July 15, 2005

Mr. Richard Anderson
Vice President
FirstEnergy Nuclear Operating Company
P.O. Box 97, A290
10 Center Road
Perry, Ohio 44081

SUBJECT: PERRY NUCLEAR POWER PLANT, UNIT 1 - FINAL ACCIDENT SEQUENCE PRECURSOR ANALYSIS OF SEPTEMBER 1, 2003, AND MAY 21, 2004, OPERATIONAL EVENTS

Dear Mr. Anderson:

Enclosed for your information is the final Accident Sequence Precursor (ASP) analysis of operational events which occurred at the Perry Nuclear Power Plant, Unit 1, on September 1, 2003, and May 21, 2004. The conditions were reported by Licensee Event Report Nos. 2003-004-01, dated January 29, 2004, and 2004-001-01, dated September 18, 2004, and documented in U.S. Nuclear Regulatory Commission (NRC) Inspection Report Nos. 50-440/2003-006, dated October 30, 2003, and 50-440/2004-008, dated August 5, 2004. This report is being issued as a final analysis since it is a non-controversial, lower risk precursor for which the ASP results are consistent with the results from the Significance Determination Process's final evaluation of the same condition. Elimination of the review and comment resolution for this event will reduce the burden for the NRC staff and the licensee.

Previously, the detailed ASP analyses were classified as "SENSITIVE - NOT FOR PUBLIC DISCLOSURE" based on the guidance provided by the Executive Director for Operations in the memorandum to the Commission dated April 4, 2002, concerning the release of information to the public that could provide significant assistance to support an act of terrorism. More recent guidance found in SECY-04-0191, "Withholding Sensitive Unclassified Information Concerning Nuclear Power Reactors from Public Disclosure," dated October 19, 2004, allows the uncontrolled release of ASP analyses that do not contain information related to uncorrected configurations or conditions that could be useful to an adversary. The NRC staff has reviewed the detailed ASP analysis according to SECY-04-0191 and has determined that it can be released to the public.

Please contact me at 301-415-3965 if you have any questions regarding the enclosure.

Sincerely,

/RA/

William A. Macon, Jr., Project Manager, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-440

Enclosure: Final Precursor Analysis

cc w/encl: See next page

July 15, 2005

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Perry Nuclear Power Plant, Unit 1

cc:

David W. Jenkins
FirstEnergy Corporation
76 South Main St.
Akron, OH 44308

Resident Inspector's Office
U.S. Nuclear Regulatory Commission
P.O. Box 331
Perry, OH 44081-0331

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
2443 Warrenville Road
Lisle, IL 60532-4531

Sue Hiatt
OCRE Interim Representative
8275 Munson
Mentor, OH 44060

Manager - Regulatory Compliance
FirstEnergy Nuclear Operating Company
Perry Nuclear Power Plant
P.O. Box 97, A210
10 Center Road
Perry, OH 44081

Director, Site Operations
FirstEnergy Nuclear Operating Company
Perry Nuclear Power Plant
P.O. Box 97, SB306
Perry, OH 44081

Mayor, Village of North Perry
North Perry Village Hall
4449 Lockwood Road
North Perry Village, OH 44081

Donna Owens, Director
Ohio Department of Commerce
Division of Industrial Compliance
Bureau of Operations & Maintenance
6606 Tussing Road
P.O. Box 4009
Reynoldsburg, OH 43068-9009

Carol O'Claire, Chief, Radiological Branch
Ohio Emergency Management Agency
2855 West Dublin Granville Road
Columbus, OH 43235-7150

Mayor, Village of Perry
P.O. Box 100
Perry, OH 44081-0100

Dennis Clum
Radiological Assistance Section Supervisor
Bureau of Radiation Protection
Ohio Department of Health
P.O. Box 118
Columbus, OH 43266-0118

Zack A. Clayton
DERR
Ohio Environmental Protection Agency
ATTN: Mr. Zack A. Clayton
P.O. Box 1049
Columbus, OH 43266-0149

Chairman
Perry Township Board of Trustees
3750 Center Road, Box 65
Perry, OH 44081

Daniel Z. Fisher
Transportation Department
Public Utilities Commission
180 East Broad Street
Columbus, OH 43215-3793

Joseph J. Hagan
Senior Vice President, Engineering and
Services
First Energy Nuclear Operating Company
76 South Main Street
Akron, OH 44308

Mr. Lew W. Myers
Chief Operating Officer
FirstEnergy Nuclear Operating Company
76 South Main Street
Akron, OH 44308

