UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
SAM NUNN ATLANTA FEDERAL CENTER
61 FORSYTH STREET, SW, SUITE 23T85
ATLANTA, GEORGIA 30303-8931

September 21, 2005

EA-05-114

Mr. Dale E. Young, Vice President
Crystal River Nuclear Plant (NA1B)
ATTN: Supervisor, Licensing &
Regulatory Programs
15760 West Power Line Street
Crystal River, FL 34428-6708

SUBJECT: FINAL SIGNIFICANCE DETERMINATION FOR A WHITE FINDING AND NOTICE OF VIOLATION (CRYSTAL RIVER UNIT 3, NRC INSPECTION REPORT NO. 05000302/2005011)

Dear Mr. Young:

The purpose of this letter is to provide you with the Nuclear Regulatory Commission's (NRC) final significance determination for a finding involving unprotected post-fire safe shutdown cables and related non-feasible local manual operator actions. The finding was documented in NRC Inspection Report No. 05000302/2005007, issued on June 16, 2005, and was assessed under the significance determination process as a preliminary "greater than Green" issue (i.e., an issue of at least low to moderate safety significance which may require additional NRC inspection). The cover letter to the inspection report informed Florida Power Corporation (FPC) of the NRC's preliminary conclusion, provided FPC an opportunity to request a regulatory conference on this matter, and forwarded the details of the NRC's preliminary estimate of the change in core damage frequency (CDF) for this finding.

At FPC's request, an open regulatory conference was conducted on July 22, 2005, to discuss FPC's position on this issue. The enclosures to this letter include the list of attendees at the regulatory conference and material presented by FPC and NRC.

During the conference, FPC presented the results of its estimate of the increase in CDF due to the performance deficiency including influential assumptions and risk analysis methodology. FPC concluded that the finding was of very low safety significance. The critical aspects of FPC's analysis and inputs that differed from the NRC's preliminary estimate included the following: (1) fully developed fires would produce enough smoke to require extensive removal efforts with a gas-powered ejector (NOTE: FPC estimated that a sufficient amount of smoke would be removed within 20 minutes to allow an operator to reset the emergency diesel generator (EDG) lockout breaker in the 3B 4160-VAC switchgear compartment and recover the 4160-VAC electrical bus.); (2) FWP-7, the non-safety-related feedwater pump, and its associated power and control circuits would remain free from fire damage and could be started from the main control room to provide and maintain secondary side heat removal; (3) the EDGs could operate unloaded without incurring damage for at least 1 hour given the potential lack of room ventilation; (4) the emergency feedwater initiation control system (EFIC) would be available for at least 2 hours instead of 30 minutes as assumed in the NRC's preliminary

estimate; and (5) FPC would use the Technical Support Center (TSC) to provide guidance to the operating and response staff for diverse emergency and auxiliary feedwater lineups and for electrical distribution alignment. FPC did not contest that the finding represented a violation of 10 CFR Part 50, Appendix R, Section III.G.2.

After considering the information developed during the inspection and the information FPC provided at the conference, the NRC has concluded that the final inspection finding is appropriately characterized as White in the mitigating systems cornerstone. In summary, the most critical differences between the NRC's assessment of the change in CDF and that of FPC's involved the likelihood of success of an operator action to reset the EDG lockout breaker to recover the 4160-VAC electrical bus and credit for use of FWP-7. The NRC ultimately concluded that the probability of failure to reset the EDG lockout was much greater than that assumed by FPC due to the extreme environmental conditions produced by the fire coupled with the very poor ergonomics associated with accomplishing a task in this situation. Therefore, possible accomplishment of this task could not be considered until smoke removal efforts were successfully employed. In considering the use of FWP-7, the NRC agreed with FPC that some credit was warranted which would result in a reduction in the NRC's preliminary estimate.

Regarding other aspects of FPC's analysis, the NRC agrees with FPC that the EDG could operate unloaded for at least 1 hour without incurring damage and that EFIC would be available for at least 2 hours. Regarding the use of the TSC, the NRC concluded that the combination of time constraints, the complexity of the emergency situation, power/communications availability, and the variability in the actual TSC response precluded TSC credit.

You have 10 calendar days from the date of this letter to appeal the staff's determination of significance for the identified finding. Such appeals will be considered to have merit only if they meet the criteria given in NRC Inspection Manual Chapter 0609, Attachment 2.

The NRC also concluded that a violation of 10 CFR Part 50, Appendix R, Section III.G.2, occurred in that the protection and metering circuits were not physically separated or protected from fire damage as required. The violation is set forth in the enclosed Notice of Violation (Notice). The circumstances surrounding the violation are described in detail in NRC Inspection Report No. 05000302/2005007 dated June 16, 2005. In accordance with the NRC Enforcement Policy, NUREG-1600, the Notice of Violation is considered escalated enforcement action because it is associated with a White finding.

The NRC has concluded that information regarding the reason for the violation, the corrective actions taken and planned to correct the violation and prevent recurrence, and the date when full compliance was achieved is already adequately addressed on the docket in NRC Inspection Report No. 05000302/2004009 dated March 14, 2005; NRC Inspection Report No. 05000302/2005007 dated June 16, 2005; and the information provided by FPC at the July 22, 2005, regulatory conference (Enclosure 3). Therefore, you are not required to respond to this letter unless the description therein does not accurately reflect your corrective actions or your position. In that case, or if you choose to provide additional information, you should follow the instructions specified in the enclosed Notice.

For administrative purposes, this letter is issued as a separate NRC Inspection Report, No. 05000302/200500011, and the above violation is identified as VIO 0500302/200500011-01, Unprotected Post-Fire Safe Shutdown Cables and Related Non-feasible Local Manual Operator Action. Accordingly, Apparent Violation 05000302/2005007-01 is closed.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosures, and your response (should you choose to provide one) will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's document system (ADAMS), which is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html. To the extent possible, any response should not include any personal privacy, proprietary, classified, or safeguards information so that it can be made available to the Public without redaction. The NRC also includes significant enforcement actions on its Web site at www.nrc.gov; select **What We Do**, **Enforcement**, then **Significant Enforcement Actions**.

Should you have any questions regarding this letter, please contact Mr. D. Charles Payne, Chief, Engineering Branch 2, Division of Reactor Safety, at (404)562-4669.

Sincerely,

William Travers

William D. Travers
Regional Administrator

Docket No.: 50-302
License No.: DPR-72

Enclosures:
1. Notice of Violation
2. List of Attendees
3. Material presented by FPC
4. Material presented by NRC

cc w/encls:
Daniel L. Roderick
Director Site Operations
Crystal River Nuclear Plant (NA2C)
Electronic Mail Distribution

Jon A. Franke
Plant General Manager
Crystal River Nuclear Plant (NA2C)
Electronic Mail Distribution

Richard L. Warden
Manager Nuclear Assessment
Crystal River Nuclear Plant (NA2C)
Electronic Mail Distribution

Michael J. Annacone
Engineering Manager
Crystal River Nuclear Plant (NA2C)
Electronic Mail Distribution

R. Alexander Glenn
Associate General Counsel (MAC - BT15A)
Florida Power Corporation
Electronic Mail Distribution

Steven R. Carr
Associate General Counsel - Legal Dept.
Progress Energy Service Company, LLC
Electronic Mail Distribution

Attorney General
Department of Legal Affairs
The Capitol
Tallahassee, FL 32304

William A. Passetti
Bureau of Radiation Control
Department of Health
Electronic Mail Distribution

Craig Fugate, Director
Division of Emergency Preparedness
Department of Community Affairs
Electronic Mail Distribution

Chairman
Board of County Commissioners
Citrus County
110 N. Apopka Avenue
Inverness, FL 36250

Jim Mallay
Framatome Technologies
Electronic Mail Distribution
FPC

Distribution w/encls:
L. Reyes, EDO
J. Dyer, NRR
W. Borchardt, NRR
L. Chandler, OGC
J. Moore, OGC
E. Julian, SECY
B. Keeling, OCA
Enforcement Coordinators
RI, RIII, RIV
E. Hayden, OPA
G. Caputo, OI
H. Bell, OIG
C. Carpenter, NRR
F. Bonnett, NRR
B. Mozafari, NRR
M. Johnson, OE
L. Trocine, OE
L. Plisco, RII
V. McCree, RII
C. Casto, RII
C. Payne, RII
W. Rogers, RII
S. Sparks, RII
C. Evans, RII
J. Munday, RII
R. Hannah, RII
K. Clark, RII
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SES for, via verbal confirmation

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ADAMS: X Yes ACCESSION NUMBER:____

OFFICE	RII:DRS	RII	RII:DRS	OE	NRR		
SIGNATURE		CE	VM				
NAME	WROGERS	CEVANS	VMCCREE				
DATE	9/1/5	9/7/05	9/7/05				
E-MAIL COPY?	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO

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OFFICE	RII:DRS	RII	RII:DRS	OE	NRR		
SIGNATURE					N/A per e-mail		
NAME	WROGERS	CEVANS	VMCCREE	C. Nolan	P. Bonnett		
DATE				09/16/05	09/12/05		
E-MAIL COPY?	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO

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NOTICE OF VIOLATION

Florida Power Corporation
Crystal River Nuclear Plant
Unit 3

Docket No. 50-302
License No. DRP-72
EA-05-114

During an NRC inspection completed on June 8, 2005, a violation of NRC requirements was identified. In accordance with the NRC Enforcement Policy, the violation is listed below:

> 10 CFR 50.48(b)(1) requires, in part, that all nuclear power plants licensed to operate prior to January 1, 1979, must satisfy the applicable requirements of 10 CFR Part 50, Appendix R, Section III.G, Fire Protection of Safe Shutdown Capability.
>
> Section III.G.2 states that, except as provided for in Section III.G.3, 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, 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.)
>
> Contrary to the above, on January 26, 2005, the licensee failed to ensure that one of the redundant trains of systems necessary to achieve and maintain hot shutdown conditions would be free of fire damage via one of the three means specified in 10 CFR Part 50, Appendix R, Section III.G.2. Specifically, cables for the electrical protection and metering circuit located in the 3A 4160-V engineered safeguards (ES) switchgear room were vulnerable to fire damage that could disable both the 3A 4160-V ES switchgear and the redundant train 3B 4160-V ES switchgear resulting in a loss of all safety-related alternating current power.
>
> This violation is associated with a White Significance Determination Process finding for Unit 3 in the mitigating systems cornerstone.

The NRC has concluded that information regarding the reason for the violation, the corrective actions taken and planned to correct the violation and prevent recurrence, and the date when full compliance was achieved is already adequately addressed on the docket in NRC Inspection Report No. 05000302/2004009 dated March 14, 2005; NRC Inspection Report No. 05000302/2005007 dated June 16, 2005; and the information provided by FPC at the July 22, 2005, regulatory conference (Enclosure 3). However, you are required to submit a written statement or explanation pursuant to 10 CFR 2.201 if the description therein does not accurately reflect your corrective actions or your position. In that case, or if you choose to respond, clearly mark your response as a "Reply to a Notice of Violation - EA-05-114," and send it to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555, with a copy to the Regional Administrator, Region II, within 30 days of the date of the letter transmitting this Notice of Violation (Notice).

If you contest this enforcement action, you should also provide a copy of your response with the basis for your denial to the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001.

If you choose to respond, your response will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's document system (ADAMS), to the extent possible, it should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the public without redaction. ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html. If personal privacy or proprietary information is necessary to provide an acceptable response, then please provide a bracketed copy of your response that identifies the information that should be protected and a redacted copy of your response that deletes such information. If you request withholding of such material, you must specifically identify the portions of your response that you seek to have withheld and provide in detail the bases for your claim of withholding (e.g., explain why the disclosure of information will create an unwarranted invasion of personal privacy or provide the information required by 10 CFR 2.390(b) to support a request for withholding confidential commercial or financial information). If safeguards information is necessary to provide an acceptable response, please provide the level of protection described in 10 CFR 73.21.

In accordance with 10 CFR 19.11, you may be required to post this Notice within 2 working days.

