UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
ARLINGTON, TEXAS 76011-4005

July 17, 2003

EA-03-016

Craig G. Anderson, Vice President,
Operations
Arkansas Nuclear One
Entergy Operations, Inc.
1448 S.R. 333
Russellville, Arkansas 72801-0967

SUBJECT: REGULATORY CONFERENCE WITH ENTERGY OPERATIONS, INC. CONCERNING THE ARKANSAS NUCLEAR ONE FACILITY

This refers to the meeting conducted in the Region IV office of the Nuclear Regulatory Commission, located in Arlington, Texas, on July 10, 2003. This meeting was held to discuss a finding which was preliminarily determined by the NRC to have a risk significance of greater than green (greater than very low safety significance) using NRC Inspection Manual Chapter 0609, "Significance Determination Process." This finding is described in NRC Inspection Report 50-313/01-06; 50-368/01-06, dated August 20, 2001, and in a letter to you from the NRC Region IV Regional Administrator, dated April 15, 2002. In a March 25, 2003, letter from the Region IV Director of the Division of Reactor Safety to you, we described the preliminary safety significance determination of the finding. An apparent violation was associated with the finding for the failure to ensure that one train of redundant safe shutdown equipment and cables in Fire Zones 98J and 99M was free of fire damage by one of the methods described in 10 CFR Part 50, Appendix R, Sections III.G.2 and III.G.3 (see Enclosure 4).

The NRC presented a description of the finding and a summary of NRC's significance determination of the finding. You and your staff provided a detailed description of your fire modeling methodology and your risk assessment of the finding, which you determined to be green (of very low safety significance). During the meeting the NRC requested additional information necessary for our understanding of your risk assessment. This request for additional information is provided in Enclosure 5 to this letter. Upon receipt, we will review this information and inform you if additional information is required.

During the meeting, you stated that you do not agree that the finding is a violation, stating it is your position that with respect to this issue, you were in compliance with your licensing basis. The NRC's final determination of the significance of the finding and the associated apparent violation will be issued separately.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosures will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Should you have any questions concerning this matter, we will be pleased to discuss them with you.

Sincerely,

/RA/ by RLN for

Charles S. Marschall, Chief
Engineering and Maintenance Branch
Division of Reactor Safety

Dockets: 50-313; 50-368
Licenses: DPR-51; NPF-6

Enclosures:
1. Agenda
2. Attendance List
3. Licensee Presentation
4. Apparent Violation
5. Request for Additional Information

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DRP Director **(ATH)**
Acting DRS Director **(ATG)**
Senior Resident Inspector **(RLB3)**
Branch Chief, DRP/D **(LJS)**
Senior Project Engineer, DRP/D **(JAC)**
Staff Chief, DRP/TSS **(PHH)**
RITS Coordinator **(NBH)**

ADAMS: **G** Yes **G** No Initials: ______
G Publicly Available **G** Non-Publicly Available **G** Sensitive **G** Non-Sensitive

RIV:TL:DRS/EMB	RIV:C:DRP/D	RIV:D:ACES	RIV:C:DRS/EMB	
RLNease;lmb	LJSmith	GFSanborn	CSMarschall	
/RA/	/RA/	/RA/	/RA/	
07/17/2003	07/17/2003	07/17/2003	07/17/2003	

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ENCLOSURE 1

Regulatory Conference Agenda

CONFERENCE WITH ENTERGY OPERATIONS, INC.,
ARKANSAS NUCLEAR ONE

JULY 10, 2003

NRC REGION IV, ARLINGTON, TEXAS

1. Introduction and Opening Remarks	Pat Gwynn, Acting Regional Administrator
2. Apparent Violation and Goals of the Regulatory Conference	Dwight Chamberlain, Acting Deputy Regional Administrator
3. Significance Determination Influential Assumptions	See-Meng Wong, Senior Reactor Analyst, NRR; Troy Pruett, Chief, Plant Support Branch, Division of Reactor Safety
4. Licensee Presentation	
5. NRC Caucus	
6. Resume Conference	
7. NRC Closing Remarks	Pat Gwynn

ENCLOSURE 2

ATTENDANCE LIST

NRC-ENTERGY REGULATORY CONFERENCE ATTENDANCE

LICENSEE/FACILITY	**Entergy, Inc.**
DATE/TIME	**July 10, 2003; 9:00 am CDT**
CONFERENCE LOCATION	**NRC Region IV** **Arlington, TX**
EA NUMBER	**03-016**

LICENSEE REPRESENTATIVES

NAME (PLEASE PRINT)	ORGANIZATION	TITLE
Craig Anderson	Entergy Operations Inc	VP, Operations ANO
Joe Kowalewski	Entergy Operations, Inc.	Director, Eng - ANO
Dale James	Entergy Operat	Manager [illegible]
Glenn Ashley	Entergy Operations Inc	Manager, Licensing ANO
Mike Krupa	Entergy Operations	Director Nuclear Safety & Lic.
Bob Eichenberger	Entergy Operations	Operations Manager - ANO
Sherrie Cotton	Entergy Operations	Director, Nuclear Safety Assurance
Jessica Walker	Entergy Operations	PSA Engineer
BIJAN NAJAFI	SAIC	MANAGER, FIRE PROTECTION SECTION
Mike Cooper	Entergy Operations	Licensing Specialist
Bill Hannaman	DS&S	Engineer
Patricia Campbell	Winston & Strawn	Attorney
RONALD RISPOLI	ENTERGY	MANAGER, LOSS CONTROL

Page 1 of 1

NRC-ENTERGY REGULATORY CONFERENCE ATTENDANCE

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DATE/TIME	**July 10, 2003; 9:00 am CDT**
CONFERENCE LOCATION	**NRC Region IV** **Arlington, TX**
EA NUMBER	**03-016**

NRC REPRESENTATIVES

NAME (PLEASE PRINT)	ORGANIZATION	TITLE
T. Pat Gwynn	NRC, Region IV	Acting Regional Administrator
Dwight Chamberlain	NRC, Region IV	Acting Deputy Regional Administrator
Art Howell	NRC, Region IV	Division Director, DRP
Tony Gody	NRC, Region IV	Acting Division Director, DRS
Troy Pruett	NRC, Region IV	Chief, Plant Support Branch, DRS
Charles Marschall	NRC, Region IV	Chief, Engineering and Maintenance Branch, DRS
Linda Smith	NRC, Region IV	Chief, Projects Branch D, DRP
Karla Smith	NRC, Region IV	Regional Counsel
Rebecca Nease	NRC, Region IV	Team Leader, Engineering and Maintenance Branch, DRS
Gary Sanborn	NRC, Region IV	Director, ACES
See-Meng Wong	NRC, Office of Nuclear Reactor Regulation	Senior Reactor Analyst
Phil Qualls	NRC, Office of Nuclear Reactor Regulation	Fire Protection Engineer
Brian Tindell	NRC, RIV	Nuclear Safety Intern
Jeff Moreno	NRC, RIV	Engineering Associate

NRC-ENTERGY REGULATORY CONFERENCE ATTENDANCE

LICENSEE/FACILITY	**Entergy, Inc.**
DATE/TIME	**July 10, 2003; 9:00 am CDT**
CONFERENCE LOCATION	**NRC Region IV Arlington, TX**
EA NUMBER	**03-016**

NRC REPRESENTATIVES

NAME (PLEASE PRINT)	ORGANIZATION	TITLE
Russ Bywater	NRC RIV	Sr Resident Inspector

NRC-ENTERGY REGULATORY CONFERENCE ATTENDANCE

LICENSEE/FACILITY	Entergy, Inc./Arkansas Nuclear One
DATE/TIME	July 10, 2003; 10:00 am EDT
CONFERENCE LOCATION	NRC Rockville, MD
EA NUMBER	03-016

NRC REPRESENTATIVES

NAME (PLEASE PRINT)	ORGANIZATION	TITLE
TOM ALEXION	NRR/DLPM	PROJECT MANAGER
Jim Bongarra	NRR/DIPM/IROB	ENGINEERING PSYCHOLOGIST
Daniel Frumkin	NRR/DSSA/SPLB	FIRE PROTECTION ENGINEER
Gareth Parry	NRR/DSSA	Senior Level Advisor for Probabilistic Risk Assessment
David Diec	NRR/DRIP	LEAD PM - MANUAL ACTIONS Rulemaking
Bob Gramm	NRR/DLPM	Section Chief
Mark Henry Salley	NRR/SPLB	Fire Protection Engineer
Naeem Iqbal	NRR/SPLB	Fire Protection Engineer
John Hannon	NRR/SPLB	Branch Chief
MIKE TSCHILTZ	NRR/SPSB	Branch Chief
Mark Reinhart	NRR/SPSB	Section Chief
JENNIFER DIXON-HERRITY	OE	SENIOR ENFORCEMENT SPECIALIST
Richard Eckenrode	NRR/DIPM/IROB	Senior Human Factors Engineer

TOTAL P.01

NRC-ENTERGY REGULATORY CONFERENCE ATTENDANCE

LICENSEE/FACILITY	**Entergy, Inc./Arkansas Nuclear One**
DATE/TIME	**July 10, 2003; 9:00 am CDT**
CONFERENCE LOCATION	**NRC Region IV Arlington, TX**
EA NUMBER	**03-016**

MEMBERS OF THE PUBLIC

NAME (PLEASE PRINT)	ORGANIZATION	TITLE
ROBERT DUKES	NISYS CORPORATION	LEAD SENIOR ENGINEER/ PROJECT MANAGER

ENCLOSURE 3

LICENSEE'S PRESENTATION

ARKANSAS NUCLEAR ONE
APPENDIX R
REGULATORY CONFERENCE

July 10, 2003

OPENING REMARKS

Craig Anderson

Vice President, ANO

INTRODUCTION

Sherrie Cotton

Director, Nuclear Safety Assurance

AGENDA

Opening Remarks	Craig Anderson VP, ANO
Introduction	Sherrie Cotton Director, NSA
Risk Assessment Methodology	Dale James Manager, EP&C
Fire Modeling	Bijan Najafi SAIC Analyst
Break	
Probabilistic Safety Assessment	Jessica Walker PSA Engineer
Overall Summary	Joe Kowalewski Director, DE
Closing Remarks	Craig Anderson VP, ANO

Risk Assessment Methodology

Dale James

Manager, Engineering Programs and

Components

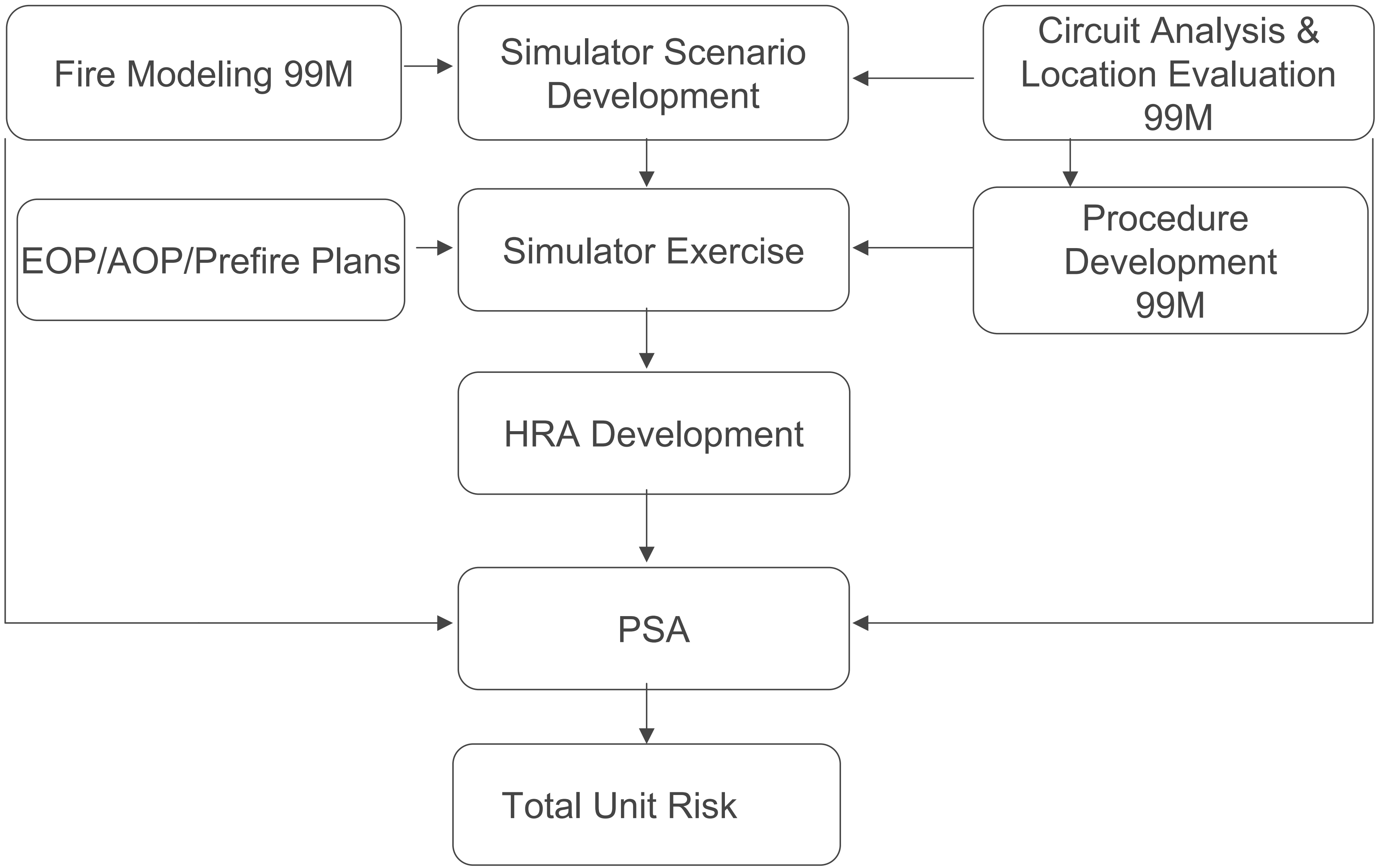
Problem Statement

- NRC Conclusions
  - ANO's reliance on manual actions in lieu of providing separation design features is in violation of Appendix R
  - ANO's strategy for implementing manual actions is inadequate

Risk Assessment Overview

- NRC's preliminary SDP evaluation concluded unacceptable (greater than green) increase in core damage frequency
- Key assumptions in NRC evaluations vs ANO's preliminary assessment
  - Heat release rate
  - Human error probability
- Subsequent site-specific in-depth assessment
  - Results incorporated into Unit 1 PSA model to derive ΔCDF

Risk Informed Strategy for Zone 99M



Risk Assessment Comparison

NRC

- 425° F cable failure temperature
- Zone wide prompt damage
- Generic HRA
  - Based on zone wide prompt damage
  - Included LOOP
- Greater than Green finding

ANO

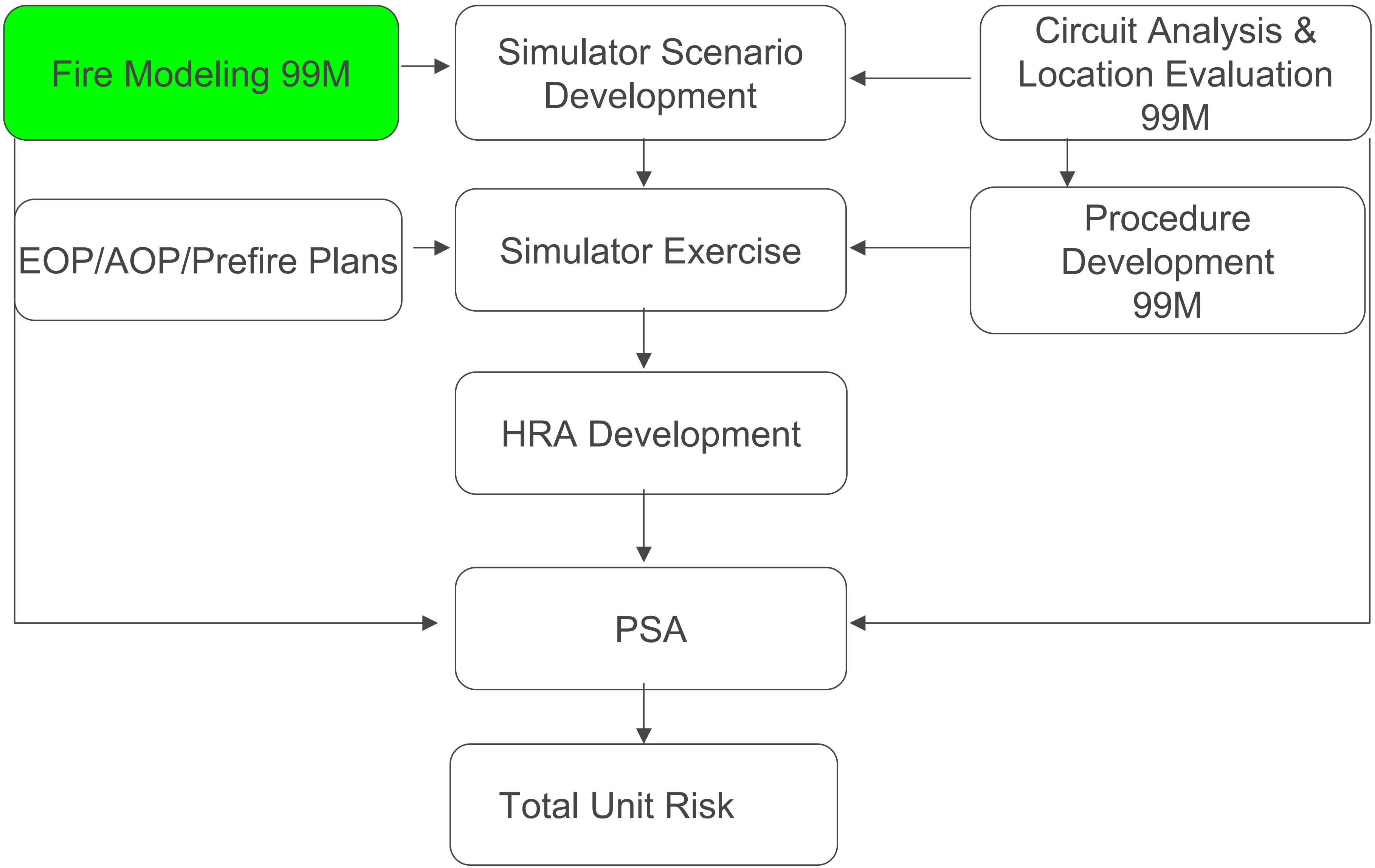
- 700° F cable failure temperature
- Limited time phased damage
- Plant specific HRA
  - Scenario specific operator actions evaluated
  - No LOOP
- Green finding