Perry Nuclear Power Plant, Unit 1

cc:

Director, Regulatory Affairs
FirstEnergy Nuclear Operating Company
395 Ghent Road
Akron, OH 44333

Final Precursor Analysis

Accident Sequence Precursor Program --- Office of Nuclear Regulatory Research

Perry Nuclear Power Plant	**ESW A Pump Failure To Run Due To Shaft Failure and Inadequate Repairs Led to a Second Failure**	
Event Dates: 9/01/2003 and 5/21/2004	**Licensee Event Report No. 440-2003-004-01 440-2004-001-01**	**ΔCDP = 1.2E-6**

April 13, 2005

Operating Condition Summary

Description. On September 1, 2003, the Perry Nuclear Power Plant (PNPP or the licensee) was in Mode 1 at 100 percent of rated thermal power. The Emergency Service Water (ESW) A pump at PNPP was started. The ESW A pump failed to run after 42 minutes, resulting in loss of flow to its loads as documented in Licensee Event Report (LER) 440-2003-004-01 (Reference 1). The control room staff observed all ESW A pump flow indications for Residual Heat Removal A, Emergency Core Cooling A, and Division 1 Emergency Diesel Generator (EDG) A at zero gallons per minute. The ESW A pump motor temperature began to rise. Operators then declared the ESW A pump to be inoperable. No motor protective trips occurred during the pump run failure event. Region III issued an inspection report on October 30, 2003 (Reference 2). The Office of Enforcement issued a final significance determination process (SDP) finding letter on January 28, 2004, on the same event (Reference 3).

On May 21, 2004, the ESW A pump failed again. This second failure event, as documented in LER 440-2004-001-01 (Reference 4), was caused by inadequate repair of the first pump failure. Region III issued an inspection report on July 2, 2004 (Reference 5).

Cause. The second ESW A pump failure was caused again by a failure of the pump shaft coupling sleeve. The shaft coupling sleeve failure was the result of improper coupling reassembly by plant maintenance personnel following the September 2003 failure. Inspection findings documented that intergranular stress corrosion cracking was the failure mechanism for shaft coupling.

The cause of the pump shaft sleeve failure was later identified to be due to licensee’s failure to follow vendor-specified reassembly instructions for the ESW pumps after maintenance events. Instead of following vendor-specified reassembly instructions, the licensee relied on knowledge-based skills of their facility maintenance personnel for reassembly of the ESW pumps after maintenance events. This contributed to the first failure of the pump. Additionally, the pump coupling was not designed for sufficient stress margin to failure and this contributed to the second failure of the pump. Not following vendor-specified instructions to fix a safety-related pump and not providing adequate stress safety margin for the pump coupling were found to be major factors to the licensee’s performance which resulted in two non-compliance findings in a one-year period.

Condition duration. On August 14, 2003, 1610 hours, the ESW A pump was started and ran successfully until August 23, 2003. On September 1, 2003, the ESW A pump was restarted. The pump ran for 42 minutes and then failed due to pump shaft coupling sleeve failure. The ESW A pump was declared inoperable on September 1, 2003, at 1717 hours. The pump was repaired and declared operable on September 5, 2003, at 1855 hours. The reactor was operating during the entire period between August 23 at 0607 hours and September 5, 2003, at 0655 hours. So, the operating condition involving the inoperable ESW A pump existed for 13.46 days (323 hours).

On May 21, 2004, 0148 hours, the ESW A pump was started for surveillance testing. The pump ran for about 2 minutes and then failed because the uppermost split ring coupling broke in half. Region III documented in their inspection finding on this second failure event that "the primary cause for this failure was related to the cross-cutting issue of problem identification and resolution in that the licensee neither understood nor corrected the design deficiencies associated with the coupling" when it failed the first time on September 1, 2003. The pump was declared inoperable on May 21, 2004, at 0152 hours. Following the failure, the licensee shut down the plant to replace the pump. The pump was repaired and declared to be operable on May 29, 2004, at 0513 hours. Region III inspectors found that the second failure event was due to the same cause as the first failure event (coupling installation error).

