ARKANSAS NUCLEAR ONE
APPENDIX R
REGULATORY CONFERENCE

July 10, 2003

EX 5

OPENING REMARKS

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Vice President, ANO

[redacted]

Risk Assessment Overview

- NRC's preliminary SDP evaluation concluded unacceptable (greater than green) increase in core damage frequency

Ex5

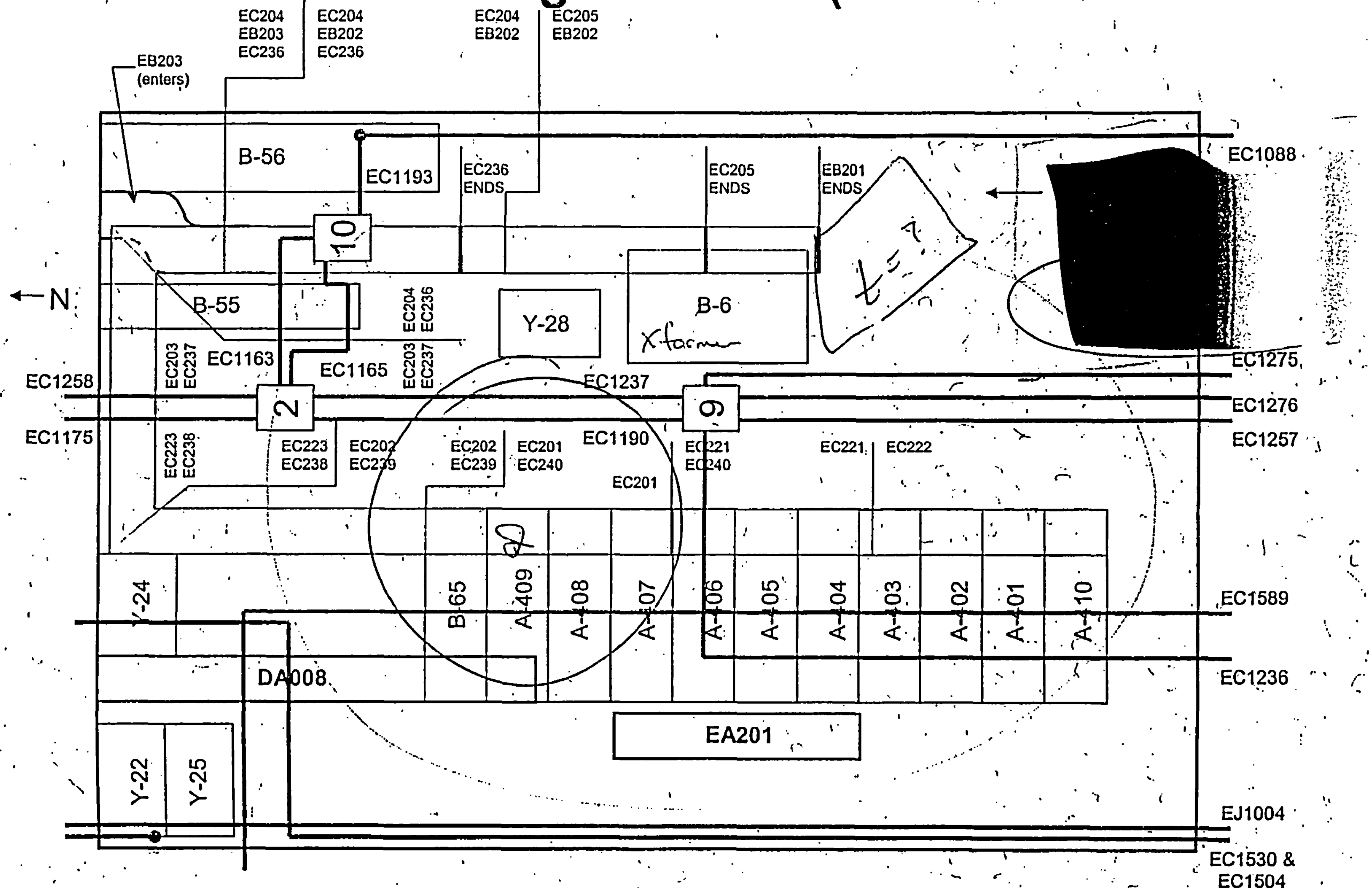
- Key assumptions in NRC evaluations vs ANO's preliminary assessment
  - Heat release rate
  - Human error probability

