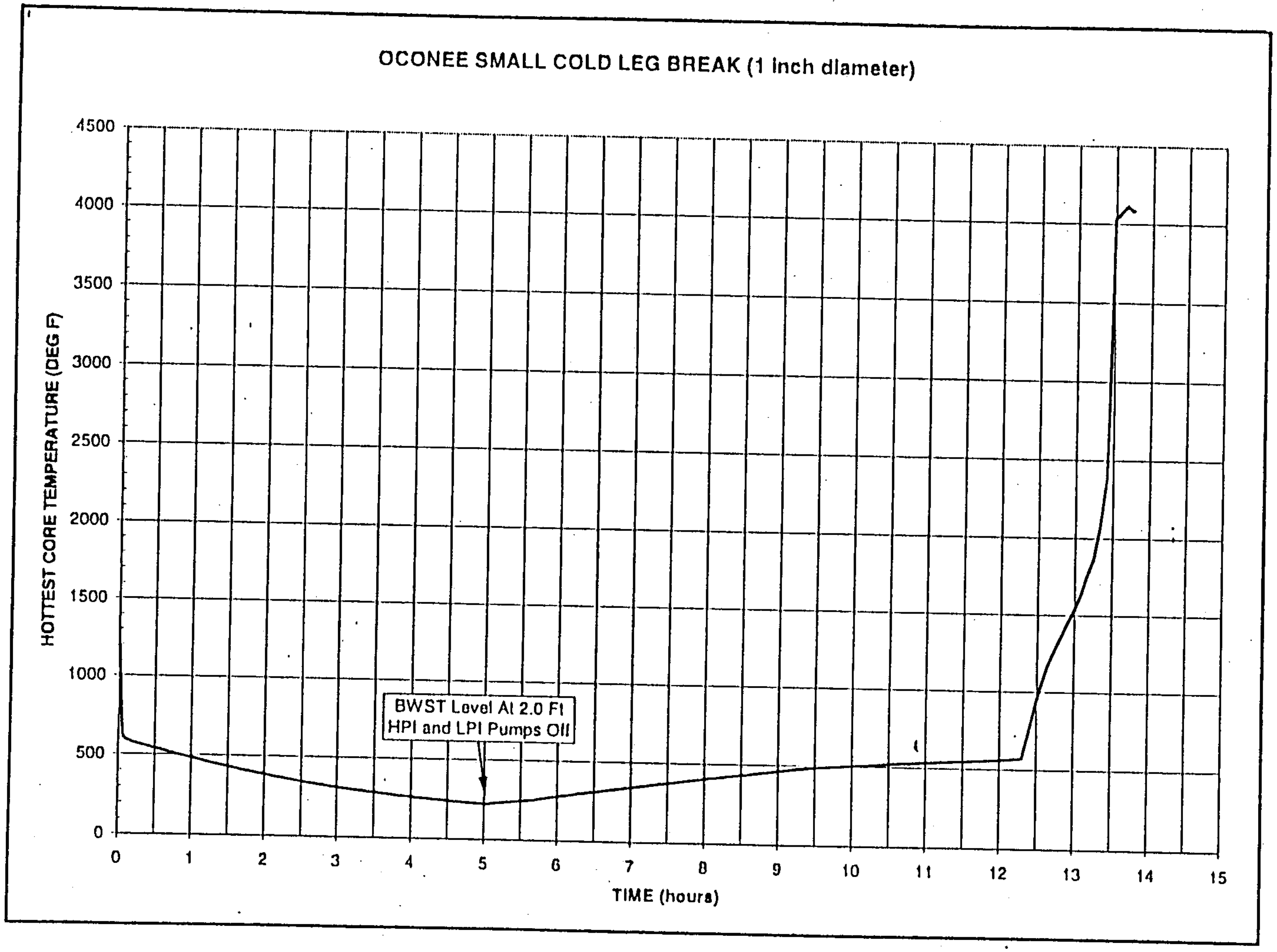


Figure 1 Hottest Core Node Temperature versus Time for the Small LOCA