Dated this 21st day of September 2005

LIST OF ATTENDEES

Nuclear Regulatory Commission:
V. McCree, Director, Division of Reactor Safety (DRS), RII
C. Payne, Chief, Engieering Branch 2, DRS, RII
J. Munday, Chief, Projects Branch 3, Division of Reactor Projects, (DRP), RII
W. Rogers, Senior Reactor Analyst, DRS, RII
R. Rodriguez, Reactor Inspector, DRS, RII
C. Evans, Enforcement Officer and Regional Attorney
S. Sparks, Senior Enforcement Specialist
R. Schin, Senior Engineering Inspector, DRS, RII
M. Maymi, Engineering Inspector, DRS, RII
D. Mas-Penaranda, Project Engineer, DRP, RII
D. Starkey, Office of Enforcement (telecon)
A. Klein, Office of Nuclear Reactor Regulation (NRR) (telecon)
P. Koltay, NRR (telecon)
B. Mozafari, NRR (telecon)
R. Gallucci, NRR (telecon)

Florida Power Corporation:
D. Young, Site Vice President, Crystal River
M. Annacone, Engineering Manager
S. Barkofski, Electrical Design Engineering Supervisor
D. Porter, Superintendent, Shift Operations
D. Miskiewicz, Lead PSA Engineer
S. Powell, Licensing Supervisor

Crystal River Unit 3
Regulatory Conference
Region II, Atlanta, GA

July 22, 2005

Progress Energy

• Introduction –	Dale Young
• Description of Finding –	Mike Annacone
• Electrical Distribution and Plant Layout –	Steve Barkofski
• Response Timeline –	Dave Porter
• Probabilistic Safety Assessment –	Dave Miskiewicz
• Conclusions -	Mike Annacone
• Closing Remarks –	Dale Young

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During the conference, FPC presented the results of its estimate of the increase in CDF due to the performance deficiency including influential assumptions and risk analysis methodology. FPC concluded that the finding was of very low safety significance. The critical aspects of FPC's analysis and inputs that differed from the NRC's preliminary estimate included the following: (1) fully developed fires would produce enough smoke to require extensive removal efforts with a gas-powered ejector (NOTE: FPC estimated that a sufficient amount of smoke would be removed within 20 minutes to allow an operator to reset the emergency diesel generator (EDG) lockout breaker in the 3B 4160-VAC switchgear compartment and recover the 4160-VAC electrical bus.); (2) FWP-7, the non-safety-related feedwater pump, and its associated power and control circuits would remain free from fire damage and could be started from the main control room to provide and maintain secondary side heat removal; (3) the EDGs could operate unloaded without incurring damage for at least 1 hour given the potential lack of room ventilation; (4) the emergency feedwater initiation control system (EFIC) would be available for at least 2 hours instead of 30 minutes as assumed in the NRC's preliminary

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You have 10 calendar days from the date of this letter to appeal the staff's determination of significance for the identified finding. Such appeals will be considered to have merit only if they meet the criteria given in NRC Inspection Manual Chapter 0609, Attachment 2.

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Should you have any questions regarding this letter, please contact Mr. D. Charles Payne, Chief, Engineering Branch 2, Division of Reactor Safety, at (404)562-4669.

Sincerely,

William Travers

William D. Travers
Regional Administrator

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110 N. Apopka Avenue
Inverness, FL 36250

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M. Johnson, OE
L. Trocine, OE
L. Plisco, RII
V. McCree, RII
C. Casto, RII
C. Payne, RII
W. Rogers, RII
S. Sparks, RII
C. Evans, RII
J. Munday, RII
R. Hannah, RII
K. Clark, RII
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OFFICE	RII:DRS	RII	RII:DRS	OE	NRR		
SIGNATURE					N/A per e-mail		
NAME	WROGERS	CEVANS	VMCCREE	C. Nolan	P. Bonnett		
DATE				09/16/05	09/12/05		
E-MAIL COPY?	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO

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NOTICE OF VIOLATION

Florida Power Corporation
Crystal River Nuclear Plant
Unit 3

Docket No. 50-302
License No. DRP-72
EA-05-114

During an NRC inspection completed on June 8, 2005, a violation of NRC requirements was identified. In accordance with the NRC Enforcement Policy, the violation is listed below:

10 CFR 50.48(b)(1) requires, in part, that all nuclear power plants licensed to operate prior to January 1, 1979, must satisfy the applicable requirements of 10 CFR Part 50, Appendix R, Section III.G, Fire Protection of Safe Shutdown Capability.

Section III.G.2 states that, except as provided for in Section III.G.3, 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, 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.)

Contrary to the above, on January 26, 2005, the licensee failed to ensure that one of the redundant trains of systems necessary to achieve and maintain hot shutdown conditions would be free of fire damage via one of the three means specified in 10 CFR Part 50, Appendix R, Section III.G.2. Specifically, cables for the electrical protection and metering circuit located in the 3A 4160-V engineered safeguards (ES) switchgear room were vulnerable to fire damage that could disable both the 3A 4160-V ES switchgear and the redundant train 3B 4160-V ES switchgear resulting in a loss of all safety-related alternating current power.

This violation is associated with a White Significance Determination Process finding for Unit 3 in the mitigating systems cornerstone.

The NRC has concluded that information regarding the reason for the violation, the corrective actions taken and planned to correct the violation and prevent recurrence, and the date when full compliance was achieved is already adequately addressed on the docket in NRC Inspection Report No. 05000302/2004009 dated March 14, 2005; NRC Inspection Report No. 05000302/2005007 dated June 16, 2005; and the information provided by FPC at the July 22, 2005, regulatory conference (Enclosure 3). However, you are required to submit a written statement or explanation pursuant to 10 CFR 2.201 if the description therein does not accurately reflect your corrective actions or your position. In that case, or if you choose to respond, clearly mark your response as a "Reply to a Notice of Violation - EA-05-114," and send it to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555, with a copy to the Regional Administrator, Region II, within 30 days of the date of the letter transmitting this Notice of Violation (Notice).

If you contest this enforcement action, you should also provide a copy of your response with the basis for your denial to the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001.

If you choose to respond, your response will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's document system (ADAMS), to the extent possible, it should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the public without redaction. ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html. If personal privacy or proprietary information is necessary to provide an acceptable response, then please provide a bracketed copy of your response that identifies the information that should be protected and a redacted copy of your response that deletes such information. If you request withholding of such material, you must specifically identify the portions of your response that you seek to have withheld and provide in detail the bases for your claim of withholding (e.g., explain why the disclosure of information will create an unwarranted invasion of personal privacy or provide the information required by 10 CFR 2.390(b) to support a request for withholding confidential commercial or financial information). If safeguards information is necessary to provide an acceptable response, please provide the level of protection described in 10 CFR 73.21.

In accordance with 10 CFR 19.11, you may be required to post this Notice within 2 working days.

Dated this 21st day of September 2005

LIST OF ATTENDEES

Nuclear Regulatory Commission:
V. McCree, Director, Division of Reactor Safety (DRS), RII
C. Payne, Chief, Engieering Branch 2, DRS, RII
J. Munday, Chief, Projects Branch 3, Division of Reactor Projects, (DRP), RII
W. Rogers, Senior Reactor Analyst, DRS, RII
R. Rodriguez, Reactor Inspector, DRS, RII
C. Evans, Enforcement Officer and Regional Attorney
S. Sparks, Senior Enforcement Specialist
R. Schin, Senior Engineering Inspector, DRS, RII
M. Maymi, Engineering Inspector, DRS, RII
D. Mas-Penaranda, Project Engineer, DRP, RII
D. Starkey, Office of Enforcement (telecon)
A. Klein, Office of Nuclear Reactor Regulation (NRR) (telecon)
P. Koltay, NRR (telecon)
B. Mozafari, NRR (telecon)
R. Gallucci, NRR (telecon)

Florida Power Corporation:
D. Young, Site Vice President, Crystal River
M. Annacone, Engineering Manager
S. Barkofski, Electrical Design Engineering Supervisor
D. Porter, Superintendent, Shift Operations
D. Miskiewicz, Lead PSA Engineer
S. Powell, Licensing Supervisor

Crystal River Unit 3 Regulatory Conference Region II, Atlanta, GA

July 22, 2005

Progress Energy

• Introduction –	Dale Young
• Description of Finding –	Mike Annacone
• Electrical Distribution and Plant Layout –	Steve Barkofski
• Response Timeline –	Dave Porter
• Probabilistic Safety Assessment –	Dave Miskiewicz
• Conclusions -	Mike Annacone
• Closing Remarks –	Dale Young

Background – NRC Triennial Inspection

- Findings related to today's presentation:
  - Single failure criteria violation for 4160V ES protective relaying.
  - B EDG lockout reset manual action not considered feasible in required time frame
- Introduced during implementation of Off-Site Power and Backup Emergency Safeguards Transformer installations (1990/1993)
- Vulnerability originally recognized in Fire Study as a Fire Protection issue (Appendix R Manual Action) but not as a Single Failure Criteria Violation

Background – Single Failure Issue

- Modifications implemented
  - Eliminating need for manual action to reset the B EDG lockout.
- Immediate extent of condition – 4160V and 480V Emergency Safeguards power distribution protective relaying and metering with no additional vulnerabilities identified
- Root Cause Analysis performed:
  - Failure to perform Failure Modes Effects Analysis during OPT/BEST modifications
  - Corrective Actions:
    - Implement FMEA process
    - Detailed Extent of Condition completed with no additional vulnerabilities identified

Highlights of NRC Findings:

- Reliance on manual actions vs. physical separation or protection
- Local Manual Action to reset B EDG Lockout not feasible:
  - Proximity to Fire location – Fire in A ES SWGR Room
    - Fire Team entry through B ES SWGR Room requires fire door between rooms to be open, No floor drains in rooms
  - Manual Action time critical – 30 minutes:
    - Restoration of ventilation and cooling to Emergency Feedwater Isolation and Control (EFIC)
  - Operator arrival at B SWGR room – 25 minutes, room not yet ventilated – smoke filled, water on floor, water mist
    - CR3 Time validated / NRC walk-down

CR-3 Insights

- 30 minute time requirement to re-establish EFIC room cooling is conservative.
  - **Fire Study 30 minute time limit conservatively chosen for simplicity**
  - **At least 120 minutes available**
  - **Steam driven EFP-2 remains available**

- Fire Study and NRC SDP do not credit use of Auxiliary Feedwater System.
  - **System free of fire damage**
  - **FWP-7 has it's own diesel generator**
  - **Emergency Operating Procedures direct system use when EFW unavailable**

CR-3 Insights

- As a result of the above items, secondary side heat removal is not lost
  - **Eliminates uncertainties in Phase II evaluation regarding:**
    - **Effectiveness of secondary side cooling following an overcooling event**
    - **Primary system response with a delay in secondary side heat removal**
- Only one scenario causes loss of power to Unit Auxiliary loads
  - **Reduces probability of normal secondary side heat removal loss**

CR-3 Insights

- At least one off-site power transformer remains available in all scenarios
- EDG availability without room cooling
  - Diesel has started and is running unloaded
  - Engine coolant and lube oil cooling remains unaffected
  - No power to EDG Room Supply Fans until ES Bus re-powered
  - Engine heat raises room ambient temperature

Electrical Distribution and Physical Layout

- CR3 Energy Complex Switchyard Layout
- Emergency Safeguards (ES) Electrical Buses
- Control Complex Physical Layout
- Photos of the ES Switchgear Rooms
- Photos of the ES Switchgear Control Cubicles
- Fire Scenarios
- Mechanical / Hydraulic Time Line
- ES Switchgear Room Fire Model
- Evaluation of Auxiliary Feed Water Pump Circuits

500KV Switchyard One-Line Diagram

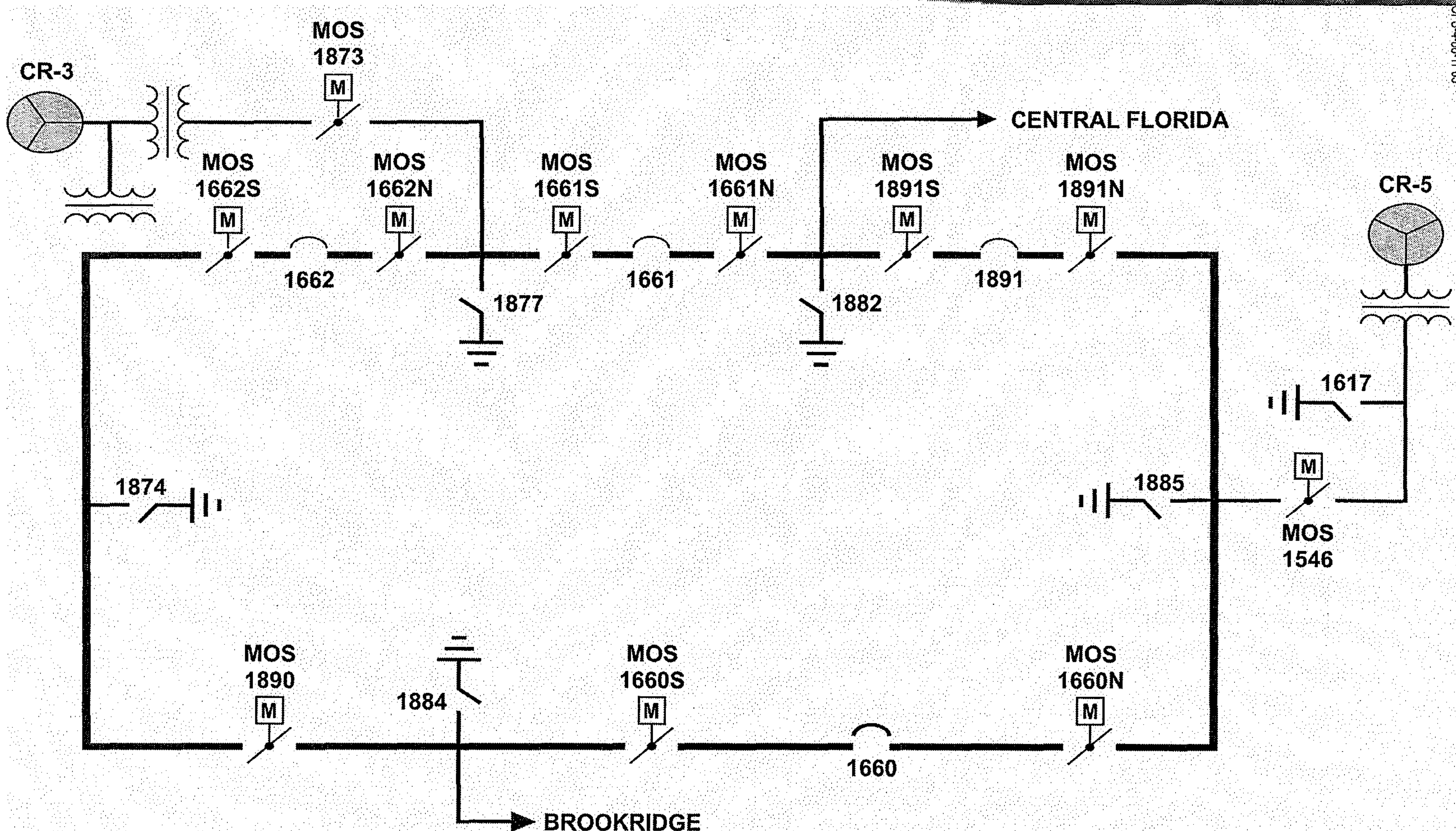

Figure 1 - Crystal River 500KV Switchyard

July 22, 2005

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230KV Switchyard One-Line Diagram