FIRE MODELING

Bijan Najafi

SAIC Analyst

Risk Informed Strategy for Zone 99M



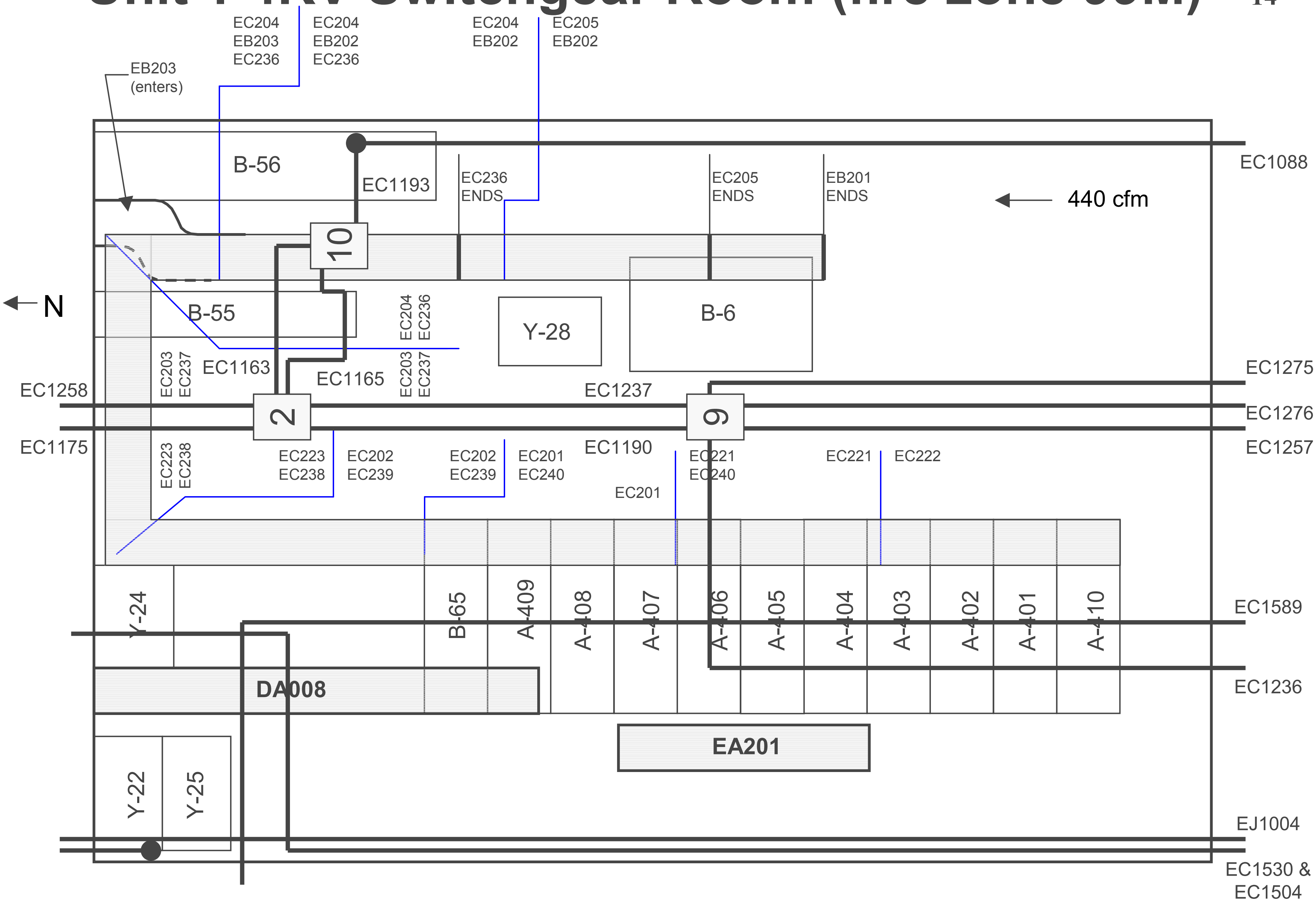
Summary

- In our analysis we will show that:
  - Damage to equipment and instruments needed for safe shutdown will be limited to portions of the room
  - Failures will occur over a period of time, and
  - No credible fire can be postulated that leads to zone-wide damage

Fire Modeling in the 4KV Switchgear Room (Fire Zone 99-M)

- Unit 1 4KV switchgear room (fire zone 99M)
- Fire scenario selection
- Fire characterization
- Fire modeling, evaluation of the consequences and timing of the fire scenarios
- Results and conclusions

Unit 1 4KV Switchgear Room (fire zone 99M)



Unit 1 4KV Switchgear Room (fire zone 99M)

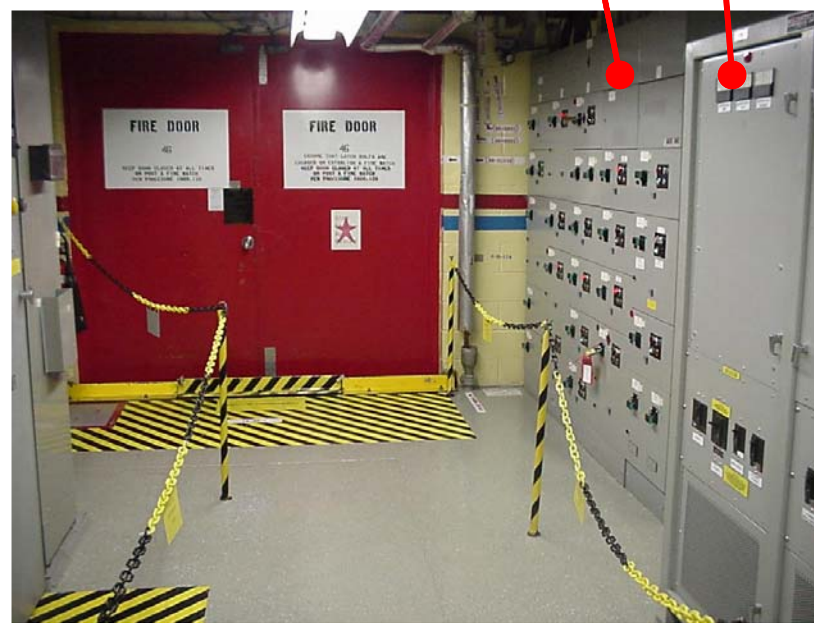
A4 Switchgear

99M – north view

B55 MCC

Y28 Inverter

99M – north view



Unit 1 4KV Switchgear Room (fire zone 99M)

Typical ANO switchgear cabinet wiring, control cubicle

99M - south view

B6 Load center

Dry-type transformer

Fire Scenario Selection: General Approach

- Fire scenarios define potential ranges of damage by a fire
  - They define sequence and timing of failures, i.e., equipment and instruments
  - Ensure that risk-significant failure sets are identified
- Considerations for selection of fire scenarios
  - Location of critical cables in the room
  - Potential characteristics of the fire sources located in the zone, thermal and high energy
  - Configuration of the combustibles in the room

Fire Scenario Selection: General Approach

- Three distinct fire scenario classifications:
  - An electrical fire (non-energetic) in any of the electrical cabinets in the room
    - Fire may spread in the cable trays, but requires considerable time
    - Circuit damage/failures follow a time-phased sequence with first damage after 10 minutes
  - A high energy arcing fault switchgear fire that may initiate secondary fire
    - The event has an initial (immediate) pressure phase that causes damage to targets and ignites exposed cables in the vicinity
    - The fire may continue in the switchgear and grow within the ignited combustibles (e.g., cable trays) in the vicinity
    - There is an initial/immediate circuit damage/failure followed by potential time-phased circuit damage/failures
  - A transient fire that may spread into cable trays
    - A transient fire between B55 and B56 was selected as the maximum expected scenario due to its potential for extent and timing of damage
    - Circuit damage/failures follow a time-phased sequence with first damage after 10 minutes

Fire Scenario Selection: Scenarios Modeled in Zone 99M

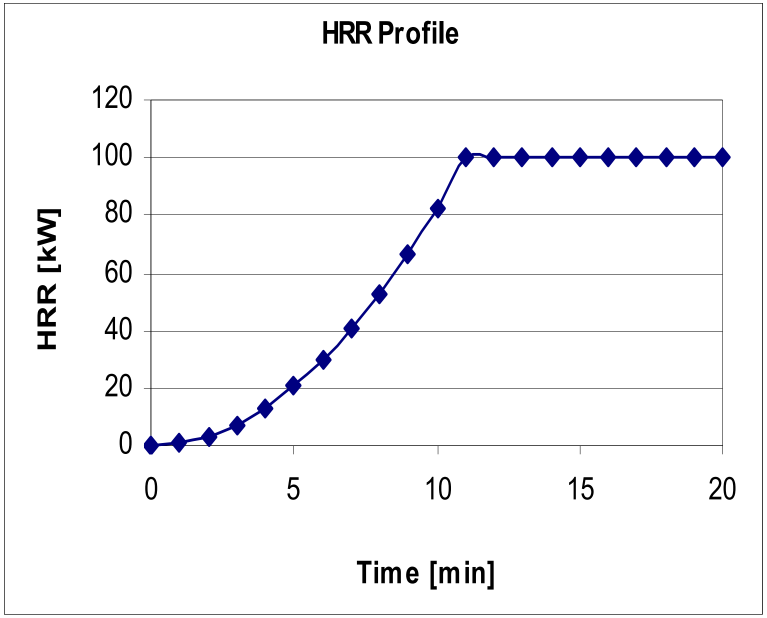
- Eight fire scenarios selected represent credible fire risks for 99M
  - Scenario 1a - Fire in A4 switchgear
  - Scenario 1b - High energy fire in A4 switchgear
  - Scenario 2 - Fire in the B55 motor control center
  - Scenario 3 - Fire in the B56 motor control center
  - Scenario 4 - Fire in the Y22 inverter
  - Scenario 5 - Fire in the B6 load center
  - Scenario 6a - Transient fire between B55 and B56 below three stack tray
  - Scenario 6b - Welding/cutting fire between B55 and B56 below three stack tray
- Illustration of these scenarios is provided in the attachment to this presentation

Fire Scenario Selection: NRC and ANO SDP Analyses

- NRC SDP fire scenarios
  - Based on fire size
    - Total room heat-up and zone-wide damage
  - Electrical cabinet and electrical equipment fires
- ANO SDP fire scenarios
  - Based on source and target-set characteristics and configuration
    - Local as well as zone-wide damage
  - Electrical cabinets in zone
  - High energy arcing faults in the 4KV switchgear (a “beyond design basis event”)
  - Transient fires including hot work

Fire Characterization

- Electrical cabinet fires
  - 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

100KW is an appropriate estimate of HRR for the type of cabinets in 99M

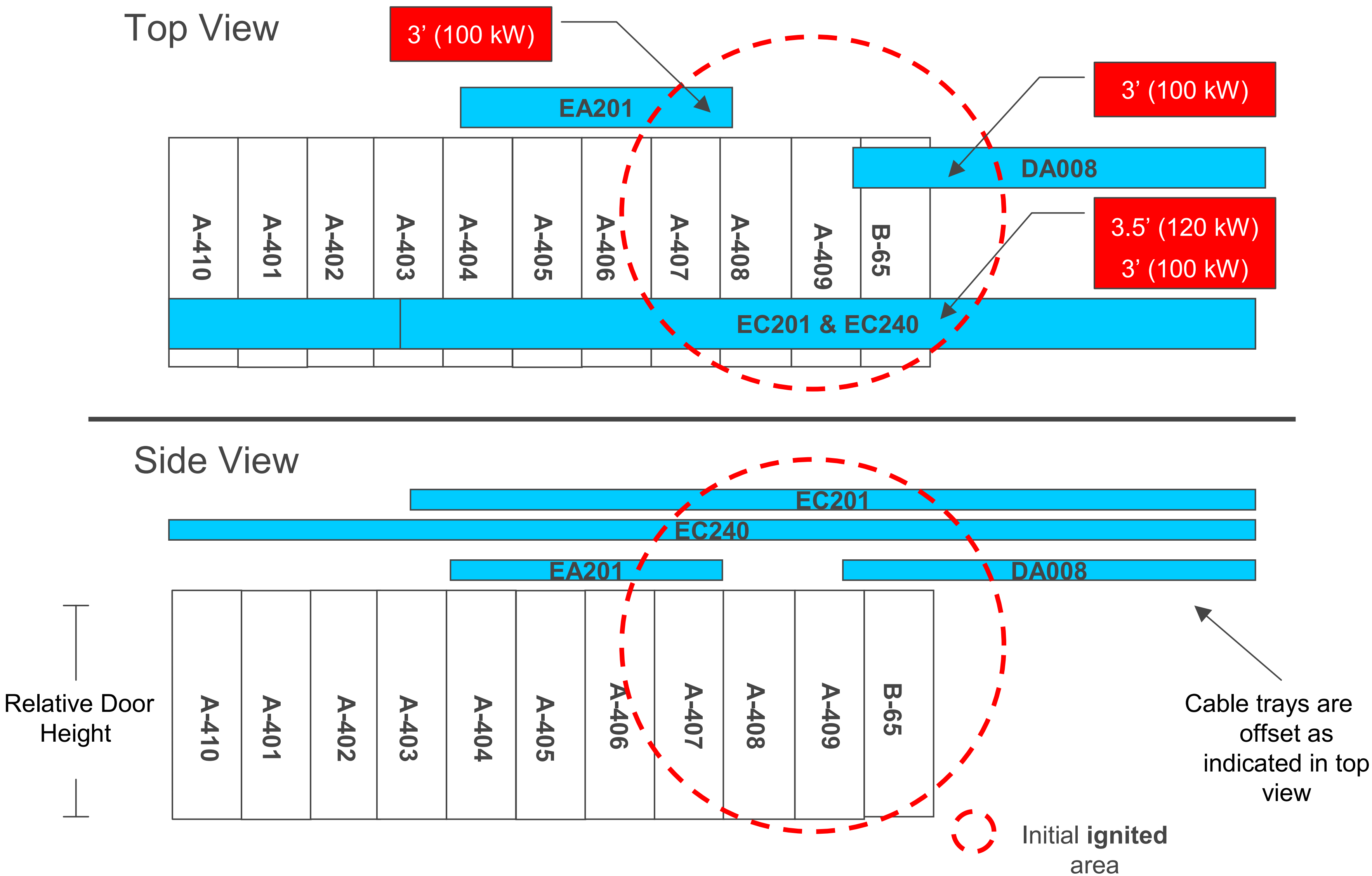
Fire Characterization (cont.)