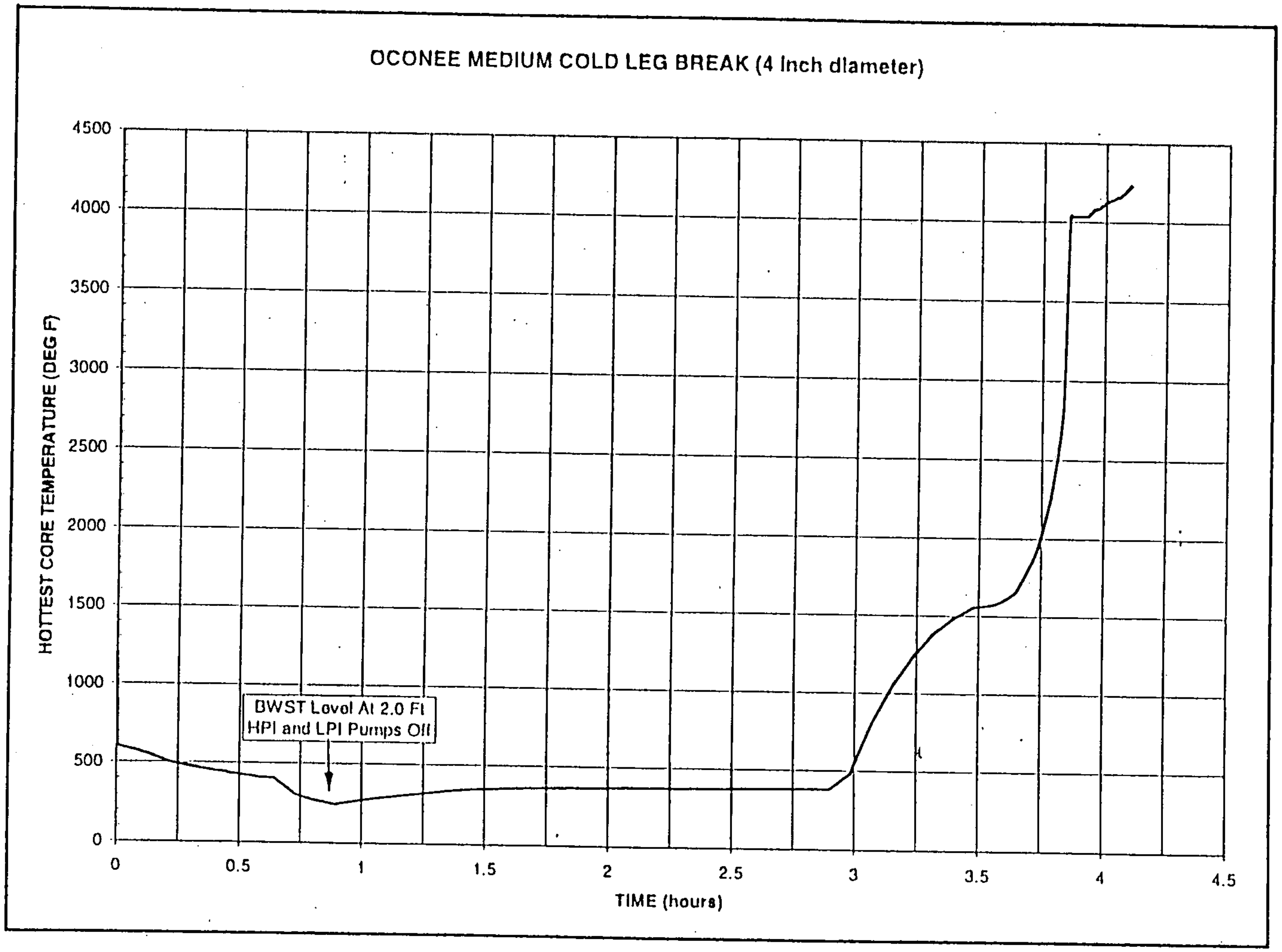


Figure 2 Hottest Core Node Temperature versus Time for the Medium LOCA