OPS-4-88-TP04

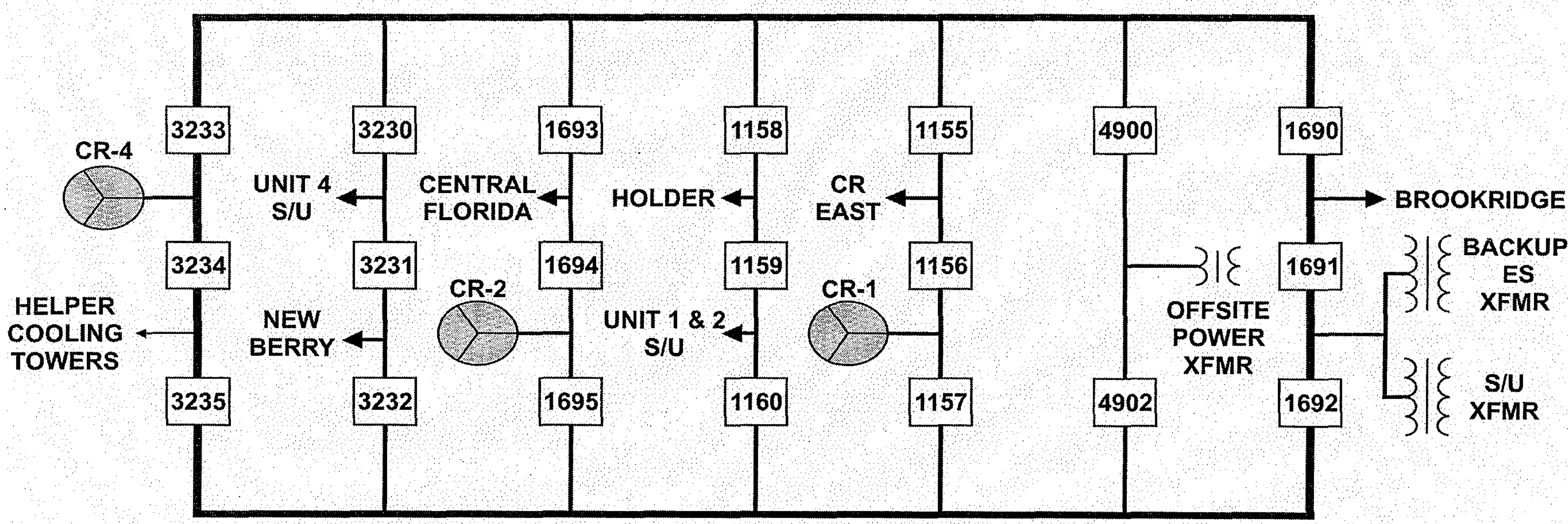
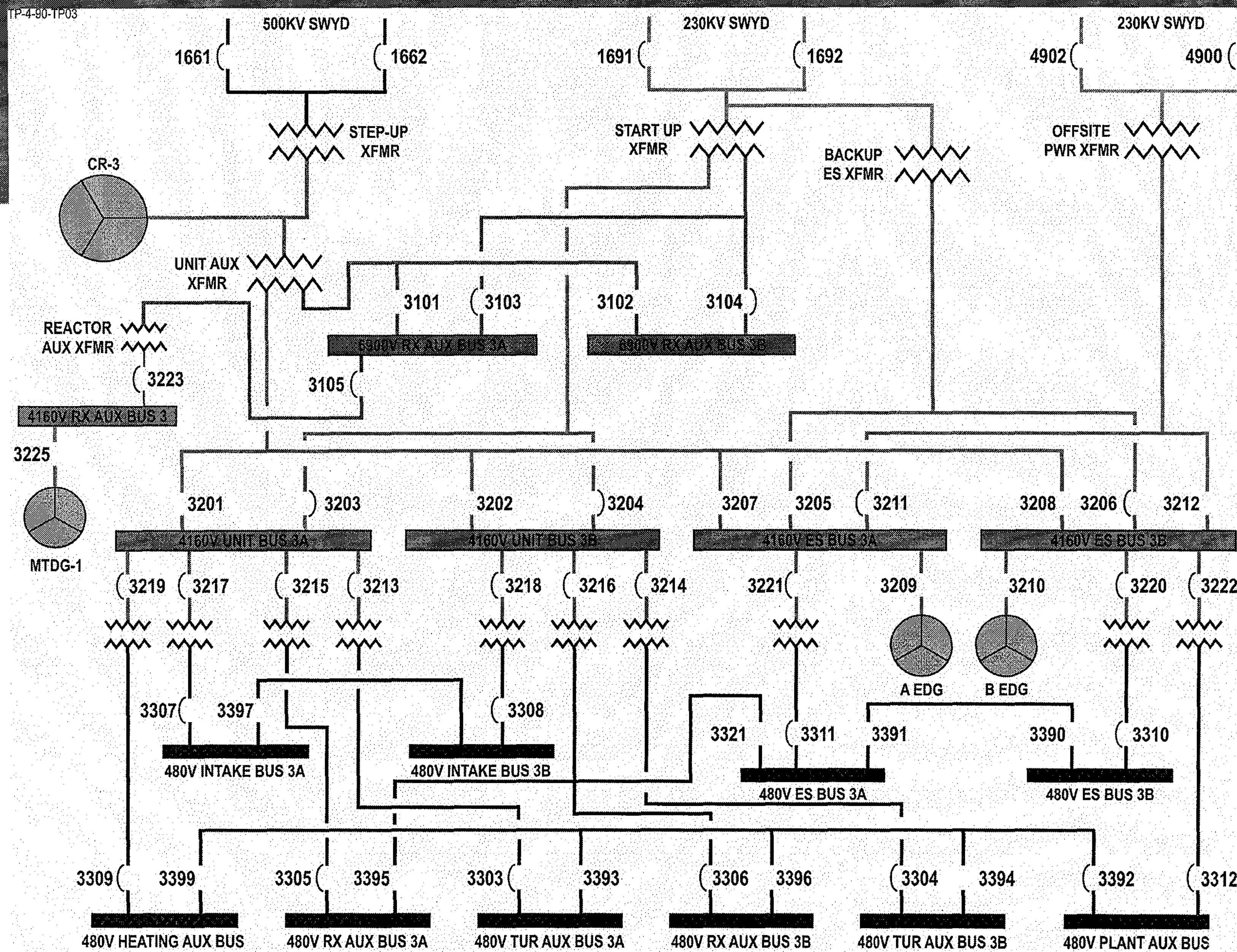
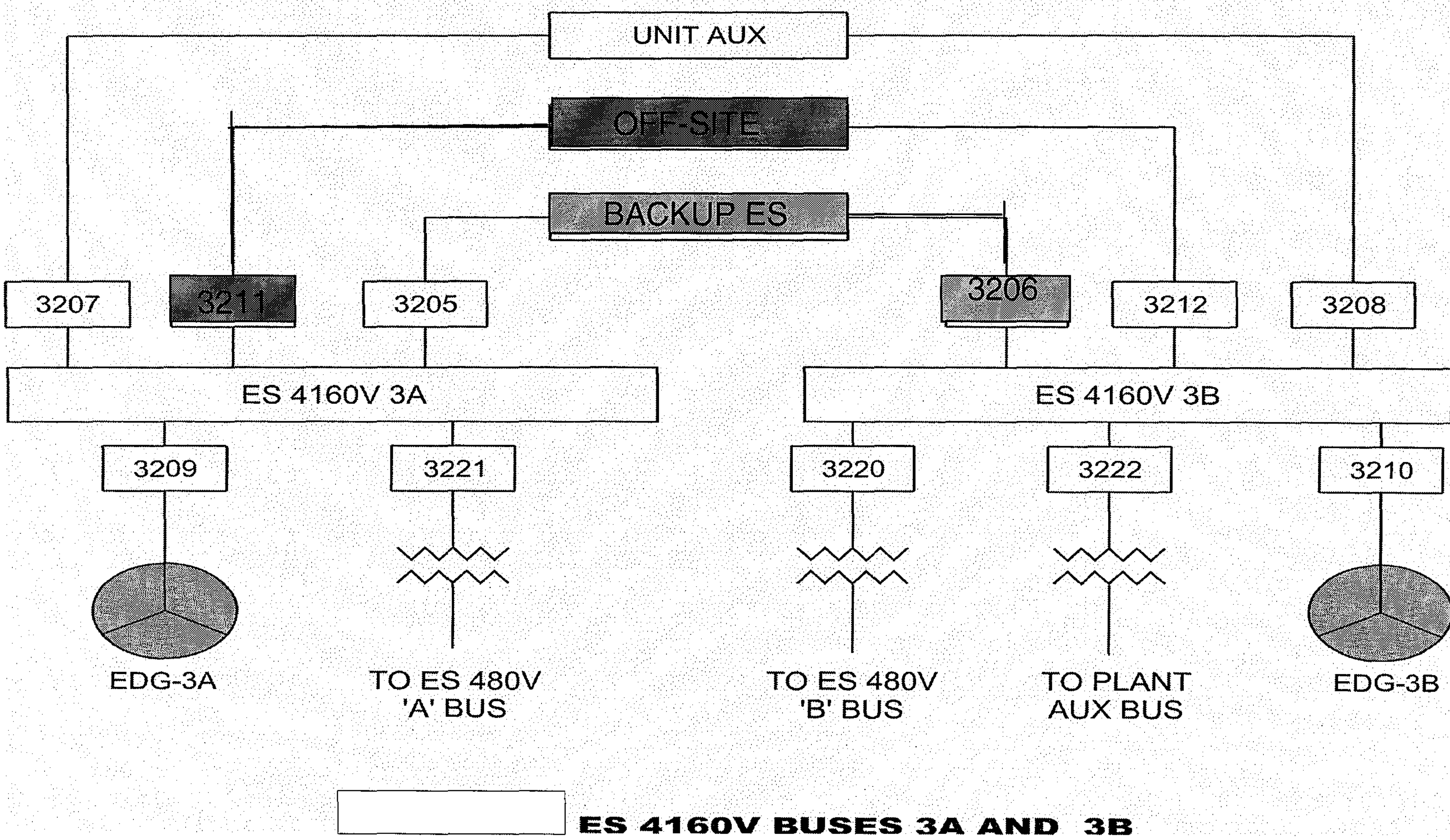