The ESW A pump ran for over 24 hours during maintenance on the control complex chillers from April 24, 2004, until 1732 hours on May 13, 2004. Between May 13 and the second failure event on May 21, 2004, the ESW A pump ran for an additional 0.46 days (11 hours) intermittently. The pump never ran for a 24-hour period continuously. So, it is questionable whether the ESW A pump would have been able to operate for a 24-hour mission time on demand during this period. From about 0435 hours on May 13, 2004, until shutdown cooling was initiated at 0307 hours on May 23, 2004, a period of 9.96 days (239 hours), it is assumed that the ESW A pump would not have been able to run for its 24-hour mission time on demand. Therefore, the second operating condition involving the inoperable ESW A pump existed for a net period of 9.5 days (9.96 days - 0.46 days) or 228 hours.

Since the same pump was inoperable due to run-failures on two separate occasions within a one-year period, the total inoperability period for the ESW A pump was found to be 22.96 days (551 hours).

Related event. None.

Recovery opportunity. Given a transient or loss of offsite power (LOOP) event, the ESW A pump shaft failure would have been considered to be non-recoverable in a timely manner.

Other related conditions or events during the condition period. A review of Region III-issued green SDP findings (finding with less than 1E-6 delta CDF values) for the same condition period was conducted in identifying potential overlapping operating conditions. It was found that none of the green SDP findings was applicable for evaluation of combined overlapping conditions for the same condition period.

Analysis Results

- **Importance**[1]

  The risk significance of the potentially inoperable ESW A pump due to shaft reassembly problem for a condition duration of 22.96 days (551 hours) was determined by subtracting the nominal core damage probability (point estimate) from the conditional core damage probability (point estimate):

Conditional core damage probability (CCDP) =	1.6E-6
Nominal core damage probability (CDP) =	4.5E-7
Importance (ΔCDP = CCDP - CDP) =	1.2E-6

  The estimated importance (CCDP-CDP) for the operating condition was 1.2E-6.

  A uncertainty analysis was conducted for the operating condition. The mean estimates for CCDP, CDP, and importance were 1.629E-6, 4.558E-7, and 1.173E-6, respectively.

- **Dominant sequence**

  Loss of condenser heat removal event followed by successful reactor scram, successful reclosure of safety relief valves, successful feedwater system, failure of suppression pool cooling, failure of the containment spray system, failure of PCS recovery, and failure of containment venting.

  Sequence LOCHS-07; importance was estimated to be 3.5E-7. The events and important component failures in this sequence were as follows:

  - Loss of condenser heat removal event
    - successful reactor scram,
    - successful reclosure of safety relief valves,
    - successful feedwater system,
    - failure of suppression pool cooling,
    - failure of the containment spray system,
    - failure of PCS recovery, and
    - failure of containment venting
  - Onset of potential core damage

  Success-failure path for dominant sequence LOCHS-07 is shown Figure 1.

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[1] Since this condition did not involve an actual initiating event, the parameter of interest is the measure of the incremental change between the conditional probability for the period in which the condition existed and the nominal probability for the same period but with the condition nonexistent and plant equipment available. This incremental change or "importance" is determined by subtracting the CDP from the CCDP. This measure is used to assess the risk significance of hardware unavailabilities especially for those operating conditions where the nominal CDP is high with respect to the incremental change of the conditional probability caused by the hardware unavailability.

- **Results tables**

  – Table 1 provides the conditional probabilities for 2 dominant sequences.
  – Table 2a provides the event tree sequence logic for the dominant sequences listed in Table 1.
  – Table 2b provides the definitions of fault trees used in event tree logic listed in Table 2a.
  – Table 3 provides the CCDP cut sets for 2 dominant sequences.
  – Table 4 provides the definitions and probabilities for added basic events and condition-affected basis events.

Modeling Assumptions

- **Assessment summary**

Assessment type - This event was modeled as an at-power condition assessment with the ESW A pump run failure for a 23-day period (551 hours).

Condition modeling and related assumptions -

1. Given a demand for the ESW A pump to run, it would have failed due to shaft failure (operating condition). The ESW A pump run failure would not have been recovered since the shaft failure could not have been recovered in a timely manner.

Model use - The Revision 3.11 Standardized Plant Analysis Risk (SPAR) model for PNPP (Reference 6) was used for this condition assessment.

Model update to Revision 3.11 SPAR model -

CFAILED (CONTAINMENT FAILURE CAUSES LOSS OF ALL INJECTION) - In the baseline plant model, this event was judged to have a probability of 0.5. No references to a physical analysis (e.g., ultimate pressure capacity analysis) and/or structure-thermal hydraulic calculations to support the 0.5 probability assignment were documented in the PNPP model documentation by Idaho National Engineering and Environmental Laboratory (INEEL) staff as part of SPAR model update project.