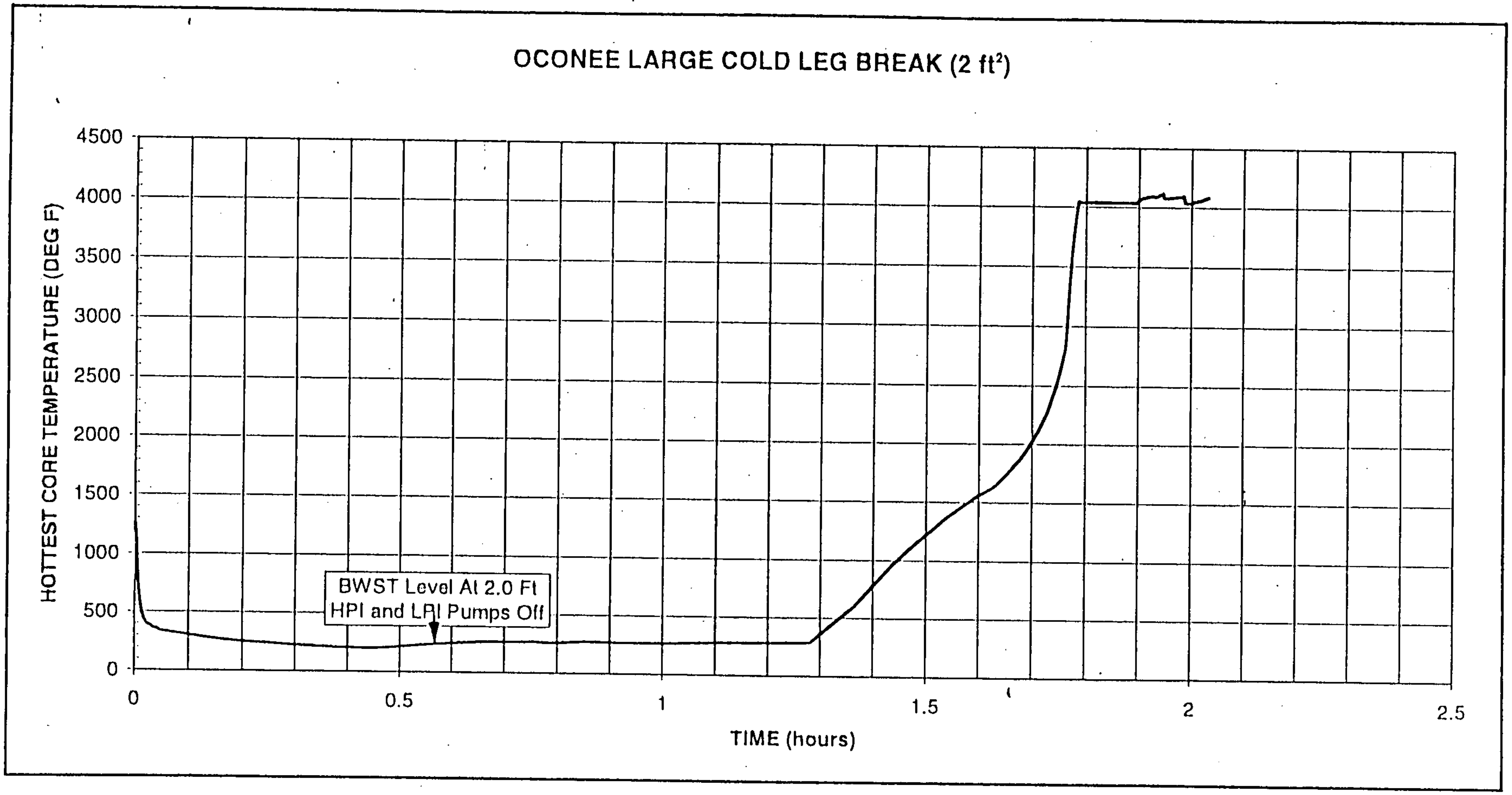


Figure 3 Hottest Core Node Temperature versus Time for the Large LOCA