Figure 2 - 230KV Switchyard



AC ELECTRICAL DISTRIBUTION SYSTEM

Emergency Safeguards (ES) Buses



July 22, 2005

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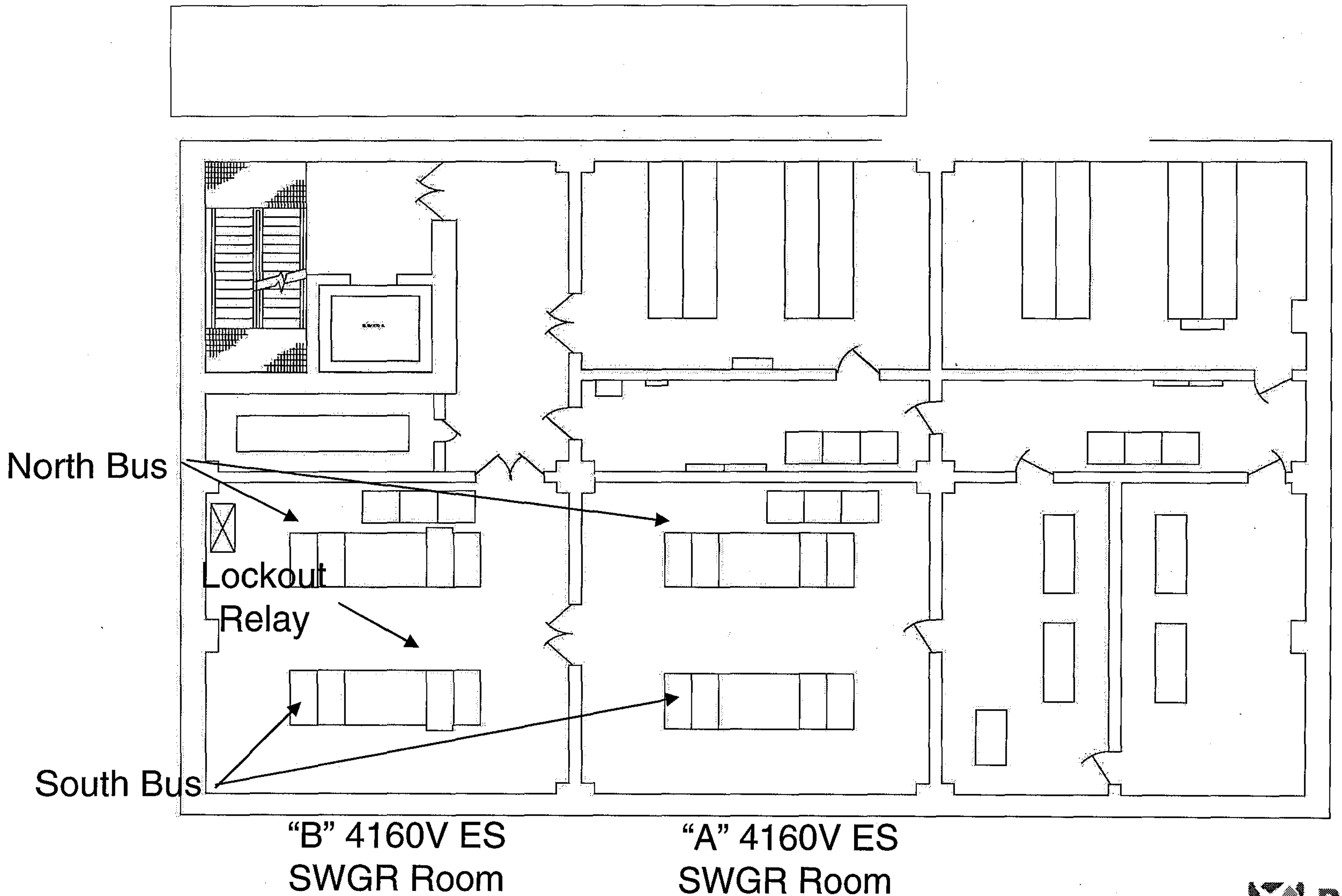
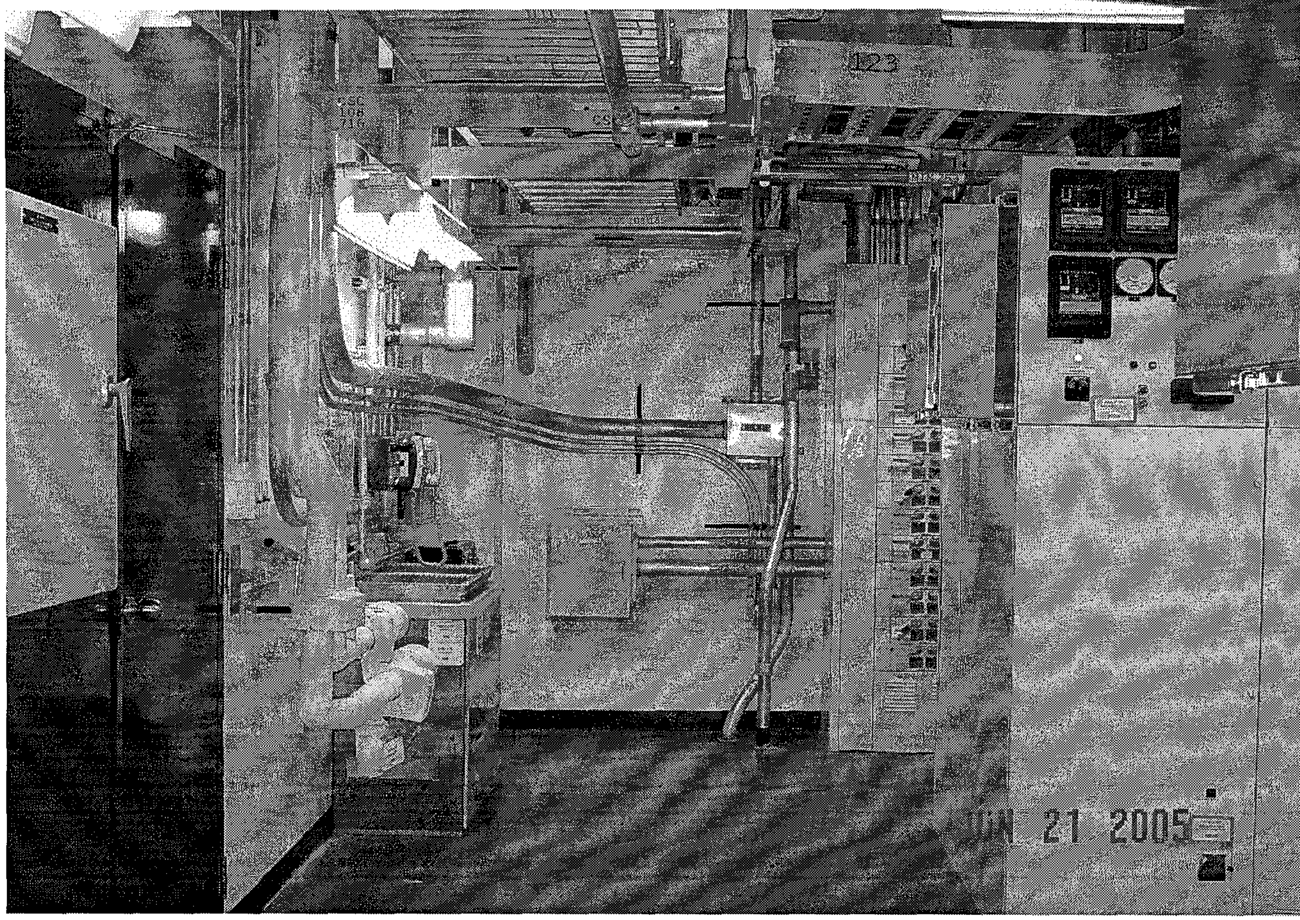
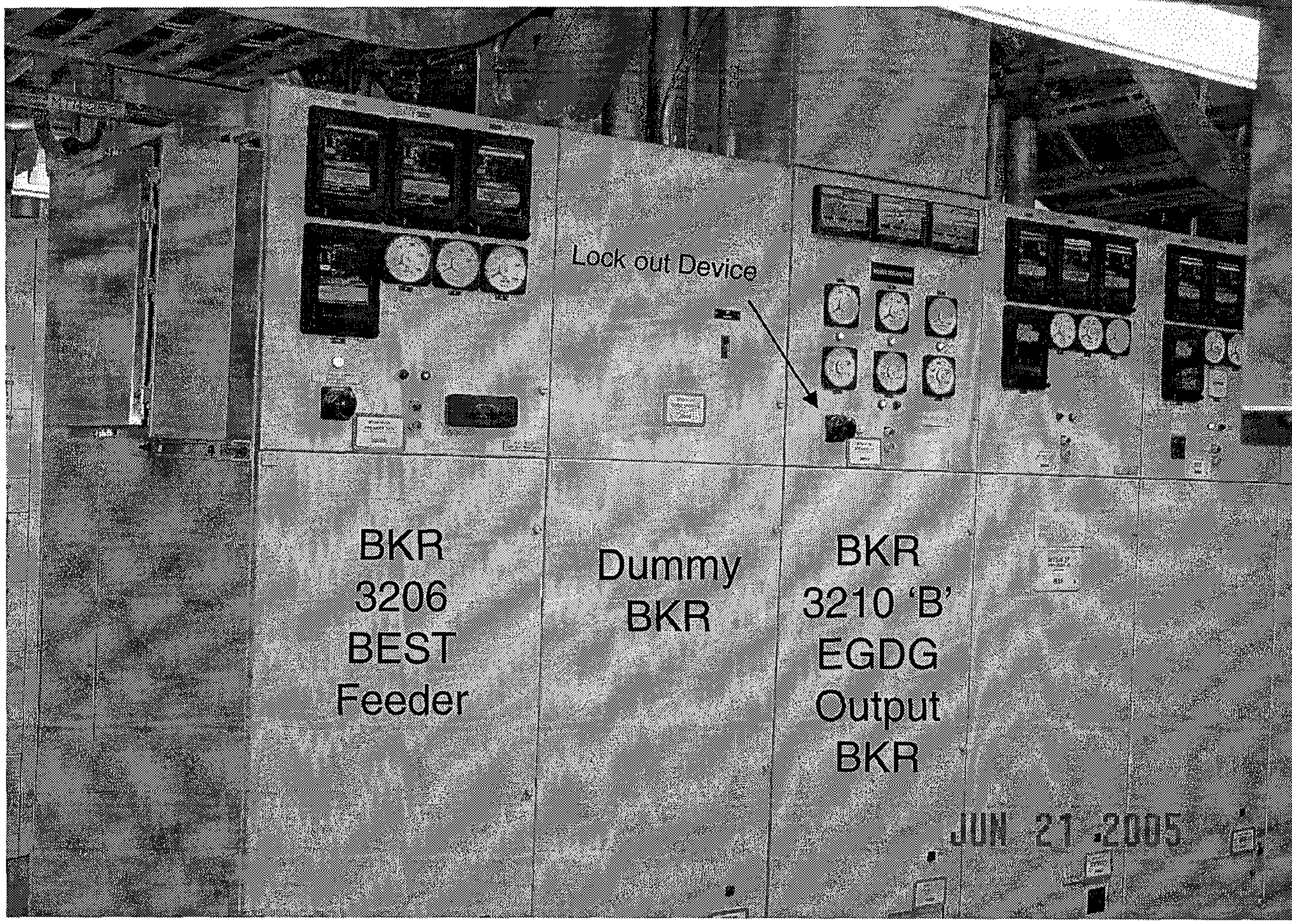
Control Complex 108' Elevation



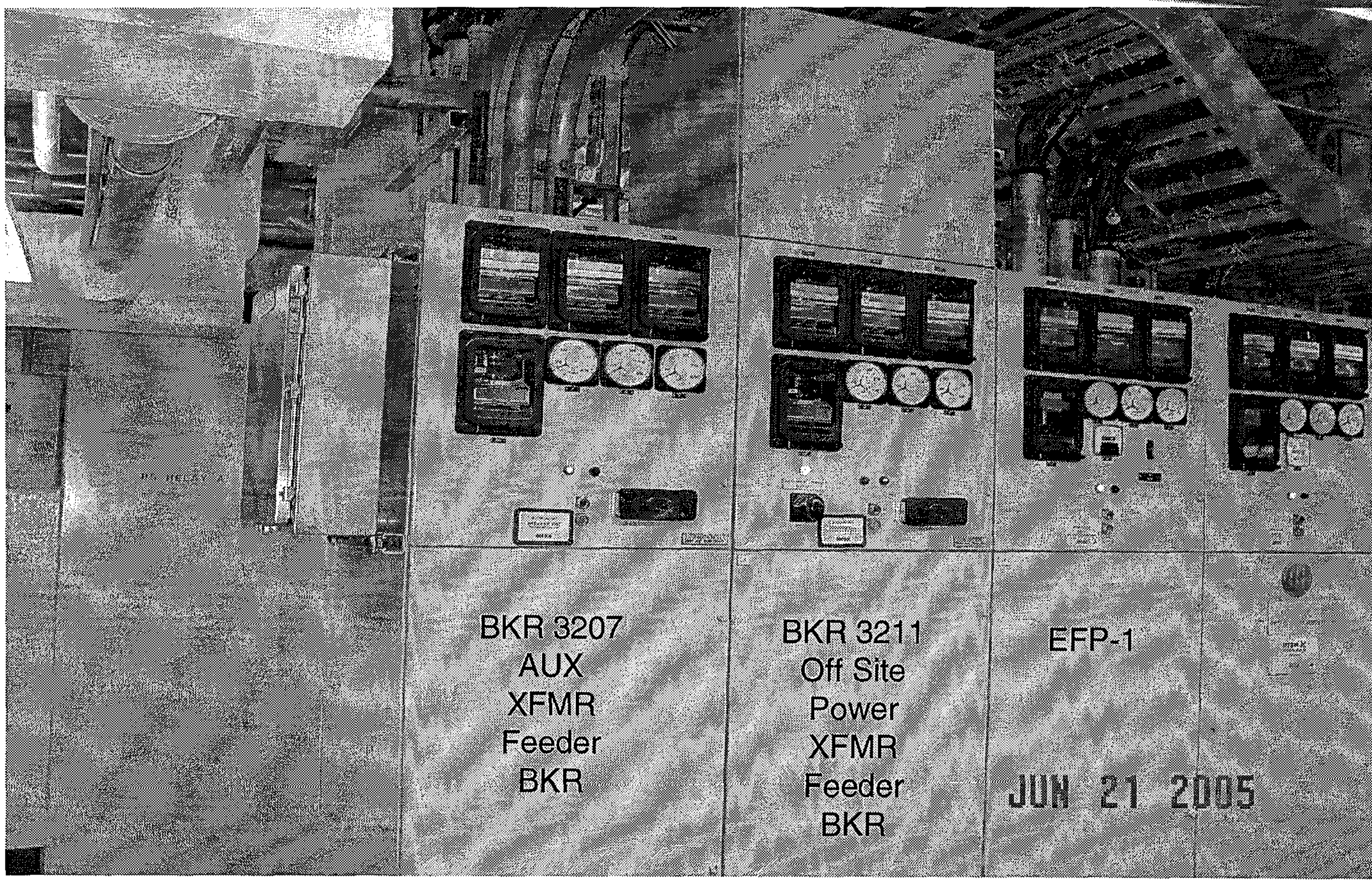
Control Complex 108' Elevation "B" 4160V Switchgear (SWGR) Room