[redacted]

- Subsequent site-specific in-depth assessment
  - Results incorporated into Unit 1 PSA model to derive ΔCDF

Unit 1 4KV Switchgear Room (fire zone 99M

EX5



A - 4.16kV

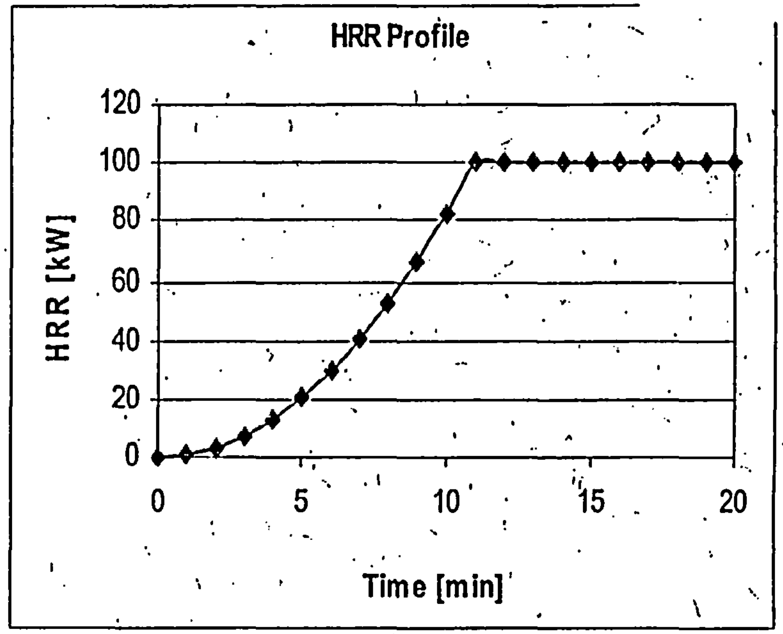
Fire Characterization

[redacted]

- Electrical cabinet fires

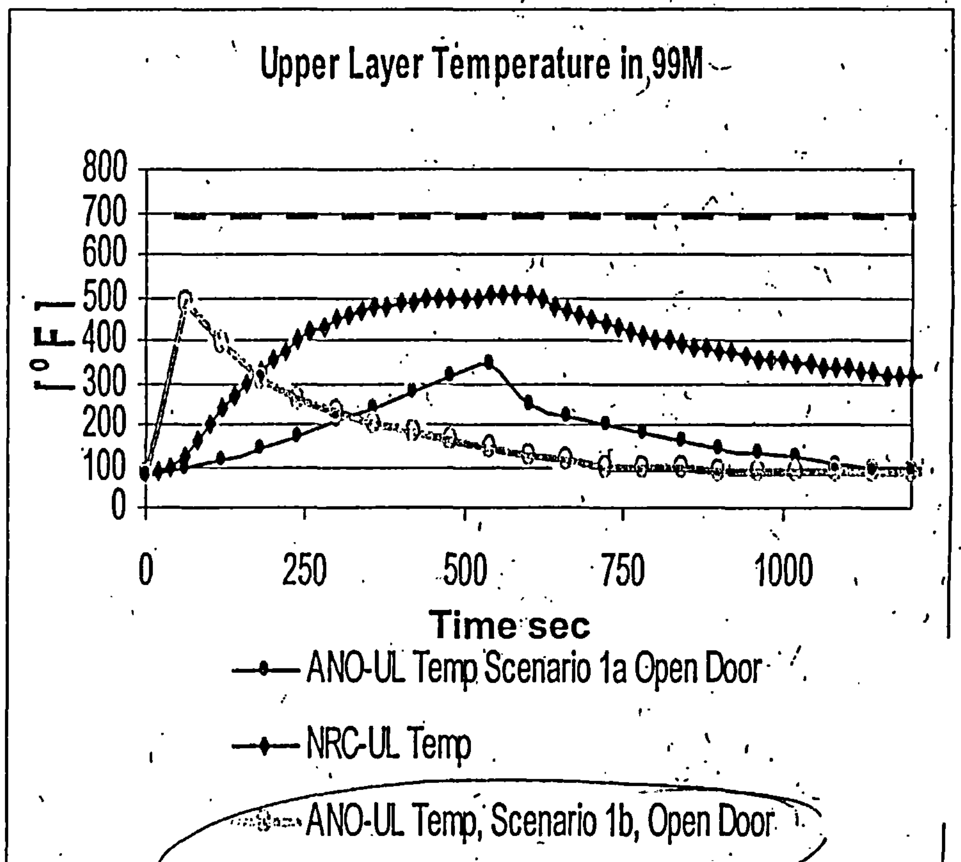
[redacted]