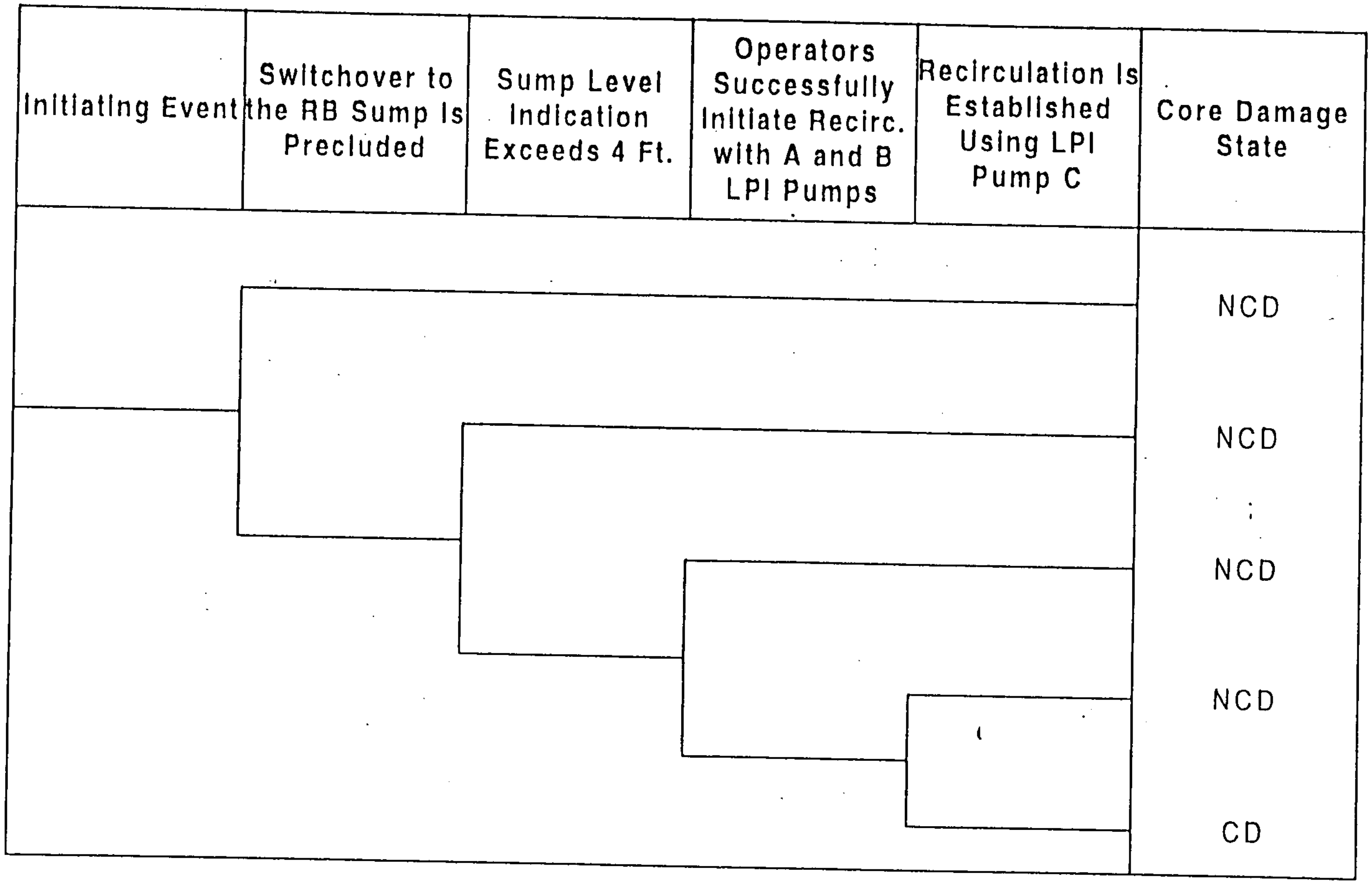


Figure 4 Core Damage Sequence Event Tree for the BWST Level