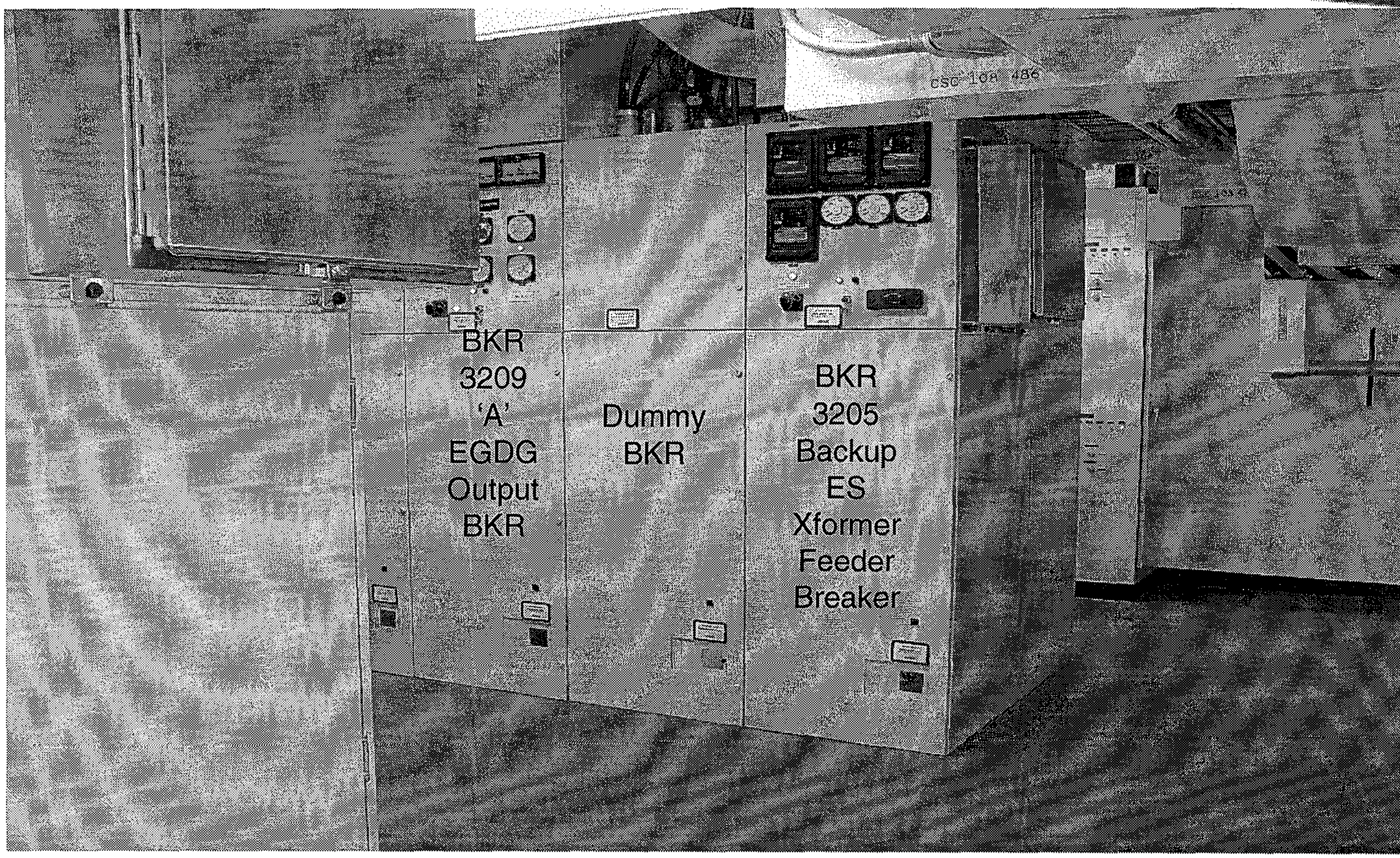
Control Complex 108' Elevation "B" 4160V SWGR



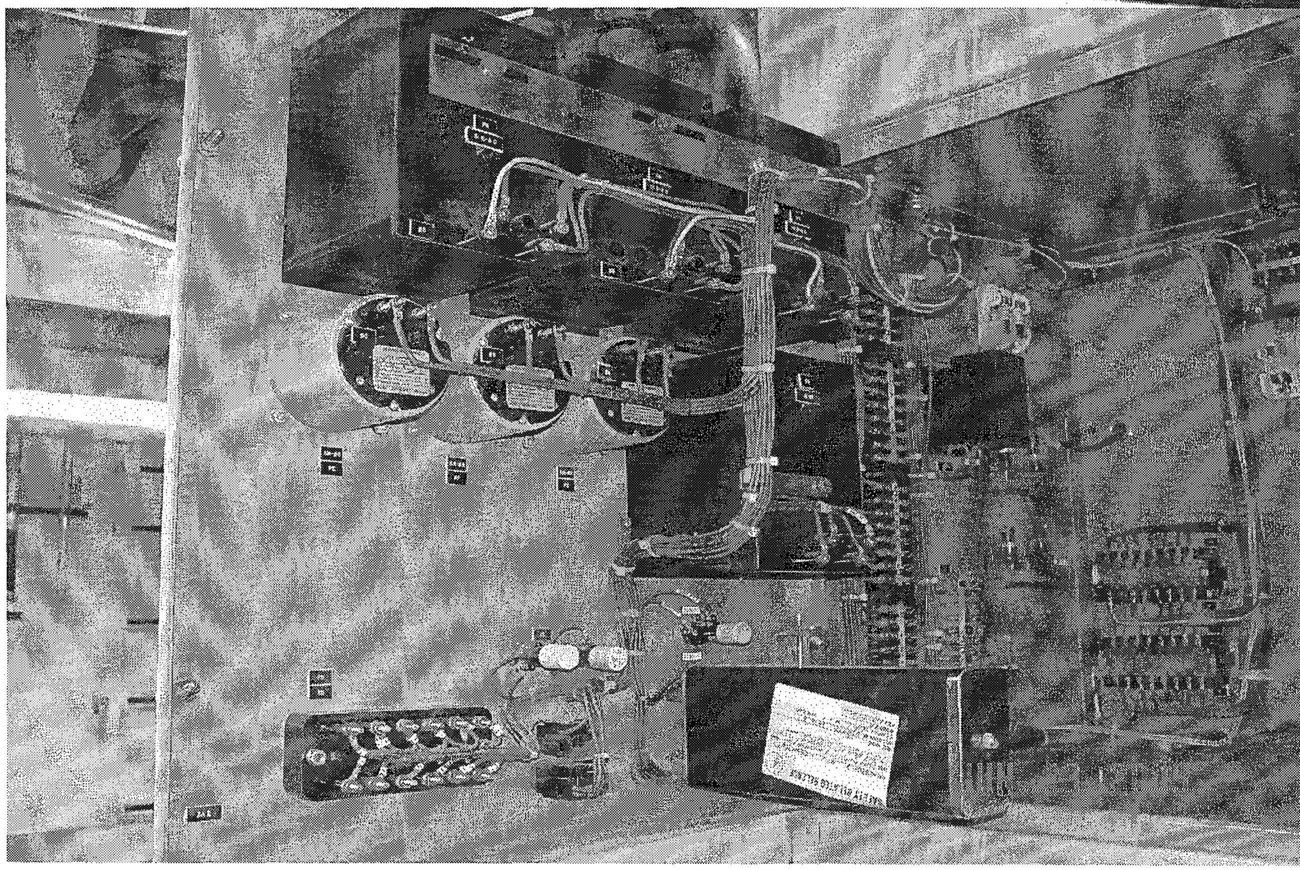
Control Complex 108' Elevation "A" 4160V SWGR



Control Complex 108' Elevation "A" 4160V SWGR

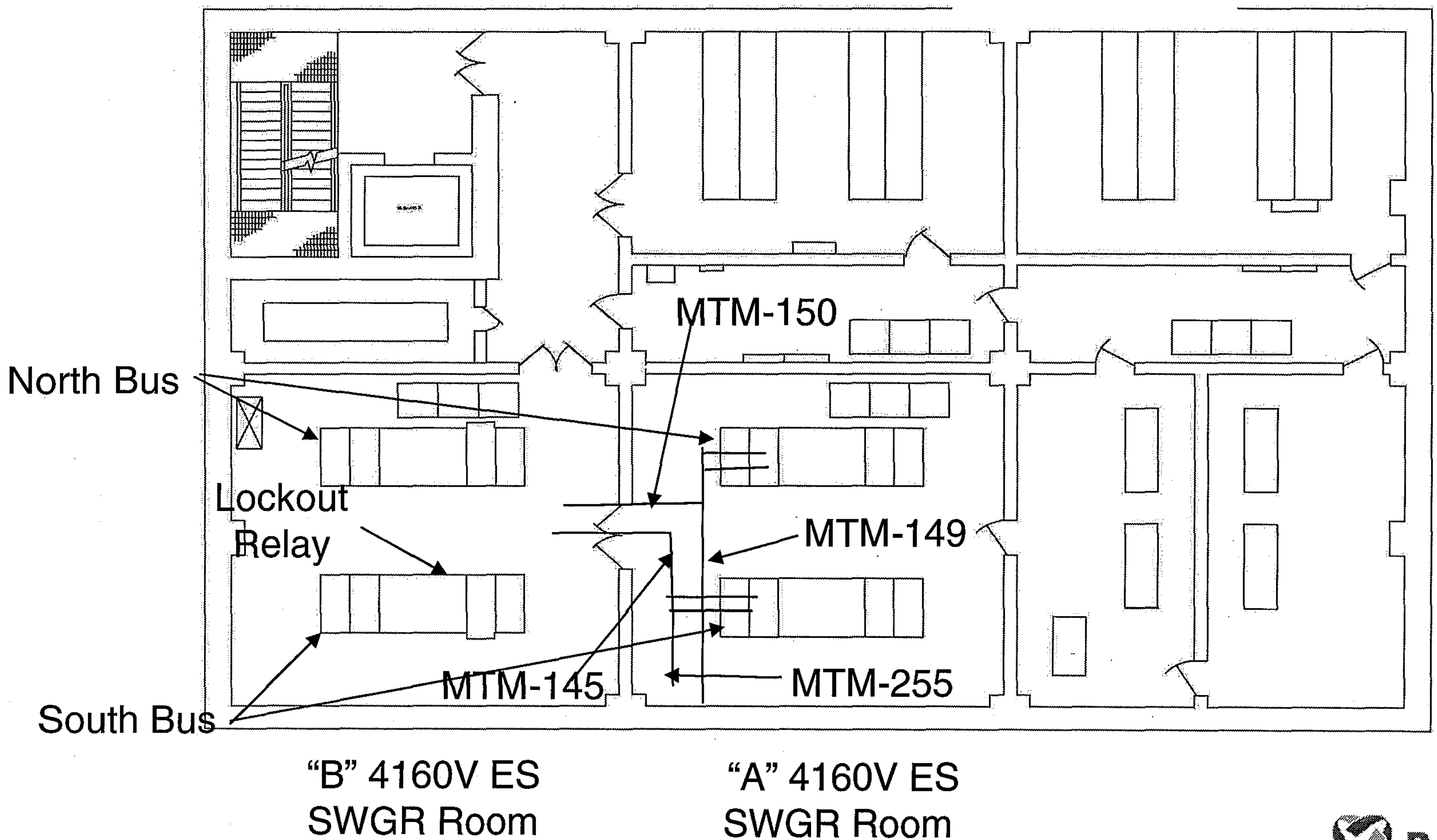


Control Complex 108' Elevation "A" 4160V SWGR



"A" 4160V SWGR – BKR-3211 Control Cubicle

Control Complex 108' Elevation



Fire Scenarios

- Evaluated Fire Scenarios in the “A” 4160V Switchgear Room
  - **Fire had to impact the CT relay circuits associated with a single failure issue.**
  - **Result in a loss of both ES Buses.**
  - **Require the manual action to reset the B-EDG lockout.**
- Validated four cabinet fires
  - **Three cabinets, 3207, 3211 and EFP-1, that are located on the north section of the A ES Bus.**
  - **One Cabinet, 3205, located on the south section of the A ES Bus.**