  - The heat release rate data profile is based on the best available fire test data
    - Sandia National Lab (NUREG/CR-4527, 87/88) and VTT (Valtion Teknillinen Tutkimuskeskus, 94/96) in Finland
    - Same test used in the NRC SDP analysis
  - The ANO HRR is based on the highest peak of ST5 (unqualified, open 110 KBTU loading) and all qualified, vertical cabinets (excluding PCT6 and test 23 with 1.5 MBTU loading)
    - The NRC HRR is based on test 23 (qualified, open 1.47 MBTU loading) and test 24 (unqualified, open, 1.44 MBTU)
  - Time-to-peak is based on the average
  - Tests are based on control panels
  - The switchgear, MCC's and load centers are enclosed with sealed penetrations
- Used for scenarios 1a, 2 - 5



Results (cont.) Comparison of NRC and ANO Results

- Damage threshold
  - NRC: 425ºF
  - ANO: 700ºF
- Heat release rate
  - NRC: 500KW
    fire peaking in 105 sec.
  - ANO: 100KW peaking in 12 min (Scenario 1a) + cable fires and high energy fault in A4 switchgear and cable fires (Scenario 1b)
- High energy arcing fault in the 4KV switchgear
  - NRC: Not analyzed
  - ANO: Limiting scenario in terms of its consequence, i.e., affected circuits and timing
- Neither analysis reaches 700°F



Ex5

[redacted]

Results:
Frequency of Fire Scenarios in Fire Zone 99M

ANO SDP Analysis Results

Scenario	Source	Generic Frequency	WFI (location weighting factor)	WIs (ignition source weighting factor)	Floor area ratio (transient fires)	Severity Factor	Ratio of HE event for a severe switchgear fire	Pns by plant personnel or fire watch	Pns by fire brigade	Results
1a	Fire in the A4 switchgear. Nominal value, 100 KW fire	1.50E-02	2.50E-01	5.88E-01	1.00E+00	1.20E-01	2.50E-01	1.00E+00	1.00E+00	6.62E-05
1b	High energy arcing fault in any of the A4 switchgear breaker cubicles	1.50E-02	2.50E-01	5.88E-01	1.00E+00	1.20E-01	7.50E-01	1.00E+00	1.00E+00	1.99E-04
2	Fire in the B55 MCC. Nominal 100 KW fire. Fires in Inverter Y28 are bounded by this scenario.	1.50E-02	2.50E-01	5.88E-02	1.00E+00	1.20E-01	1.00E+00	1.00E+00	1.00E+00	2.65E-05
3	Fire in the B56 MCC. Nominal 100 KW fire	1.50E-02	2.50E-01	5.88E-02	1.00E+00	1.20E-01	1.00E+00	1.00E+00	1.00E+00	2.65E-05
4	Fire in the Y22 Inverter. Base case, 100 KW fire. Fires in Y24 and Y 25 are bounded by this scenario.	1.50E-02	2.50E-01	5.88E-02	1.00E+00	1.20E-01	1.00E+00	1.00E+00	5.00E-01	1.32E-05
5	Fire in the Load Center B6. 100KW nominal HRR	1.50E-02	2.50E-01	5.88E-02	1.00E+00	1.20E-01	1.00E+00	1.00E+00	2.00E-01	5.29E-06
6a	Transient fire in areas of the room where cable trays are exposed to a floor-based fire. Nominal Value of 150KW.	3.60E-02	2.00E+00	1.80E-02	1.00E-01	1.00E+00	1.00E+00	5.00E-01	1.00E+00	6.48E-05
6b	Cable fire caused by welding and cutting in areas of the room where cable trays are exposed to a floor-based fire. Nominal Value of 150KW.	1.30E-03	2.00E+00	2.00E-02	1.00E-01	1.00E+00	1.00E+00	5.00E-02	1.00E+00	2.60E-07

NRC SDP Analysis Results (May 15, 2003 Supplemental Letter Page 25)

Source	Frequency
Electrical cabinets	2.3E-04
Transformers	1.6E-05
Ventilation Subsystems	4.4E-06

Key Systems Affected in the Risk-Significance Determination (Fire Zone 99M)

