UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

August 21, 2001

MEMORANDUM TO: ACRS Members

FROM: Amarjit Singh, P.E., Senior Staff Engineer
ACRS/ACNW Technical Support Staff

SUBJECT: CERTIFICATION OF THE MINUTES OF THE ACRS SUBCOMMITTEES PLANT OPERATIONS AND FIRE PROTECTION, JUNE 28, 2001, ARLINGTON, TEXAS

The minutes of the subject meeting, issued on July 12, 2001, have been certified as the official record of the proceedings of the meeting. A copy of the certified minutes is attached.

Attachment: As stated

cc: J. Larkins
S. Bahadur
H. Larson

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

July 12, 2001

MEMORANDUM TO: Amarjit Singh, P.E., Senior Staff Engineer
ACRS/ACNW Technical Support Staff

FROM: John D. Sieber, Chairman
Plant Operations and Fire Protection Subcommittees

SUBJECT: CERTIFICATION OF THE MINUTES OF THE ACRS SUBCOMMITTEE MEETING, ON JUNE 28, 2001, AT REGION IV, ARLINGTON, TEXAS

I hereby certify that, to the best of my knowledge and belief, the Minutes of the subject meeting issued July 12, 2001, are an accurate record of the proceedings for that meeting.

John D. Sieber, Chairman

July 12, 2001
Date

GT260-185

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

July 12, 2001

MEMORANDUM TO: Mr. John D. Sieber, Chairman
Fire Protection and Plant Operations

FROM: Amarjit Singh, P.E., Senior Staff Engineer
ACRS/ACNW Technical Support Staff

SUBJECT: WORKING COPY OF THE MINUTES OF THE ACRS SUBCOMMITTEES MEETING ON FIRE PROTECTION AND PLANT OPERATIONS, JUNE 28, 2001, ARLINGTON, TEXAS

A working copy of the minutes for the subject meeting is attached for your review. I would appreciate your review and comments as soon as possible. Copies are being sent to the ACRS members who attended the meeting for information and/or review.

Attachment: As stated

cc: G. Apostolakis
D. Powers
G. Leitch
R. Uhrig

Issued: July 12, 2001
Certified By: John D. Sieber
July 12, 2001

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
MINUTES OF THE JOINT MEETING OF THE ACRS SUBCOMMITTEES
ON PLANT OPERATIONS AND ON FIRE PROTECTION
JUNE 28, 2001
REGION IV OFFICE, 611 RYAN PLAZA DRIVE SUITE 400
ARLINGTON, TEXAS

INTRODUCTION

The Advisory Committee on Reactor Safeguards (ACRS) Subcommittees on Plant Operations and Fire Protection held a joint meeting on June 28, 2001, at the Region IV Office, 611 Ryan Plaza Drive, Suite 400, Arlington, Texas. The purpose of this meeting was to discuss Region IV activities and other items of mutual interest, including significant operating events and fire protection issues. The entire meeting was open to the public. Mr. Amarjit Singh was the cognizant ACRS staff engineer for this meeting. The meeting was convened at 8:30 a.m. and adjourned at 3:30 p.m.

ATTENDEES

ACRS

G. Apostolakis, Chairman
D. Powers,Member
G. Leitch, Member
J. Sieber, Member
R. Uhrig, Member
J. Larkins, ACRS Staff
H. Larson, ACRS Staff
A. Singh, ACRS Staff
M. Weston, ACRS Staff
I. Schoenfeld, EDO Staff

NRC Region IV Participants

P. Gwynn, Deputy Regional Administrator
K. Brockman, Director, Division of Reactor Projects (DRP)
A. Howell, Director, Division of Reactor Safety (DRS)
K. Kennedy, Senior Reactor Analyst (SRA), DRS
T. Pruett, SRA, DRS
G. Pick, Senior Project Engineer, DRP
G. Good, Chief, Plant Support Branch, DRS

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

SUBCOMMITTEES MEETING ON PLANT OPERATIONS AND FIRE PROTECTION

JUNE 28, 2001
Date

NRC STAFF SIGN IN FOR ACRS MEETING
PLEASE PRINT

NAME	NRC ORGANIZATION
Omid Tabatabai	NRC/NRR/DRIP/RLSB
MAGGALEAN W. WESTON	NRC/ACRS
Amarjit Singh	NRC/ACRS/ACNW
John D. Sieber	NRC/ACRS
GEORGE APOSTOLAKIS	ACRS
DANA A. POWERS	ACRS
JOHN T. LARKINS	ACRS
GRAHAM M. LEITCH	ACRS
ROBERT E. UHRIG	ACRS
HOWARD J. LARSON	ACRS/ACNW
Isabelle Schoenfeld	OEDO/NRC
Jeffrey Clark	NRC/R-IV
Charles Marschall	NRC/R-IV
Kriss M. Kennedy	NRC/RIV
DAVID LOVELESS	NRC/RIV
Clyde Osterholtz	NRC/RIV
Wayne Sifre	NRC/RIV
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TROY PRUETT	NRC/RIV

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

SUBCOMMITTEES MEETING ON PLANT OPERATIONS AND FIRE PROTECTION

JUNE 28, 2001
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Jeff Moreno	DRP/Branch E/Engineering Associate
Alberto Garcia	DRS- EMB
Greg Pick	DRP/Branch E/Senior Proj Eng
Don Allen	DRP / Branch A / Senior Project Eng.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

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JOINT MEETING OF THE ACRS SUBCOMMITTEES
ON PLANT OPERATIONS AND FIRE PROTECTION
JUNE 28, 2001
REGION IV OFFICE, 611 RYAN PLAZA DRIVE
ARLINGTON, TEXAS

ACRS CONTACTS:
Amarjit Singh
301-415-6899
Maggalean W. Weston
301-415-3151

REGION IV CONTACT
Greg Pick
817-860-8270

- PROPOSED SCHEDULE -

TOPIC	PRESENTER	TIME
1. Opening Remarks by the Subcommittee Chairman	Sieber	8:30-8:35 A.M.
2. Region IV Organizational Responsibilities/Accomplishments	Gwynn	8:35-9:00 A.M ~~8:35-8:50 A.M.~~
3. Reactor Oversight Program Implementation	Brockman/Clark	9:00-10:35 A.M ~~8:50-9:50 A.M.~~
4. Senior Reactor Analyst Role In Risk Assessment Significance Determination Process Implementation in Region IV	Kennedy/Pruett	~~9:50-10:20 A.M.~~ 10:50-11:36 A.M
	BREAK	10:35-10:50 A.M ~~10:20-10:35 A.M.~~
5. Plant Operations Experience in IV	Brockman	~~10:35-11:30 A.M.~~
Scram Trends	Johnson	
Callaway ALARA	Good	
Callaway Grid Experience	Pruett	
	LUNCH	~~11:30-12:30 P.M.~~ 11:36-12:15 P.M

TOPIC	PRESENTER	TIME
6. Plant Experience in Region IV (Continued)	Brockman	~~12:30-1:15 P.M.~~ 12:15-1:55 P.M
California Grid	Marschall	
Electrical Design & Operations	Loveless/Clark	
Issues at Cooper		
		2:05-2:35 PM
7. Fire Protection Experience in Region IV	Howell	~~1:15-2:15 P.M.~~
SONGS Electrical Fire	Pick/Osterholtz	2:35-2:50 PM 1:55-2:05 pm
	BREAK	~~2:15-2:30 p.m.~~
8. Region IV Responsibilities Under COOP/COG	Andrews/Pellet	~~2:30-3:00 P.M.~~ 2:50-3:10 P.M
9. Closing Remarks	Sieber	~~3:00-3:15 P.M.~~ 3:10-3:30 PM
	ADJOURNMENT	~~3:15 P.M.~~ 3:30 PM

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

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JUNE 28, 2001
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Excerpt from Integrated Report Diablo Canyon 01-03

4. OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems (71152, 71153)

a. Inspection Scope

The licensee performed inspections of 12 kV Auxiliary Bus 2-1 and 4 kV Auxiliary Bus 2-2 as described in Licensee Event Report 50-275/2000-004-01 and prescribed in Action Request A0510971. (This event was also the subject of NRC Inspection Report 50-275; 323/00-09). During the inspection of 12 kV Auxiliary Bus 2-1, an Overheated Splice Joint (No. 309) was discovered (refer to Attachment 2). The inspectors reviewed work instructions, scope of the planned work activities, and field drawings. The inspectors evaluated the work history for the Unit 2 12 kV bus, evaluated previous corrective actions, and reviewed the design requirements for the 12 kV bus. The inspectors walked down the bus bars prior to disassembly to independently evaluated the condition of the buses. The inspectors interviewed craft personnel performing the work, system engineers implementing the corrective actions, design engineers who performed the root cause evaluation, and management personnel who were responsible for effective resolution of the bus bar work. The inspectors attended management meetings related to resolution of the bus bar deficiency.

b. Assessment

.1 Bus Bar Physical Condition

The licensee determined that an overheated splice joint was located on the center phase of the 12 kV bus bar from Auxiliary Transformer 2-1 to Buses D and E. The splice joint was located outside of the turbine building approximately 10 feet from the transformer. The inspectors verified that there were no other bus ducts in close proximity to the bus duct at the degraded splice joint. The inspectors noted that the adjacent joints on either side of the degraded joint had evidence of overheating, as well as the Raychem insulation on the center phase. As seen in Attachment 2, a second Splice Joint (307A) had also degraded because of overheating.

The inspectors evaluated the condition of additional splice joints inside the switchgear room since this location mirrored the Unit 1 configuration, which had previously failed (refer to Inspection Report 50-275; 323/00-09). The inspectors noted that the aluminum tee had copper bus bar routed to Switchgears D and E after the first joint. The inspectors determined that the zinc-chromate grease remained in good condition, the splice joints had both single and double splice plates, the joints had no evidence of overheating (contrary to the condition found outside the turbine building). Because of the good condition of the bus bar inside the Unit 2 switchgear room, the inspectors concluded that a fault on the Unit 2 12 kV Auxiliary Bus would not have been as severe as Unit 1 and would not have affected the startup buses.

.2 Work History

Unit 1

The inspectors reviewed the scope of work completed during Refueling Outage 1R10, as documented in Action Request A0510961. The inspectors confirmed that the licensee had completed the following work activities:

a. For Startup Bus 1-1 from Startup Transformer 1-1 to the Startup Switchgear (3750A), for nontaped connections full face 3x6" splice plates were installed and torqued to 50 ft-lbs; booted connections had Raychem boots installed to replace the polyvinyl chloride boots and taped connections at 90 degree joints were visually inspected for signs of overheating; and connections had new bolts installed including Belleville washers.

b. For Auxiliary Bus 1-2 from Auxiliary Transformer 1-2 to Switchgears D, E and Vital Buses (F, G, and H), measured the as-found torque on selected joints and torqued all joints to 50 ft-lbs; replaced the straight polyvinyl chloride boots with Raychem boots since 90 degree Raychem boots were not available; and replaced the installed aluminum bus bar with copper bus bar.

c. For Auxiliary Bus 1-1 (replaced in May 2000), the licensee identified, during planned inspection of the bus, that a Raychem boot had discolored because of heat at the bolt locations at the high point. Subsequently, the licensee installed ventilation louvers instead of the breather tube to eliminate heat at this high point.

The inspectors reviewed the completed work packages and noted the following: (1) the as-found torque values for a small number of joints were less than 20 ft-lbs; (2) the licensee identified the as-built configuration of all joints inspected; (3) the licensee identified the joints not inspected so that corrective actions could be implemented during a future inspection; and (4) identified some evidence of overheating as revealed by some leaching of elasticizer from a small number of bus bar joints. The inspectors considered these corrective actions reasonable considering the short period of time allowed to plan the work and procure the necessary parts for the unique configurations. The joints not inspected were either taped (i.e. had an unusual shape that required other than a straight Raychem boot) or were located in a fire barrier.

The inspectors determined that these corrective maintenance activities, combined with the operating conditions for the buses (continuously loaded and heavily loaded), reduced the risk of another fault occurring on Unit 1.

Unit 2

From review of the Unit 2 maintenance records, the inspectors determined that the licensee had worked on the affected joint in April 1996, while replacing the existing insulation with Raychem insulation. As indicated in Action Requests A0400090 and A0400295, the licensee found cracked insulation on the 12 kV buses. The licensee inspected the Unit 2 bus bars after finding cracked insulation pieces in the Unit 1 buses

following a 1996 transformer failure. The licensee found hairline cracking of the insulation on the 4 kV bus bars. The licensee taped the 4 kV bus bars with insulating tape in accordance with vendor recommendations since the hair line cracks did not affect the integrity of the bus bar insulation. The inspectors noted that the work order indicated that personnel torqued the joints to 50 ± 5 ft-lbs, which agreed with the vendor manual bolting requirements. The licensee inspected the condition of the 4 kV bus bar insulation cracks during this outage. The licensee noted no significant degradation.

In March 1998, the licensee inspected the protective boots in this bus duct to look for degradation. The records indicated that the inspections found the protective boots in good condition.

From May 3-6, 2001, the inspectors observed disassembly, inspection, and torquing of selected splice plates for the bus bars for Auxiliary Transformers 2-1 and 2-2. The inspectors verified that the scope of the planned work agreed with the commitments contained in Licensee Event Report 50-275/2000-004-01 and Nonconformance Report N0002112. The inspectors reviewed the as-found torque, micro-ohm, and splice joint condition for the joints sampled by the maintenance personnel. The inspectors noted that the joints did not indicate any significant deficiencies other than the joint that had been found overheated.

The inspectors determined that these corrective maintenance activities, once completed, combined with the operating conditions for the buses (continuously loaded and heavily loaded), reduced the risk of another fault occurring on Unit 2.

.3 Design Requirements

As described in Inspection Report 50-275; 323/00-09, the vendor could not provide qualification test reports to demonstrate viability of the various bus bar configurations at Diablo Canyon. The licensee concludes in Nonconformance Report N0002112 that the operating experience at Diablo Canyon provides the best basis for the ampacity ratings of the installed 3750A Bus Bars. This engineering judgement was provided after the licensee could not locate a test report that would support qualification of these bus bars. The licensee found no evidence that the bus bars had experienced any significant degradation. In addition, the 3750A Bus Bars are normally unloaded, rarely loaded, and the maximum design load is 3400A. Because of the lack of identified degradation and establishment of a preventive maintenance task to periodically inspect the bus bars, the licensee concluded that the bus bars could be used. The licensee concluded the preventive maintenance program combined with the refurbished bus bars in Refueling Outages 1R10/2R10, 1R11/2R11, and 1R12/2R12 provide assurance that any future degradation would be identified prior to reaching unacceptable levels.

.4 Corrective Actions

Following the May 2000, bus bar failure, the licensee established a priority for evaluating the buses based upon factors that increased the risk of a splice joint failure, as identified in Nonconformance Report N0002112. The licensee attributed the failure to a heavily loaded bus (relative to the rated load and to the size/area of the splice plates), whether the bus is continuously loaded or normally unloaded (the auxiliary versus the startup

buses), and inconsistent silver plating. Other interactions that may have contributed to the failure included off-gassing of the polyvinyl chloride boot. These conditions resulted in the following buses being identified as most susceptible: 12 kV buses from Auxiliary Transformer 2-1, 4 kV auxiliary buses for Units 1 and 2, and the 12 kV startup bus for Unit 1.