Establishing Ventilation Cooling

- Appendix R Fire Study
  - Mechanical Hydraulic Timeline
  - Identifies time critical functions to ensure safe shutdown
  - Meeting the time line is one of the methods of establishing the feasibility of manual actions
- Engineering Evaluation 61671
  - Evaluated margin HVAC Calculation
  - Temperature Rise timeline modeled
  - Critical equipment design temperatures are not exceeded for 140 minutes

Establishing Ventilation

- Summary
  - For a fire in the "A" ES 4160V Switchgear Room, the loss of ventilation will cause the temperature to increase in the Control Complex
  - Modeling of the Control Complex shows that EFIC Room equipment will not be challenged for at least 140 minutes after loss of all ventilation.
  - 120 minutes to reset lockout relay provides additional 20 minutes to restore ventilation

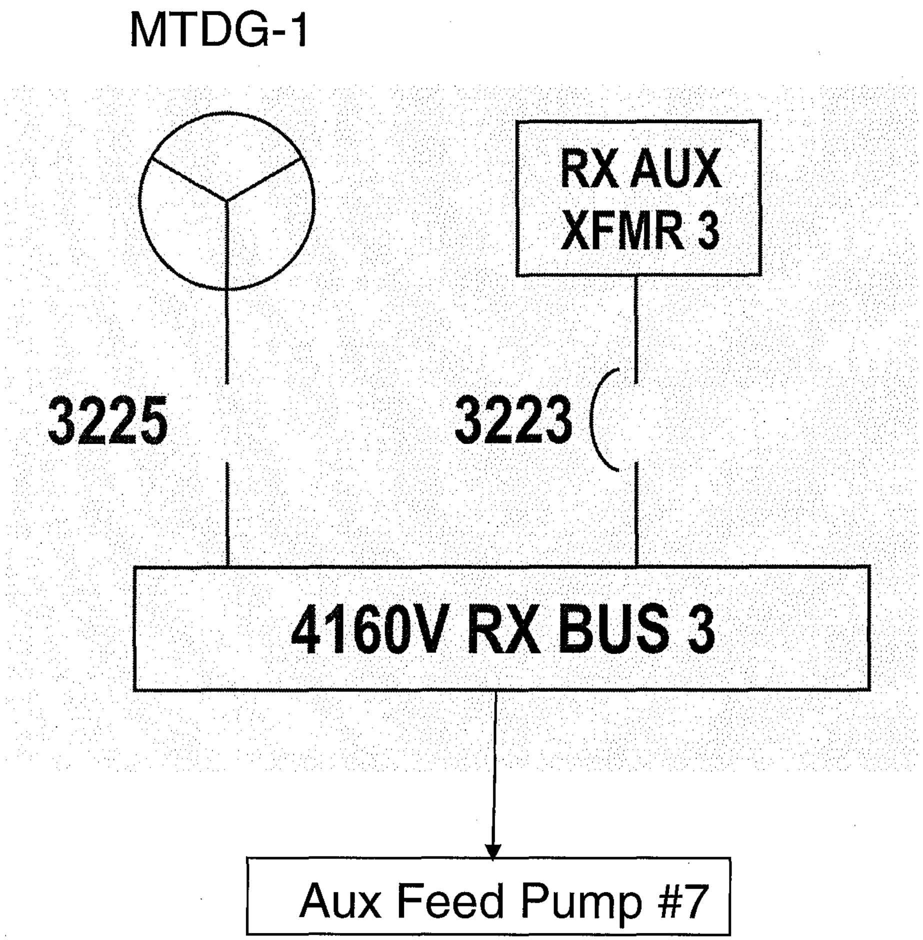
Fire Model

- Conditions of Habitability in the Switchgear Rooms
- A Fire Model was prepared by an independent consultant
  - Modeled the conditions in the “A” 4160V Switchgear Room for credible fire scenarios
  - Evaluated the habitability of the “B” 4160V Switchgear Room

Fire Model

- Results of the Fire Model:
  - No Hot Gas Layer formed
  - Visibility restored within 60 minutes except for smoldering fire
  - Toxic gas and oxygen levels remain acceptable in the “B” Switchgear Room

Auxiliary Feed Water Pump – FWP-7



Auxiliary Feed Water Pump Circuits

- **Engineering Disposition 60385 evaluated:**
  - **Power and control circuits for FWP-7**
  - **Power and control circuits for MTDG-1**
- **Conclusions:**
  - **FWP-7 and MTDG-1 power and control circuits remain free from fire damage**
  - **Can be started from the control room**

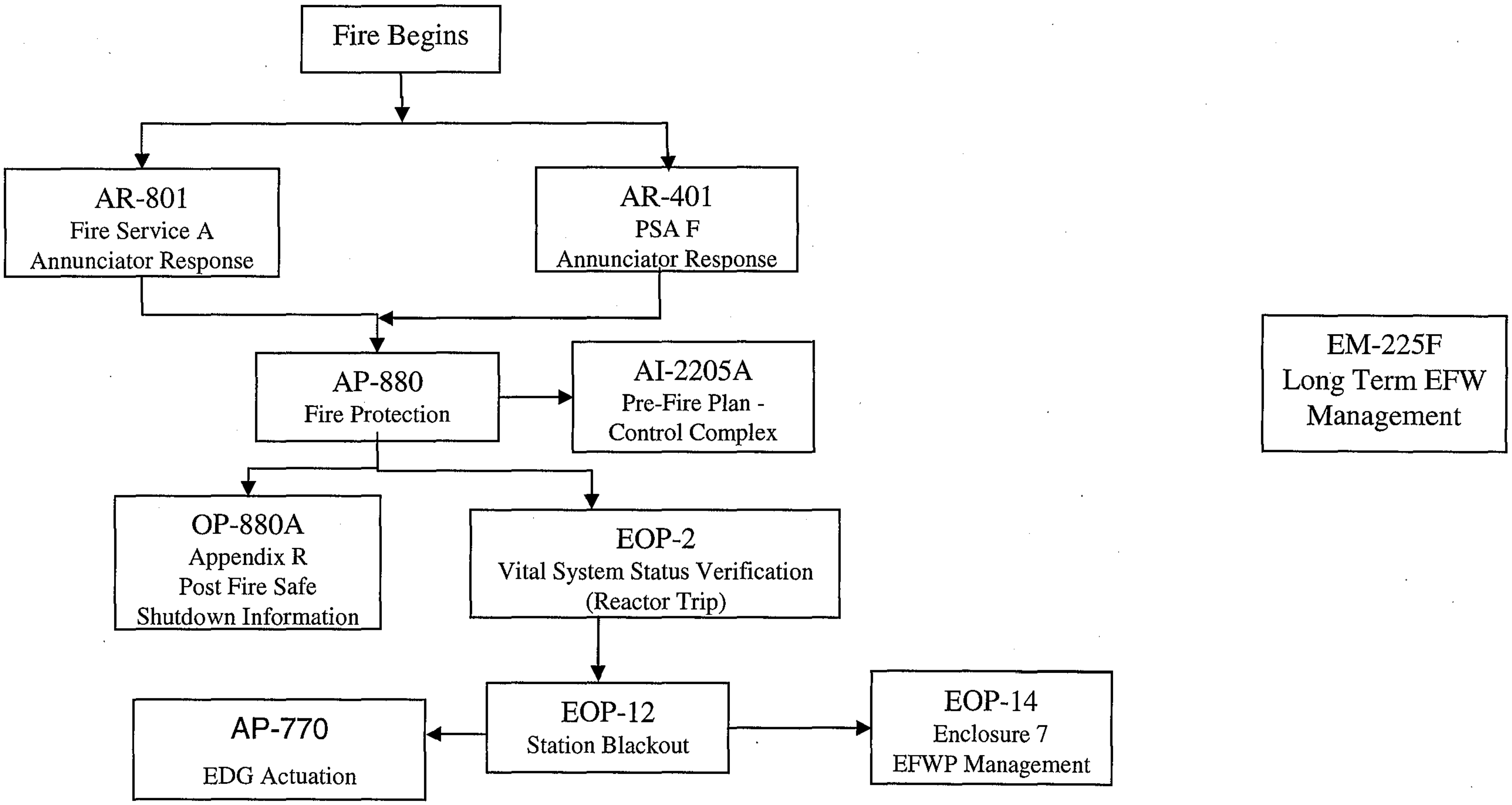
Electrical Distribution and Physical Layout Summary

- CR3 has a robust switchyard
- CR3 has modified the protective relaying circuits by removing the watt-hour meter, thus removing the single failure mechanism
- The modeling of the control complex temperatures shows that there is time available to accomplish the manual action.
- Fire modeling supports the ability of the operator to reset the lockout in the “B” Switchgear Room
- FWP-7 and its emergency power source MTDG-1 are unaffected by fires in the “A” Switchgear Room

Fire Response

- Five Man On-Site Brigade
  - Team Leader is a Licensed Operator
  - Cart Driver is a Non licensed Operator
- Site Emergency Response Coordinator
  - Responds to provide assistance and act as Emergency Medical Technician
- Security provides scene control
- Local Fire Departments
  - Provides backup support

Response Procedures



T0 - T2 Plant Response

- Times used are conservative for EFIC room ventilation restoration (12 minutes was added for safety significance determination of fire)
- Alarm actuates
  - Control Room (CR) Reviews Annunciator Response Procedure (AR) -801
  - CR Reviews AR-401
- Fire Team Leader (FTL) responds to scene
  - Notifies CR of local conditions
  - Will attack with extinguisher if conditions warrant

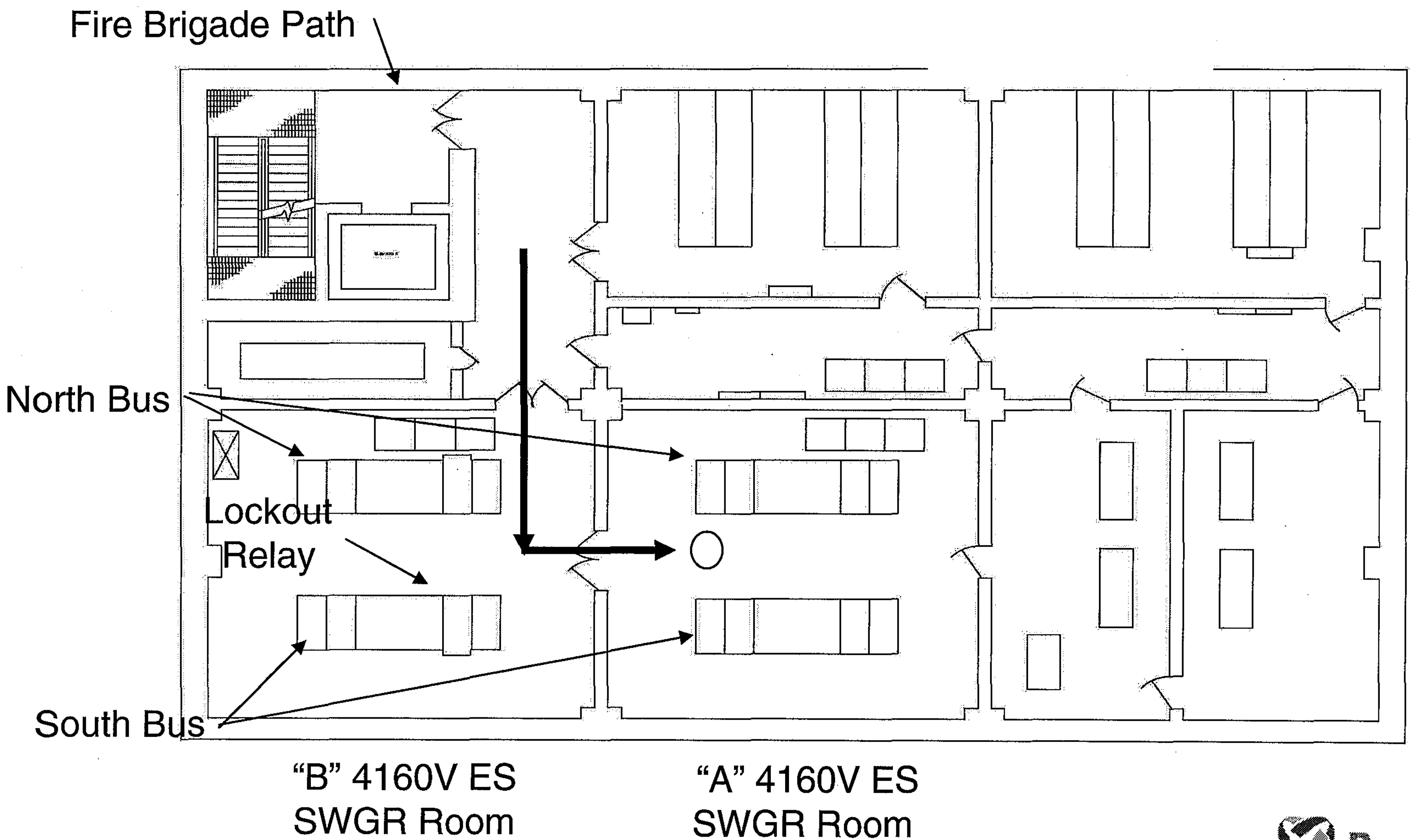
T3 – T5 Plant Response

- CR Enters Abnormal Procedure (AP) -880, Fire Protection and performs the following:
  - Sound fire alarm/muster Fire Brigade
  - Secure ventilation
  - Isolate PORV