After a containment venting system (CVS) failure event occurs, no more coolant injection could be provided to vessel. Saturated pool water may not cool the core adequately. MARK III containment may not be in intact once containment (drywell) failure pressure reaches beyond the design pressure due to venting failure. A core damage event would be onset if CVS failure would occur. This CVS failure-based core damage finding is consistent with the plant models for other BWR-6/MARK III plants (e.g., Clinton, River Bend Station, Grand Gulf). So, basic event CFAILED was set to TRUE in the baseline plant model.

<u>Basic event probability changes</u> -

Table 4 provides the basic events that were modified to reflect the operating condition being analyzed. The bases for these changes are as follows:

Given a demand (Transient or a LOOP event), the ESW A pump might have failed to run after its successful start. Operators could have been forced to use the ESW B pump for EDG B cooling and other decay heat removal cooling through the ESW B pump and its cooling loop.

SSW-MDP-FR-1A - This was set to TRUE to reflect the operating condition (the ESW A pump failed to run due to its shaft failure).

<u>Uncertainty analysis and range for total importance due to operating condition</u> -

The parameter estimates and the uncertainties regarding the numerical estimates of the parameters used in the model (parameter uncertainty) are calculated. These data and uncertainty distributions are then propagated through the modified version of the Revision 3.11 SPAR model for PNPP (Reference 6) to produce statistical uncertainty estimates.

An uncertainty analysis of the operating condition, along with parameters, was performed using the SAPHIRE code (Version 7.22). Default distribution types for applicable initiating events (e.g. LOOP, transients) and basic events for components were documented in the Revision 3.11 SPAR model for PNPP. These uncertainty estimates and uncertainty estimates for condition-affected basic events were used in estimating mean condition-CDP values and mean condition-CCDP values. Other statistical values such as point estimates, 5 percent estimates, and 95 percent estimates were also calculated for CDP and CCDP analysis cases. Estimated statistical values for the operating condition are shown in Table 5.

References

1. FirstEnergy Nuclear Operating Company, LER 440-2003-004-01, "Emergency Service Water Pump Upper Shaft Coupling Sleeve Failure," dated January 29, 2004.

2. USNRC, Region III, "Perry Nuclear Power Plant, NRC Integrated Inspection Report 50-440/2003-006," dated October 30, 2003. (ADAMS Accession No. ML033040217)

3. USNRC Office of Enforcement, "Final Significance Determination for a White Finding (NRC Inspection Report 50-440/2004-005) (Perry Nuclear Power Plant) - EA-03-197," dated January 28, 2004. (ADAMS Accession No. ML040280577)

4. FirstEnergy Nuclear Operating Company, LER 440-2004-001-01, "Emergency Service Water Pump Upper Shaft Coupling Sleeve Failure," dated September 18, 2004.

5. USNRC, Region III, "Perry Nuclear Power Plant, NRC Special Inspection Report 50-440/2004-011," dated July 2, 2004. (ADAMS Accession No. ML041900080)

6. Robert Buel, et al., "Standardized Plant Analysis Risk (SPAR) Model for Perry Nuclear Power Plant (Version 3.11)," by Idaho National Engineering and Environmental Laboratory, December 2004.

Table 1. Conditional probabilities (point values) for dominant sequences

Event tree name	Sequence no.	Conditional core damage probability (CCDP)	Core damage probability (CDP)	Importance (CCDP - CDP)[2]
LOCHS	07	6.8E-7	3.4E-7	3.5E-7
LOOP	34-09	3.2E-7	1.5E-8	3.0E-7
Total (all sequences)[1]		1.6E-6	4.5E-7	1.2E-6

Notes:
1. Total CCDP and CDP includes all sequences (including those not shown in this table).
2. Importance is calculated using the total CCDP and total CDP from all sequences of all applicable event trees. Sequence level importance measures are not additive.

Table 2a. Event tree sequence logic for dominant sequences

Event tree name	Sequence No.	Logic ("/" denotes success; see Table 2b for top event names)
LOCHS	07	/RPS * /SRV * /MFW * SPC * CSS * PCSR * CVS * LI01
LOOP	34-09	/RPS * EPS * /SRV * HCS * /RCI * /DEP * /VA01 * AC-07HR