- Cable tray fire heat release rate: $\dot{Q}_{ct} = 0.45 \cdot \dot{q}_{bs} \cdot A_o$
  - Widely used model from Society of Fire Protection Engineers (SFPE) handbook
    - Used for scenarios that include ignited cable trays
- Transient fires: 150KW
  - Typical refuse based on fire tests at SNL/LLNL documented in EPRI Fire PRA Guide
    - Used for scenarios 6a and 6b

Fire Characterization (cont.) High-energy Switchgear Arcing Fire

- The damage/ignition zone of the initial pressure phase is derived from US nuclear experience (next slide) (EPRI SU105928 Supp to EPRI Fire PRA Guide)
- Ensuing electrical cabinet fires (the switchgear or others exposed to its arcing fault) follow the same behavior as the non-energetic electrical cabinet fires
- Potential ensuing cable fires spread horizontally and spread faster vertically through cable tray stacks
- Observations:
  - Experience of the US nuclear industry indicates that damaging/severe switchgear fires tend to be of the energetic arcing fault type
- Used in scenario 1b

Fire Characterization (cont.)
ZOI of the High-energy Switchgear Arcing Fire



Fire Characterization: NRC and ANO SDP Analyses

- Electrical cabinets
  - NRC: 200-500KW, peaking in 105 seconds
  - ANO: 100KW, peaking in 12 minutes
- High energy fires in switchgear
  - NRC: Assumed covered by the range of HRR
  - ANO: Empirical model based on experience (previous slide), damage/ignition within five ft.
- Transient fires
  - NRC: Out of scope
  - ANO: 150KW, peaking in 10 minutes

Fire Modeling:
Model for Prediction of Fire Growth

- Hand calculations used to calculate time to localized target damage
  - Target immersed in flame
  - Target in the fire plume
  - Target in the ceiling jet, and
  - Target in the flame radiation zone
- CFAST used to calculate room temperature and target damage/ignition due to hot gas layer
  - CFAST and simple correlations such as Heskestad, are validated and widely used for the range of fire conditions expected in zone 99M
- Cable fires: Used fire tests for both growth through stacks and horizontal cable tray
  - An empirical model used to determine the extent and timing of the spread through the stack (based on SNL tests documented in NUREG/CR5384 and in the EPRI Fire PRA Guide)
  - Ten linear ft/hr is the generally used available model for fire propagation along horizontal cable tray (EPRI NP 7332)

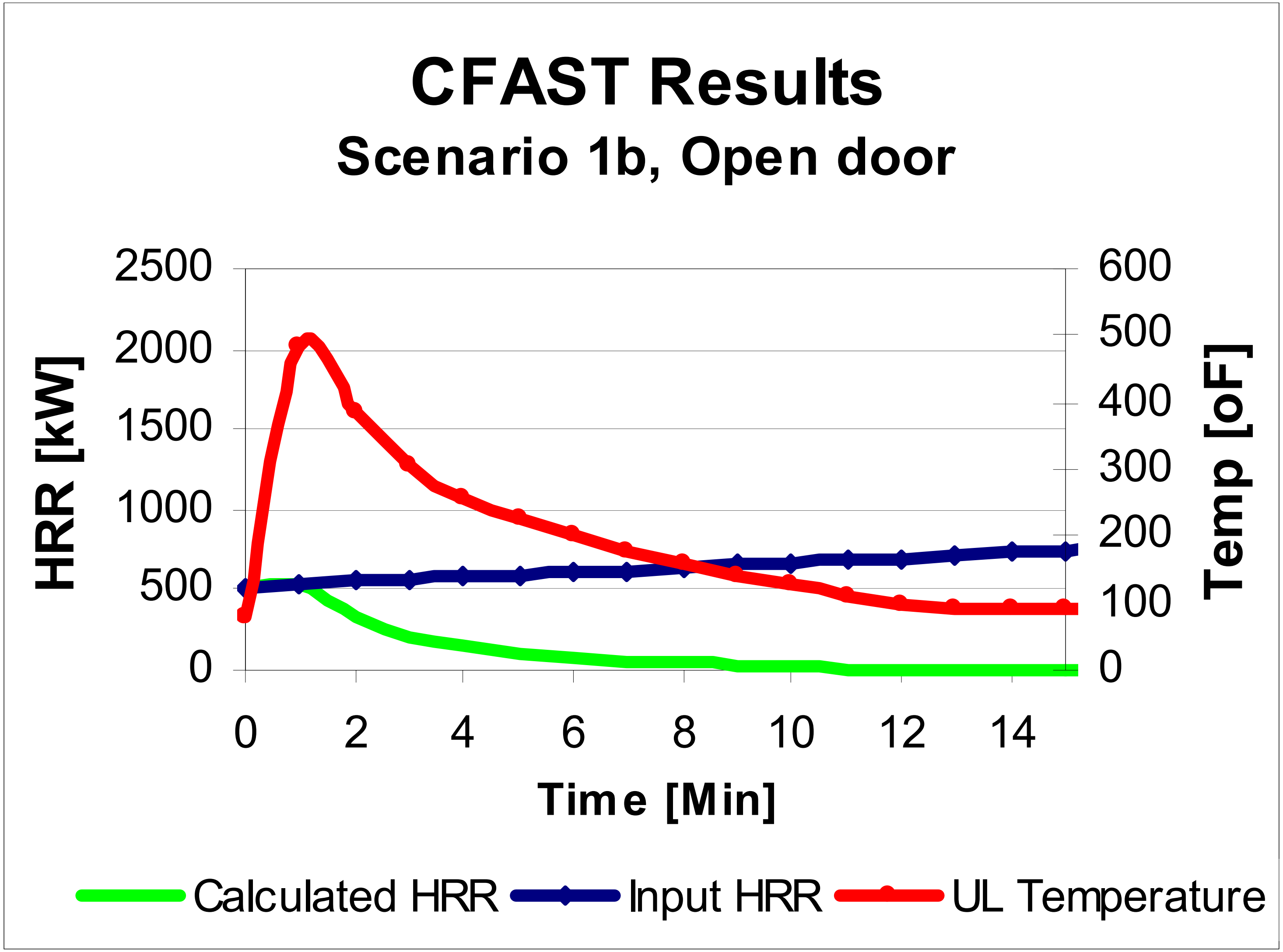
Fire Modeling: Target Damage/Ignition

- Targets (cables) are considered damaged or ignited when their surface temperature reaches 700°F
  - Thermoset insulated cable predominantly used in the plant as verified through original and current design and installation specifications
  - Thermoplastic insulated cables are not used in ANO Unit 1 high risk zones
- This is the critical difference between the NRC and ANO analysis as it relates to the extent and timing of fire damage
  - 425°F vs 700°F
  - Critical to extent and timing of damage and fire growth

Fire Modeling: Target Damage/Ignition

- Assumed cables inside metal conduits damaged at the same critical temperature, but will not contribute to room heat-up
- High-energy arcing fire
  - Assumed raceways and cabinets in the zone-of-influence are damaged with exposed cables (trays) ignited
  - Assumption conservative for the conduits (if stainless or galvanized steel pipes) where they are likely to survive the short-lived (seconds) initial pressure spike
- Spurious operation of damaged circuits were modeled. In some cases, the likelihood of the spurious actuation was obtained from the EPRI Expert Elicitation report (EPRI 1006961) which was estimated in part based on the data from EPRI/NEI circuit failure characterization fire tests

Results


CFAST Results
Scenario 1b, Open door
HRR [kW]
Temp [oF]
Time [Min]
2500
2000
1500
1000
500
0
600
500
400
300
200
100
0
0
2
4
6
8
10
12
14
Calculated HRR
Input HRR
UL Temperature

Results (cont.)

- A high-energy switchgear fire (scenario 1b) is the maximum expected fire scenario
  - Initial HE phase could lead to ignition of as much as 12 linear ft of cable tray
  - After the initial HE phase, ensuing cable fire may grow although at a very slow rate
- The floor-based sources of fire in fire zone 99M are electrical cabinets and transients
  - The likely location of electrical cabinet fires (flame) is below 5ft off the floor once the breaker cubicle is open in the high-energy event
  - The floor-based fire intensity needed to generate damaging (700^{o}F) HGL is ~2MW.
  - None of the floor-based fires are capable of such intense heat
- Only cable fires are potentially capable of generating such intensity if enough cable is involved
  - Cable tray fires are elevated fires (none of the cable trays in fire zone 99M are located below the 8ft door opening)
  - Cable fires are expected to be in the smoke layer once the smoke layer reaches the top of the door. Once in the smoke layer, intensity of the cable fire will be controlled by the oxygen availability, which is not enough to sustain the combustion process
- With an elevated cable fire that grows at a rate of 10 linear ft/hr as input
  - The oxygen depletion occurs very quickly, regardless of open or closed door
  - The cable fire does not grow beyond the initial 12 ft
  - The temperature peaks at 500-535°F
  - The fire has to be below the settled smoke layer (4-5 ft) for the cable fire to continue to grow

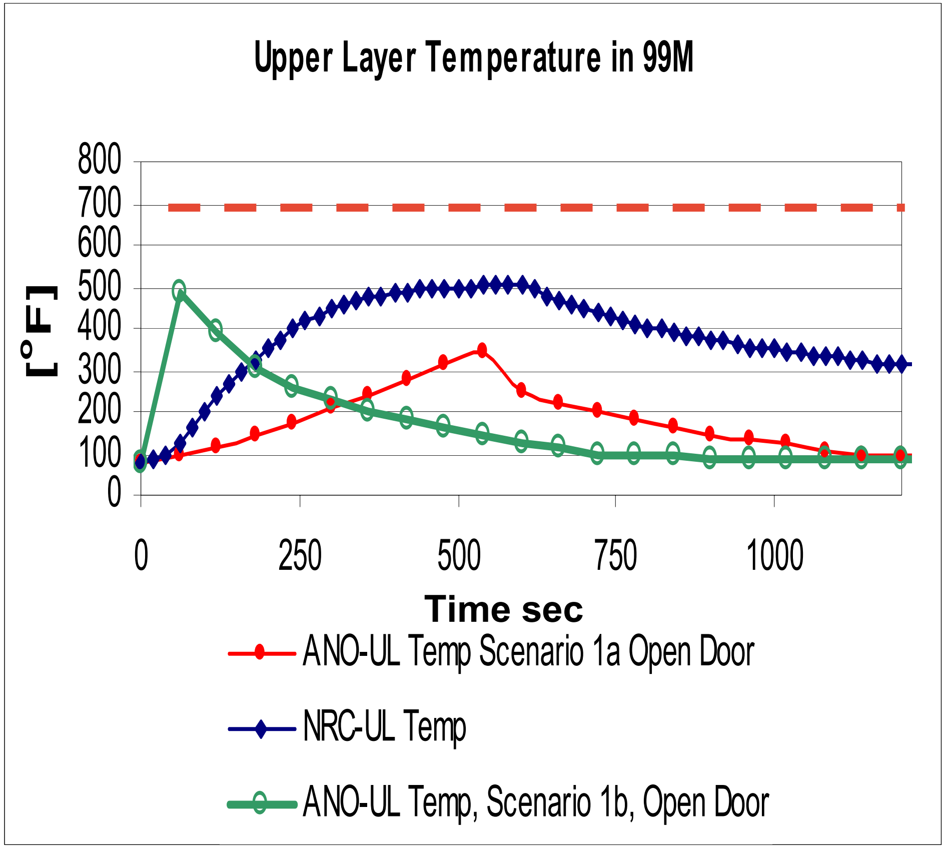
Results (cont.)

- The limiting fire scenario, one that can generate a damaging HGL, is not credible
  - The non-suppression probability by the brigade for very long duration cable fires (100 minutes for the high-energy switchgear event) is 0.01 (per EPRI Fire PRA Guide)
  - Fuel depletion, cables ignited earlier have burned out
  - Parts of the cable trays are coated with flamastics which both delays ignition and slows propagation of cable fires
  - Continued growth of the non-piloted cable fire for a long time is not likely. (Tests reported in NUREG/CR-5387 state that cable fires, "spreading horizontally only as it progressed from level to level")

- **Maximum expected fire is a high-energy switchgear fire**
- **No credible fire reaches 700^{o}F in this room (limiting fire scenario)**

Results (cont.)
Comparison of NRC and ANO Results

- Damage threshold
  - NRC: 425°F
  - ANO: 700°F
- 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



Frequency of Fire Scenarios

- Fire risk = $\sum$ (Scenario Frequency) $_i$ x CCDP $_i$
- Scenario Frequency is derived from multiplication of:
  - Generic fire frequency
    - Based on the EPRI FIVE method (EPRI TR 105928 page 4-7)
  - Severity Factors
    - Based on type and location of fire (EPRI TR 105928)
  - High energy weighting factor for the 4KV switchgear
    - Based on operating experience (EPRI fire data base)
  - Prompt suppression of transient fires by plant personnel or fire watch
    - Based on operating experience (EPRI TR 105928, Appendix K)
  - Manual suppression by fire brigade
    - For scenarios that critical target is beyond plume, ceiling jet or flame radiation zone

- Next presentation discusses development of the $CCDP_i$ and fire risk

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

Fire Modeling Summary

- Maximum expected fire scenario in fire zone 99M is a high energy switchgear fire
  - Immediate damage caused by high energy event will be limited to portions of the room
  - Followed by time delayed failures caused by secondary cable fires
- Credible fires will not result in a hot gas layer (limiting fire scenario) in excess of the cable failure temperature
  - Zone wide damage is not credible
  - Adequate margin

Probabilistic Safety Assessment

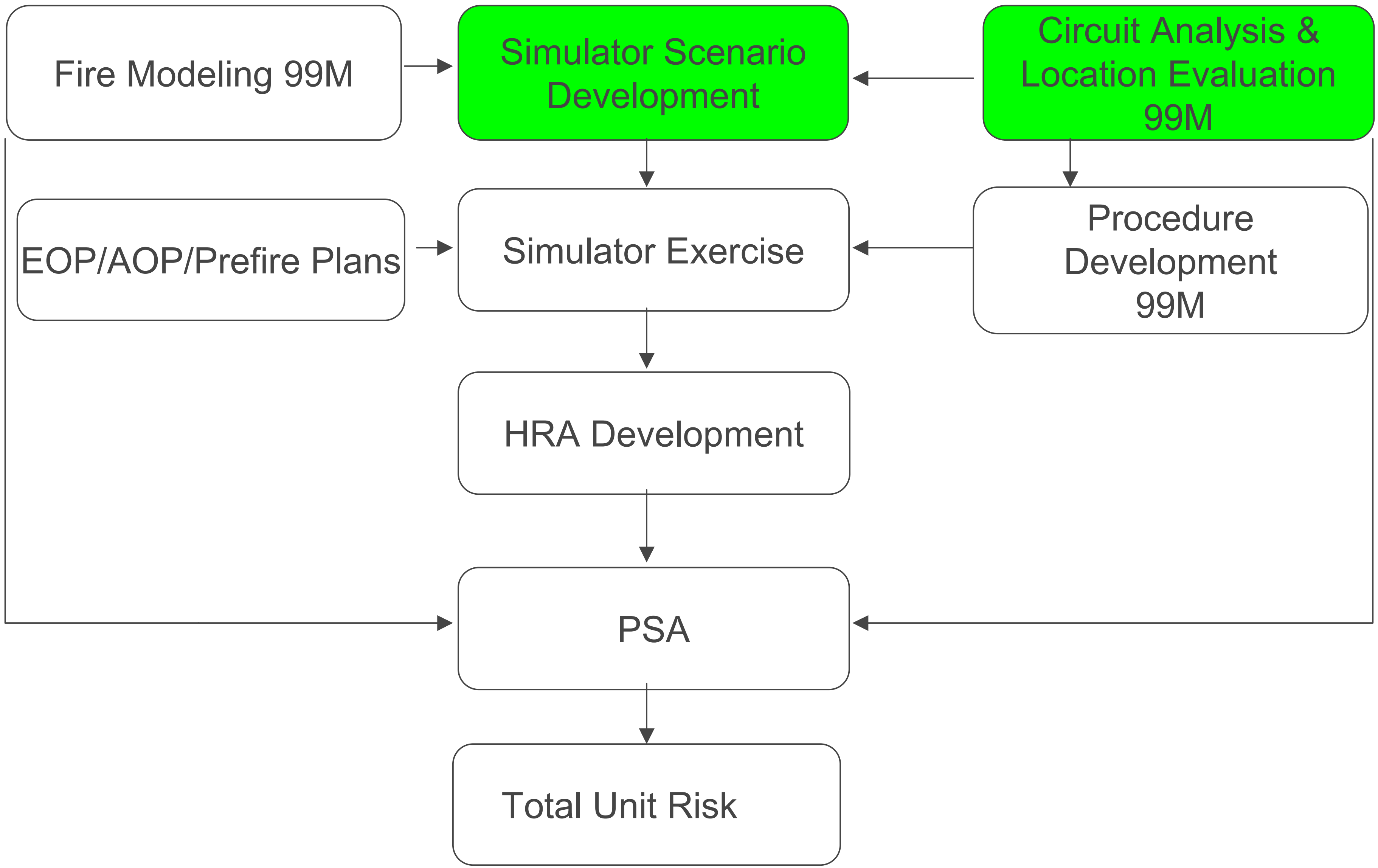
Jessica Walker

PSA Engineer

Introduction

- Key system failures in 99M
- Affected components due to cable failure
- Key operator action/response
- Simulator scenario/results
- Operator action probabilities
- CCDP calculation
- Delta CDF determination

Risk Informed Strategy for Zone 99M



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 usable

Circuit Analysis

- Detailed circuit analysis performed on zone 99M
- Investigation of cables located in the trays and conduits associated with the target sets
- Analysis showed no loss of offsite power associated with zone 99M
  - NRC evaluation did use loss of offsite power
- Analysis of associated failure modes for affected cables
- Failures unrelated to safe shutdown also examined to provide accurate portrayal of the risk caused by the fire

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

- Scenario specific failures are based on cable location; subsets of the following are impacted for each scenario
  - EFW flow control valves
    - Loss of power to these valves will fail them open (desired state)
    - Subsequent spurious operation not probable
  - DC control power to the A3 switchgear fails
    - Breakers to remain static and require manual closure at the switchgear
  - P-7B (motor driven EFW pump) suction valve could spuriously close
    - Cable in conduit; spurious operation not probable but assumed in evaluation
  - Steam admission valves for P-7A (turbine-driven EFW pump)
    - Requires local action to start P-7A
  - Aux-lube oil pumps for the unaffected HPI train
    - Requires local start of HPI pump when affected

Operator Actions/Response in the Risk-Significance Determination (Fire Zone 99M)

- A subset of the following operator actions are required in each scenario
  - Starting turbine-driven EFW pump P-7A manually and the positioning of its associated valves
  - Controlling EFW (A or B) flow to prevent overfill
  - Local closure of A3 switchgear breakers for P-7B and HPI A.
  - Starting HPI for make-up (long term action)
    - May require local start of pumps depending on fire scenario

- Emergency diesel generator recoveries were not necessary due to the lack of a LOOP event

Previous Procedures vs Zone Specific Procedures

- Previous procedures
  - Combination of EOP/AOP/Pre-Fire plan
  - Opportunistic approach
  - Plant condition determines action
- New procedures
  - Zone-specific fire procedures
  - Tactical approach
  - Reduces impact and probability of spurious operations

Summary of Procedural Guidance

#	Key Action	Previous Procedures	New Procedure
1	Starting EFW P-7A manually and positioning associated valves	The previous procedures discuss this in great detail. Spurious and false indicators are not mentioned which could delay operator response.	Discussion in new procedure includes functional indicators.
2	Controlling EFW (A or B) to prevent overfill	Previous procedures discuss this local or control room action.	Lack of adequate and correct indication is directly discussed in the new procedure which makes this action more likely in the new procedure.
3	Local closing of bus A3 switchgear for P-7B and HPI A (e.g., inverter fires)	This action not explicitly discussed in the normal operating procedures but is discussed in Alternate Shutdown.	The new procedure explicitly addresses locally closing these breakers.
4	Starting HPI Makeup	Discussed in previous procedures. The timing of this action depends on when letdown is isolated.	The new procedure addresses the possibility of starting the HPI pump locally.
5	Isolation of letdown to avoid needing HPI (Makeup) sooner	In both the previous and new procedures, this action is discussed and can be performed in the control room.	In both the previous and new procedures, this action is discussed and can be performed in the control room.
6	Switch to recirculation long-term cooling	In both the previous and new procedures, this action is discussed and can be performed in the control room.	In both the previous and new procedures, this action is discussed and can be performed in the control room.