The inspectors noted that the licensee expressed surprise at the amount of overheating and degradation identified at Joint 309. The licensee concluded that the joint would not have lasted another operating cycle without failing. The licensee estimated that the joint could have failed within 4-6 months since the failure mechanism (increased resistance resulted in increased heating that increased the degradation/resistance, et cetera) was increasing exponentially. The inspectors concluded that the licensee had missed a failure by a small margin. The inspectors noted that had a failure occurred that the power supply to the reactor coolant pumps and circulating water pumps would have tripped. An anticipatory reactor trip would have occurred. The event would have resulted in a loss of load reactor trip with loss of normal heat removal. This type of failure is tracked by the performance indicators.

Following the failure of the Unit 1 bus bar in May 2000, the inspectors agreed with the licensee decision to delay inspecting Unit 2 since: (1) the bus bars had been recently torqued (5 years on Unit 2 versus 20+ years on Unit 1) and (2) an inspection 2 years previously had identified no degradation.

As of the end of this inspection, the licensee had not received the analysis results from their offsite vendor for the likely cause of Joint 309 overheating. The inspectors determined that the licensee had provided portions (e.g., bus bar pieces, splice plates, boot, and insulation) of both the overheated joints and a good joint for analysis. The inspectors will review the results of this analysis once completed for any additional regulatory response that may be required.

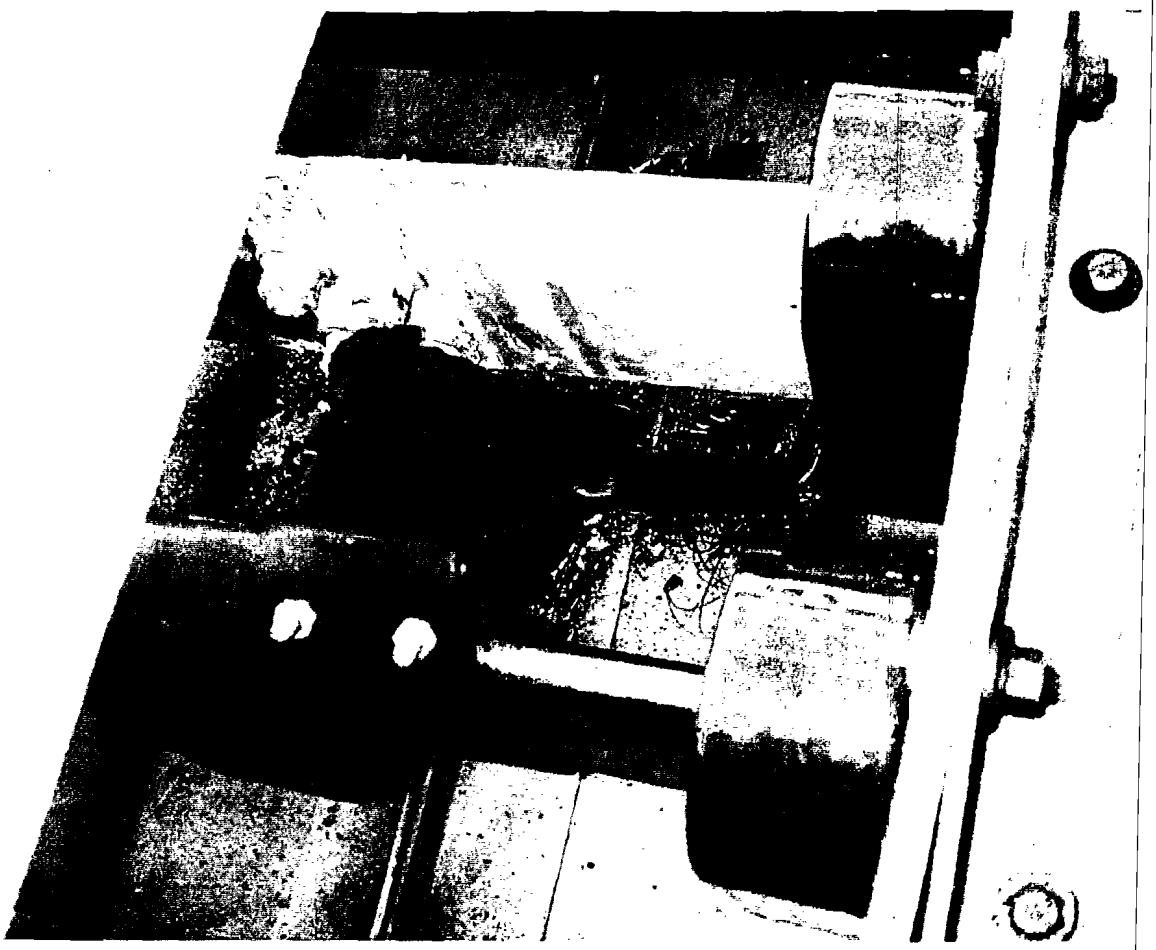
Work activities outstanding that need to be completed in Refueling Outages 1R11/2R11 include: (1) Startup Transformer 1-1 replace taped connections and use larger splice plates; (2) Auxiliary Transformer 1-2(2-2) to Buses D and E and vitals replace taped connections; (3) Startup Transformer 1-2(2-2) to Buses D and E and vitals replace taped connections; and (4) 12 kV startup switchgear to Startup Transformer 1-2(2-2) replace taped connections. The licensee documented the work scope for the Unit 1 and 2 outages on Action Requests A0510972 and A05100973, respectively.

.5 Risk Assessment

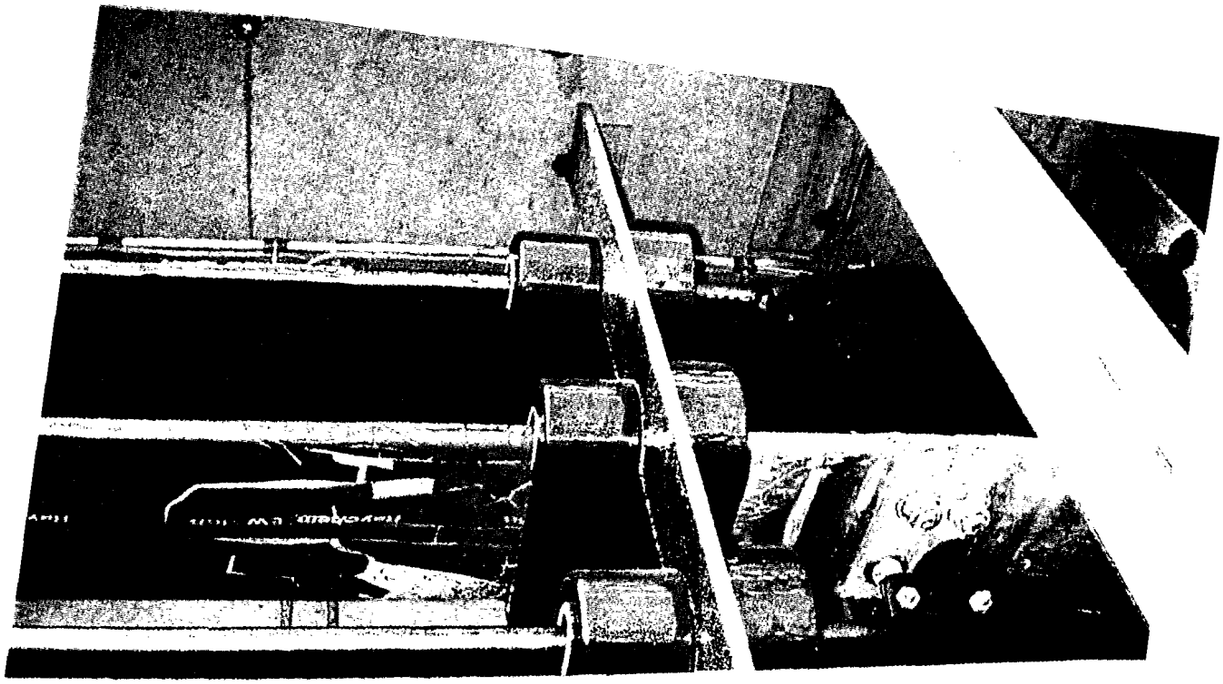
Had this degraded condition not been corrected, a failure would have resulted in a loss of load reactor trip with a loss of the power conversion system. This event would have been captured by the performance indicators for reactor scram with loss of the power conversion system.

c. Observations and Findings

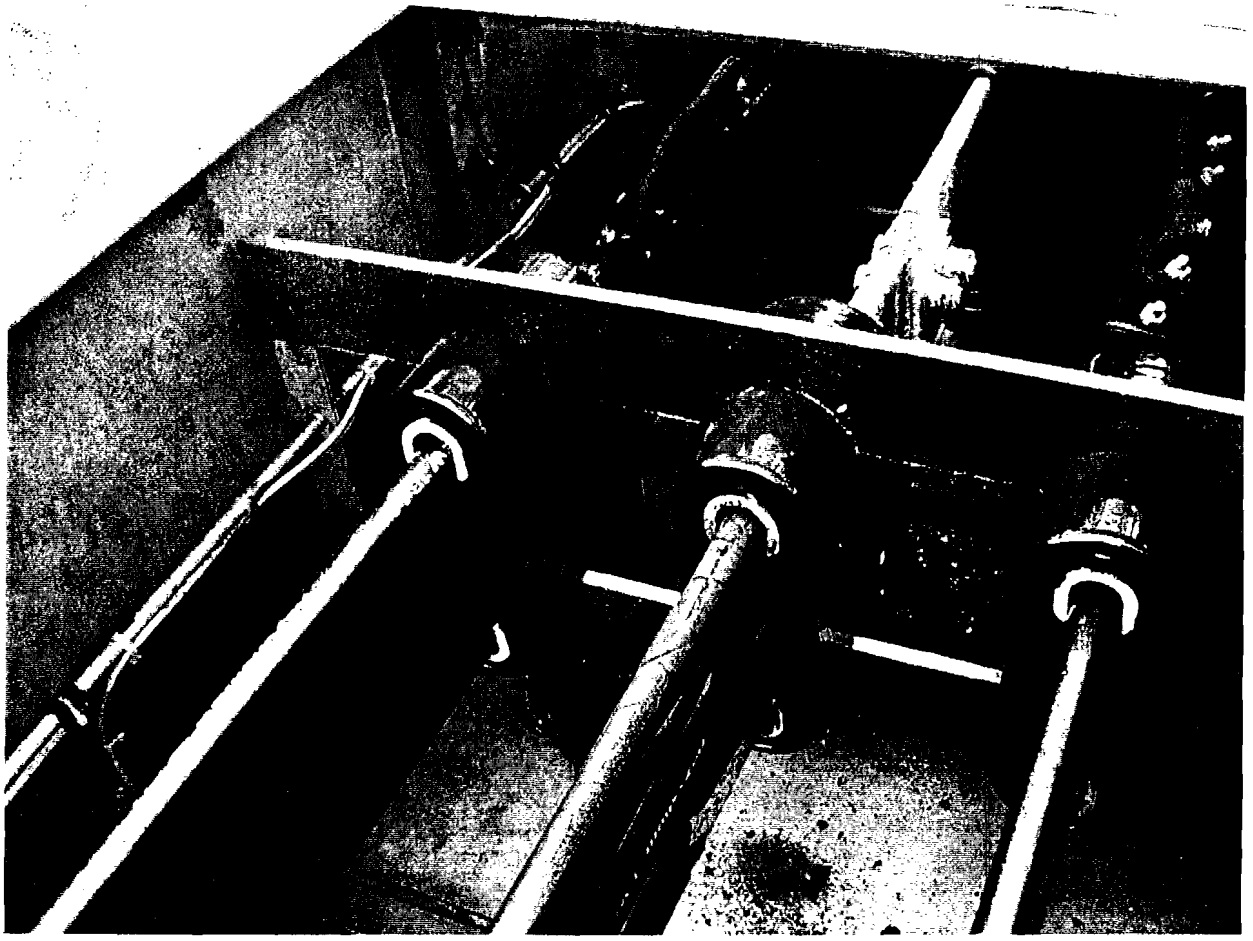
No findings of significance were identified.



View 1 of BUS 309 Showing degraded Insulation on Bus Bar.



View 2 of BUS 309 Showing degraded Insulation on Bus Bar



View 1 of BUS 309 Showing degraded Insulation on Bus Bar

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV

VISIT OF THE ACRS SUBCOMMITTEES ON PLANT OPERATIONS AND FIRE PROTECTION

JUNE 28, 2001

U. S. Nuclear Regulatory Commission
Region IV

- E.W. Merschoff, Regional Administrator, NRC Region IV
  - E. L. Williamson, Director, Office of Investigation
  - T. P. Gwynn, Deputy Regional Administrator
    - B. W. Henderson, Public Affairs Officer
    - K. D. Smith, Regional Counsel
    - C. A. Hackney, Regional State Liaison Officer
    - T. H. Andrews, Incident Response Coordinator
    - G. F. Sanborn, Director, Allegation Coordination / Enforcement Staff
    - D. D. Chamberlain, Director, Division of Nuclear Materials Safety
      - Linda Howell, Senior Materials Analyst
      - Chuck Cain, Chief, Nuclear Materials Licensing Branch
      - Mark Shaffer, Chief, Nuclear Materials Inspection Branch
      - Blair Spitzberg, Chief, Fuel Cycle & Decommissioning
    - A. T. Howell III, Director, Division of Reactor Safety
      - Dale Powers, Senior Technical Analyst
      - Charles Marschall, Chief, Engr. & Maint. Branch
      - Dale Powers, Chief, Operations Branch
      - Gail Good, Chief, Plant Support Branch
    - K. E. Brockman, Director, Division of Reactor Projects
      - Elmo Collins, Deputy Director
      - Phil Harrell, Chief, TSS
      - *David Graves, A/Chief, Branch A
        - Comanche Peak
        - South Texas
        - Grand Gulf
      - William Johnson, Chief, Branch B
        - Callaway
        - Wolf Creek
        - River Bend
      - *David Loveless, Chief, Branch C
        - Cooper NS
        - Fort Calhoun
        - San Onofre NGS
      - Linda Smith, Chief, Branch D
        - Arkansas NO
        - Palo Verde
      - William Jones, Chief, Branch E
        - CGS
        - Diablo Canyon PP
        - Waterford
    - K. J. Hamill, Director, Division of Resource Mgmt. & Admin.
      - Karen League, RPO, Human Resources Staff
      - John Pellet, Chief, Information Resource Management
      - Peter Krayer, Chief, Financial Resource Management Branch