T5 – T10 Plant Response

- **AP-880 –** Secondary Plant Operator (SPO) Charges fire header for Control Complex
- **AP-880 -** CR Closes Borated Water Storage Tank (BWST) valves
- **AP-880 -** CR Transfers both ES 4160V Buses to Offsite Power Transformer
  - **FTL will request "A" ES 4160V de-energization**
- Fire Brigade is dressed with Primary hose charged
  - **Secondary hose being charged**

Control Complex - 108' Elevation



T10-T15 Fire Brigade Response

- Primary team enters "A" ES 4160V SWGR room with fog nozzle.
  - Second nozzle man trained to carry extinguisher
- Secondary team is in ready status at muster area with charged backup line
- Limiting extinguishing time is smoldering fire
  - Takes ~ 20 minutes to extinguish
  - Requires opening upper cabinets to locate fire

T10 -T15 Plant Response

- Trip reactor if fire is impacting safe operation
- Perform EOP-2, Reactor Trip, Immediate Actions
  - **Ensure Reactor is shut down**
  - **Ensure Turbine valves are closed**
- Transition to EOP-12, Station Blackout
- **AP-880 Enclosure 1** CR Initiates both Trains of EFW
- **AP-880 Enclosure 1** CR Isolates Main feedwater and Main steam to both steam generators

T15-T20 Plant Response

- **EOP-12** CR Isolates Main Steam to both steam generators
- **EOP-12** Isolate losses to reactor coolant system
- **EOP-12** CR Ensures EFW is operating (EFP-3, EFP-2 or FWP-7)
  - **FWP-7 and its diesel (MTDG-1) can be started and controlled from Control Room**
- **EOP-12** SPO Aligns Backup air to atmospheric dump valves
- **EOP-12** CR Manages battery loads

Emergency Feedwater (EFW) and Auxiliary Feedwater (AFW) Systems

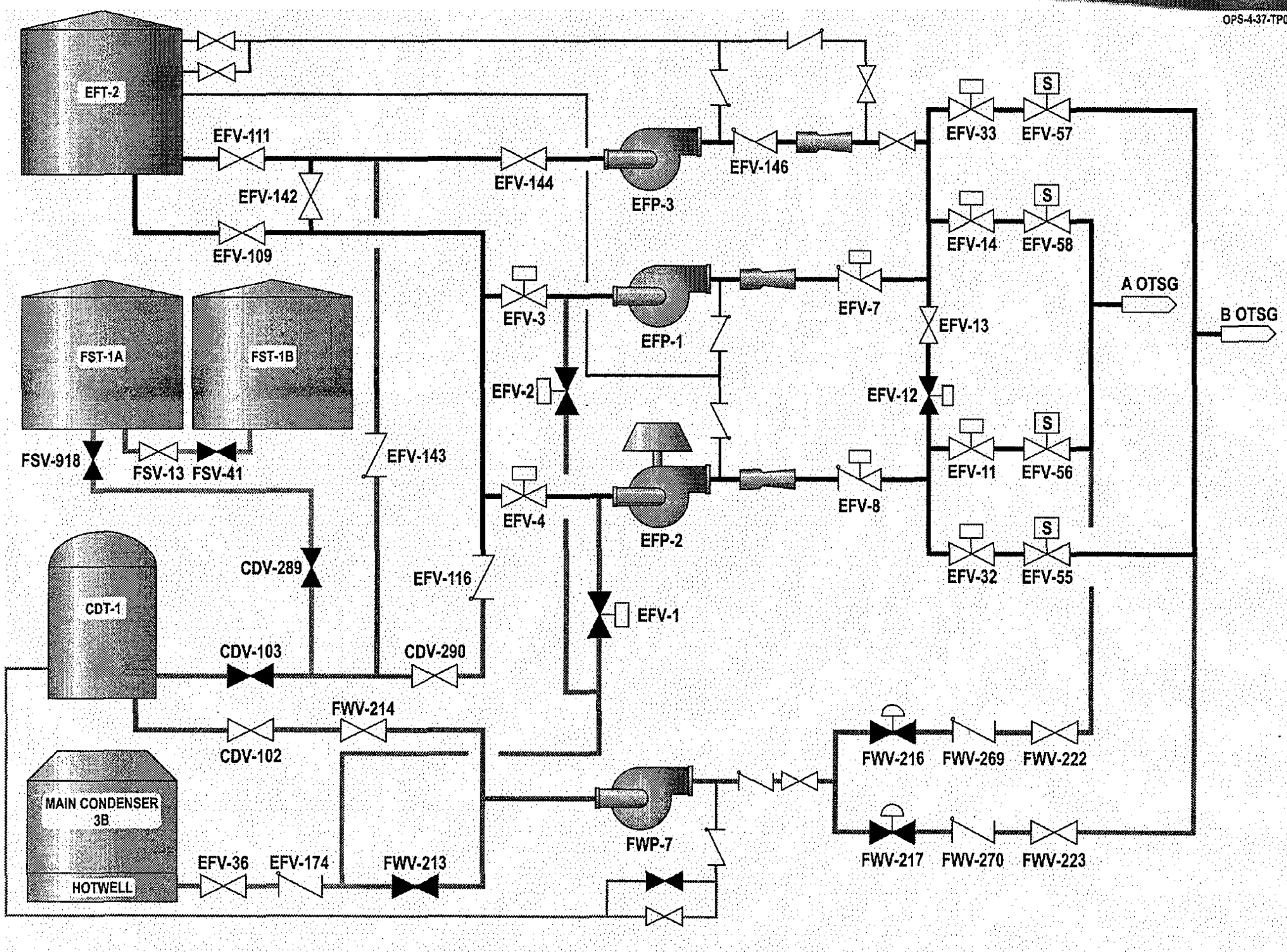


FIGURE 1 - EMERGENCY AND AUXILIARY FEEDWATER

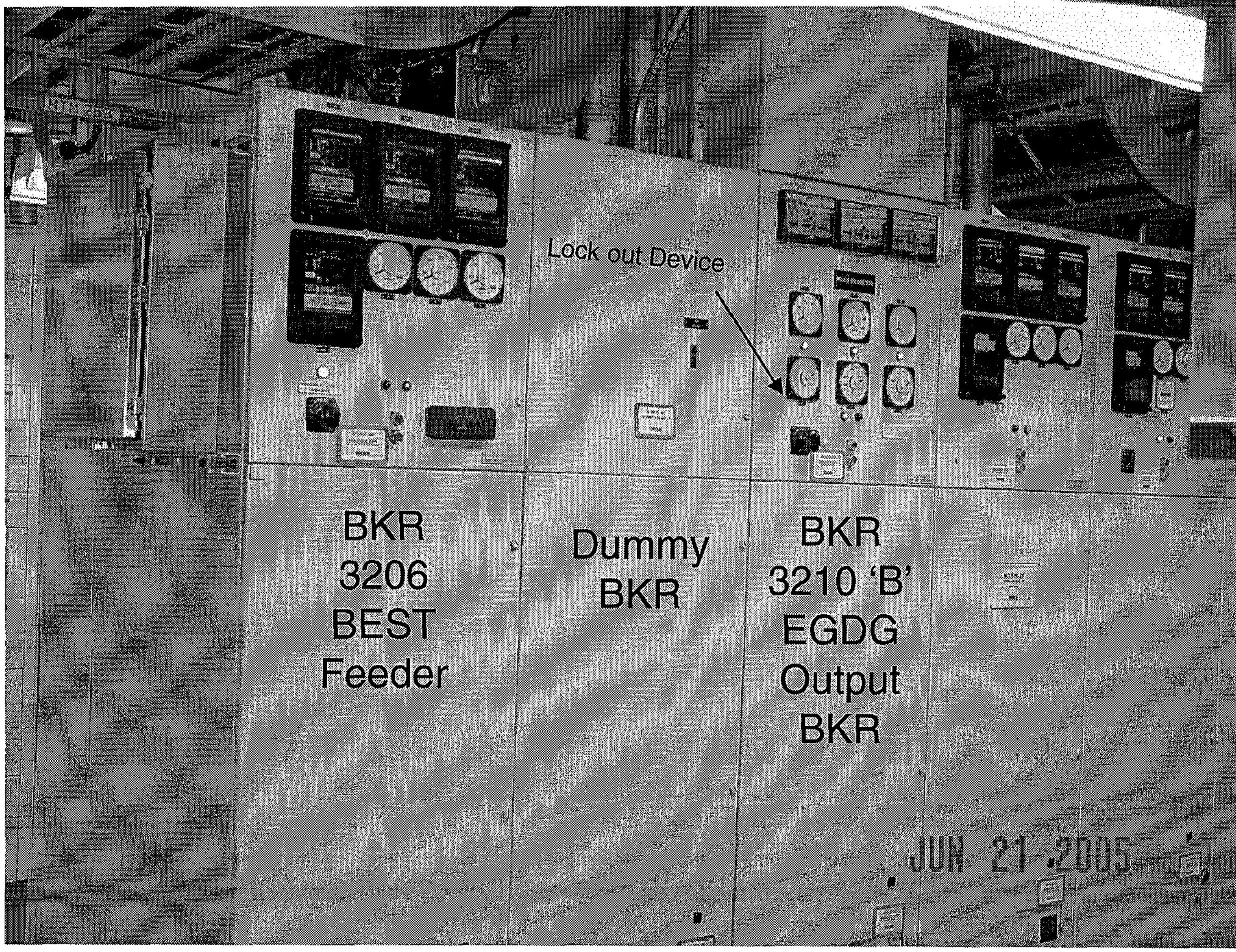
T20 –T35 Plant Response

- **OP-880A** PPO aligns EFP-2 flow path to prevent spurious valve closure (T20)
- **OP-880A** PPO aligns EFP-3 flow path to prevent overfill (T32)
- **OP-880A** PPO Opens Breakers for BWST valves (T35)
- Fire is out (T35)

T35-T60 Plant Response

- **OP-880A** PPO is available to reset "B" EDG Lockout (T-37)
  - Smoke should clear to 4 ft visibility in 20 minutes after SWGR room door is closed
  - If habitability of room is impaired, the PPO has SCBA in local area and full bunker gear available in Fire Brigade dress out area
  - IF "B" ES 4160V SWGR room is inaccessible for PPO, CR would notify FTL to have Cart Driver (Operator) perform action

Control Complex 108' Elevation "B" 4160V SWGR – South Bus



Operator Manual Action

- Only two of this type lockouts in "B" ES 4160V SWGR room
  - Second is for HPI pump ES select
- Proper lockout operation provides immediate feedback (EDG output breaker closure)
  - IF lock out reset is unsuccessful, task can be re-performed
- Fire brigade members are in electrically rated boots.
- High voltage gloves are staged just outside SWGR rooms

Operator Manual Action

- Post Fire Room Conditions
  - **Smoke diminishing**
    - **Natural or forced ventilation**
  - **Water in SWGR room is less than 1”**
    - **Trained to use Primary hose to divert water to hallway**
    - **Water drains to Control Complex stairwell**
    - **Water absorbing devices are on fire cart for water management**
  - **Could be steam in atmosphere**
    - **Trained to minimize time “B” to “A” SWGR door is opened**

Operator Manual Action

- Establishing EFIC Room Cooling
  - Following Power restoration
    - **CR starts EFIC room fan (1 minute)**
    - **SPO starts Appendix R Chiller (5 minutes)**
    - **Total time for EFIC ventilation restoration is less that 66 minutes from fire initiation**

Technical Support Center

- Staffed at maximum of 75 minutes
- Provides support and guidance outside of EOPs and APs
- EM-225F provides guidance for diverse EFW/AFW lineups (EFP-3)
- Provide guidance for electrical distribution alignment

Summary

- Reset of “B” EDG lockout is feasible
- Restoration of EFIC room ventilation can be accomplished well before equipment temperature limits are exceeded
- Primary heat removal is maintained with EFP-2
- FWP-7 provides a readily available source of backup to emergency feedwater
- EFP-3 and Offsite Power available via Technical Support Center guidance