Simulator Scenario for Zone 99M

- Fire damage chosen to provide HRA information for multiple operator actions
  - Fire model beginning with an A4 switchgear fire
  - Fire propagated throughout zone causing wide-spread cable damage
  - Damage for scenario extends beyond credible fires
- Realistic control room communication challenges
  - Fire brigade leader communication
    - Timelines based on actual fire drill
    - Included need to contact local fire department
  - In plant auxiliary operator used for operator actions
    - Radio and telephone communications used

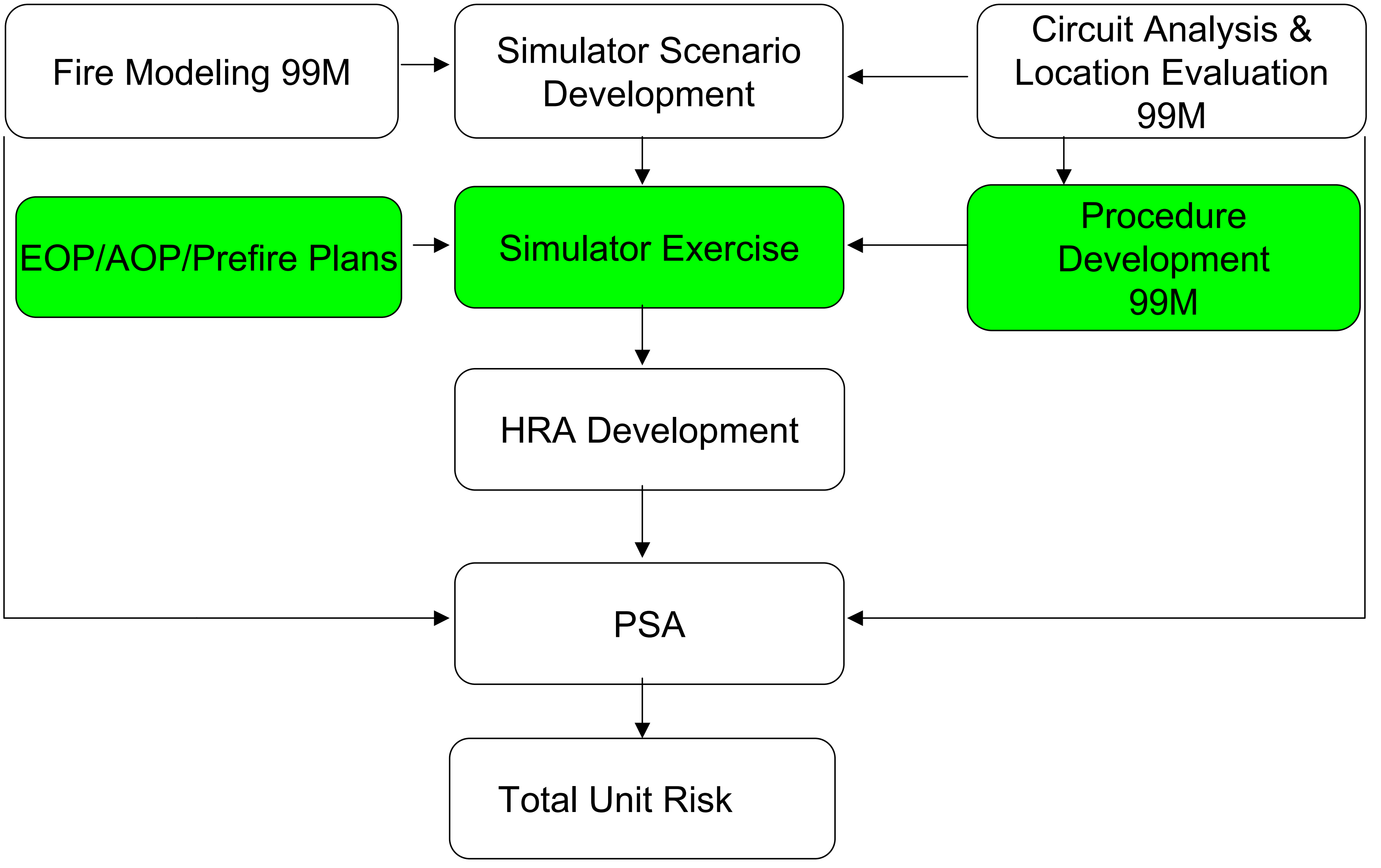
Simulator Scenario for Zone 99M

- Simulator scenario failures included:
  - Direct failures
    - A4 switchgear (4KV)
    - B6 load center (480 VAC)
    - EFW flow control valve power failure
    - HPI aux-lube oil pump power failure
  - Included spurious operations
    - P-7B EFW suction valve closed at T=15
  - Included failed and incorrect indications
    - Multiple panel indications failed (EFW, HPI, Power)
    - Random annunciators spuriously alarmed

Simulator Scenario for Zone 99M

- Three crews ran simulator with previous procedures
- Two crews ran simulator with training on zone specific fire procedure
- One crew with each procedure contained operators in the plant simulating local actions
- Controllers were present in the field to evaluate local manual actions
    - Time to location
    - Potential hazards
    - Communication barriers
- Observers in the simulator to evaluate control room actions
    - Including time to perform in control room actions
    - Procedure usage
    - Work practices due to loss of indications

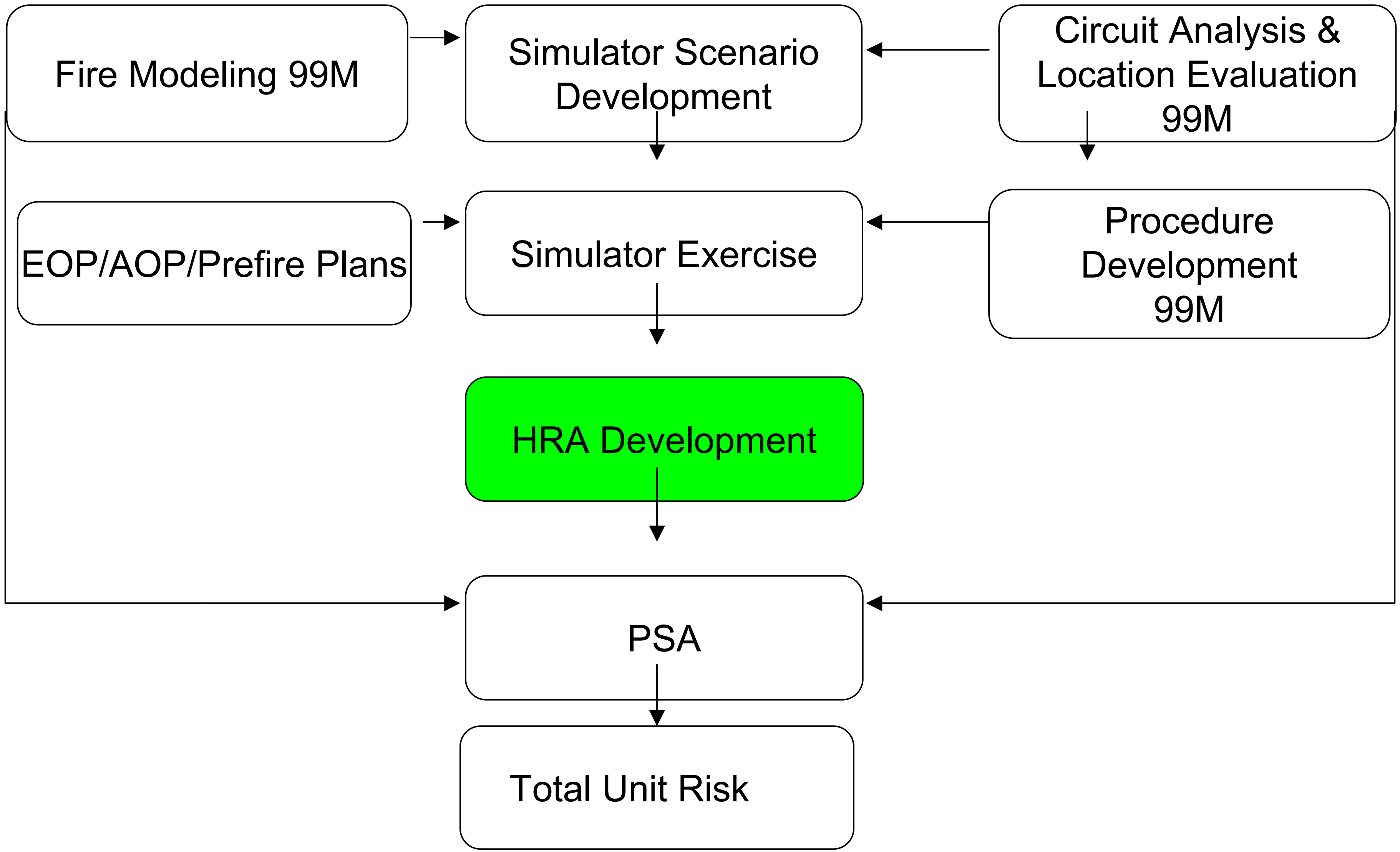
Risk Informed Strategy for Zone 99M



Simulator Scenario Results

- Simulator runs using previous EOP/AOP/ Pre-Fire plan approach and unrehearsed crews (3)
  - EOP approach for plant trip provided adequate initial core cooling
  - Pre-fire plan used to show faulty indications and possible local actions
  - Crews responded appropriately and in a timely manner
  - Plant maintained at a safe stable state
- Simulator runs with crews trained on new tactical procedure approach (2)
  - EOP for plant trip still used until fire confirmed
  - Using new procedures, crews directly implemented local control of core cooling
  - Plant maintained at a safe stable state
- Crew performance using either previous or new procedures met Appendix R requirements for achieving safe shutdown

Risk Informed Strategy for Zone 99M



Manual Actions are Reliable

- HRA methods for quantification demonstrate there is an impact of fire on reliability of human actions
- Previous vs new procedures for shutdown
  - Previous procedures use an opportunistic approach to control, where crews respond to cues and symptoms by selecting EOPs for that condition with the aid of pre-fire plans
  - New procedures assist crew to respond using a more tactical control process
- Use of either approach demonstrated
  - Identifying symptom or cue will generate appropriate response for either procedure
  - Ability to recover from spurious actuations
    - Enhanced in new procedures

Method for Updating HRA Assessments to Account for Fire

- The current Unit 1 model for human recovery actions in internal events PRA is based on a time reliability curve
- HRA accounts for operational context by adjusting parameters such as:
  - Rule-based vs knowledge-based behavior
  - No burden vs burden
  - Other performance influencing factors
- In current assessment, effects of fire are not addressed nor are model parameters available; therefore, a different adjustment method was identified
- EPRI HRA calculator used to assess differences of fire

EPRI Calculator

- Industry sponsored method provides a process for book keeping HRA evaluations
- Addresses HRA requirements in ASME PRA Standard 2002
- Includes several methods for quantification
  – Industry and NRC sponsored
  – Generic data quantitatively differentiate human error probabilities (HEP's) for key characteristics of procedures and man machine interface
- HRA analyst judgment is still required

Evaluation of Fire Impact on Probability

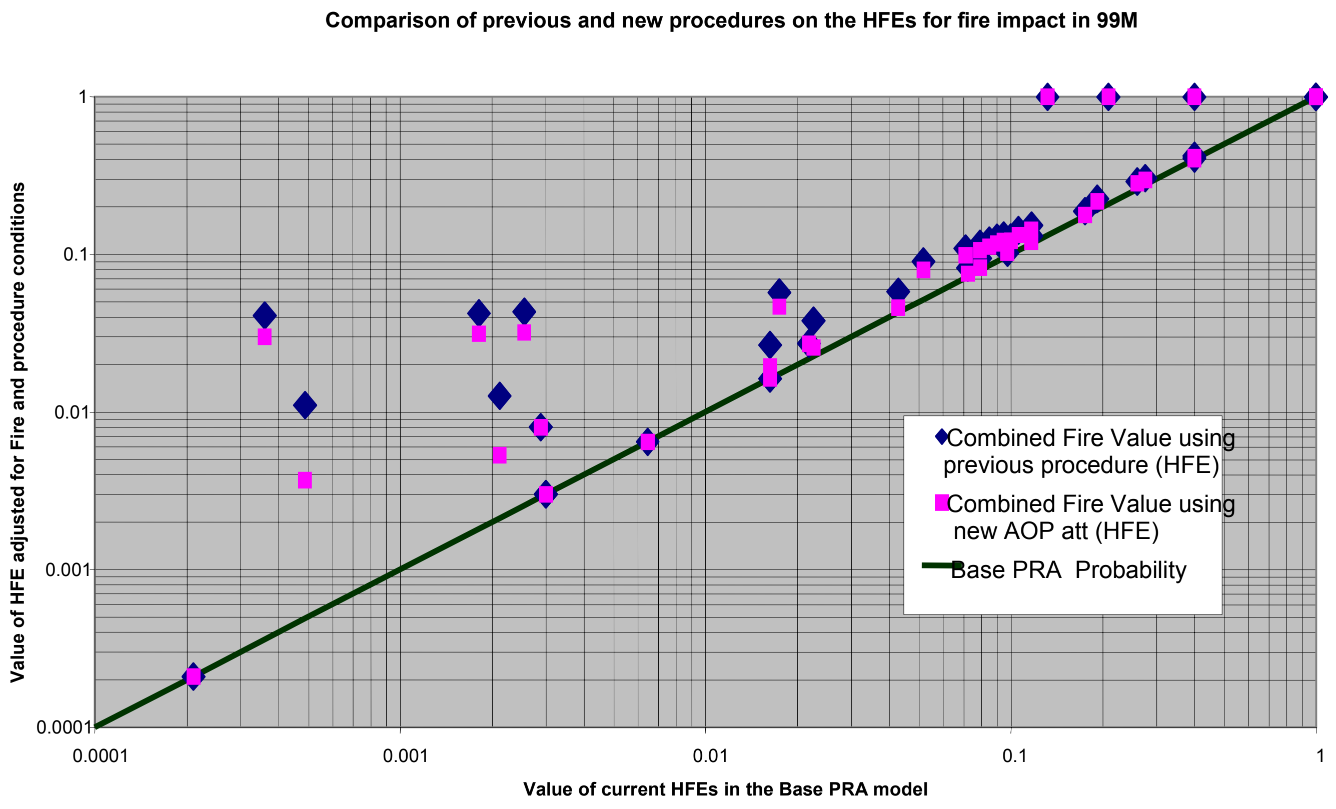
- Seven cognitive assessments on differences in procedures
  - Availability of information
  - Failure of attention
  - Misread/miscommunication data
  - Information misleading
  - Skip a step in procedure
  - Misinterpret instruction
  - Misinterpret decision logic
- Probability of execution also calculated for fires based on inputs in the HRA calculator

Summary of ΔHEP Applications Due to Fire

Case	Event ID	Basic Event Description	ΔP_{cog}	ΔP_{exe}	ΔHEP_{fire}
1	**FIREOLDP**	Actions are carried out within the control room - previous	9.8E-03	7.50E-04	1.1E-02
2	**FIRENEWP**	Actions are carried out within the control room - new	2.6E-03	6.10E-04	3.2E-03
3	**99-MFIRECR**	Realistic fire in 99M decisions in control room with local manual actions	9.8E-03	2.00E-02	3.0E-02
4	**99-MFIRECRE**	Realistic fire in 99M early control room actions	4.7E-03	4.3E-04	5.1E-03
5	**99-MFIRELOCAL**	Local actions taken by field operators	1.5E-02	2.6E-02	4.1E-02
6	**Not Feasible**		1	1	1
7	**No Change**		0	0	0

Comparing HFEs from PRA baseline with HFEs in 99M fire

- Fire in 99M increases human failure event (HFE) for typical feasible actions over initial internal events PRA from zero to a value in range of 3E-3 to 4E-2 for various scenarios and conditions
- If action is not feasible, then HFE assessment is set at 1.0
- Very small difference in impact of previous versus new procedures