Revised 06-26-2001
* Acting
+ Reporting in the future
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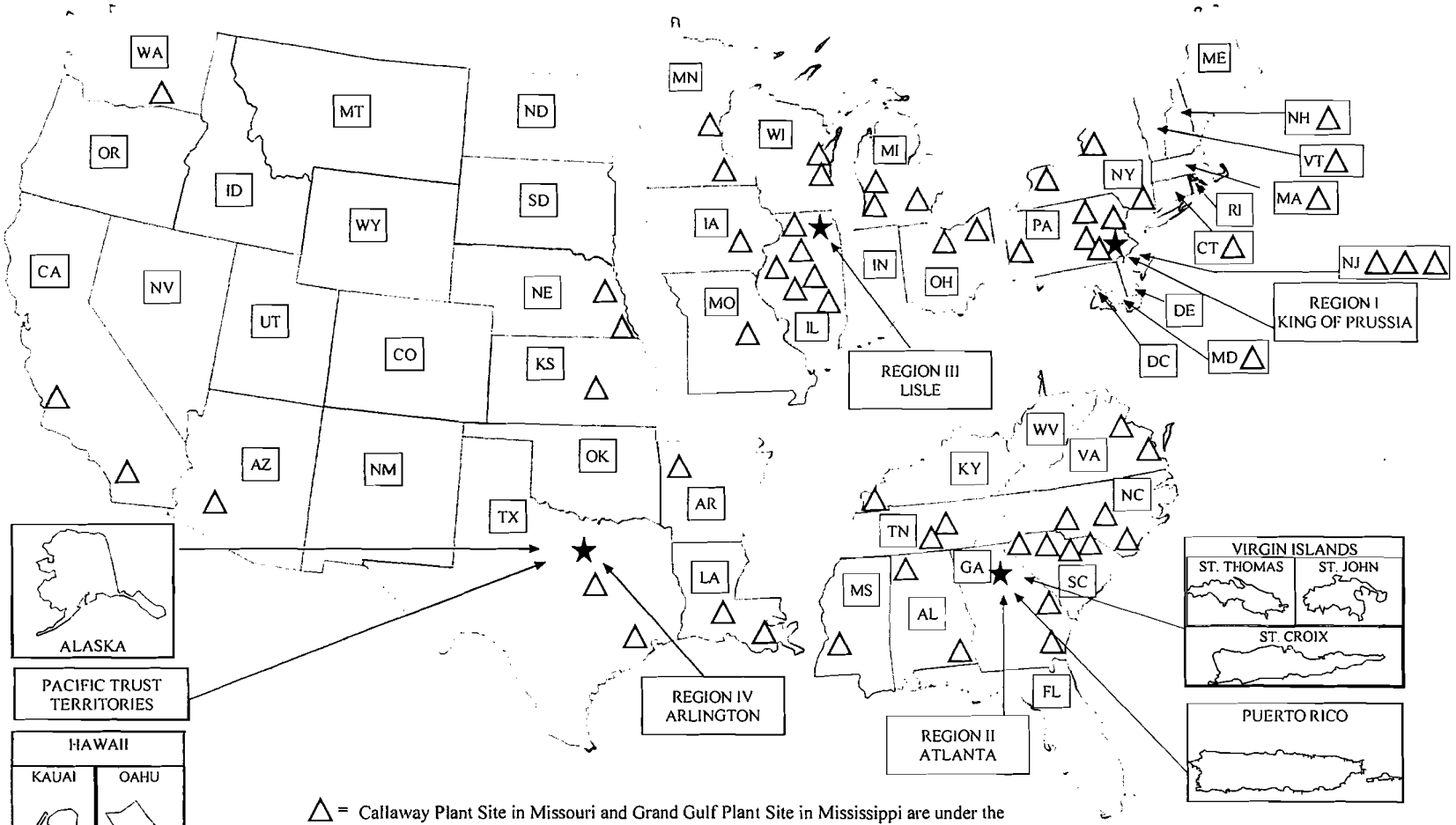


△ = Callaway Plant Site in Missouri and Grand Gulf Plant Site in Mississippi are under the purview of Region IV.

△ = The Paducah Gaseous Diffusion Plant in Kentucky is under the purview of Region III.

Individual State Assignments for NRC Region IV State Liason Officer and State Agreement Officers

- ☐ Hackney
- ☐ Campell
- ☐ McLean

For issues related to power reactors, research reactors, low level waste and high level waste, government agencies, and Indian affairs activities, Mr. Hackney has notification responsibilities for all States within Region IV.

California Electric Grid

Presenter:

David P. Loveless

Topic Outline

- History of California Electric Grid
- Current Situation under Deregulation
- Region IV Response
- Summary

Historical Causes

- California is a Power Importer
- The BANANA Principle
- Electric Power Growth

Current Situation

- Average Cost - $138 wholesale
- Retail Cost - $60 regulatory cap
- PG&E: Chapter 11 Bankruptcy
- SCE: Memorandum of Understanding

Region IV Response

- SRI Financial Impact Observables
- Management of Monthly Visits
- Shorter Report Periods & More Detailed
- Additional Public Meetings
- Weekly SRI Calls
- Bi-Weekly Utility Calls

SRI Observations

- Staffing
- Maintenance
- Outage and Plant Modifications
- Emergency Preparedness
- Grid Stability

Summary

- Current Safety Impacts:
  - None
- Working with Bankruptcy Judge and MOU
- Continued Vigilance Required:
  - Safety Maintenance
  - Public Confidence

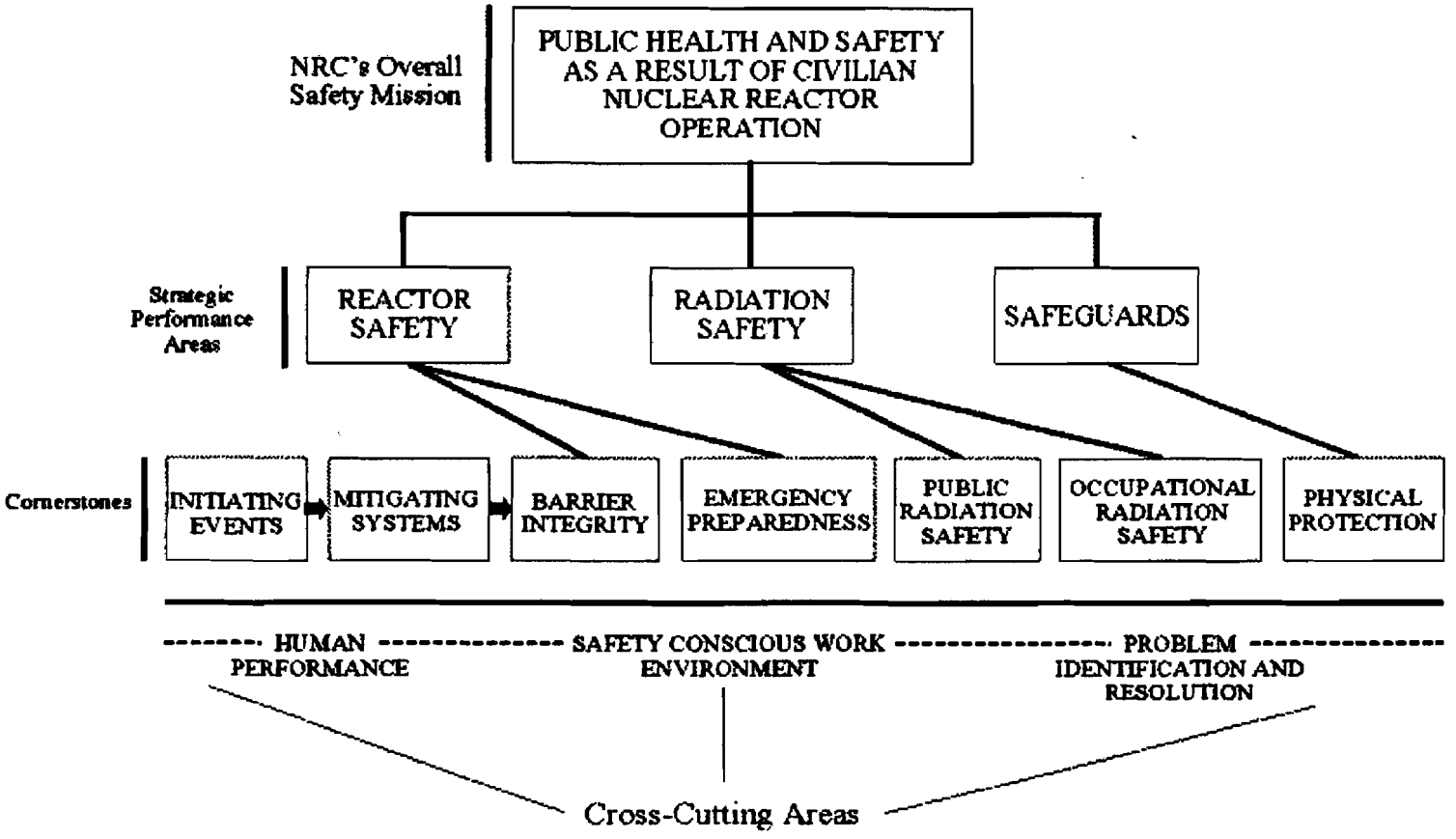
REVISED REACTOR OVERSIGHT PROGRAM
REGION IV IMPLEMENTATION

Ken Brockman
Jeff Clark

- ⇨ Process Overview
- ⇨ Timeline
- ⇨ Inspection & Assessment
- ⇨ Region IV Plant Results
- ⇨ Action Matrix
- ⇨ Conclusions
- ⇨ Q & A

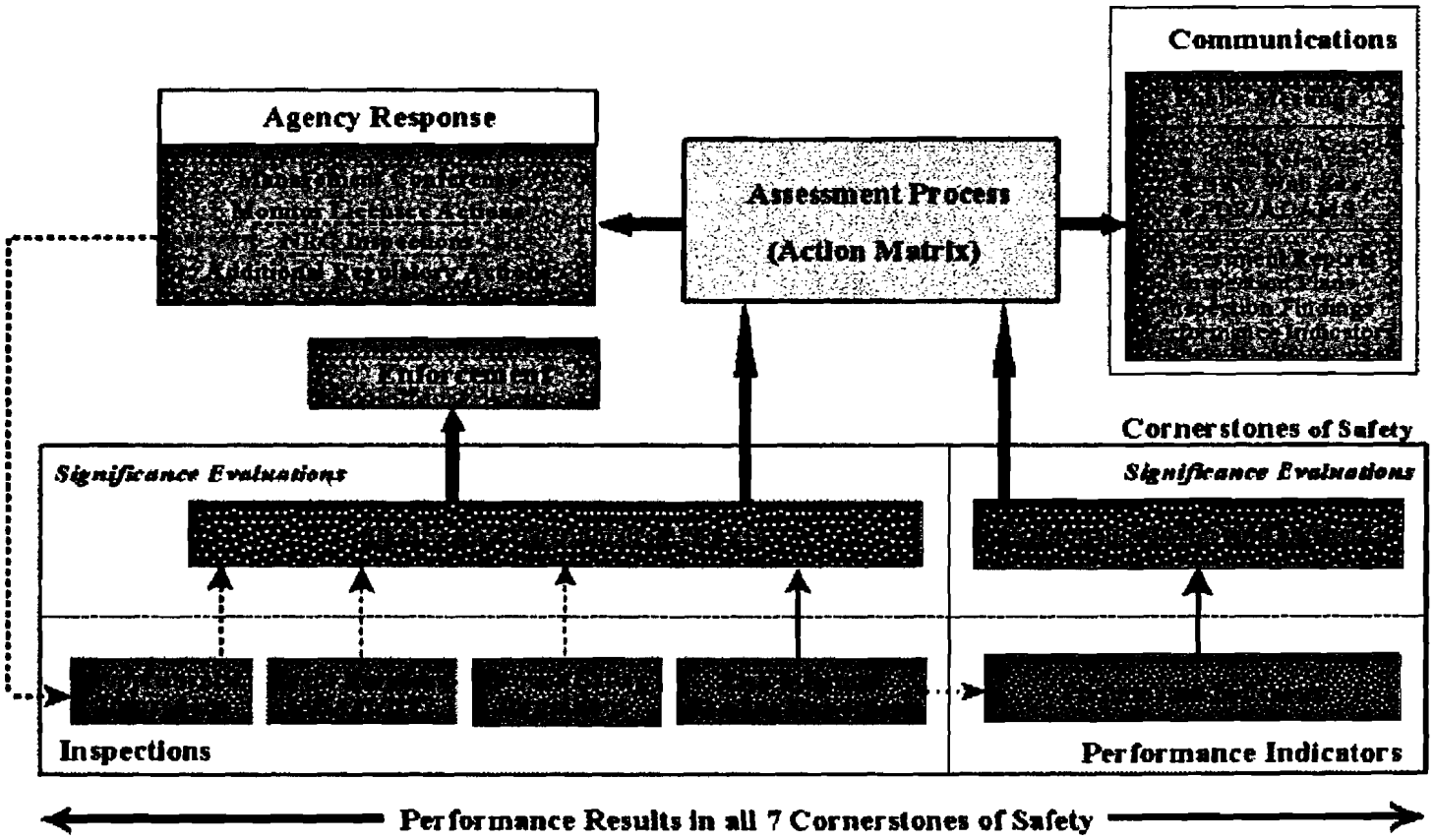
Slide #1

REGULATORY FRAMEWORK

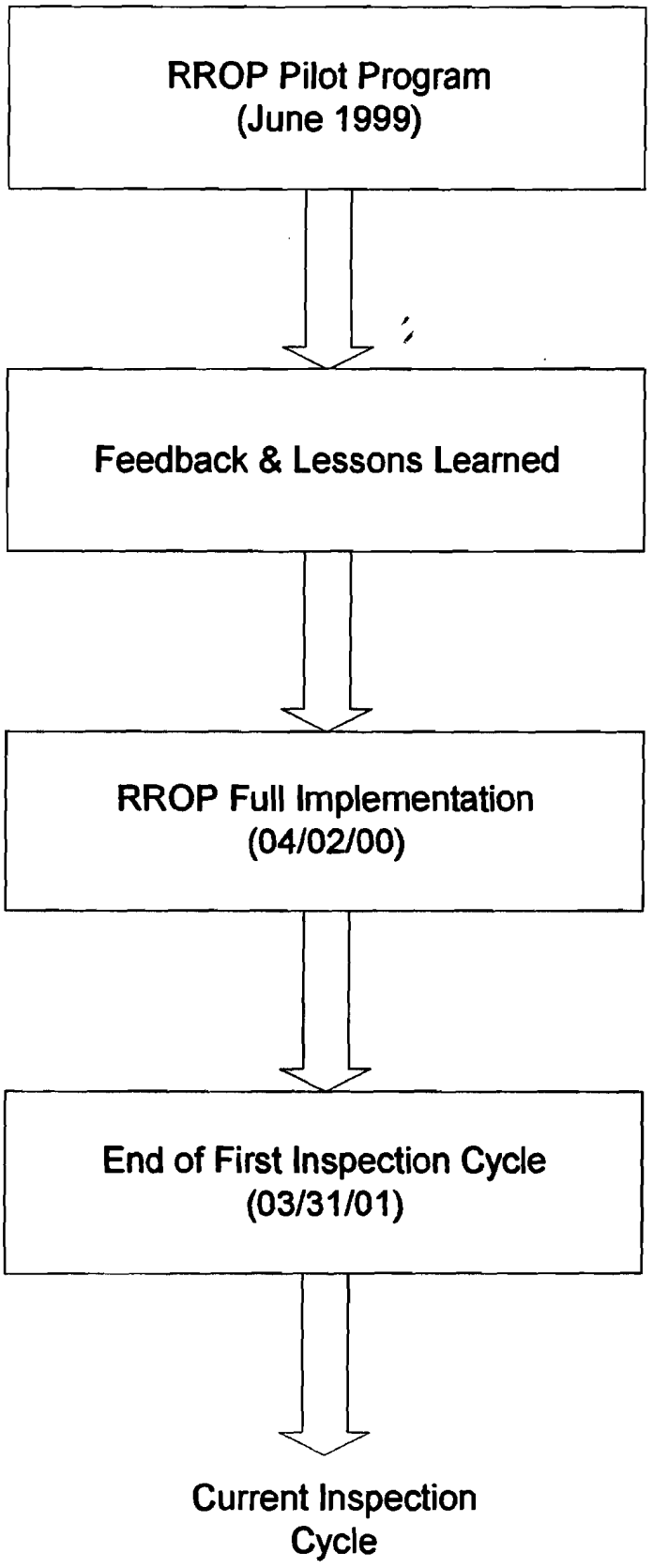


Slide #2

REACTOR OVERSIGHT PROCESS



RROP IMPLEMENTATION TIMELINE



Reactor Oversight Process Pilot Program

ROP Home Page

Plant Assessment Results

Initial Implementation Evaluation Panel

ROP "Plain "Language" Description

Meeting Notices & Schedules

ROP Program Documents

Slides from Regional Public Workshops

Pilot Program

Additional Information

Glossary

The NRC tested the revised reactor oversight process (ROP) during a six month pilot program at 13 plants at 9 sites across the country, beginning in June 1999. The plants represented a cross section of the nuclear industry, featuring different plant designs and varying levels of performance. In this program, utilities voluntarily collected performance indicator data and reported it to the NRC, and the NRC inspected and assessed the plants the plants using the new oversight process. In addition, the NRC took appropriate enforcement action using the new enforcement policy guidelines.