- The following systems/trains are directly failed due to fire induced power losses of A4 and B6
  - One train and the swing pump of service water
  - One train and the swing pump of HPI (makeup)
  - The A4 associated diesel is no longer usabl[redacted]

EX 5

Human Error Probability Comparison

- NRC approach assumes zone wide damage at time zero
- NRC approach included loss of offsite power

Operator Action	NRC Value No Procedure	NRC Value W/Procedure	ANO Value Previous	ANO Value New
Establish EFW (A3 local start)	1	0.6	0.11	0.098
Establish EFW (Control EFW)	1	0.6	0.038	0.026
Establish Feed & Bleed	0.75	0.55	0.008	0.008
Establish Feed & Bleed (A3 Local Start)	0.75	0.55	0.11	0.098
Secure Diesel with no Service Water	0.75	0.55	Not needed due to no loss of offsite power	

EMPLOYMENT HISTORY:

Mr. Najafi is the Manager of the Fire Protection Program at SAIC responsible for overseeing a business area that includes domestic and international nuclear utilities, DOE facilities and commercial/industrial facilities. He is one of the principal investigators for Electric Power Research Institute (EPRI) fire risk analysis and fire protection projects. These projects included development of EPRI's Fire PRA Implementation Guide and Fire-Induced Vulnerability Evaluation (FIVE) methodology and application of these technologies to US nuclear power plant support. Over the past decade Mr. Najafi has been instrumental in development of the fire research program at EPRI to support nuclear power industry move towards a Risk-Informed/Performance-Based (RI/PB) fire protection rule. Under this program data and methods are being developed a more engineering-based (as opposed to prescriptive-based) approach to fire protection. Several methods where also developed to demonstrate use of the technology, such as "Methods for Evaluating Cable Wrap Fire Barrier Performance."

As part of this process of continuous enhancement of technology, Mr. Najafi is currently the principal technical manager of a joint project between EPRI and USNRC office of Research for development of the next generation of Fire Risk Analysis Methods ~~that can support the fire~~ protection industry in RI/PB rule. This is a ground braking exercise in cooperative research between EPRI and NRC and key to improving the environment for risk-informed rule in fire protection. Mr. Najafi is the key in providing goals and directions to this program that includes the development of the first documented methodology for assessment of fire risk during low power and shutdown modes of operation.

Between 1991 and 1997, Mr. Najafi managed Fire PSA projects at over eighteen (18) U.S. nuclear plants in response to NRC's Individual Plant Examination for External Events (IPEEE) as well as Dodewaard Plant in the Netherlands. The experience was part of the process to improve the Fire PSA data and methods developed by EPRI (with Mr. Najafi as the Project Manager).

Between 1988 and 1993, Mr. Najafi served as SAIC Project Manager for GE's ABWR/SBWR Level 1 PRA, Comanche Peak Level I/II PRA support, Project Engineer (Technical Project Manager) for the Turkey Point Nuclear Power Plant (PWR-W) Units 3 and 4 Level 2 PRA with external events (excluding seismic), and Systems Analysis Task Leader for the River Bend Station (BWR) Level 1 PRA. He also served as an instructor in a course on Seismic PRA and Unresolved Safety Issue (USI) A-46, "Seismic Qualification of Equipment in Operating Plants," for the Omaha Public Power District staff.

During 1987-1988, he was the manager of a project to update the PRA for the Indian Point Unit 3 plant and perform a SAIC/Utility-conducted Level 1 PRA for a BWR-4 plant (confidential client). Mr. Najafi was involved in the N-Reactor Safety and Reliability Evaluation program as the task leader responsible for analyzing the Confinement, Reactor Trip, HVAC, and Emergency Core Cooling Systems.

Mr. Najafi was one of the principal authors of the Reliability-Centered Maintenance studies for the Diesel Generator Systems at the Catawba (PWR-W) and Palo Verde (PWR-CE) Nuclear Power Plants, and the River Water Makeup System for the Susquehanna Steam Electric Station (BWR).

During 1985, Mr. Najafi was one of the principal authors of a PRA study for the Peach Bottom plant (BWR) as part of the NUREG-1150 program for Sandia National Laboratories. He was primarily responsible for the modeling of the plant Safety Support Systems including Electric Power and Service Water Systems.

During 1985 and 1986, Mr. Najafi directed an NRC-sponsored work to develop a methodology for assessment of uncertainties in the phenomenological events (back-end). This effort involved development of