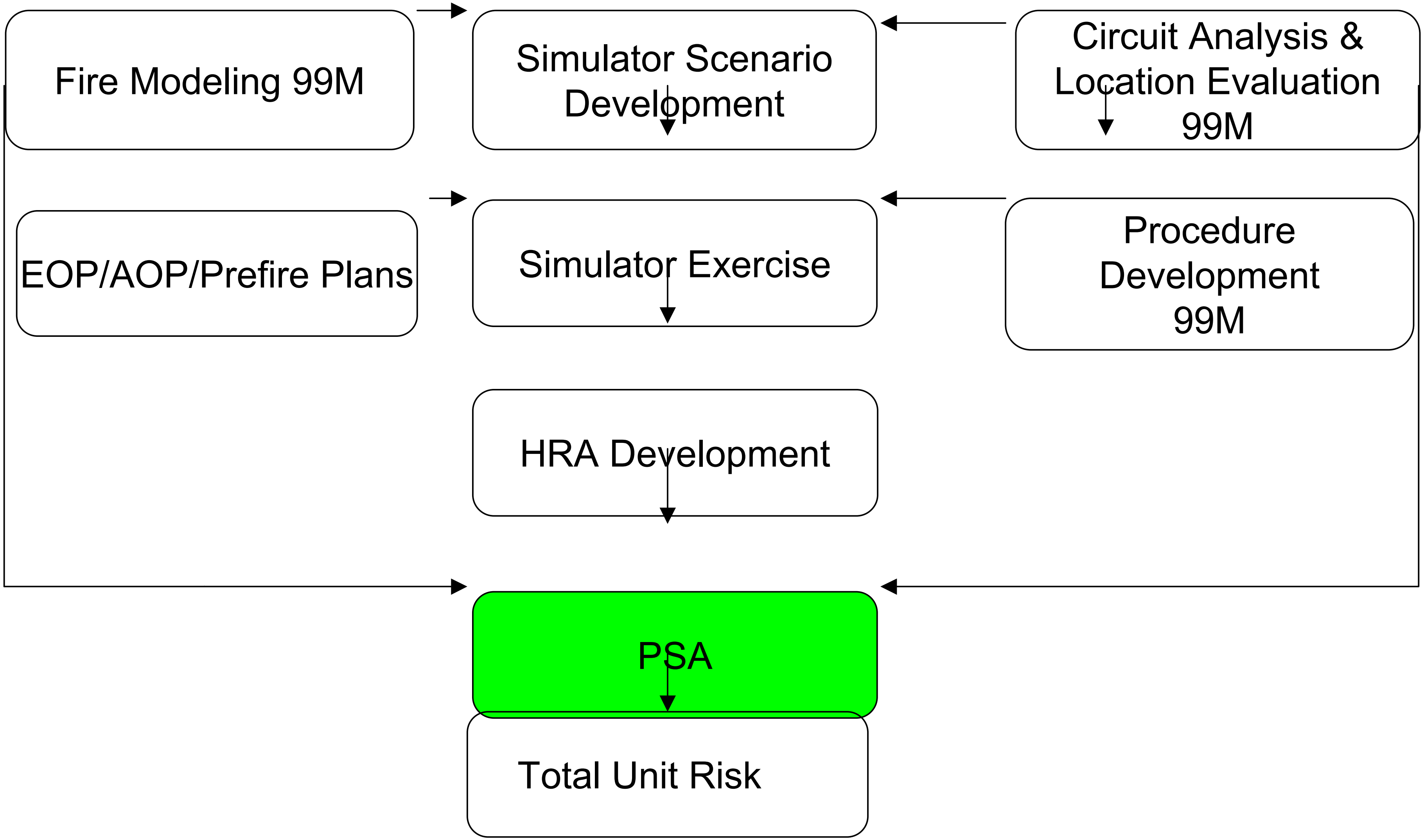


Human Error Probability Comparison

- 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	

Risk Informed Strategy for Zone 99M



CCDP Determination for Zone 99M

- Eight fire scenarios in zone 99M quantified
  - Current Unit 1 PSA model used
- Fire modeling targets used to determine failed components
- Spurious operation probabilities used in high-energy electrical fault scenario 1b
  - All other scenarios conservatively assume the spurious operation will occur
- All components failed together (conservative)
  - Timings only used to disallow spurious operation of components whose control cable would be lost after power loss
- HRA values for the previous and new procedures used to recover the baseline CCDP values for 99M

SDP Process Review

- Created eight fire scenarios
- Used fire modeling/characterization
  - Determined failures for each scenario
- Used simulator exercises and industry experts to determine reliability of necessary operator actions
- Combined interaction into plant specific PSA model
  - Calculated change in risk between the previous and new procedures

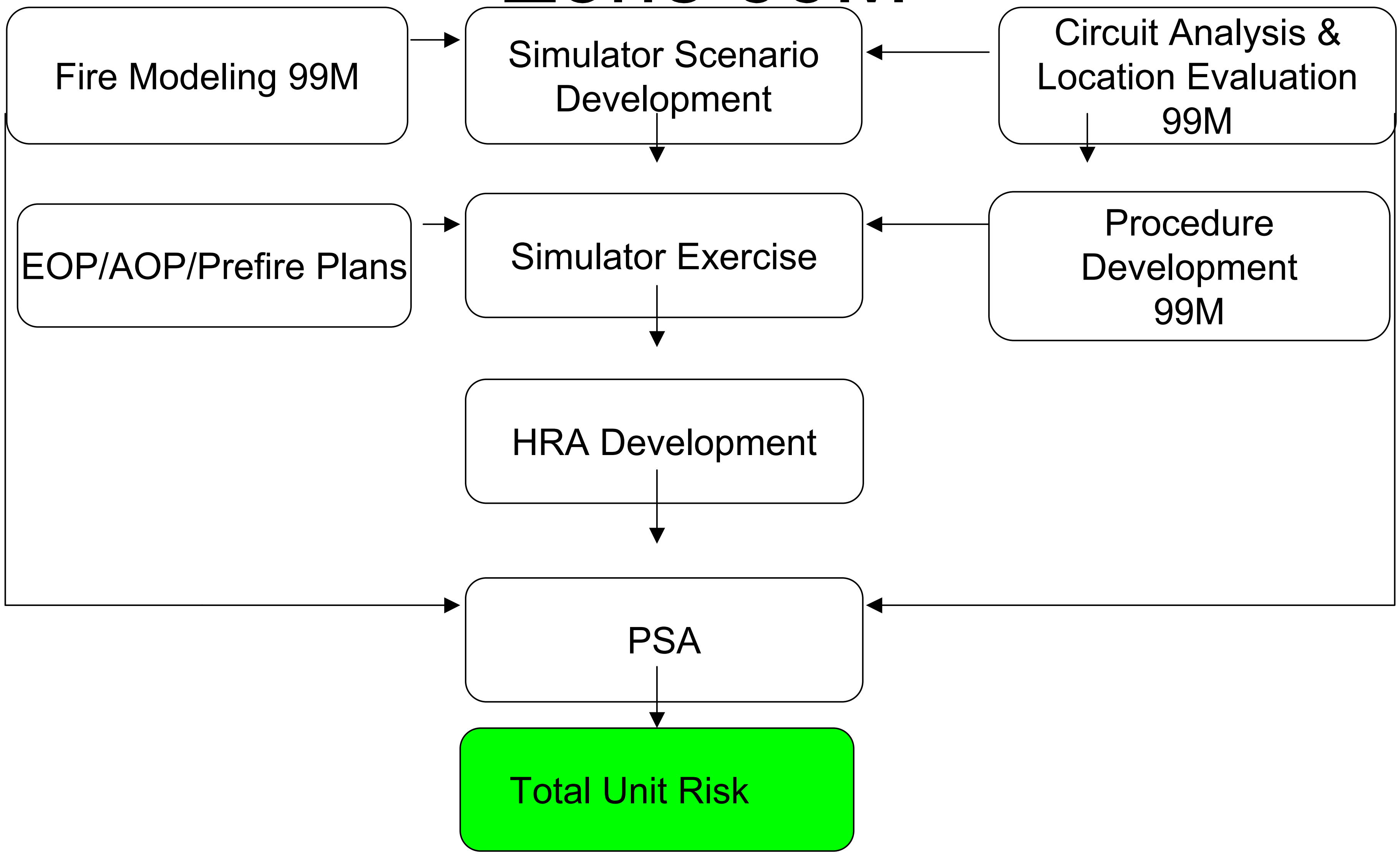
Fire Risk in Zone 99M

Scenario	Source	Fire ignition frequency	CCDP previous	CCDP new	CDF previous	CDF new	Delta CDF
1a	Fire in the A4 switchgear. Nominal value, 100 KW fire	6.62E-05	3.12E-04	2.06E-04	2.06E-08	1.37E-08	6.98E-09
1b	High energy arcing fault in any of the A4 switchgear breaker cubicles.	1.99E-04	1.28E-03	9.01E-04	2.54E-07	1.79E-07	7.55E-08
2	Fire in the B55 MCC. Nominal 100 KW fire. Fires in Inverter Y28 are bounded by this scenario.	2.65E-05	2.78E-04	1.79E-04	7.35E-09	4.74E-09	2.61E-09
3	Fire in the B56 MCC. Nominal 100 KW fire.	2.65E-05	2.78E-04	1.79E-04	7.35E-09	4.74E-09	2.61E-09
4	Fire in the Y22 Inverter. Base case, 100 KW fire. Fires in Y24 and Y25 are bounded by this scenario.	1.32E-05	3.98E-05	3.86E-05	5.27E-10	5.10E-10	1.60E-11
5	Fire in Load Center B6. 100KW nominal HRR.	5.29E-06	3.02E-02	1.88E-02	1.60E-07	9.93E-08	6.07E-08
6a	Transient fire in areas of the room where cable trays are exposed to a floor-based fire. Nominal value of 150KW.	6.48E-05	3.24E-03	2.12E-03	2.10E-07	1.37E-07	7.25E-08
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.	2.60E-07	3.24E-03	2.12E-03	8.41E-10	5.50E-10	2.91E-10
99M					6.61E-07	4.39E-07	**2.21E-07**

Critical Risk Inputs

- Time-phased fire-induced failures are a critical element
  - Realistic assessment of fire progression, failures in 0 - 60 minutes (targets of the high-energy switchgear damage immediate, the rest time-dependent failures are from ensuing cable fire)
- Critical cable insulation used is thermoset
  - 700ºF cable damage temperature
- Operator action probabilities
  - New procedures offer slight HEP improvement over previous procedures
  - Human reliability analysis: CCDP indicates that impact of ΔHEP is measurable but small

Risk Informed Strategy for Zone 99M



Total Unit Risk

- Focus on zones that have delta risk due to the difference in manual actions between two types of procedures
- Qualitative review of zones where manual actions are utilized
  - Alternate shutdown zones screened
  - Zones with automatic suppression screened
    - Agrees with NRC SDP approach – Suppression provides at least one order of magnitude in risk results and provides time for operator actions to be performed
  - Zones with MFW unaffected screened
    - MFW greatly extends time needed for EFW local actions
  - Zones with one complete train of core cooling unaffected screened
    - Control room operation of equipment removes impact of local operator actions
- Similar to NRC results, the following zones remain:
  - 100N
  - 104S

Total Unit Risk

- The assessment of fire risk in 99M was extrapolated to two other Unit 1 fire zones:
  - Each was assessed with walkdown and examination of the potential fire scenarios threatening the other train raceways (e.g., red train raceway in a green train room)
  - Unit 1 A3 4KV switchgear room (100N)
    - Similar to 99M in combustibles and fire sources
    - Considerably less redundant train cable routed through zone
  - Unit 1 electrical equipment room (104S)
    - Lack of high energy switchgear
    - Considerably less redundant train cable routed through zone
- Each zone is bounded by the results of 99M
  - Conservative estimated fire risk (ΔCDF) for this condition
    - Unit 1 < 6.6E-07/yr

Unit 2 Risk

- The four Unit 2 zones identified as risk significant by NRC were qualitatively evaluated
- Each was assessed with walkdown and examination of the potential fire scenarios threatening the other train raceways (e.g., red train raceway in a green train room)
- Conclusion
  – The four Unit 2 zones contain similar characteristics to the Unit 1 zones
    - Two switchgear rooms
    - Two rooms containing MCCs similar to 104S
  – The results from 99M bound these zones

Summary of Risk Assessment

- ANO risk assessment concluded that:
  - Realistic fires will not achieve whole-zone damage as originally assumed in NRC evaluation
  - Realistic fires will result in time-phased damage of cables
  - Manual actions required to achieve safe shutdown for a fire in zone 99M are credible
    - Simulator scenarios validated that operators could achieve safe shutdown
      - Met Appendix R requirements for achieving safe shutdown
- Conclusion
  - Delta CDF
    - Unit 1 < 6.6E-07/yr

OVERALL SUMMARY

Joe Kowalewski
Director, Design Engineering

Overall Summary

- Detailed analysis of zone 99M
  - Credible fires result in time-phased failures without zone-wide damage (700°F damage temperature for thermoset cables)
  - Detailed circuit analysis indicates there is not a loss of offsite power from any fire scenario
  - Simulator scenarios provided realistic data for assessment of operator reliability in the use of previous and new procedures
  - ΔCDF for 99M is 2.2E-07/yr
- Total Unit Risk
  - Two additional zones considered risk significant for Unit 1
  - Risk assessment of zone 99M conservative with respect to other zones
  - Conservative estimate of total unit ΔCDF is < 6.6E-07/yr
- The significance of the use of manual actions to achieve safe shutdown has very low safety significance and should be characterized as <u>GREEN</u>

Overall Summary (cont.)

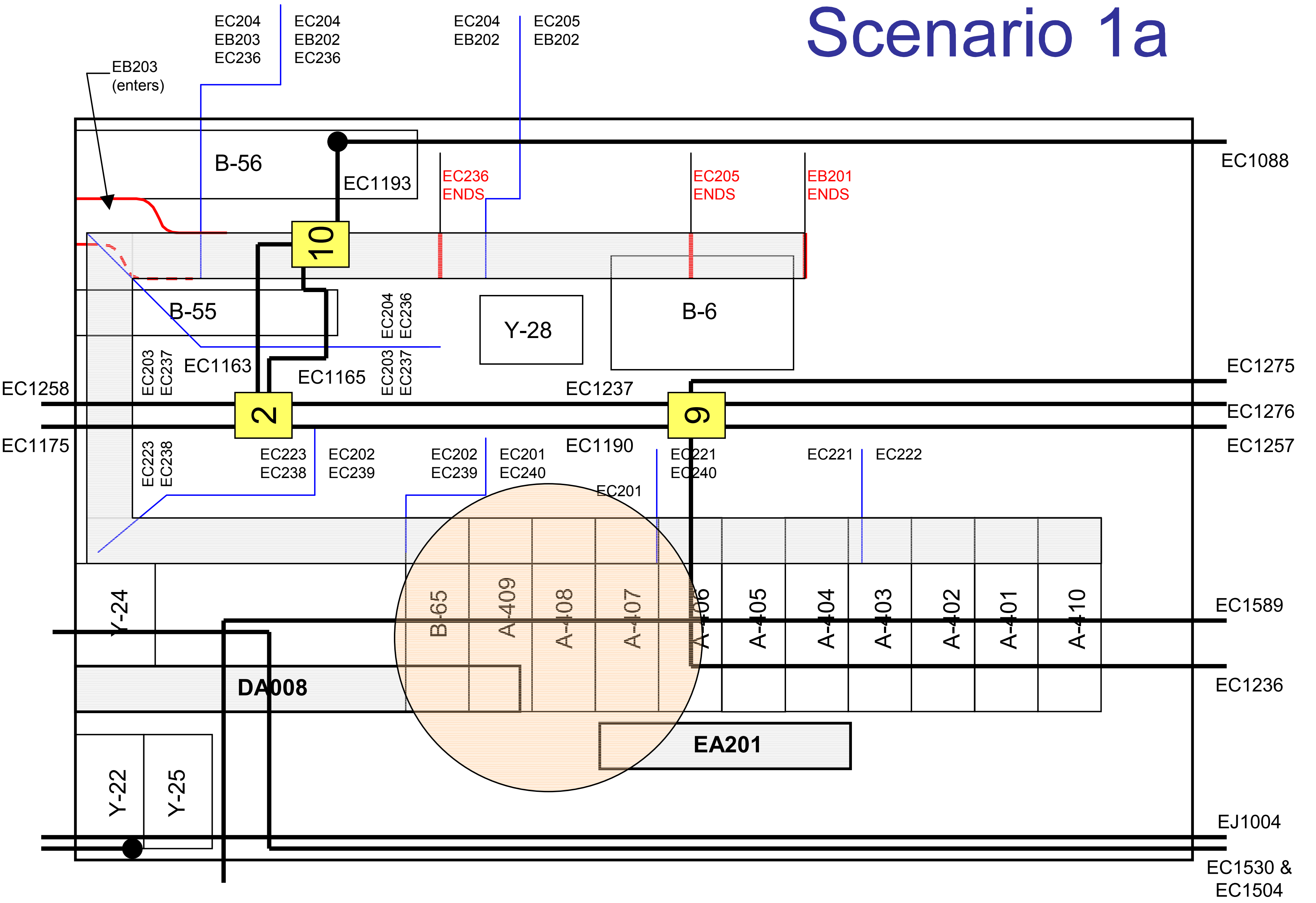
- ANO fire protection program
  - Defense in depth strategy to prevent and mitigate fires
  - Explicit control of combustibles
  - Fire brigade effectiveness
- Primarily rely on barriers or physical separation for equipment required for safe shutdown
  - Fire detection and suppression
  - Limited use of manual actions utilized for Appendix R compliance
- Actions taken to further reduce risk
  - Validated circuit analysis
  - Feasibility evaluation of manual actions (IE 71111.05)
  - New procedures developed to enhance operator response
  - Fire detection reliability improved
- ANO can successfully achieve safe shutdown in the event of a fire in any zone