The pilot program was intended to test how effectively the revised ROP worked and to identify possible problems. The program measured the resources required by the NRC and by the industry to implement the new system. A panel of senior NRC managers and external stakeholders assessed the pilot program for readiness for full implementation, and documented their conclusions in the Pilot Program Evaluation Panel final report *(pdf)*. Based on lessons learned from the pilot, the NRC made revisions to the process before extending the overnight process to all commercial nuclear power plants in April 2000. The NRC staff reported the results of the pilot program to the Commission in SECY-00-049, "Results of the Revised Reactor Oversight Process Pilot Program" *(html)(pdf)(wpd)*; the Commission approved the staff's recommendation to implement the revised ROP for all plants in SRM for SECY-00-049 *(html)(wpd)*. The plants in the pilot program are listed below.

Region I	Region III
Hope Creek Public Service Electric & Gas Co.	Prairie Island 1 and 2 Northern States Power Co.
Salem 1 and 2 Public Service Electric & Gas Co.	Quad Cities 1 and 2 Commonwealth Edison Co.
FitzPatrick N. Y. Power Authority	
Region II	**Region IV**
Shearon Harris Carolina Power and Light Co.	Ft. Calhoun Omaha Public Power District
Sequoyah 1 and 2 Tennessee Valley Authority	Cooper Nebraska Public Power District

Updated July 14, 2000

The assessment program collects information from inspections and performance indicators (PIs) in order to enable the agency to arrive at objective conclusions about the licensee's safety performance. Based on this assessment information, the NRC determines the appropriate level of agency response,including supplemental inspection and pertinent regulatory actions ranging from management meetings up to and including orders for plant shutdown. The Action Matrix Summary listed below reflects overall plant performance and is updated regularly to reflect inputs from the most recent performance indicators and inspection findings. Notes have been added to some plants to explain the reasons that these plants have changed Action Matrix columns from the previous quarter. Plant oversight for D.C. Cook 1 and D.C. Cook 2 is being conducted in accordance with IMC 0350 " Oversight of Operating Reactor Facilities in Shutdown Condition with Performance Problems" and is not listed below. This page will be updated as necessary to reflect changes in licensee performance.

Licensee Response Column	Regulatory Response Column	Degraded Cornerstone Column	Multiple/Repetitive Degraded Cornerstone Column	Unacceptable Performance Column
Arkansas Nuclear 1	Calvert Cliffs 1	Callaway	Indian Point 2	
Arkansas Nuclear 2	Cooper	Kewaunee		
Beaver Valley 1	Fermi 2[1]	Millstone 2		
Beaver Valley 2	FitzPatrick			
Braidwood 1	Harris 1[2]			
Braidwood 2	North Anna 2[3]			
Browns Ferry 2	Oconee 1			
Browns Ferry 3	Point Beach 1			
Brunswick 1	Prairie Island 1[4]			
Brunswick 2	Prairie Island 2[5]			
Byron 1	Quad Cities 1[6]			
Byron 2	Quad Cities 2[7]			
Calvert Cliffs 2	Summer[8]			
Catawba 1	Susquehanna 1[9]			
Catawba 2	Susquehanna 2[10]			
Clinton				

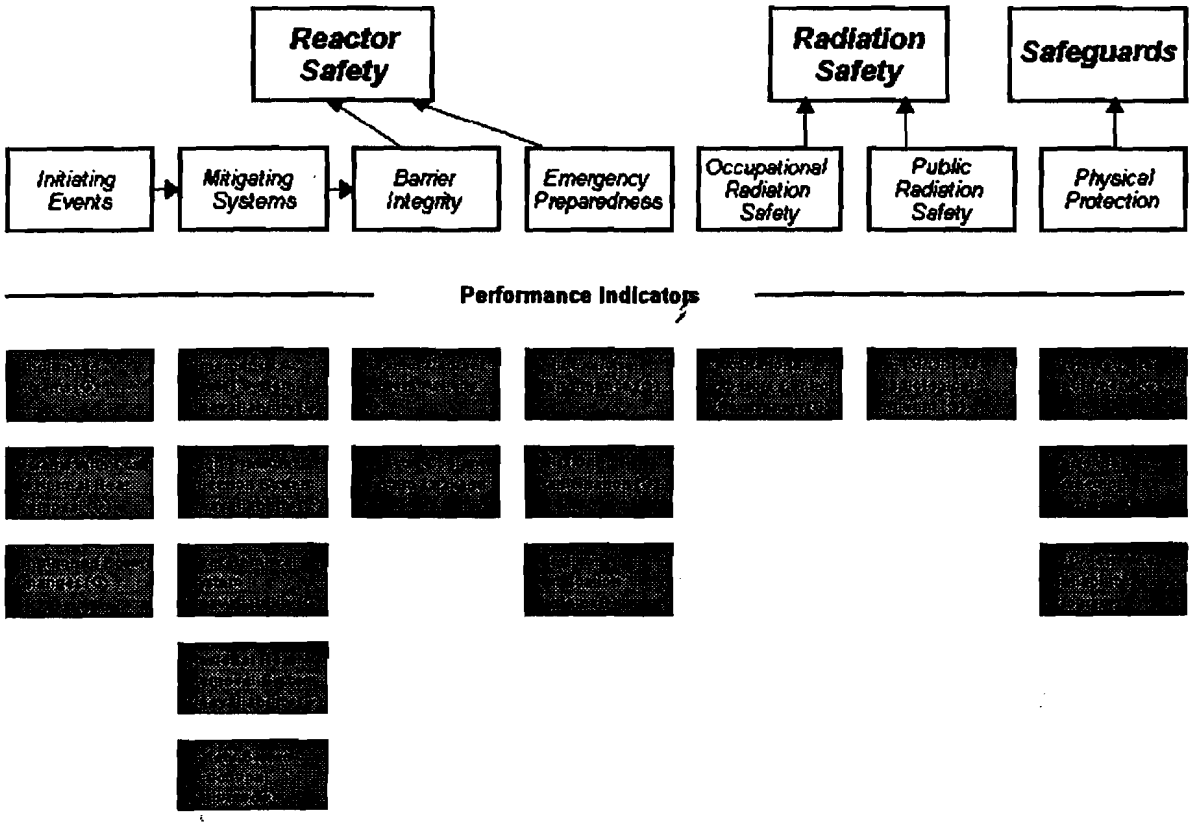
Reactor Oversight Process **Performance Indicators Summary**

Plants	IE 01	IE 02	IE 03	MS 01	MS 02	MS 03	☒	MS 05	BI 01	BI 02	EP 01	EP 02	EP 03	OR 01	PR 01	PP 01	PP 02	PP 03
Catawba 1	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Catawba 2	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Clinton	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Columbia Generating Station	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Comanche Peak 1	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Comanche Peak 2	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Cooper	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Crystal River 3	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
D.C. Cook 1	G	G	W	I	I	I	I	G	G	G	I	G	G	G	G	G	G	G
D.C. Cook 2	G	G	G	I	I	I	I	G	G	G	I	G	G	G	G	G	G	G
Davis-Besse	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Diablo Canyon 1	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Diablo Canyon 2	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Dresden 2	G	G	G	G	G	T	G	G	G	G	G	G	G	G	G	G	G	G
Dresden 3	G	G	G	G	G	T	G	G	G	G	G	G	G	G	G	G	G	G
Duane Arnold	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G

Inspection Findings & Additional Information

Current Action Matrix Column: Degraded Cornerstone Column

Callaway
1Q/2001 Performance Summary



Legend: R=Red W=White T=Thresholds under development N=Not Applicable
Y=Yellow G=Green I=Insufficient data to calculate PI U=Unique Design

	Initiating Events →	Mitigating Systems →	Barrier Integrity	Emergency Preparedness	Occupational Radiation Safety	Public Radiation Safety	Physical Protection
Most Significant Inspection Findings							
1Q/2001							
4Q/2000							
3Q/2000					W (3)		
2Q/2000							
Miscellaneous findings							

Additional Inspection & Assessment Information

Assessment Reports/Inspection Plans:

- 1Q/2001
- 4Q/2000
- 3Q/2000
- 2Q/2000

List of Inspection Reports

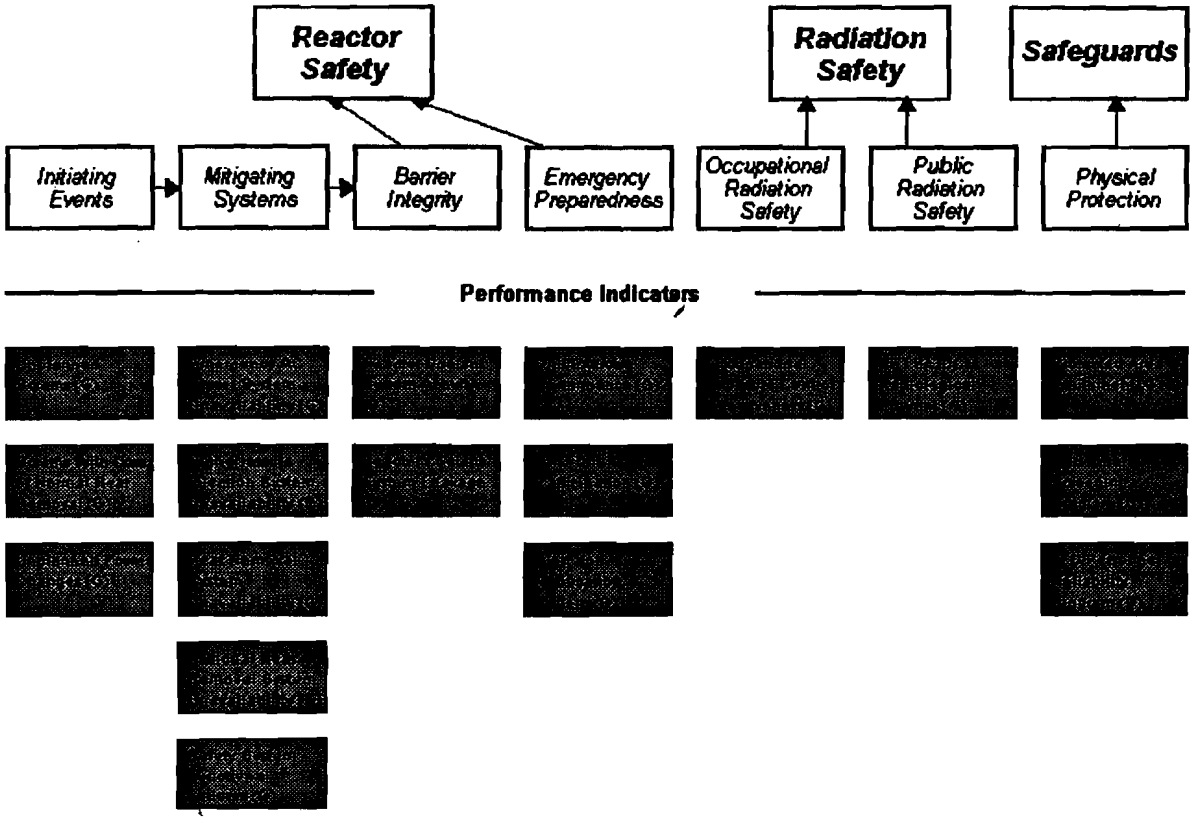
PI Summary | Inspection Findings Summary | Action Matrix Summary | Plant Assessment Results

Last Modified: June 12, 2001

Inspection Findings & Additional Information

Current Action Matrix Column: Regulatory Response Column

Cooper
1Q/2001 Performance Summary



Legend: R=Red W=White T=Thresholds under development N=Not Applicable
Y=Yellow G=Green I=Insufficient data to calculate PI U=Unique Design

	Initiating Events →	Mitigating Systems →	Barrier Integrity	Emergency Preparedness	Occupational Radiation Safety	Public Radiation Safety	Physical Protection
Most Significant Inspection Findings							
1Q/2001							
4Q/2000				W (1)			
3Q/2000							
2Q/2000							