PSA Model Inputs and Methodology

PSA Analysis

- Fire Modeling
- Initial Conditions
- Initiator Selection
- Appendix R Procedure Impacts
- Human Reliability Analysis (HRA)
- Core Damage Frequency
- Conservatisms
- Sensitivities

PSA Model Inputs and Methodology

Fire/Smoke Model

- Considered Thermal and High Energy Fires
- Suppression times assumed out to 35 minutes from alarm
- Habitability ("Cleared") conditions based on:
  - visibility (4ft)
  - carbon monoxide (500 ppm)
  - oxygen (16%)
  - temperature (116F)
  - radiant heat flux (2.5kW/m2)

PSA Model Inputs and Methodology

Fire/Smoke Model

- Thermal Fires
  - 200kw and 65kw
  - Initial Damage limited to cubicle (can propagate)
  - No hot gas layer (HGL)
  - Smoke "cleared" within 60 minutes for all cases except smoldering fires
- High Energy Arcing Faults (HEAF)
  - All targets within 3ft (H) and 5ft (V) are failed at T=0
  - No HGL
  - Smoke "cleared" within 60 minutes

PSA Model Inputs and Methodology

Initial Conditions

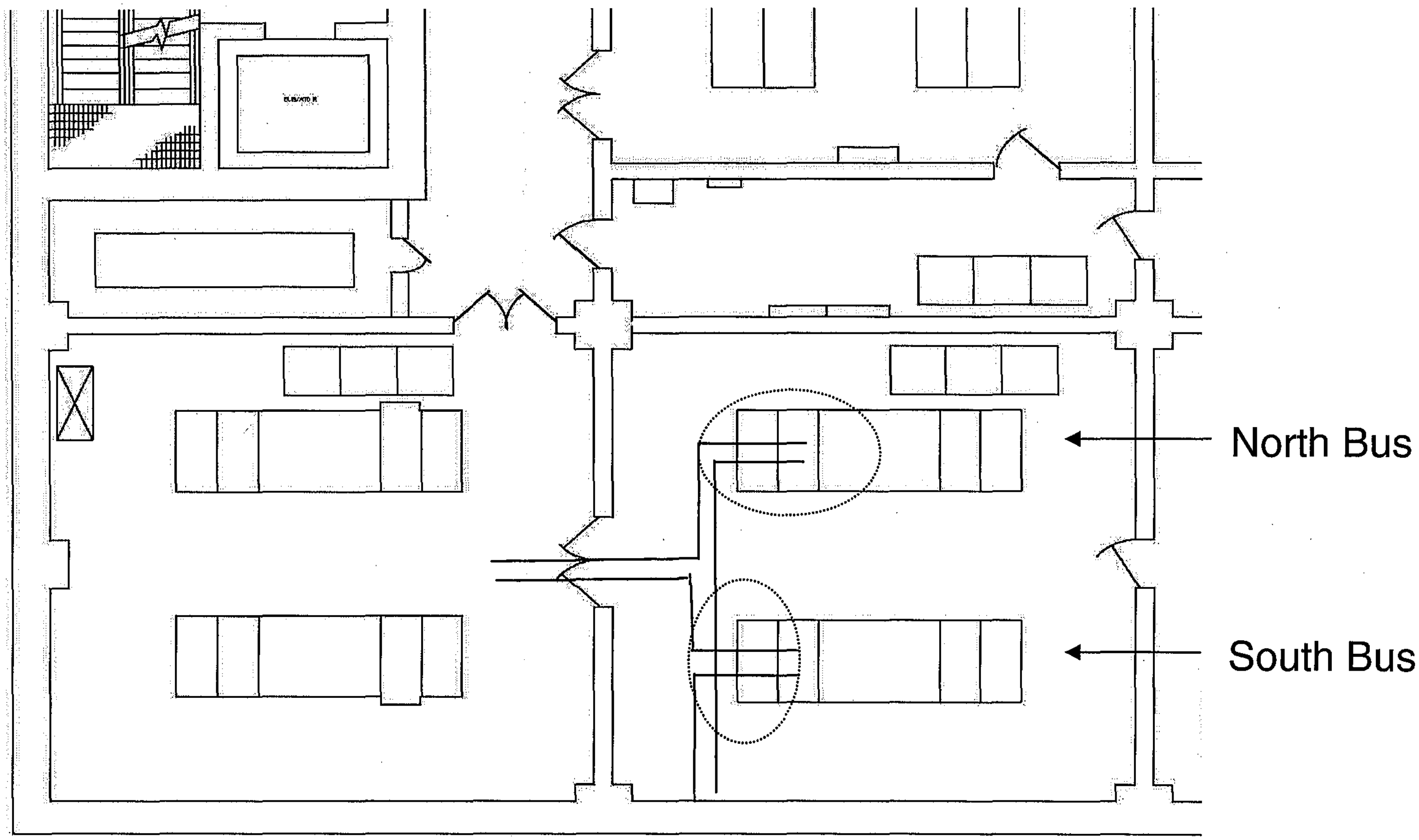
- On-line 100% power
- "A" 4160V ES Bus aligned to OPT (BKR 3211)
- "B" 4160V ES Bus aligned to BEST (BKR 3206)
- Operating equipment
  - MUP-1B
  - RWP-1, SWP-1C (non-safety related)
  - "A" train HVAC

PSA Model Inputs and Methodology

Initiator Selection

- FMEA of single failure scenarios was performed
- Abnormal bus alignments can be screened out based on time spent in these configurations (<1%)
- With normal bus alignment the fire must create:
  - ES “A” bus fault
  - CT path open with ground present on ESA side of OPT circuits
- Initiators limited to cubicles containing or close to the CT circuits connecting the OPT feeds to breakers 3211 & 3212

PSA Model Inputs and Methodology



"B" 4160V ES
SWGR Room

"A" 4160V ES
SWGR Room

PSA Model Inputs and Methodology

Two fire initiators modeled

- **Fire 1 – North Bus Breaker cubicles 3207,3211,EFP-1**
  - **HEAF and Thermal fires (1.86E-04/yr)**
    - **Conservatism, HEAF in 3207 is less likely based on data**
    - **Conservatism, Thermal fire in EFP-1 cubicle needs to propagate**
  - **Fails both ES buses at T=0**
    - **Control Complex HVAC stops**
    - **No Makeup (incl. RCP seal injection)**
    - **Emergency Diesels can not load due to fault**
    - **Plant trip assumed (manual or 3207 protective circuitry)**
    - **Startup transformer continues supplying offsite power to unit loads (RWP-1, SWP-1C ,RCPs, Battery Chargers, IA, MFW)**
    - **BEST available**

PSA Model Inputs and Methodology

Two fire initiators modeled (cont.)

- Fire 2 – South Bus Breaker cubicle 3205
  - HEAF fire only (1.42E-05/yr)
    - **Conservative, HEAF is less likely based on data**
  - Fails both ES buses at T=0
    - **Control Complex HVAC stops**
    - **No Makeup (incl. RCP seal injection)**
    - **Emergency Diesels can not load due to fault**
    - **Loss of Startup transformer**
    - **OPT available**

PSA Model Inputs and Methodology

Other modeled impacts due to Appendix R Fire Procedures

- EFP-3 injection lines closed and de-energized
- PORV-block closed and de-energized
- MSIVs closed, MFW tripped

PSA Model Inputs and Methodology

HRA Impacts

- **No credit for local actions outside control room**
  - **EFP-3 recovery due to HVAC**
  - **Local start/control of FWP-7**
- **Reduced Credit for time critical control room actions**
  - **Early start of FWP-7 to limit RCS re-pressurization**
  - **Trip RCPs following loss of SW cooling**
- **Appendix R actions**
  - **Restore "B" ES power by resetting EGDG-1B lockout**
- **TSC actions**
  - **EFP-3 (if EFP-2 and FWP-7 unavailable)**
  - **Offsite Power (if Diesel generator unavailable)**

PSA Model Inputs and Methodology

Timeline for HRA

- T=0 min., fire initiation/alarm, AP-880
- T=12 min., diagnosis compete, enter EOPs, trip RX
- T=18 min., operator dispatched to perform Appendix R manual actions
- T=35 min., fire extinguished
- T=37 min., operator available to reset lockout
    - Typically simple action (< 1 min to perform), complicated by environmental conditions
    - Fire brigade members available to assist, Qualified operators
    - Smoke "cleared" @ T=60 for most cases
- T=60 min., lockout reset ("B" 4160V power restored)
    - EGDG-1B operation may be impacted
- T=66 min., EFIC room cooling restored
- T=75 min., TSC operational
    - Begin efforts to align offsite power if EDG unavailable
- T=120 min., last opportunity to restore EFIC cooling
- T=140 min., EFIC failure (ends credit for EFP-2)
    - Start FWP-7 (EOP action)
    - Attempt other recovery (TSC support)
- T=200 min., Core damage 1 hr after loss of all core cooling

PSA Model Inputs and Methodology

Appendix R Manual Action

- Timeline
  - Tsw = 120 minutes
  - $T_{1/2}$ = 12 minutes
  - Tm = 48 minutes
- Probabilities
  - 1.0E-01 (typical screening value)
  - 6.7E-02 (traditional HRA methodology, with unfavorable PSFs to account for fire condition)
  - 4.4E-02 (credit applied for fire brigade assistance*)
  - 2.1E-02 (unfavorable PSFs, no fire complications)

PSA Model Inputs and Methodology

TSC Recovery Actions

- EFP-3, (EM-225F)
  - Open EFV-12,13 to feed through “B” train injection path
  - Open EFV-14,33 to feed through “A” train injection path
- BEST, (AP-770, OP-880A)
  - Available for fire scenarios involving North “A” bus
  - Availability obvious due to continued operation of Startup Transformer
  - Simple control room action
- OPT, (AP-770, OP-880A)
  - Available for fire scenarios involving South “A” bus
  - Availability would need to be deliberately determined
  - Simple control room action
- Completion any of these actions within 1 hour from loss of core cooling (0.3)

PSA Model Inputs and Methodology

Conservatisms

- Fire frequencies
  - not all modeled fires will create the subject faults
    - Smoldering fires (high smoke production) are less likely to cause the fault before suppressed
    - propagation of low energy fires between cabinets is less likely before suppression
  - HEAFs in normally open breakers less likely
- 4 hour battery life
  - CR3 2004 LOOP event demonstrated > 8hrs (non-1E)

PSA Model Inputs and Methodology

CDF = 1.47E-07/yr

- Emergency Diesel available
- Initiating Event Frequency (2.0E-04)
- Appendix R manual action (4.4E-02)
  - Fire brigade assistance credited
- FWP-7 (EOP directed, HEP = 5.6E-03)
  - Full credit for control room action
- Other recoveries (TSC support, HEP = 0.3)
  - EFP-3
  - Offsite power

PSA Model Inputs and Methodology

- Sensitivities (base = 1.47E-07/yr)

1	No propagation of low energy fires between cabinets (65kw fire in EFP-1 cubicle)	**1.14E-07**
2	Decreased HEAF in 3205, 3207 (50%)	**1.41E-07**
3	FWP-7 HEP x10	**2.80E-07**
4	App. R screening credit (HEP = 0.1 * 0.3)	**3.34E-07**
5	No App. R credit (HEP= 1.0 * 0.3) (same as failed emergency diesel)	**3.34E-06**
6	No TSC Recovery credit (HEP = 4.4E-2)	**4.90E-07**
7	Cases 5 and 6 (HEP = 1.0)	**1.13E-05**
8	Cases 5 and 3	**6.37E-06**

Conclusions

- Unit Auxiliary Loads lost in only one fire scenario
- At least 120 minutes available before EFIC is inoperable
  - Room conditions able to be improved, or more time for dress-out
  - Time for repeated attempts to reset the EDG lockout
- Auxiliary Feedwater and EFP-2 remain available – secondary side heat removal not lost
- EFP-3 can be restored with TSC Guidance
- Operator action is simple, trained on, proceduralized, and provides immediate feedback
- Fire brigade members may be used for manual action after fire out
- Offsite power can be restored if EDG unavailable

Closing Remarks

AGENDA

OPEN REGULATORY CONFERENCE

CRYSTAL RIVER NUCLEAR PLANT

JULY 22, 2005

NRC REGION II OFFICE, ATLANTA, GA.

I. OPENING REMARKS, INTRODUCTIONS AND MEETING INTENT
   Mr. V. McCree, Director, Division of Reactor Safety

II. NRC REGULATORY CONFERENCE POLICY
   Mr. V. McCree, Director, Division of Reactor Safety

III. STATEMENT OF THE ISSUE WITH RISK PERSPECTIVES
   Mr. V. McCree, Director, Division of Reactor Safety

IV. SUMMARY OF APPARENT VIOLATIONS
   Mr. V. McCree, Director, Division of Reactor Safety

V. LICENSEE RISK PERSPECTIVE PRESENTATION

VI. LICENSEE RESPONSE TO APPARENT VIOLATIONS

VII. BREAK / NRC CAUCUS
   Mr. V. McCree, Director, Division of Reactor Safety

VIII. CLOSING REMARKS
   Mr. V. McCree, Director, Division of Reactor Safety