CLOSING REMARKS

Craig Anderson
VP, ANO

ADDITIONAL INFORMATION

Scenario 1a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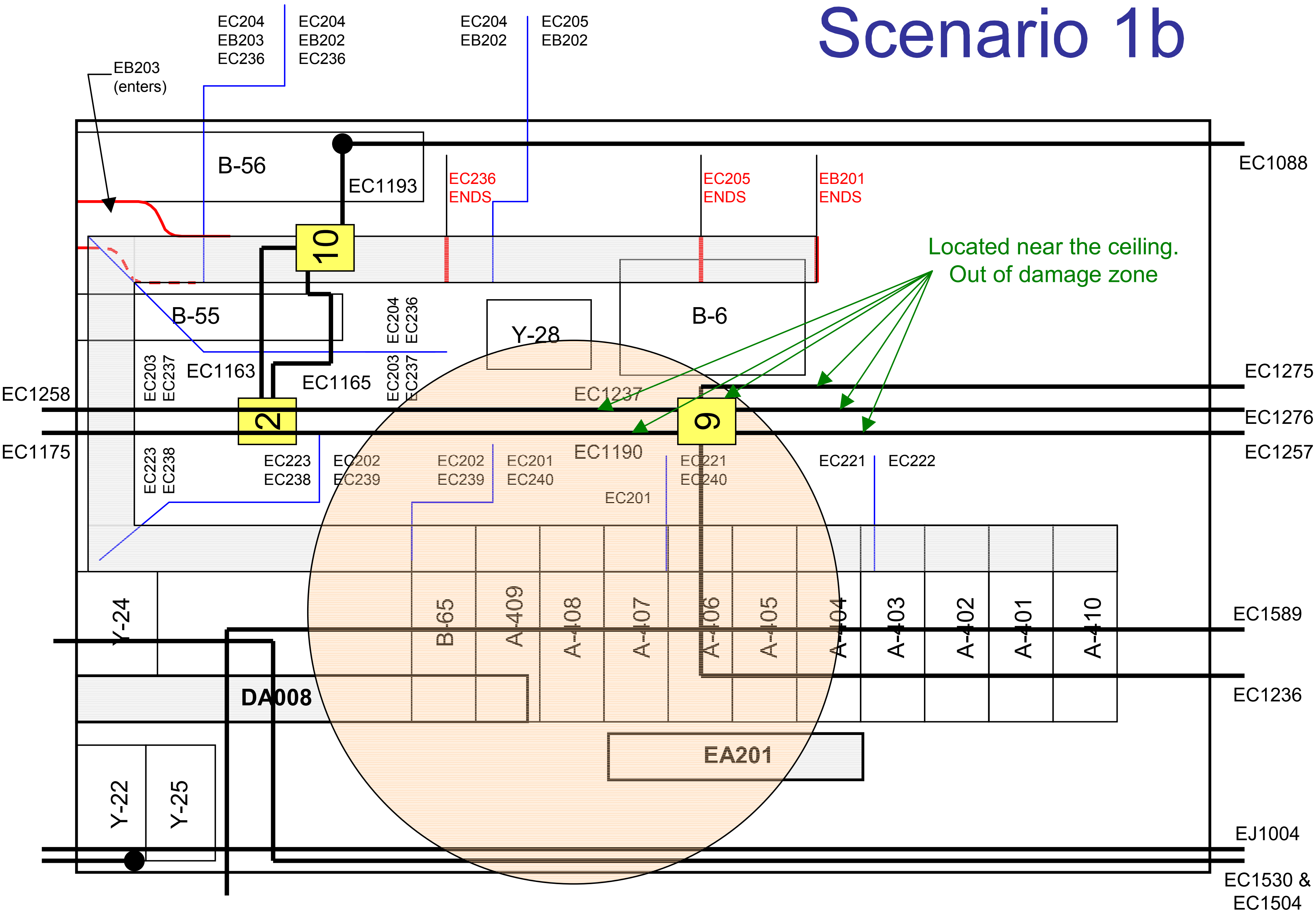
Scenario 1b



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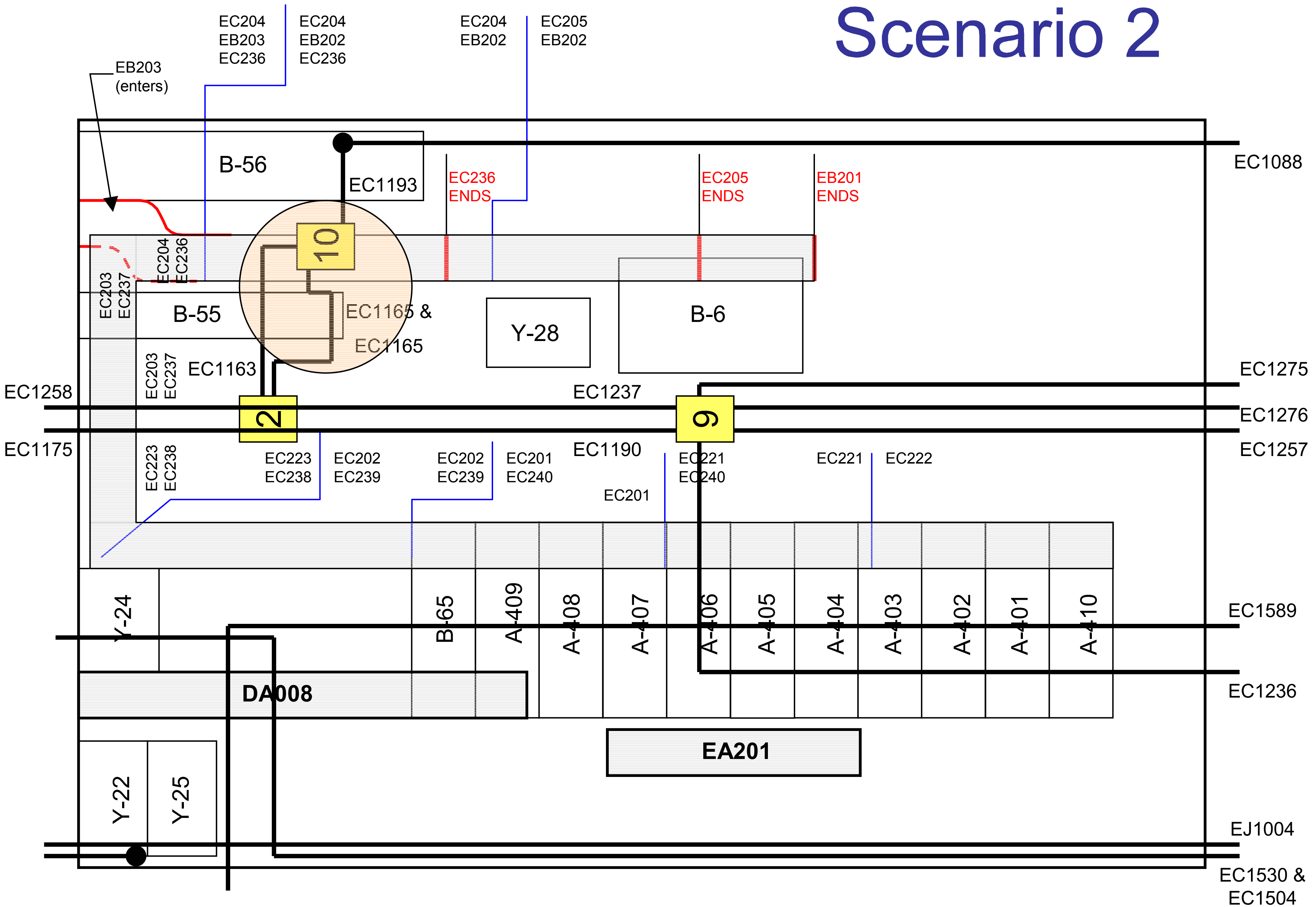
Scenario 2



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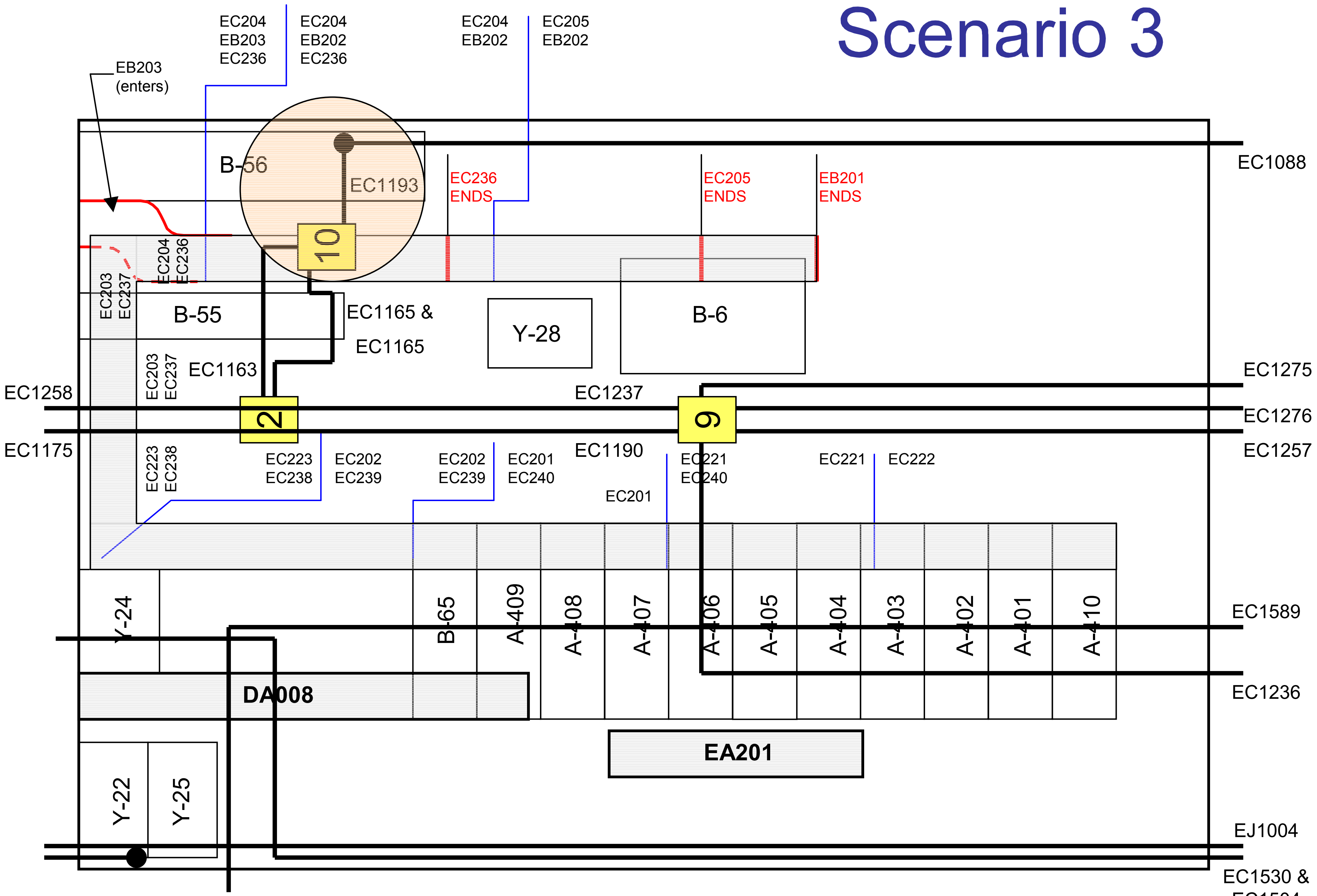
Scenario 3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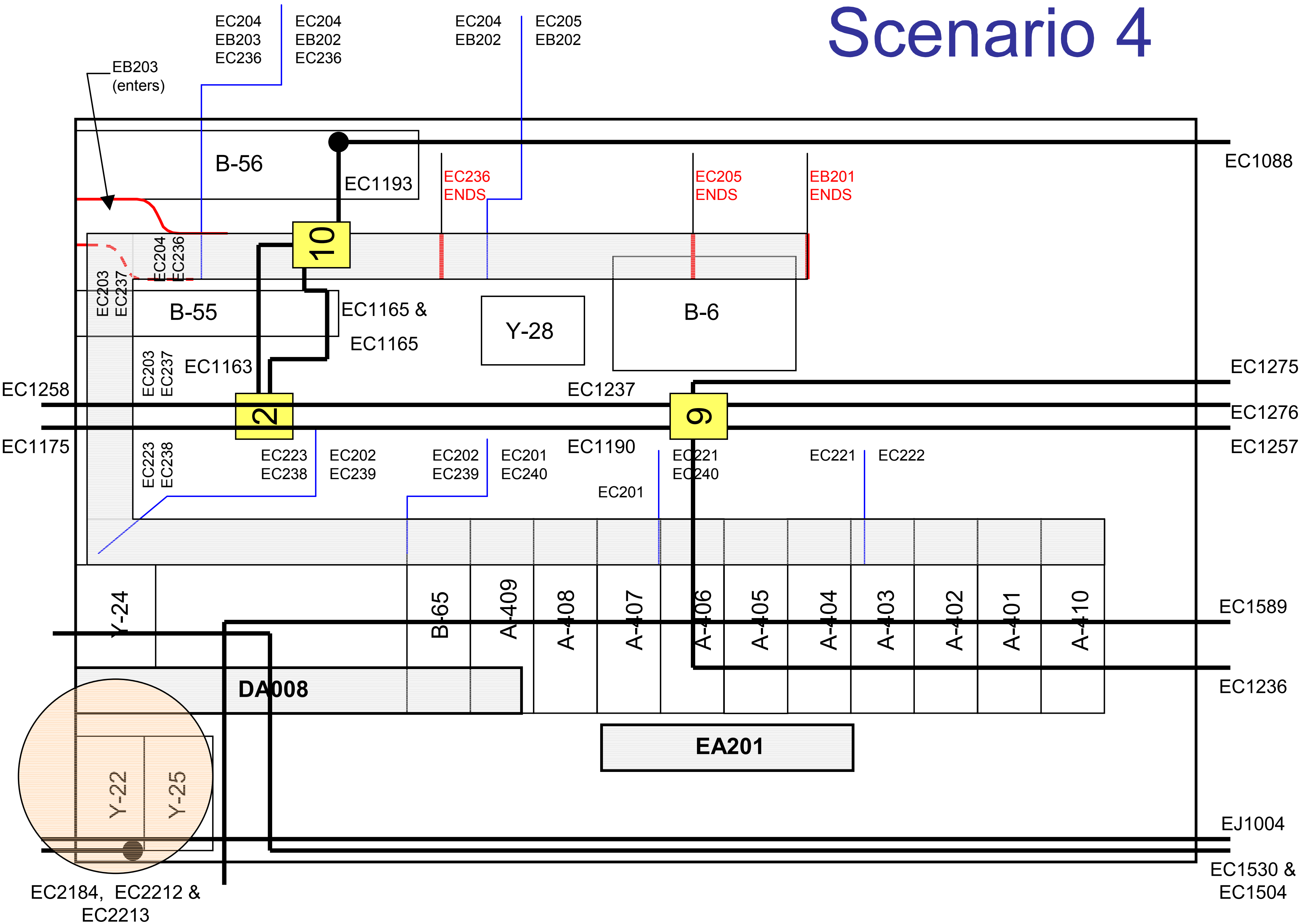
Scenario 4