Miscellaneous findings

Additional Inspection & Assessment Information

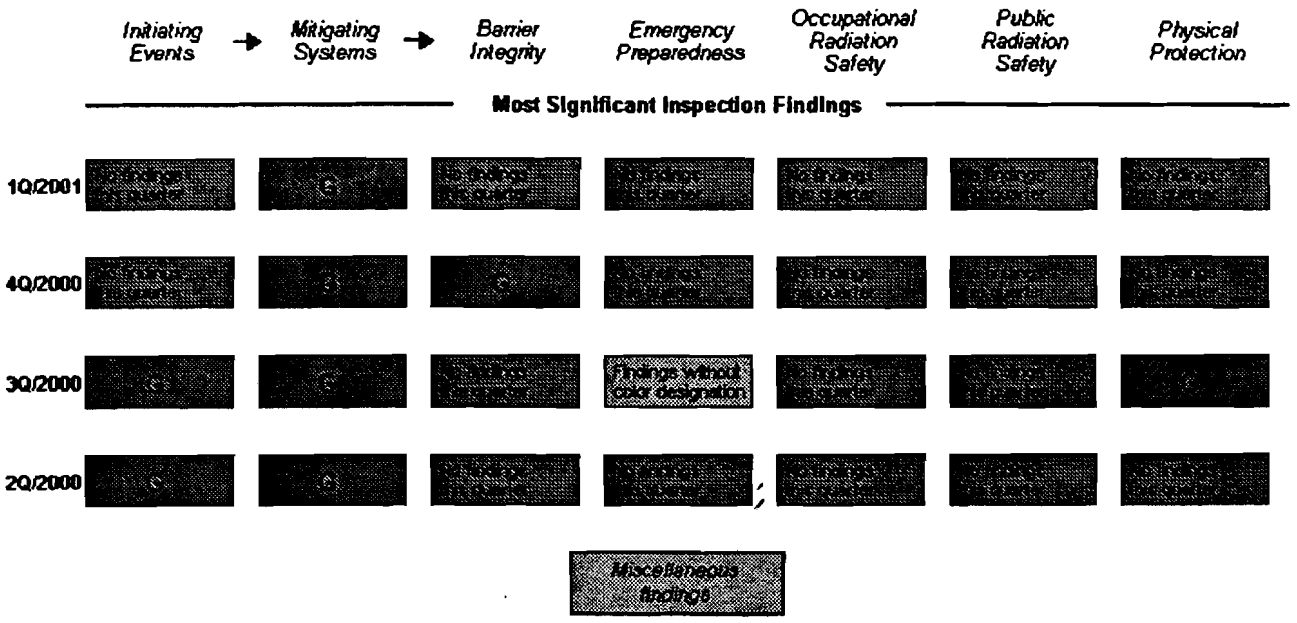
Assessment Reports/Inspection Plans:

- 1Q/2001
- 4Q/2000
- 3Q/2000
- 2Q/2000

List of Inspection Reports

PI Summary | Inspection Findings Summary | Action Matrix Summary | Plant Assessment Results

Last Modified: June 12, 2001



Additional Inspection & Assessment Information

Assessment Reports/Inspection Plans:

- 1Q/2001
- 4Q/2000
- 3Q/2000
- 2Q/2000

List of Inspection Reports

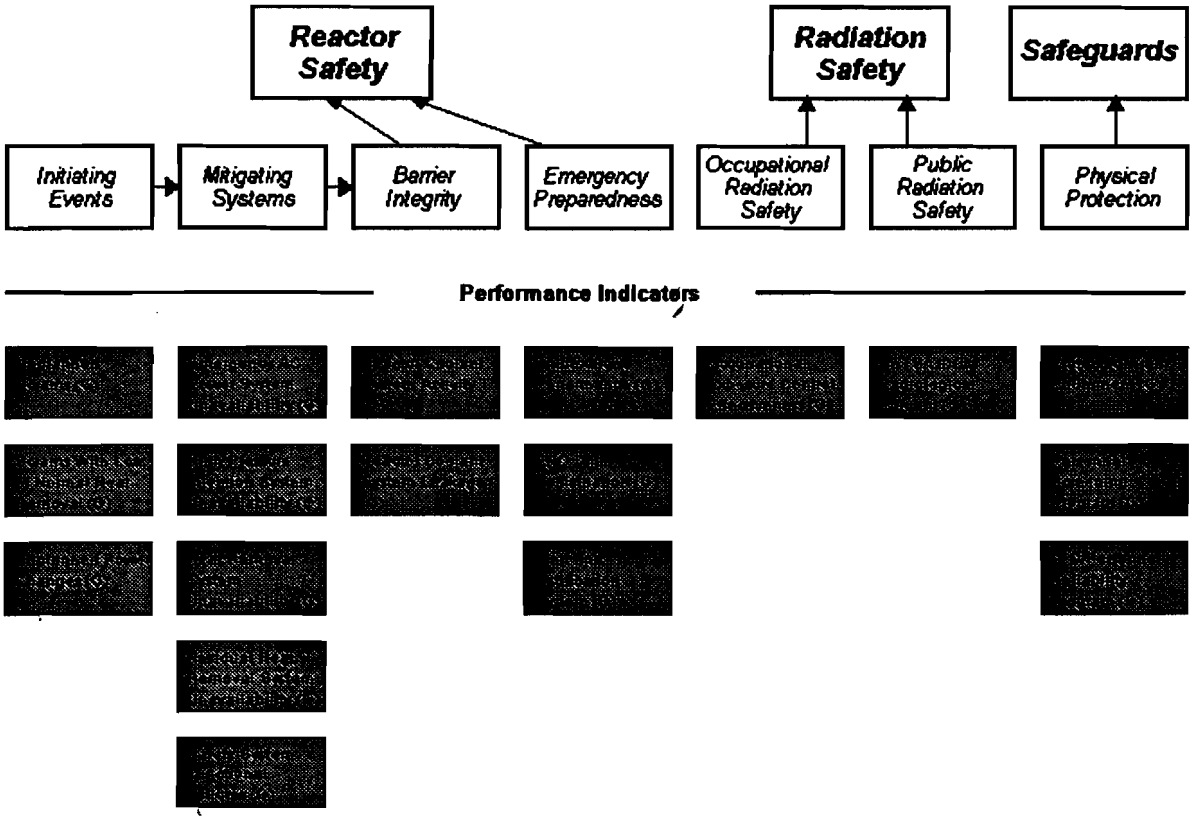
PI Summary | Inspection Findings Summary | Action Matrix Summary | Plant Assessment Results

Last Modified: June 12, 2001

Inspection Findings & Additional Information

Current Action Matrix Column: Licensee Response Column

San Onofre 2
1Q/2001 Performance Summary



Legend: R=Red W=White T=Thresholds under development N=Not Applicable
Y=Yellow G=Green I=Insufficient data to calculate PI U=Unique Design

Assessment Process Table

Level of Review	Frequency/Timing	Participants (*indicates lead)	Desired Outcome	Communication
Continuous	Continuous	SRI*, RI, Regional Inspectors, Analysts	Performance awareness	None required, notify licensee by an Assessment Follow Up letter only if thresholds crossed
Quarterly	Once per quarter/Five weeks after end of quarter	DRP:BC*, PE, SRI, RI	Input/verify PI/PIM data, detect early trends	Update data set, notify licensee by an Assessment Follow Up letter only if thresholds crossed
Mid-Cycle	At mid-cycle/Six weeks after end of second quarter	Division of Reactor Safety (DRS) or DRP DD*, DRP and DRS BCs	Detect trends, plan inspection for twelve months	Inspection look ahead letter
End-of-Cycle	At end-of-cycle/ Six weeks after end of assessment cycle	DRS or DRP DD*, RAs, NRR representative, BCs, principal inspectors	Assessments of plant performance, oversight and coordination of regional actions	Assessment letter and inspection look ahead letter
Agency Action Review	Annual/ Two weeks after end-of-cycle review	DIR NRR*, RAs, DRS/DRP DDs, DISP, OE, OI, other HQ offices as appropriate	Oversight and coordination of agency-level actions	Commission briefing, followed by public meetings with individual licensee's to discuss assessment results

Action Matrix

		Licensee Response Column	Regulatory Response Column	Degraded Cornerstone Column	Multiple Repetitive Degraded Cornerstone Column	Unacceptable Performance Column
Results		All assessment inputs (performance indicators (PI) and inspection findings) Green; cornerstone objectives fully met	One or two White inputs (in different cornerstones) in a strategic performance area; Cornerstone objectives fully met	One degraded cornerstone (2 White inputs or 1 Yellow input) or any 3 White inputs in a strategic performance area; cornerstone objectives met with minimal reduction in safety margin	Repetitive degraded cornerstone, multiple degraded cornerstones, multiple Yellow inputs, or 1 Red input[1]; cornerstone objectives met with longstanding issues or significant reduction in safety margin	Overall unacceptable performance; plants not permitted to operate within this band, unacceptable margin to safety
Response	Regulatory Conference	Routine Senior Resident Inspector (SRI) interaction	Branch Chief (BC) or Division Director (DD) meet with Licensee	DD or Regional Administrator (RA) meet with Licensee	EDO (or Commission) meet with Senior Licensee Management	Commission meeting with Senior Licensee Management
	Licensee Action	Licensee Corrective Action	Licensee corrective action with NRC oversight	Licensee self assessment with NRC oversight	Licensee performance improvement plan with NRC oversight	
	NRC Inspection	Risk-informed baseline inspection program	Baseline and supplemental inspection 95001	Baseline and supplemental inspection 95002	Baseline and supplemental inspection 95003	
	Regulatory Actions	None	Document response to degrading area in assessment letter	Document response to degrading condition in assessment letter	10 CFR 2.204 DFI 10 CFR 50.54(f) letter CAL/Order	Order to modify, suspend, or revoke licensed activities
Communications	Assessment Report	BC or DD review / sign assessment report (w/ inspection plan)	DD review / sign assessment report (w/ inspection plan)	RA review / sign assessment report (w/ inspection plan)	RA review / sign assessment report (w/ inspection plan) Commission informed	
	Public Assessment Meeting	SRI or BC meet with Licensee	BC or DD meet with Licensee	RA discuss performance with Licensee	EDO (or Commission) discuss performance with Senior Licensee Management	Commission meeting with Senior Licensee Management
	Increasing Safety Significance →					

[1] It is expected in a few limited situations that an inspection finding of this significance will be identified that is not indicative of overall licensee performance. The staff will consider treating these inspection findings as exceptions for the purpose of determining appropriate actions.

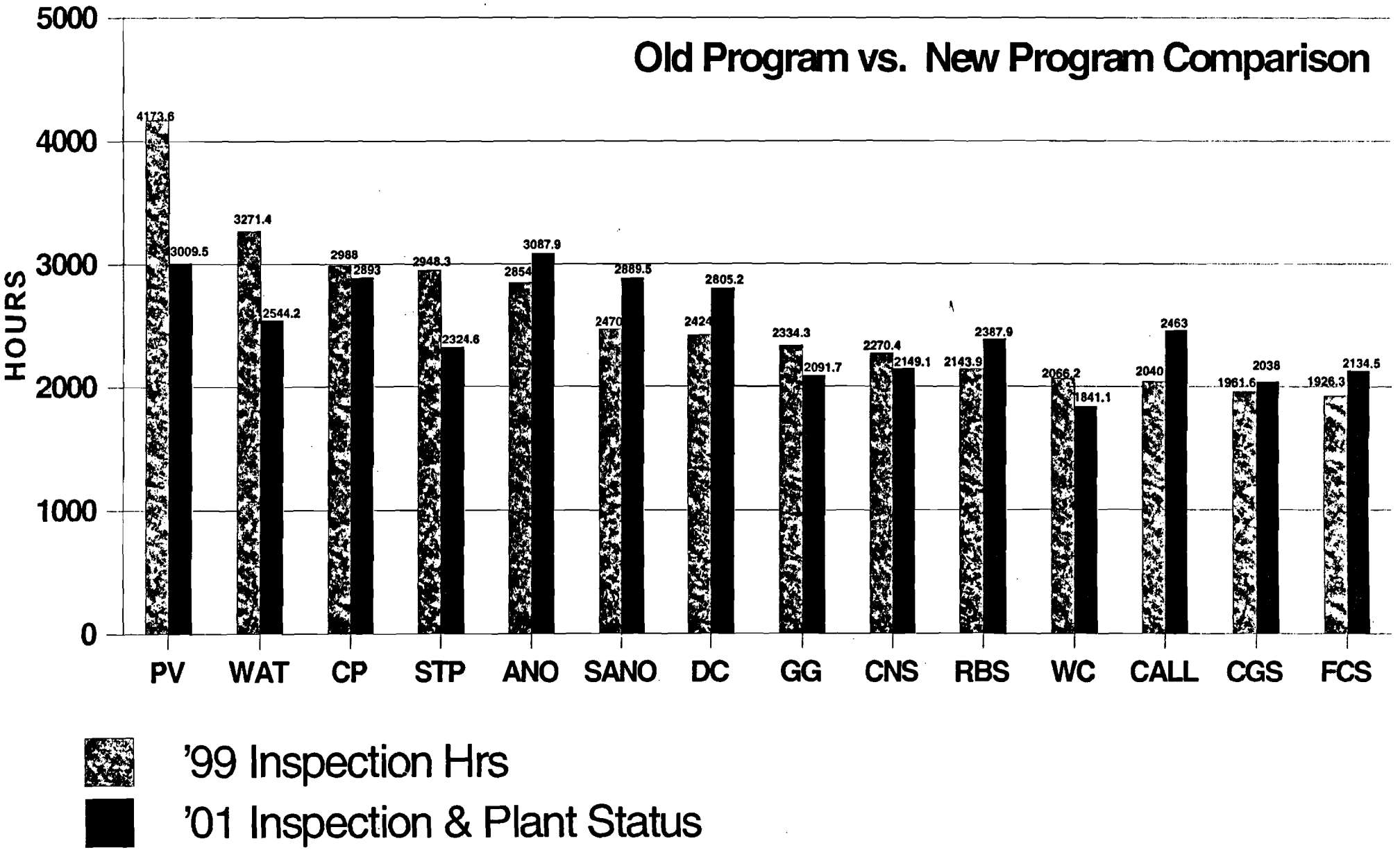
Event Response

- The resident inspector shall perform an initial determination of facility status and licensee actions to mitigate the event in accordance with inspection procedure 71153 *Event Follow Up* including:
  - observing plant parameters and current plant status
  - evaluating the performance of mitigating systems and actions taken by the licensee
  - confirming that the licensee has properly classified the event in accordance with the emergency plan and made timely off-site notification of any event when required.
- A risk analyst will estimate the risk significance of the event using the best available PRA tools and insights.
- The initial risk determination will be used to consider appropriate followup inspection resources per MD 8.3.

Event Response
(continued)

- Any resulting performance issues will be characterized for risk significance utilizing the Significance Determination Process (SDP).

- These performance issues will be combined with performance indicators and inspection findings to determine appropriate agency response per the Action Matrix.