Table 2b. Definitions of fault trees used in event tree logic listed in Table 2a

IE-LOCHS	LOSS OF CONDENSER HEAT SINK
IE-LOOP	LOSS OF OFFSITE POWER
RPS	REACTOR SHUTDOWN FAILS
HCS	HIGH PRESSURE CORE SPRAY FAILS
MFW	FEEDWATER FAILS
PCSR	POWER CONVERSION SYSTEM RECOVERY FAILS
EPS	EMERGENCY POWER SYSTEM FAILS
AC-07HR	OPERATOR FAILS TO RECOVER AC POWER IN 7 HOURS
DEP	MANUAL REACTOR DEPRESSURIZATION FAILS
RCI	REACTOR CORE ISOLATION COOLING FAILS
SRV	ANY ONE SRV FAILS TO RECLOSE
LI01	LATE INJECTION FAILS
VA01	FIREWATER INJECTION FAILS
SPC	SUPPRESSION POOL COOLING FAILS
CSS	CONTAINMENT SPRAY FAILS
CVS	CONTAINMENT VENTING FAILS

Note:
1. "/" indicates that top event is a success event in the event tree logic

Table 3a. CCDP cut sets for LOCHS Sequence 07

CCDP	Percent contribution	Minimal cut sets[1]
Event Tree: LOCHS, Sequence 07		
3.208E-07	46.91	PCS-XHE-XL-LTTRAN * RHR-XHE-XM-ERROR * CVS-XHE-XM-VENT1
8.806E-08	12.88	PCS-XHE-XL-LTTRAN * RHR-MDP-TM-TRNB * CVS-XHE-XM-VENT
6.800E-07	Total[2]	

Table 3b. CCDP cut sets for LOOP Sequence 34-09

CCDP	Percent contribution	Minimal cut sets[1]
Event Tree: LOOP, Sequence 34-09		
1.384E-07	42.56	SSW-MDP-CF-RUN * EPS-XHE-XL-NR07H * OEP-XHE-XL-NR07H
3.774E-08	11/94	EPS-DGN-FR-DGB * EPS-DGN-FR-DGC * EPS-XHE-XL-NR07H * OEP-XHE-XL-NR07H
3.200E-07	Total[2]	

1. See Table 4 for definitions and probabilities for the basic events.
2. Total CCDP includes all cut sets (including those not shown in this table).

Table 4 - Definitions and probabilities for added basic events and condition-affected basis events

Basic event name	Description	Added to base Model	Prob.	Modified to reflect condition	Note
PCS-XHE-XL-LTTRAN	OPERATOR FAILS TO RECOVER THE MAIN CONDENSER	NO	1.000E+0	NO	
RHR-XHE-XM-ERROR	OPERATOR FAILS TO START/CONTROL RHR	NO	5.000E-4	NO	
CVS-XHE-XM-VENT1	OPERATOR FAILS TO VENT CONTAINMENT (DEP EVT)	NO	5.100E-2	NO	
RHR-MDP-TM-TRNB	RHR TRAIN B IS UNAVAILABLE BECAUSE OF MAINTENANCE	NO	7.000E-3	NO	
CVS-XHE-XM-VENT	OPERATOR FAILS TO VENT CONTAINMENT	NO	1.000E-3	NO	
SSW-MDP-CF-RUN	ESW PUMPS FAIL FROM COMMON CAUSE TO RUN	NO	8.230E-7	NO	
EPS-XHE-XL-NR07H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 7 HOURS	NO	2.970E-1	NO	
OEP-XHE-XL-NR07H	OPERATOR FAILS TO RECOVER OFFSITE POWER IN 7 HOURS	NO	1.365E-1	NO	
EPS-DGN-FR-DGB	DIESEL GENERATOR B FAILS TO RUN	NO	2.117E-2	NO	
EPS-DGN-FR-DGC	DIESEL GENERATOR C FAILS TO RUN	NO	2.117E-2	NO	
CFAILED		NO	TRUE	NO	1
SSW-MDP-FR-PUMPA	SSW PUMP A FAILS TO RUN	NO	TRUE	YES	2

NOTE:
1. Basic event probability is changed in the baseline plant model. Bases for change is documented in Basic event probability changes section of this report.