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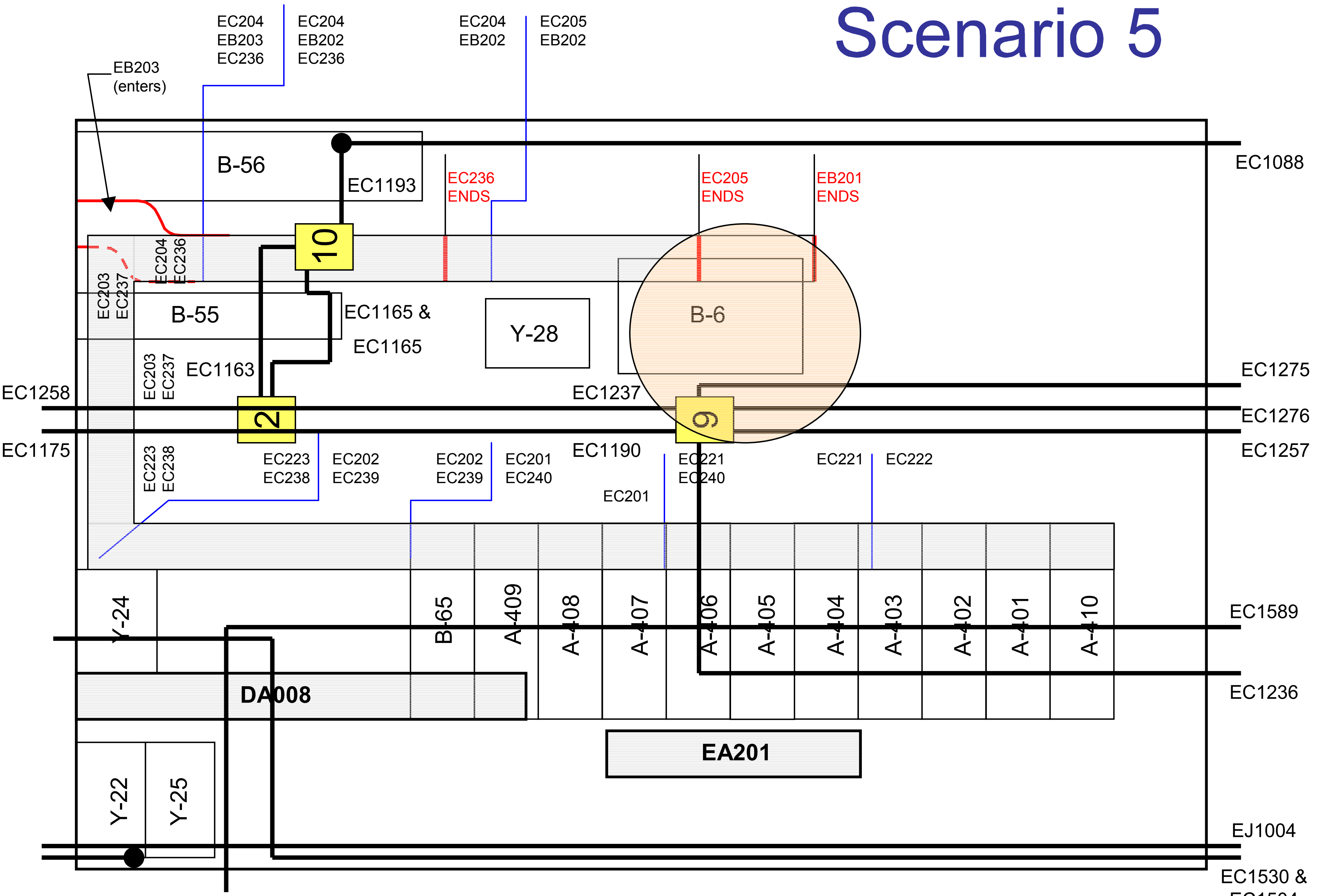
Scenario 5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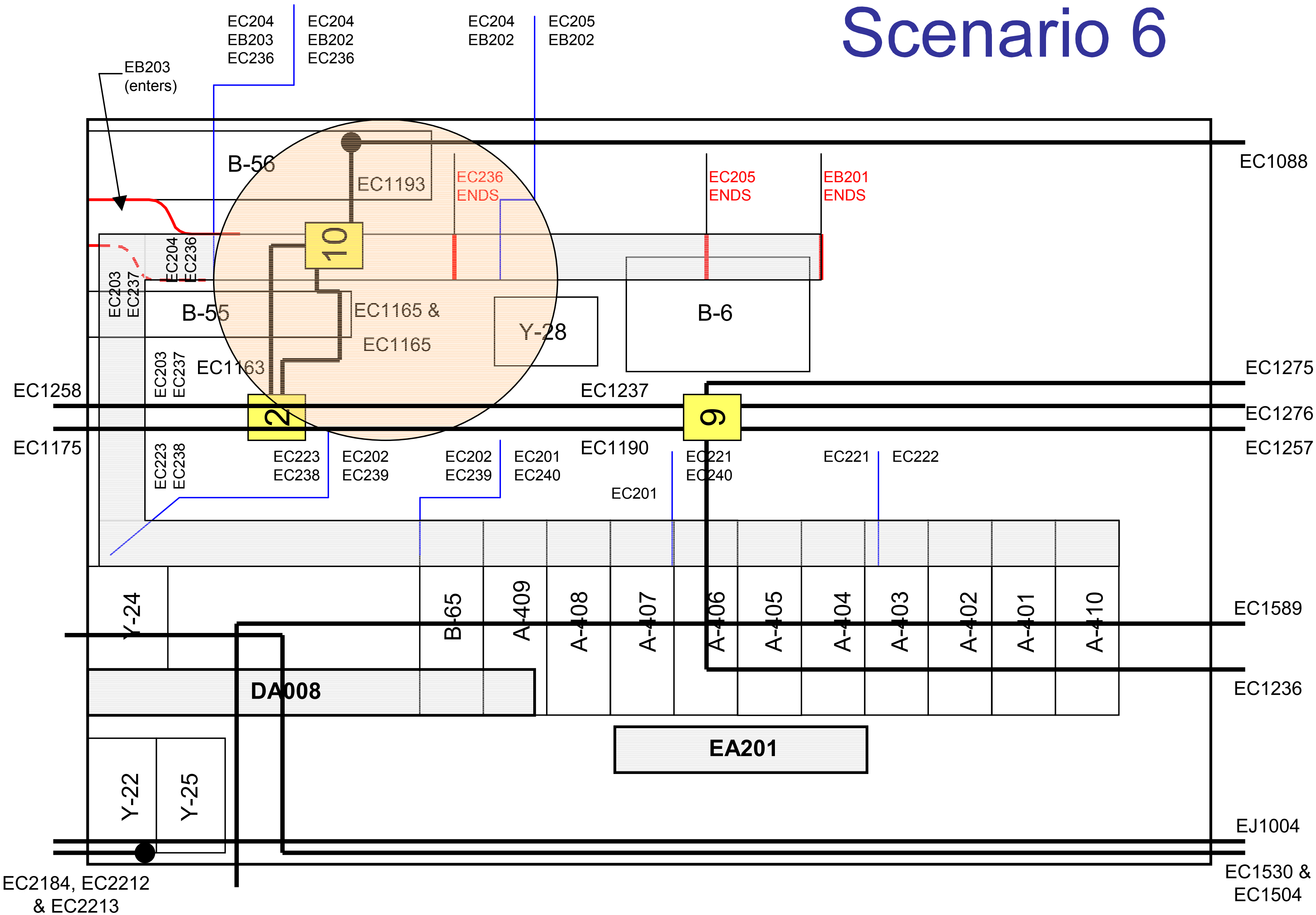
Scenario 6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A-402
A-401
A-410
EC1589
EC1236
DA008
EA201
Y-22
Y-25
EJ1004
EC1530 &
EC1504
EC2184, EC2212
& EC2213

INDIVIDUAL QUALIFICATIONS

SAIC SCIENCE APPLICATIONS INTERNATIONAL CORP.

BIJAN NAJAFI, P.E.
MANAGER, FIRE PROTECTION SECTION

EDUCATION:

University of Washington:	M.S., Nuclear Engineering, 1979
Shiraz University:	B.S., Electrical Engineering, 1976

Registered Professional Mechanical Engineer, State of California

SUMMARY OF CURRENT POSITION:

Mr. Najafi is the Manager of the Fire Protection Section at SAIC responsible for overseeing a program that includes domestic and international nuclear utilities, DOE facilities and commercial/industrial facilities.

EXPERIENCE:

Mr. Najafi is a nuclear engineer with over 23 years of experience, emphasizing Reliability, Risk Assessment, Fire Protection and Systems Analysis. His background includes development of methods for risk assessment and fire protection as well as application of these techniques in solving plant-specific problems.

Mr. Najafi is the SAIC Manager for Electric Power Research Institute (EPRI) fire risk analysis and fire protection program. Over the past decade he has been instrumental in development of the EPRI fire risk technology currently in use in the U.S. nuclear power industry. This technology has also been used internationally in Europe and parts of Asia and South America. Mr. Najafi has conducted training courses in U.S. and Europe on Fire Technology, most recently a series of Fire Modeling courses for nuclear power plant fire protection engineers.

Mr. Najafi is an active member of the fire protection community. His contributions include:

- Principal member of the National Fire Protection Association (NFPA) Technical Committee on Fire Protection for Nuclear Facilities (801/805)
- Principal member of the American Nuclear Society's committee for the development of the Fire PRA Standards
- Participating member of various taskforces at Nuclear Energy Institute including the circuit failures issues taskforce in the development of the NEI-00-01, "Guidance for Post-fire Safe Shutdown Analysis." Invited panelist at the NRC-Industry fire-induced circuit failures workshop on February 19, 2003.
- Member of the NRC-Industry team for the revision of fire protection Significance Determination Process (SDP)
- Member of the NRC's "International Collaborative Project to Evaluate Fire Models for Nuclear Power Plant Applications."
- Member of the Society of Fire Protection Engineers (SFPE) task Groups for development of the, "SFPE Engineering Guide to Performance-Based Fire Protection," completed in 2000 and "Risk Assessment Guidelines," in progress.

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 a computer-based probabilistic framework to integrate the vast body of knowledge that exists regarding LMFBR core disruptive accidents and their inherent uncertainty. The methodology not only estimates the uncertainties, but also can display the nature and extent to which the state of knowledge (or lack of knowledge) contributes to them. The potential application of the methodology to the PWR steam explosion events in a large, dry containment was investigated. The results of this study were published in the Nuclear Science and Engineering Journal.

Over the period 1982-1984, Mr. Najafi was the principal investigator of several system safety studies on the Clinch River Breeder Reactor Plant (LMFBR) that were presented to the Advisory Committee on reactor safeguards as part of a technical assistance effort for the NRC staff. This effort covered a variety of limited-scope studies for both systems and consequence evaluations, including radioactivity release frequencies, unprotected reactivity insertion accidents, reliability analysis of the Decay Heat Removal System, and Core Disruptive Accident Energetics. He was also involved in review of the CRBRP Reliability Assurance program for the NRC to ensure that the LWR licensing requirements and associated Regulatory Guides that are applicable to LMFBR's are being applied to CRBRP.

During 1980-1981, Mr. Najafi acted as the task manager for the SAIC team to perform the probabilistic systems analysis part of the probabilistic risk analysis study for the SNR-300 (LMFBR) Nuclear Power Station in Kalkar, West Germany. The objective of this two-year project was to provide safety-oriented information to a special commission of the German Parliament that was considering appropriate energy policies for West Germany, including continuation of the SNR-300 construction.

Mr. Najafi has been one of the principal participants in the risk reduction program conducted by the Nuclear Safety Analysis Center to investigate the PRA methodology for estimating incremental changes in plant reliability and risk due to modifications. The methodology was validated using VEPCo's Surry (PWR-W, with several shared systems) plant by estimating the incremental change in system reliability and plant safety as the result of the modification in system design and operation and specifications implemented since the original WASH-1400 study. He was also the Task Manager and conducted the probabilistic analysis part of the accident evaluation chapter for the Seabrook Nuclear Power Plant (PWR-W) Environmental Report. This study was prepared for Yankee Atomic Electric Company in support of the Seabrook Station licensing.

Mr. Najafi was the principal investigator of a Heat Rate Improvement Study performed by SAIC. A steady-state model of the Morgantown plant using the PEPSE computer code was developed covering the boiler, turbine and balance of plant systems. A limited sensitivity analysis was performed to investigate the sensitivity of plant heat rate to different plant operational conditions. The long-term objective of this project was to provide optimum operating strategies to be used as part of a plant performance monitoring system.

On several occasions Mr. Najafi has served as a lecturer for the reliability and safety analysis courses conducted by Argonne National Laboratories on the application of probabilistic techniques for accident sequence quantification in nuclear power plants.

Joining SAIC in 1979, Mr. Najafi participated in the system model development as part of the Seismic Safety Margin Research Program (SSMRP) for the Lawrence Livermore National Laboratories, where he developed the models for Emergency Core Cooling System and Residual Heat Removal System for the Zion Nuclear Power Station Unit 1 (PWR-W). Later he developed a fault tree model for the auxiliary feedwater system for San Onofre Nuclear Generating Station Unit 1 (PWR-W) to predict the systems reliability under seismic loading.

SELECTED PUBLICATIONS:

1. "Fire Modeling Guide for Nuclear Power Plant Applications," EPRI 1002981, August 2002.
2. "Fire Events Database for U.S. Nuclear Power Plants: Fire Initiation and Trends", EPRI 1003111, December 2001.
3. "A Pilot Plant Evaluation Using NFPA-805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants", EPRI 1001442, May 2001
4. "NFPA 805: Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants", National Fire Protection Association, 2001 Edition (Contributing Author)
5. "Fire Barrier Penetration Seal Handbook," EPRI 100196, July 2000
6. "SFPE Engineering Guide to Performance-Based Fire Protection", Society of Fire Protection Engineers, First Edition 2000 (Contributing Author)
7. "Planning for Risk-Informed/Performance-Based Fire Protection at Nuclear Power Plants", EPRI TR-108799, December 1997
8. "Reducing Operations and Maintenance Costs of Nuclear Power Plant Fire Protection Programs", EPRI TR-107337, December 1996
9. "Methods for Evaluating Cable Wrap Fire Barrier Performance", EPRI TR-106714, August 1996
10. "Fire Ignition Frequency Model at Shutdown for U.S. Nuclear Power Plants", EPRI TR-105929, December 1995
11. "Fire Probabilistic Risk Assessment Implementation Guide", EPRI TR-105928, December 1995 (and Supplement EPRI SU-105928)
12. "Fire-Induce Vulnerability Evaluation (FIVE) Software", EPRI AP-100530, February 1994
13. "Automatic and Manual Suppression Reliability Data for Nuclear Power Plant Fire Risk Analyses", NSAC-179L, February 1994
14. "Fire Risk Analysis Code, FRANC", EPRI AP-103733, January 1994
15. "Fire-Induced Vulnerability Evaluation (FIVE)", EPRI TR-100370, May 1992 (Contributing Author)
16. "Fire Events Database for U.S. Nuclear Power Plants", NSAC-178L, December 1991
17. "Reference Plant Accident Sequence Likelihood Characterization: Peach Bottom, Unit 2," NUREG/CR-4550, Volume 3. (With Alan Kolaczkowski, et al.)
18. "An Assessment of Steam Explosions Induced Containment Failure," NUREG/CR-5030, February 1989 and Nuclear Science and Engineering, December 1987.
19. "On the Probabilistic Aspects of α-Mode Containment Failure," T.G. Theofanous, B. Najafi and E. Rumble, Nuclear Science and Engineering, November 1985.
20. "Incorporation of Phenomenological Uncertainties in Safety Analysis - Application to LMFBR Core Disruptive Accident Energetics," Proceedings of ANS/ENS International Topical Meeting on Probabilistic Safety Methods and Applications, Vol. 1, San Francisco, CA, February 1983.
21. "SSMRP, Phase I, Systems Analysis," NUREG/CR-2015, November 1981 (with J.E. Wells, et al.).

Data Systems & Solutions

G. WILLIAM HANNAMAN, PHD

EDUCATION:

PhD, Nuclear Engineering, Iowa State University, 1974

MS, Nuclear Engineering, Iowa State University, 1971

BS, Electrical Engineering, Iowa State University, 1965

WORK SUMMARY:

Dr. Hannaman is a Professional Engineer with over 25 years of progressive consulting experience in solving electrical and nuclear engineering problems for a wide range of nuclear reactor types, process plants and industrial facilities. Applied educational background and experience to resolve technical issues using reliability and probabilistic risk assessment (PRA) techniques during the design process and on operating plants. Developed and applied human reliability assessment (HRA) methods to consider the impact of operator interactions before and during accident conditions. Supporting elements include data collection from training simulators, database development and integrating the results into risk and reliability studies to identify management priorities for enhanced design, operation, and maintenance.

PROFESSIONAL EXPERIENCE:

1999 to present, Senior Staff Engineer, Data Systems & Solutions
1988 to 1999 Senior Staff Engineer, Science Applications International Corporation

Recent Projects

- Support for EPRI projects in the following areas:
  - Development of simplified trip monitor for use in generation risk modeling of nuclear power plants.
  - Support Probabilistic Risk Assessment Evaluation of Spent Fuel Dry Storage Bolted Cask Designs in the area of initiating events, HRA and data evaluations.
  - Write guideline for efficiently developing derate and trip monitors for use by control room operators.
  - Support development of a procedure for addressing HRA in fire PSAs
  - Upgraded Monte Carlo Simulation software (STEIN) for evaluating the impact of NDE measures on structural integrity
  - Developed template for performing Human Reliability Analysis – lesson plans
  - Support project on methods for evaluation of organizational factors
- Independent safety reviewer for CANDU plant PSA in Romania. - Peer review of PSA modeling results to recommend changes and upgrades. Also supported HRA training and applications.
- Developed uncertainty analysis tools for predicting the quality of glass/ nuclear waste mixtures for DOE/Bechtel.

Steam Generator Assessment Software development

- Evaluate primary safety valve reliability under severe accident conditions given a leaking SG tube.
- Compare EDF COMPRIS software code with STEIN to identify areas for enhancement in addressing PWSCC through the use non-destructive (NDE) test measures.
- Product manager for establishing EPRI web site for SG SGDSM for maintaining and updating a quality assured (10CFR50) electronic database containing data from tests on pulled SG tubes to

support burst and leak rate correlations. The secure web site developed under an ISO-9000 and 10CFR50 approved quality program supports data searches.

- Product manager for development of the STEIN Monte Carlo code for use in evaluating Steam Generator ODSCC NDE results to predict operational assessment and condition monitoring criteria.
- Developed methodology using Monte Carlo Simulation of uncertainties for assessing margin between an allowed I^{131} dose and a predicted accidental release from degraded steam generator tubes.

Human Reliability Assessments

- Planned and documented human reliability assessments (HRAs) for four utilities as part of their IPEs.
- Developed and delivered a weeklong training course on HRA to Eletronuclear in Brazil.
- Supported update of VC Summer IPEEE fires assessment as HRA task leader under SAIC and VCS quality assurance programs. Evaluated risk of using fire emergency procedures for the current control room configuration. HRA methodology used NUREG/CR 4772, &1278 and EPRI-TR-100259.
- Contributor to development of ASME PSA standard HRA and data sections.
- Instructor on the subject of human reliability for Argonne National Labs Inter-regional Training Course on Prevention and Management of Accidents at Nuclear Power Plants
- Managed 3-year project to extract data from events to enhance human reliability for activities during less than full power operation. Reviewed the operator event data collection programs, updated the Systematic Human Action Reliability Procedure (SHARP), presented examples and information at EPRI's human reliability assessment workshop, and applied SHARP1 on specific accident sequences (e.g., Interfacing System Loss of Coolant Accidents).
- Developed procedures, guidelines and project instructions for performing HRA in two PRAs.
- Supported use of control room training simulators in HRA studies for six utilities including Hope Creek and Laguna Verde.