Old Program vs. New Program Comparison
HOURS
5000
4000
3000
2000
1000
0
4173.6
3009.5
3271.4
2544.2
2988
2893
2948.3
2324.6
2854
3087.9
2470
2889.5
2424
2805.2
2334.3
2091.7
2270.4
2149.1
2143.9
2387.9
2066.2
1841.1
2040
2463
1961.6
2038
1928.3
2134.5
PV
WAT
CP
STP
ANO
SANO
DC
GG
CNS
RBS
WC
CALL
CGS
FCS
'99 Inspection Hrs
'01 Inspection & Plant Status

SIGNIFICANCE DETERMINATION PROCESS IMPLEMENTATION IN REGION IV

Kriss Kennedy

SENIOR REACTOR ANALYST ROLE IN RISK ASSESSMENT

- REGION IV RISK ANALYST RESOURCES
  - Troy Pruett

    Materials Inspector, RI at Waterford, SRI at Clinton and River Bend
  - Kriss Kennedy

    License Examiner, RI at Comanche Peak, SRI at ANO
- SRAs are assigned to the Division of Reactor Safety
- The SRAs are the focal point for risk informed activities in the region
- Additional resources include DRP Branch Chief qualified SRA and 3 enrolled in the Advanced Risk Training Program

SRA FUNCTIONS

- Support Region IV management and technical staff in the area of risk informed regulation
- Development of comprehensive risk informed resources
- Disseminate risk information
- Review requests for enforcement discretion
- Integrate risk perspectives into the enforcement process
- Assess the risk associated with potentially risk significant inspection findings, operational data, and events
- Perform evaluations of plant equipment configurations and online maintenance
- Review outage plans

SRA FUNCTIONS

- Develop and integrate risk into inspection planning, prioritization and operator licensing activities
- Perform risk informed team inspections
- Participate in mid-cycle and end-of-cycle plant performance review meetings
- Support NRR in development of risk assessment tools (Phase 2 benchmark visits, BNL visits)
- Conduct periodic meetings with PSA personnel from Region IV plants
- Review Plant data for risk insights and adverse trends

STATUS OF RISK TOOLS

- Risk-Informed Inspection Notebooks (SDP Phase 2 Worksheets)

  - Issued 11/15 (73%)
  - Benchmarked 3/15 (20%)

- Standardized Plant Analysis Risk (SPAR) Models - Revision 3

  - Completed 8/15 (53%)
  - QA'd 0

Results of Benchmark Visits

SONGS	Rev 0	w/fixes	SPAR
SDP Non-conservative	13%	4%	64%
SDP Conservative	22%	9%	23%
SDP Same Results	65%	87%	13%

Diablo Canyon	Rev 0	w/fixes	SPAR
SDP Non-conservative	15%	3%	29%
SDP Conservative	36%	36%	17%
SDP Same Results	49%	61%	54%

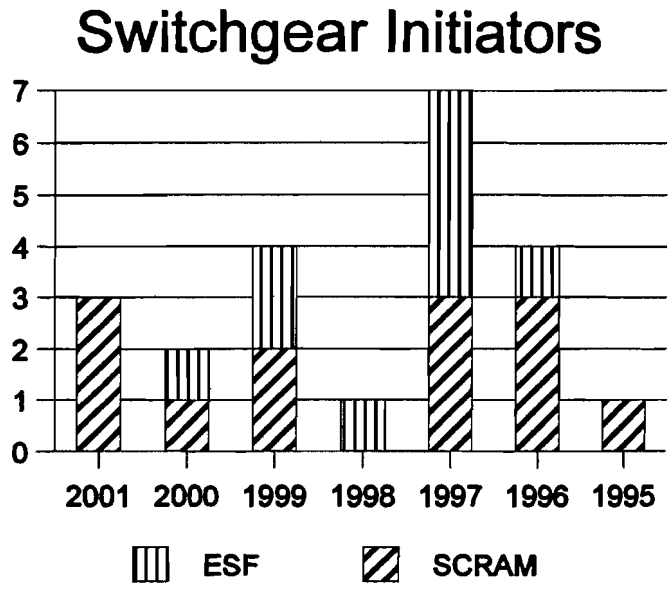
Effect of fire, flood, and seismic initiators could increase significance determination of inspection findings by about one order of magnitude.

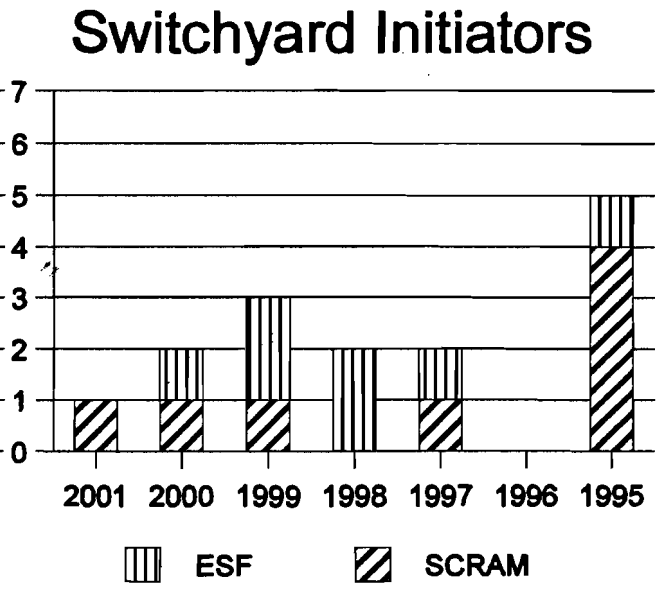
South Texas	Rev 0	w/fixes (external)
SDP Non-conservative	-	9% (14.3%)
SDP Conservative	-	29% (21.4%)
SDP Same Results	-	62% (64.3%)

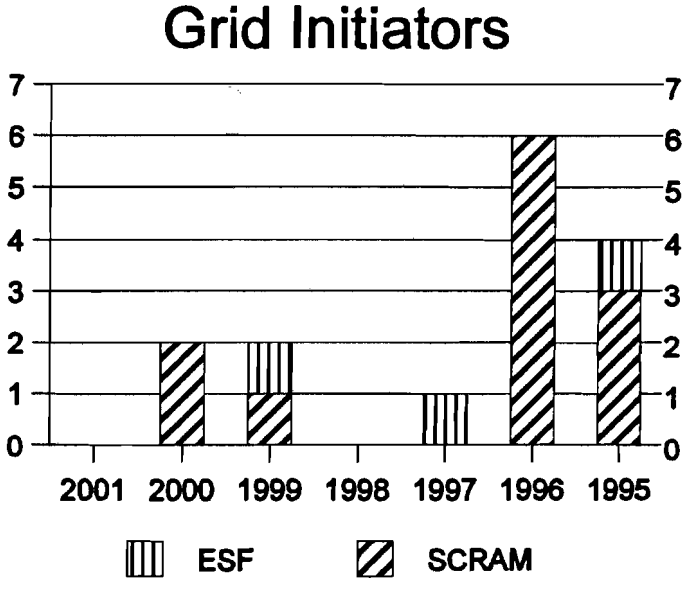
CHALLENGES

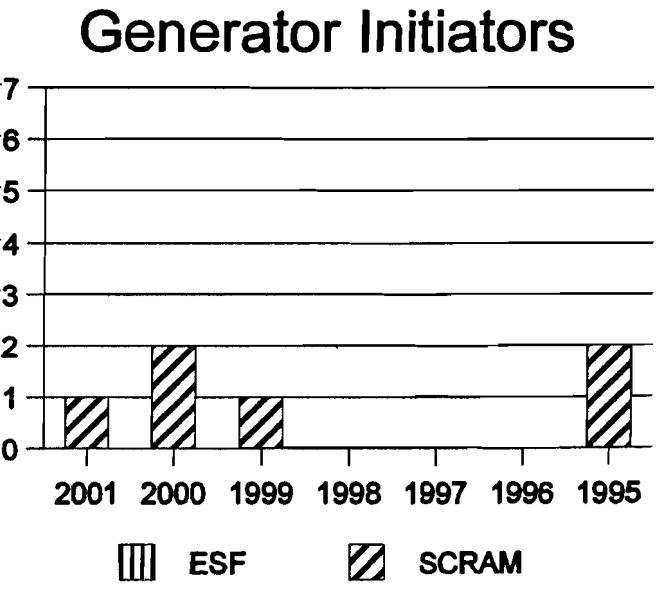
- Accuracy of SDP Phase 2 worksheets
- Availability/Accuracy of SPAR models
- Application of:
  - MC 0609, Appendix F, Fire Protection SDP
  - MC 0609, Appendix G, Shutdown Operations SDP
  - MC 0609, Appendix H, Containment Integrity SDP