2. Basic event probability is changed to reflect the operating condition.

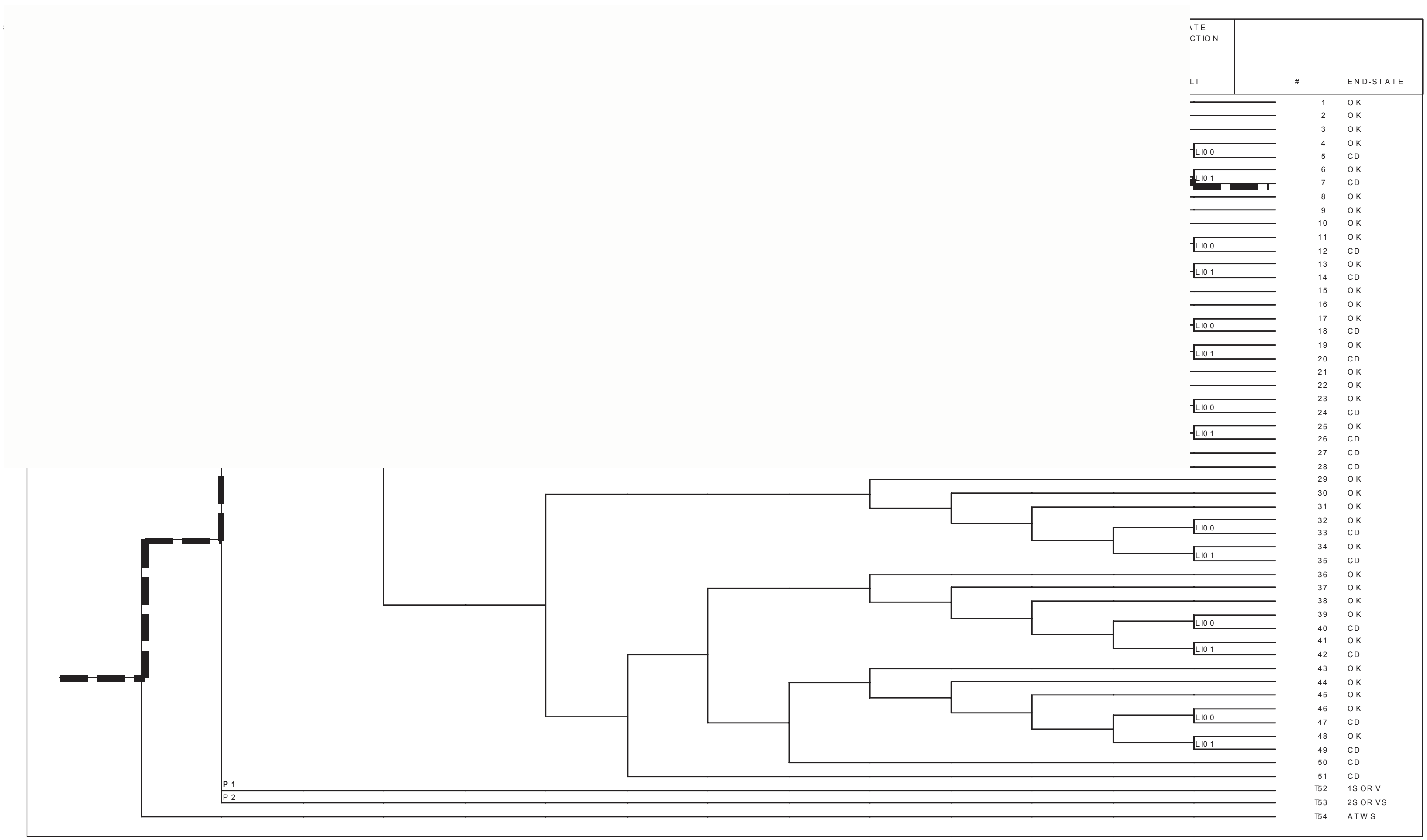
Table 5 - Uncertainty estimates for the operating condition

Plant: Perry Nuclear Power Plant
IR ID: 50-440/2003-006, 50/440/2004-011
SDP: EA-03-197
LER ID : 440-2003-004-01, 440-2004-001-01

Analysis type = Monte Carlo
Samples = 10000; Seeds = 97453

Initiating event (IE)	IE ID	Point estimate	mean estimate	5% estimate	50% estimate	95% estimate
All internal initiating events	CCDP for 1 year	2.550E-05	2.590E-05	2.448E-06	1.556E-05	8.340E-05
	CDP for 1 year	7.031E-06	7.247E-06	2.069E-07	2.098E-06	2.981E-05
	CCDP for 551 hours	1.604E-06	1.629E-06	1.540E-07	9.787E-07	5.246E-06
	CDP for 551 hours	4.422E-07	4.558E-07	1.301E-08	1.320E-07	1.875E-06
	Importance for 551 hours	1.162E-06	1.173E-06	1.410E-07	8.468E-07	3.371E-06

Figure 1 - Perry Nuclear Power Plant - Transient Event Tree Showing Sequence LOCHS 07



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