Calibration Issue

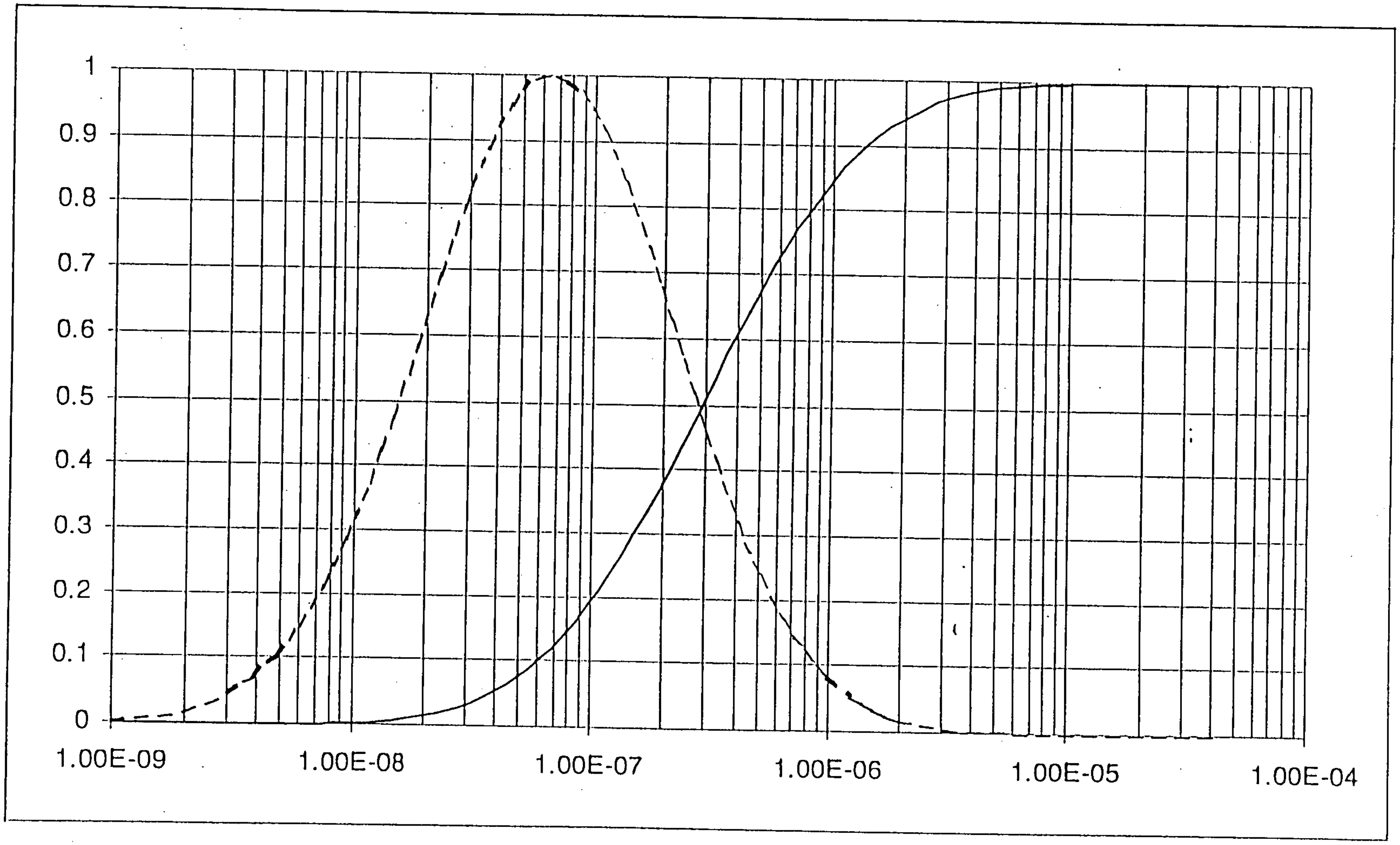


Figure 5 Log-normal Distribution of the CDF Results