SCRAM AND ESF DATA
1995 TO FIRST QUARTER 2001



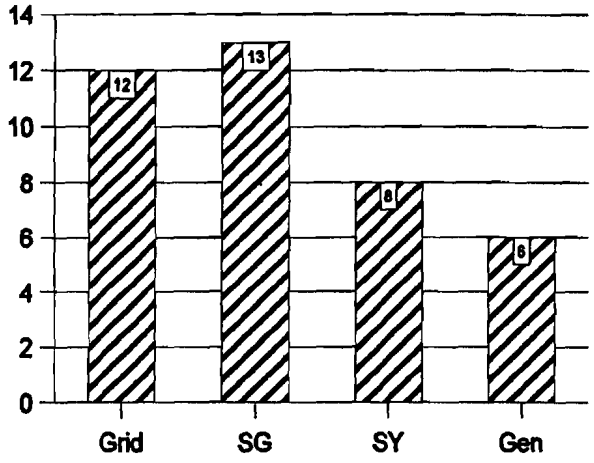




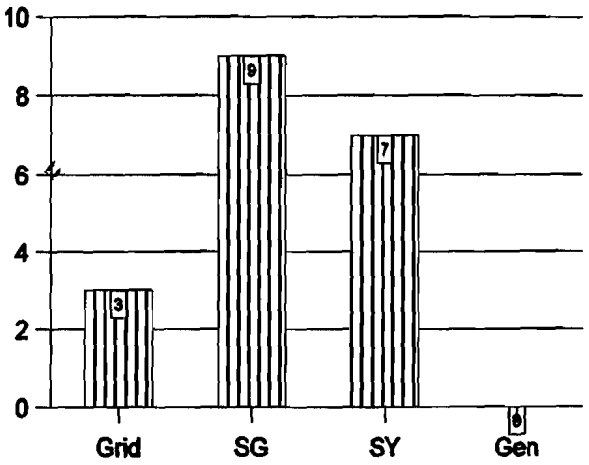


SCRAM AND ESF DATA: 1995 THROUGH FIRST QUARTER 2001

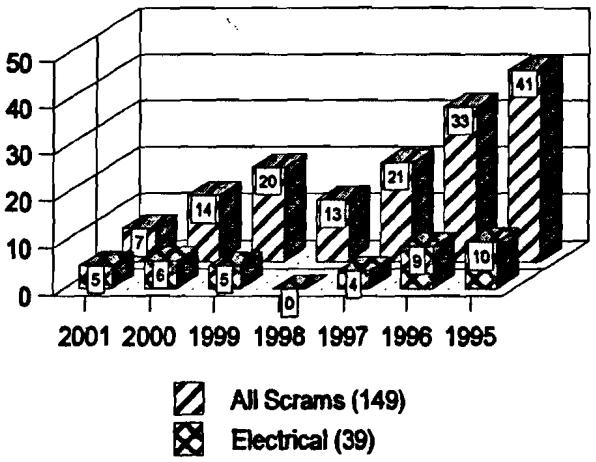
Electrical Scrams



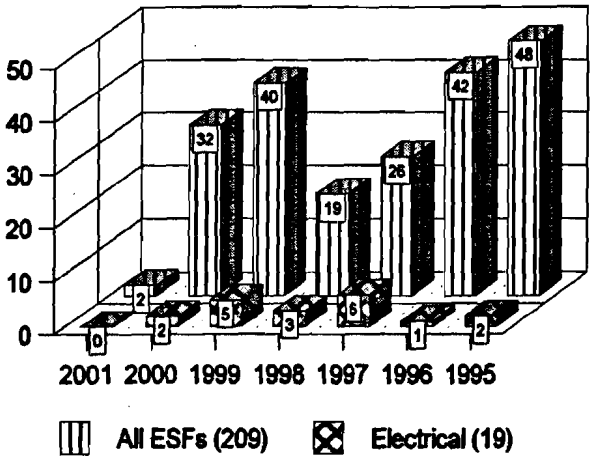
Electrical ESFs



Scrams

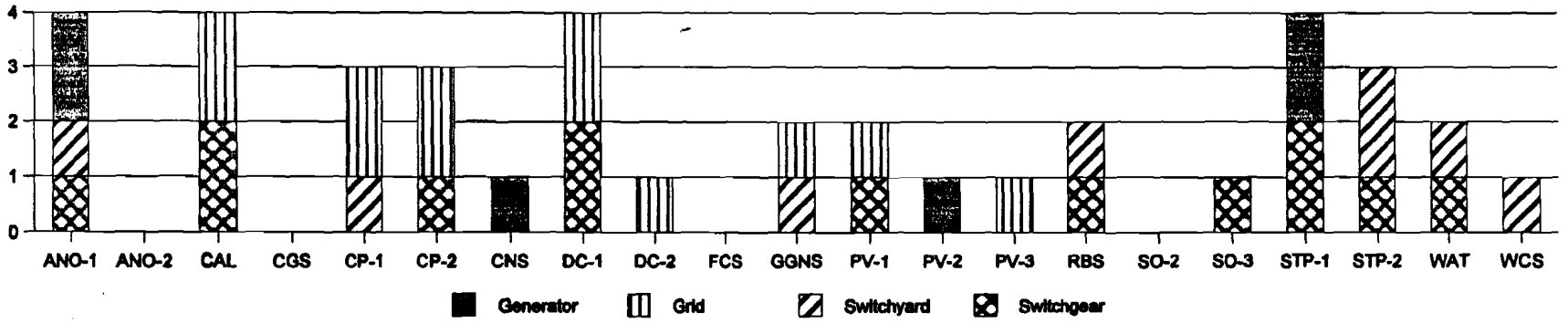


ESFs

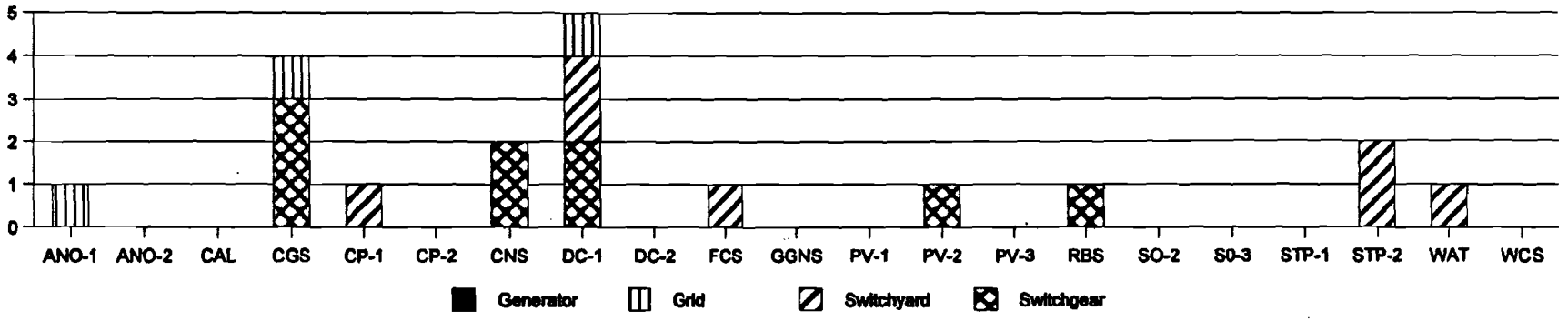


PLANT SCRAM AND ESF DATA

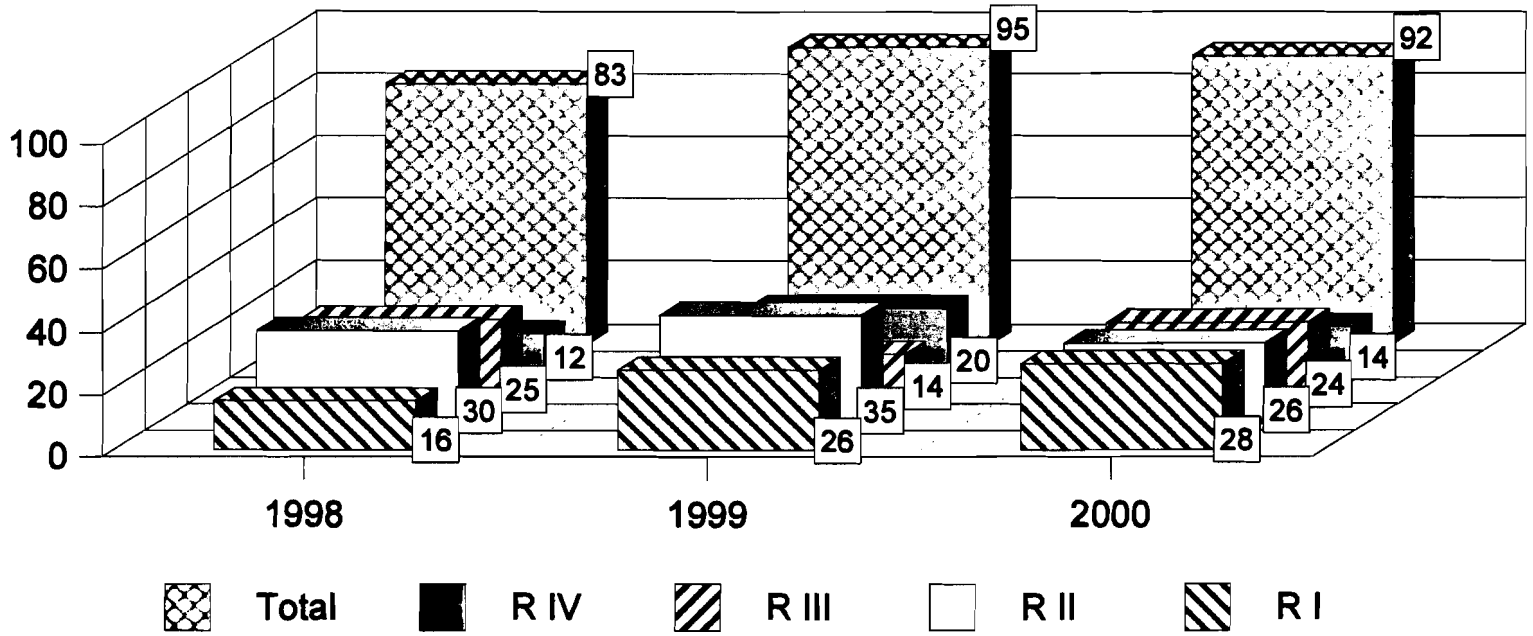
SCRAM DATA



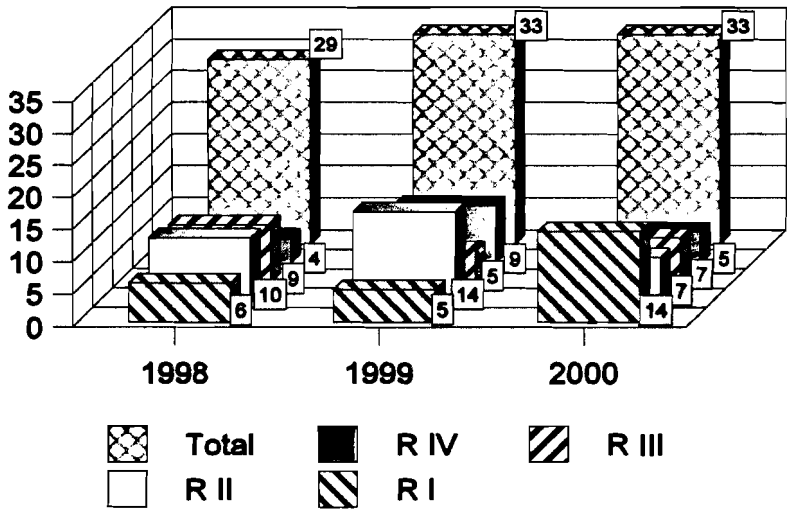
ESF DATA



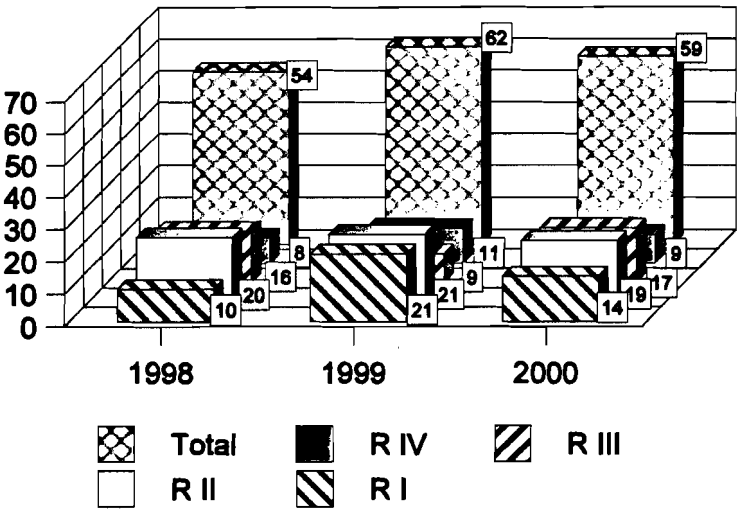
Total Number of Scrams



Number of Manual Scrams



Number of Auto Scrams



3. ASP Program Results

The below graphs are provided for illustration only. A detailed explanation is contained in SECY-01-0034, "Status Report on Accident Sequence Precursor Program and Related Initiatives."

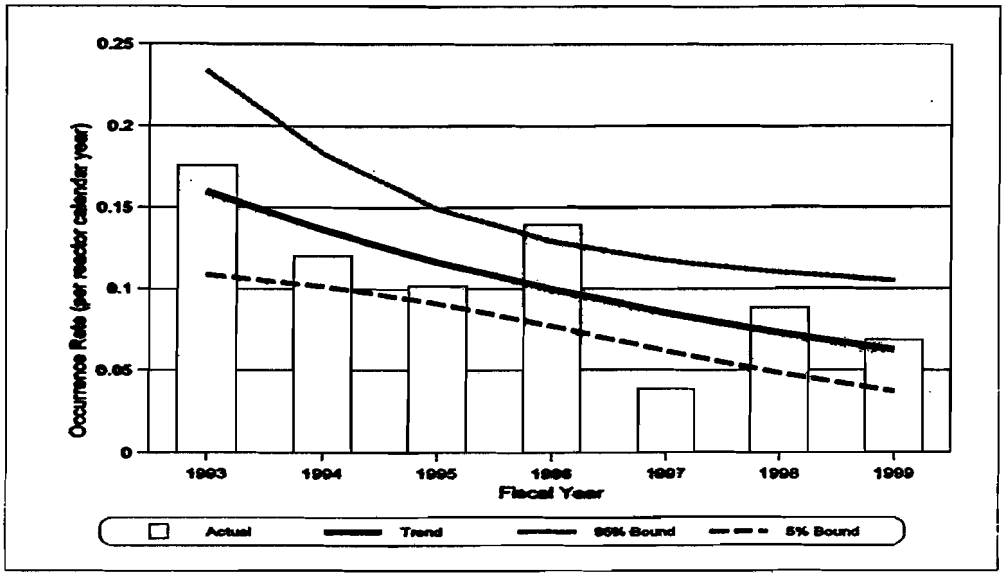


Figure 1. Precursor occurrence rate for 1993-1999 plotted against fiscal year. The trend is statistically significant (p-value = 0.0068). The result for 1999 is preliminary.

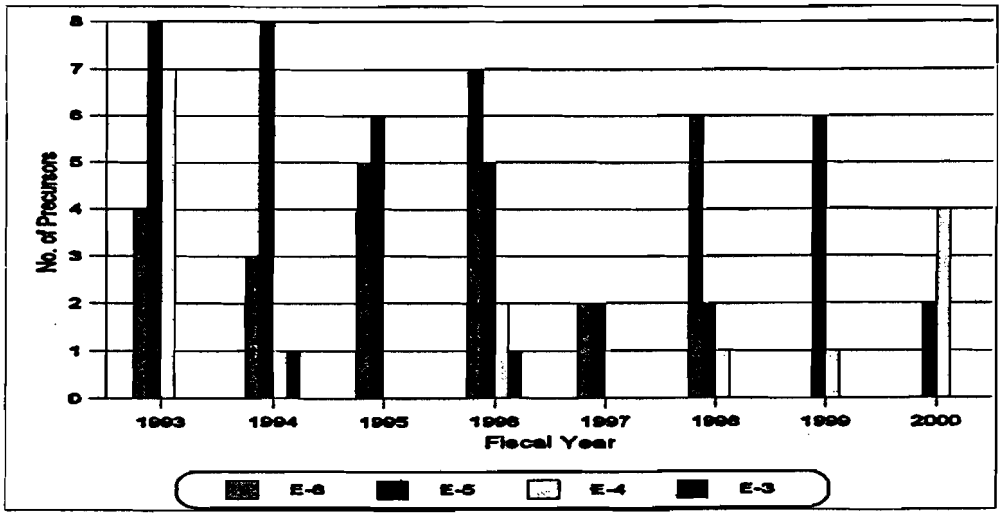
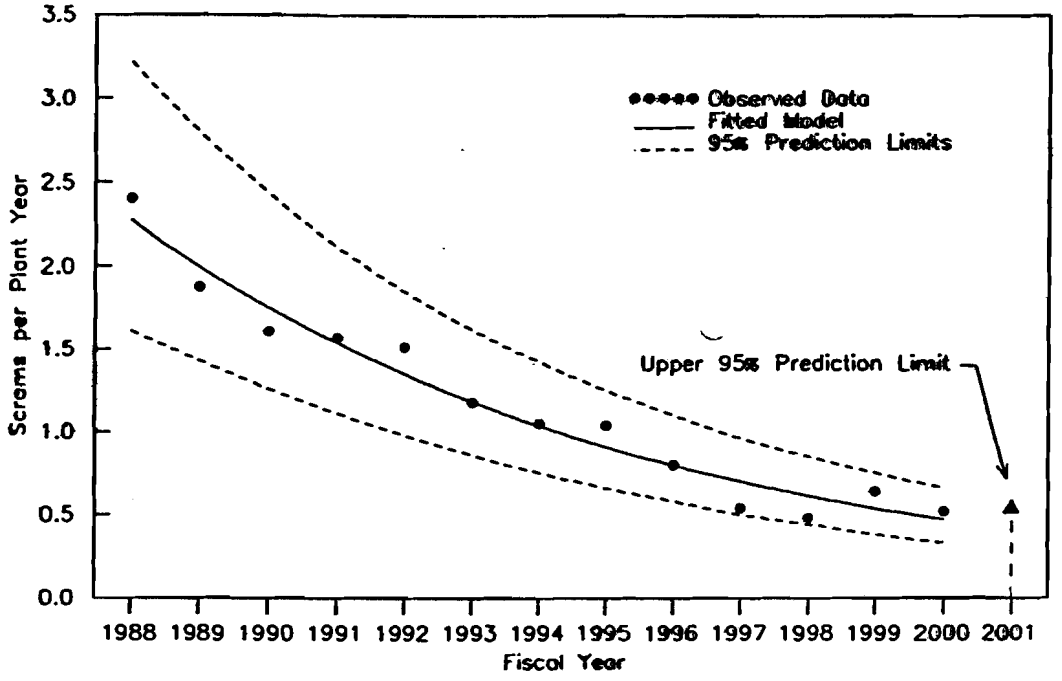


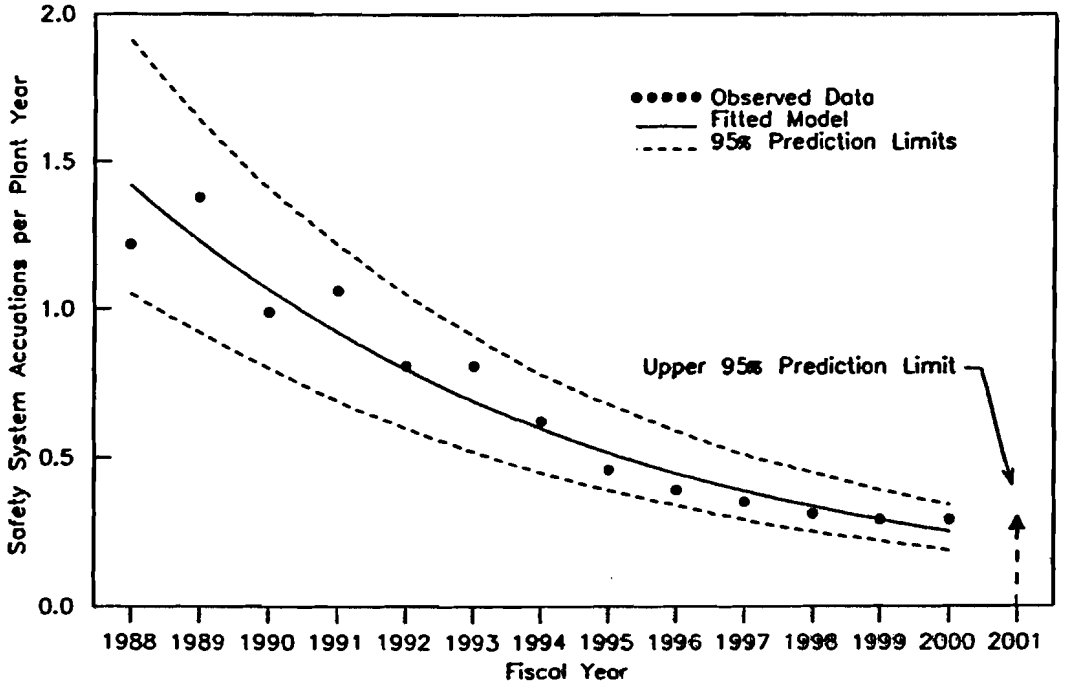
Figure 2. Conditional core damage probability results from ASP Program (1993-2000) for each of the CCDP bins (*E-3*: $\geq 1 \times 10^{-3}$; *E-4*: 9.9×10^{-4} to 1.0×10^{-4}; *E-5*: 9.9×10^{-5} to 1.0×10^{-5}; *E-6*: 9.9×10^{-6} to 1.0×10^{-6}). Results for FYs 1999 and 2000 are preliminary.