PRA and Risk Informed Applications

- For Entergy Operations, assisting in update of PRAs for ANO-2 (accident sequence overview), and Waterford nuclear power plants (ISLOCA and ATWS support).
- Applied time dependent integration of system recovery assumptions and human reliability models with thermal hydraulic transient output to produce estimates of large early release frequencies in severe accidents for use in evaluating the risk of operating steam generators with degraded tubes.
- Supported Entergy (ANO2) and SCE (SONGS) in evaluating human reliability during severe accidents to support risk informed evaluation of steam generator tube integrity including review of SAMGs, EOPs, plant interfaces, and simulator training. Presentations on results were given to the NRC.
- Performed multi-compartment fire risk analysis in support of the IPEEE at Quad Cities.
- For CEGA contributed to guidelines for PRA application during the NPR-MHTGR design process. Provide mini PRA study for the Environmental Impact Statement for the NPR-MHTGR.

Risk management

- Supported development of methodology for blending risk-informed PSA with deterministic rules to demonstrate compliance with NRC's regulations governing steam generator operation.
- Developed qualitative risk assessment methodology and delivered training course on qualitative safety assessments including consideration of HRA for non-reactor facilities as part of a Sandia National Labs project to comply with DOE orders 5480.23, 5481.1B, and standards 1027-92 and 3009-94.
- Applied methodology on two facilities (Rocket launch and Accelerator). Results support safety documentation suitable for a facility safety analysis report in a risk-based format.
- For DOE used PRA and HRA methods to support reviews of DOE reactor projects and facility operations.

Reliability Database development

- Establish a reliability and safety database for use during the MHTGR design process.
- Developed data based mechanical reliability models for safety relief valves using test demands and flow conditions to improve risk assessment results.

Ram analysis

- Supported the MHTGR conceptual design through incorporation of applicable operational experience, development of technical position papers to demonstrate that lessons learned from previous operating experience were considered in the advanced design, and updated safety, availability, and plant capacity factor reports working with Stone and Webster Availability Assessment team. This involved building reliability block diagrams for various systems to evaluate reliability and risk.

Oversight projects

- Served as secretary on senior review committee to evaluate selection criteria for the NPR-MHTGR containment.
- Project manager for independent reviews of PSA/HRA and human factors for Union Fensoa on a Spanish Reactor to identify cost effective risk reduction upgrades for control room interface
- Review of a spent fuel processing design for a DOE site.
- Performed review of human reliability assessments in the IPEs,
- Performed independent safety reviews of safety analysis reports and risk assessments including analysis of spray leaks during tank transfer operations, and evaluation of two different pump system operating lifetimes for Westinghouse Hanford using FMECAs, fault trees, aging models and data evaluations.
- Performed independent review of INEEL's ISLOCA methodology.

1981 to 1988, Senior Executive Engineer, NUS Corporation

- Principle Investigator for EPRI projects included development of a human reliability analysis framework, (SHARP), human cognitive reliability (HCR) models, and international HRA benchmark projects.
- Project leader for integration of HRA models to support simulator training, and model verification studies involving collection of data at control room simulators (e.g., for boiling water reactors (BWR's) at ComEd, PP&L, and PE). Supported use of simulator data gathering for verification of BWR EOPs.
- Technology transfer of HRA/PRA methods to clients performing in US and internationally (e.g., EdF).
- Transferred technology via: (1) seminars, (2) reviews of PRAs and HRAs, (3) HRA task definition and supervision of analysts and (4) guidebook development such as PRA procedures guide and HRA guidelines for specific projects (5) performing benchmark comparisons, (6) performing analysis, (7) reviewing work, (8) planning risk related projects, and (9) recommending programs.
- Reviewed use of the newly designed symptom based procedures in response to steam generator tube rupture and small break LOCAs to identify key operator actions.
- Probabilistic risk accident analysis of fires for the Limerick BWR.
- Detailed safety reviews of design concepts such as the advanced modular gas turbine reactor.

1974 to 1981 Staff Engineer, General Atomics

- Performed probabilistic safety analysis, reliability and availability assessments and evaluations on all of GAs operating and proposed plant designs.
- Developed and operated a computerized data base system of component and system reliability measures to analyze Fort St. Vrain availability experience as a way of improving new designs, including the Gas Turbine-HTGR, steamer, fusion designs and others.
- Lead engineer for Chemical and Process System Analysis Group on a 6-man-year effort to collect data and develop reliability evaluation methods including reliability block diagrams for process system hazard analysis reliability allocations, reliability predictions, availability, and maintainability quantification.
- Provided training seminars on probabilistic risk assessment for PRA practitioner training and for shift technical advisors.
- Performed system reliability analysis to support qualification of reactor protection, control, heat removal, main power systems, circulators and support systems for the large HTGR.
- Team member and key author of the PRA study known as the Accident Initiation Progression Analysis.

- Established and maintained the component and system reliability data bank supporting the quantification of event- tree/fault-tree scenario frequencies and uncertainties.
- Developed and applied probabilistic operator models and common-cause failure models.

1970 to 1974, Graduate Assistant and Senior Reactor Operator, Iowa State University

- Obtained licenses for reactor operator and senior reactor operator through the NRC on a university training reactor, with over 100 startups and shutdowns.
- Taught lab courses and helped prepare and present training course for Duane Arnold Energy Center operators in support of NUS training.

1965 to 1970, Supervisor, Westinghouse Electric Corporation Apparatus Repair Division

- Planned repairs and directed maintenance crews on chemical, utility and industrial sites and in repair plants for over 10,000 unique power system equipment failures.
- Designed and implemented an I&C temperature protection system for large electrical motors, and design of a transformer oil storage and transfer system.
- Developed procedures, criteria, and equipment for testing, welding, and evaluating insulation and mechanical structures for serviceability and, if needed on the basis of predicted failures, applied methods for repairing, balancing and testing electrical and mechanical apparatus including electric motors, breakers, controls, transformers, generators, turbines compressors, magnets etc.
- While in Westinghouse's Graduate Student Program performed rotating assignments in manufacturing facilities for transformers and apparatus repair.

COMPUTER PROFICIENCY:

Language/Tools: Microsoft Office Software, Math software, Monte Carlo Simulation, CAFTA

Hardware Systems: PC, and Mac

Operating Systems: Windows 95, 2000, XT; OS8, and DOS

MISCELLANEOUS

Professional Associations and Memberships:

State of California - Professional Nuclear Engineering Registration NU 1948 Since 1982

Member American Nuclear Society -

San Diego Section chairman 1979
San Diego Section executive committee, various years
Technical program chairman for Embedded topical meeting on Advanced Nuclear Installation Safety, 2000,
Assistant Technical Program Chairman for Risk Management -Expanding Horizons 1992.
Human Factors Division, Executive Committee, 1987.
Safety Division Program Committee 2000.
Organized and chaired numerous technical sessions for ANS.
Paper reviewer for Nuclear Technology

Member of Institute of Electrical and Electronics Engineers

Corresponding member of the Nuclear Engineering Subcommittee SC-5 on human factors and reliability responsible for standards on reliability methods. 2000 -2003
SC-5 Committee member on Reliability 1976 to 1980,
SC-7 Committee member on Human Performance 1984-1986.
Organized and chaired technical sessions at an IEEE meeting

Society for Risk Analysis

Executive committee of the Southern California Chapter in 1989.
Organized and chaired technical session at PSAM II

Patents, Selected Publications, and Awards:

- Elected to Sigma XI, the research honor society in 1973
- Elected to National Academy of Sciences 6-member panel on cooperation with USSR on reactor safety to identify needs and means for enhancing reactor safety. 1987
- Elected to Strathmore's Who's Who 1996-03
- Outstanding technical paper awards in ANS Meetings (e.g., ANS Midwest student conference 1974 and ANS summer meeting Human Factors Division 85, 88, and 93).
- Toastmaster CTM and ATM levels and Toastmaster of the year for Area 17 District 5 1999-2000
- Academic credit for
  Reliability Assurance, UCLA 1975
  Global Business Management, University of Phoenix 1998

Reports

Hannaman, G. W. and I. B. Wall, "Lesson Plans for Human Reliability Assessments in PSAs," EPRI 1003329 June 2002.

Hannaman G. W (DS&S), V. Durbec and C. Bauby (EdF), "Feasibility Study for the Integration of EDF's models for PWSCC into EPRI's STEIN code," Joint EDF and DS&S Report to EPRI, May 19, 2002.

Mickey M. B., G. W. Hannaman, B. W. Johnson, K. M. Batemen," Verification of IHLW Product Quality by Analysis of Uncertainty and Reliability in the HLW Process Control System," Data Systems & Solutions Report to Bechtel National Inc. May 2001.

Hannaman G. W., and S. A. Fleger, Evaluation of HCR Methodology Implementation in PSA and Control Room Human Factors Review for Josė Cabrera Nuclear Power Plant, EPRI, Palo Alto, CA, April 2000, 000000000001000028.

Hannaman, G. W., B. W. Johnson, Maureen K. Coveney, "Methodology For Steam Generator Condition Monitoring and Operational Assessment, Applying Monte Carlo Simulation," SAIC-97/1078, Science Applications International Corporation, San Diego, CA Dec 1998.

E. Fuller, E. Rumble, G. W. Hannaman, and M Kenton, "Risk Assessment Methodology for Complying with NRC Regulations on Steam Generator Tube Integrity: Diablo Canyon as an Example Plant" LR EPRI 550-7, Sept. 1997.

Hannaman G. W., M. Lloyd, B. Putney, G. Klopp, B. Johnson, A. Farruk, E. Fuller, and G. Pod "PSA Support For Steam Generator Degradation Specific Management" SAIC-1326, EPRI 550-7, March 1996.

A Dabiri, F. Johansen, B. Johnson, and B. Hannaman, "241-Y-101 Mixer Pump Lifetime Expectancy, for Westinghouse Hanford Company Richland, Washington, Nov. 1995.

Mahn J. A., G. W. Hannaman and P. M. Kryska, "Qualitative Methods for Assessing Risk," Sandia National Laboratories, SAND95 -0320, Albuquerque, New Mexico, May 1995.

Hannaman, G. W. "Transforming PRA Results into Performance-based Criteria for PWR steam generator Inspections and Management" White paper on EPRI project 550-07, March 1995.

Otis, M. D. D. A. Bradley and G. W. Hannaman, Technical Basis for Considering Uncertainties in I131 Release and Dose Limits for a Postulated Accident. EPRI TR-103878. EPRI, Palo Alto, CA March 1996.

Hannaman G. W., W. Parkinson, and C. Donahue, Lessons Learned from Documented Events about Human Reliability during Less Than Full Power Operations, EPRI report TR-104783, Sept. 1993.

Hannaman G. W. C. G. Donahoe and E. M. Dougherty, Insights from Human Reliability Assessments Performed during Less Than Full Power Operations, EPRI report SAIC-92/1056, SAIC San Diego CA, March 1992.

NSAC 154 "ISLOCA Evaluation Guidelines," HRA methodology, EPRI, Palo Alto, CA Sept. 1991.

Hannaman G. W. and J. Forester Analysis of Initiation of Boron Injection in Response to an ATWS, SAIC-91/1132 SAIC Report for Task 2 of Gulf States Utilities River Bend project, April 22, 1991.

Hannaman G. W. and R. J. Budnitz, "Case Study on the use of PSA methods: Human Reliability analysis," IAEA-TECDOC-592 International Atomic Energy Agency Vienna Austria, April 1991.

SHARP1 - A Revised Systematic Human Action Reliability Procedure, (with G. Parry, A. Spurgin and D. Wakefield), EPRI NP-7183-M, December 1990.

Contributor to Operator Reliability Experiments Using Power Plant Simulators, EPRI NP-6937 Volumes 1, 2, and 3, July 1990.

Hannaman G. W. Application of SHARP1 to Interfacing System Loss of Coolant Accidents (ISLOCAs), SAIC-90-1351, Science Applications International Corporation Report on EPRI Project 3206-14, September 19, 1990.

P. Lobner, L. Goldman, G. W. Hannaman and S. Langer Preliminary Risk Assessment of the NPR-MHTGR, App. B, Generic Reactor Plant Description and Source Terms, Environmental Impact Statement, EG&G-NPR-8522, June 1989.

Atefi, B., M. Drouin, W. Hannaman and J. Young, "Perspective on Application of Probabilistic Modeling Techniques to the Heavy Water and modular High Temperature Gas-Cooled New Production Designs," SAIC-89/1146, McLean, VA Sept. 29, 1989.

Models and Data Requirements for Human Reliability Analysis (with A. D. Swain, G. Mancini, L. Lederman, et al.). IAEA-TECDOC-499, Technical Document issued by the International Atomic Energy Agency, Vienna, 1989.

Hannaman G. W. F. S. Dombek and Y. D. Lukic, Evaluation of Key Human Interaction Postulated for EDF 1300 Mw(e) Nuclear Plants STGR and SBLOCA Accident Sequences. EDF Project, NUS Report 5105, May 1988.

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ENCLOSURE 4

APPARENT VIOLATION*

REGULATORY CONFERENCE

ENTERGY, INC.

JULY 10, 2003

**NOTE: THE APPARENT VIOLATION DISCUSSED AT THIS REGULATORY CONFERENCE IS SUBJECT TO FURTHER REVIEW AND MAY BE REVISED PRIOR TO ANY RESULTING ENFORCEMENT ACTION.*

APPARENT VIOLATION

10 CFR 50.48, "Fire protection," Section (b) states,

> *"Appendix R to this part establishes fire protection features required to satisfy Criterion 3 of Appendix A to this part with respect to certain generic issues for nuclear power plants licensed to operate before January 1, 1979. ... With respect to all other fire protection features covered by Appendix R, all nuclear power plants licensed to operate before January 1, 1979, must satisfy the applicable requirements of Appendix R to this part, including specifically the requirements of Sections III.G, III.J, and III.O."*

10 CFR Part 50, Appendix R, Paragraphs III.G.2 and III.G.3 state,

2. Except as provided for in paragraph G.3 of this section, where cables or equipment, including associated non-safety circuits that could prevent operation or cause maloperation due to hot shorts, open circuits, or shorts to ground, of redundant trains of systems necessary to achieve and maintain hot shutdown conditions are located within the same fire area outside of primary containment, one of the following means of ensuring that one of the redundant trains is free of fire damage shall be provided:

a. Separation of cables and equipment and associated non-safety circuits of redundant trains by a fire barrier having a 3-hour rating. Structural steel forming a part of or supporting such fire barriers shall be protected to provide fire resistance equivalent to that required of the barrier;

b. Separation of cables and equipment and associated non-safety circuits of redundant trains by a horizontal distance of more than 20 feet with no intervening combustible or fire hazards. In addition, fire detectors and an automatic fire suppression system shall be installed in the fire area; or

c. Enclosure of cable and equipment and associated non-safety circuits of one redundant train in a fire barrier having a 1-hour rating. In addition, fire detectors and an automatic fire suppression system shall be installed in the fire area;

3. Alternative or dedicated shutdown capability and its associated circuits, independent of cables, systems or components in the area, room or zone under consideration, shall be provided:

a. Where the protection of systems whose function is required for hot shutdown does not satisfy the requirement of paragraph G.2 of this section; or ..."

Contrary to this requirement, in Fire Areas 98J and 99M in Arkansas Nuclear One, Unit 1, the licensee failed to ensure that cables of redundant trains of systems necessary to achieve and maintain hot shutdown conditions were free of fire damage by one of the means specified in

THIS APPARENT VIOLATION IS SUBJECT TO FURTHER REVIEW AND MAY BE REVISED

10 CFR Part 50, Appendix R, Paragraph III.G.2, or by alternative means specified in 10 CFR Part 50, Appendix R, Paragraph III.G.3.

ENCLOSURE 5

REQUEST FOR ADDITIONAL INFORMATION

ANO REGULATORY CONFERENCE QUESTIONS

The following are what, in the regulatory conference, we asked the licensee to provide. Please review and confirm:

1. List the cables in Unit 2 that are thermoplastic, in which fire zones they appear, and the effect that having thermoplastic versus thermoset cables would have on your ability to achieve and maintain hot shutdown conditions in the event of a fire in these fire zones.

2. Provide the thermal/hydraulic time line that defines when critical safety functions must be established for all operator recovery actions for Fire Zone 99M. In addition, please provide data sheets from your simulated operator actions, including the times recorded.

3. Provide cable construction information (i.e., insulation and jacket material, such as XLPE/PVC) for all cables installed in cable trays or exposed (such as air drops) in Fire Zone 99M, including vendor and/or manufacturer.

4. Provide the extent to which cables and cable trays in Fire Zone 99M are coated with Flamemastic 71A. Include a list of which cables are coated, the amount of Flamemastic installed, and date of installation, ignition temperature, and heat release rate of Flamemastic 71A.

5. Please provide the CFAST model results for Fire Zone 99M, assuming forced ventilation is not secured and continues to supply air to the fire throughout the duration. In addition, please provide the input files you used in the CFAST fire simulation in all fire scenarios for Fire Zone 99M.