1. Ex-AEOD Indicators (long-term graphs only)

Automatic Scrams While Critical



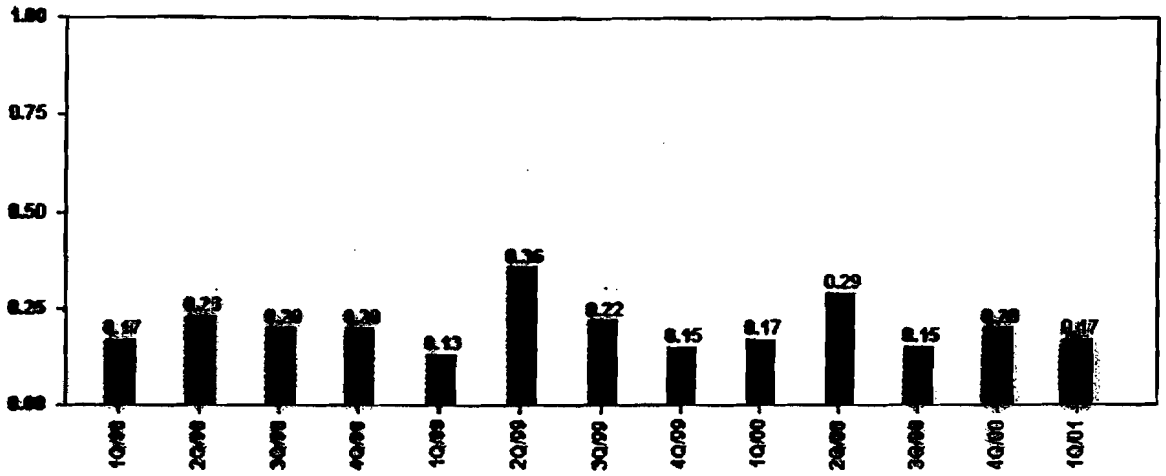
Safety System Actuations



2. ROP Performance Indicators (short-term graphs only)

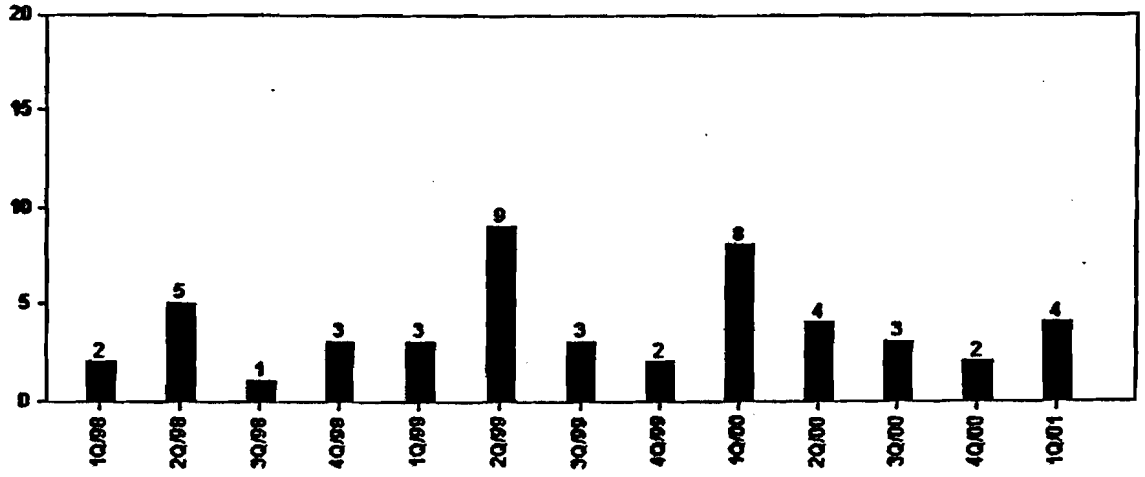
Initiating Events Cornerstone - Industry Trends **1Q/2001**

Unplanned Scrams per 7000 Annual Critical Hrs



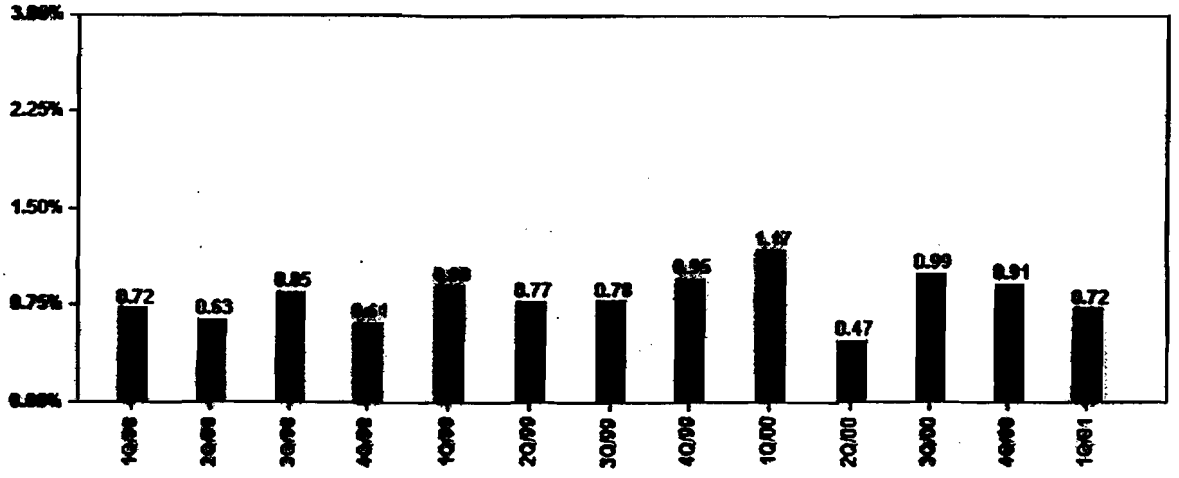
Descriptions

Scrams with Loss of Normal Heat Removal



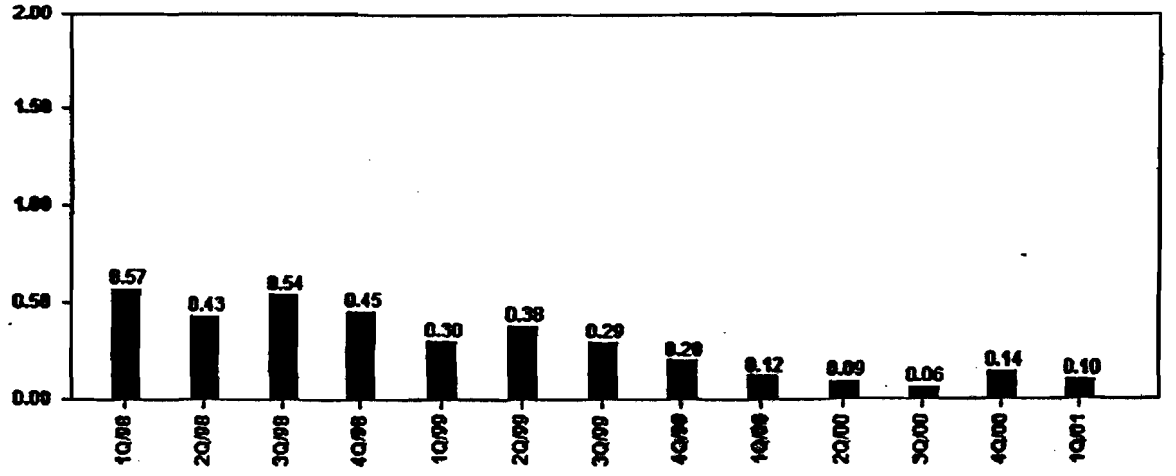
Descriptions

Safety System Unavailability, Residual Heat Removal System (BWR)



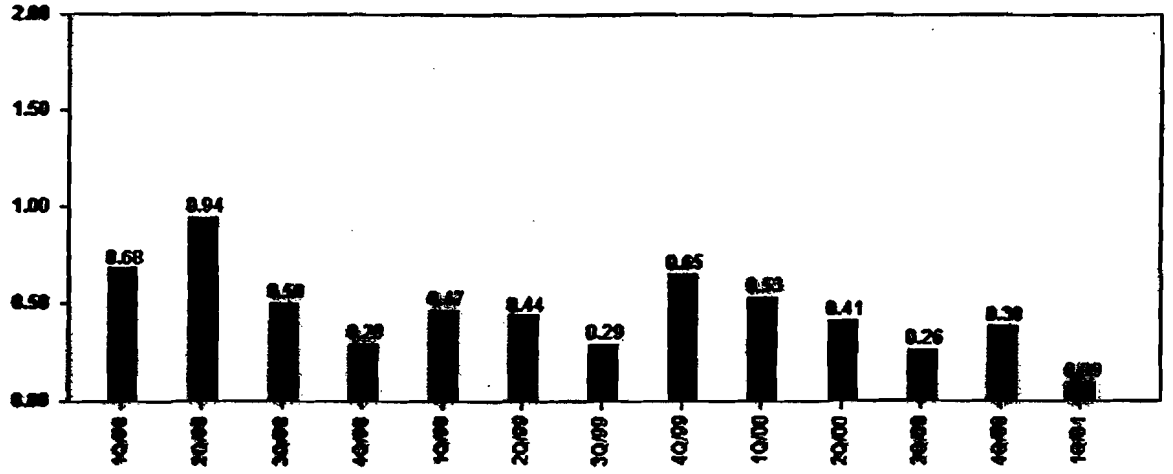
Descriptions

Safety System Functional Failures (PWR)



Descriptions

Safety System Functional Failures (BWR)

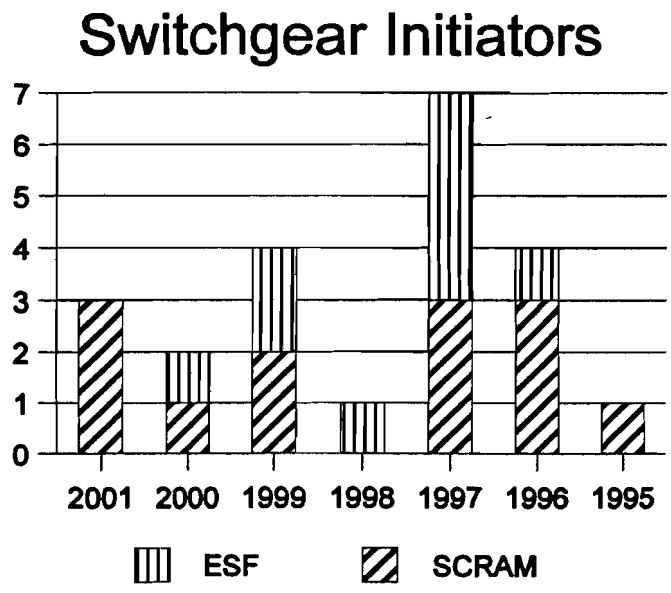


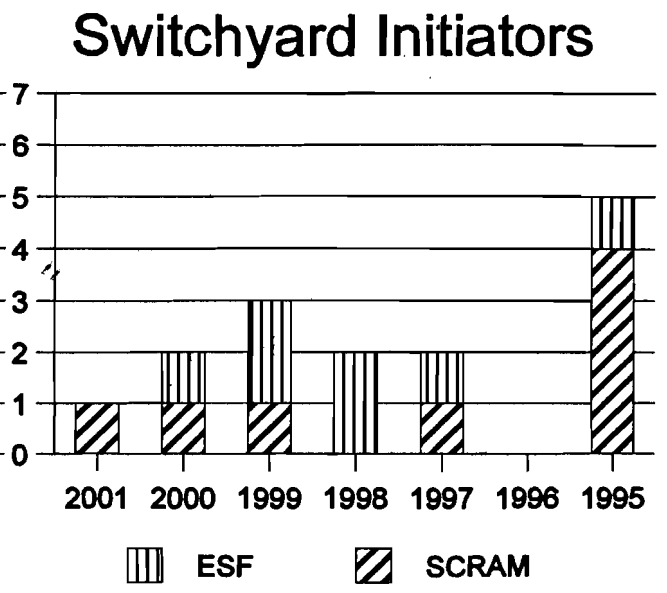
Descriptions

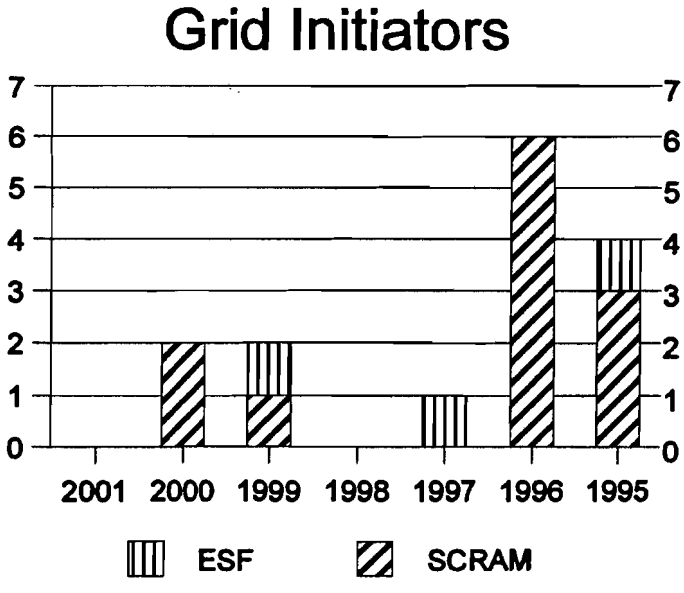
Initiating Events Industry Trends *Barrier Integrity Industry Trends*

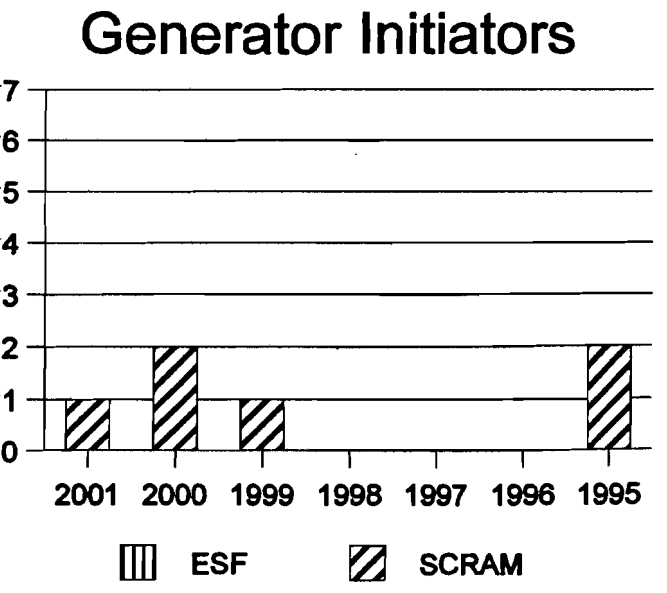
Last Modified: May 20, 2001

SCRAM AND ESF DATA
1995 TO FIRST QUARTER 2001



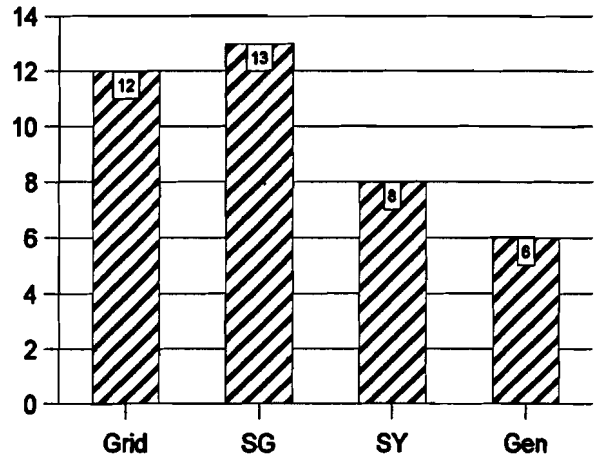




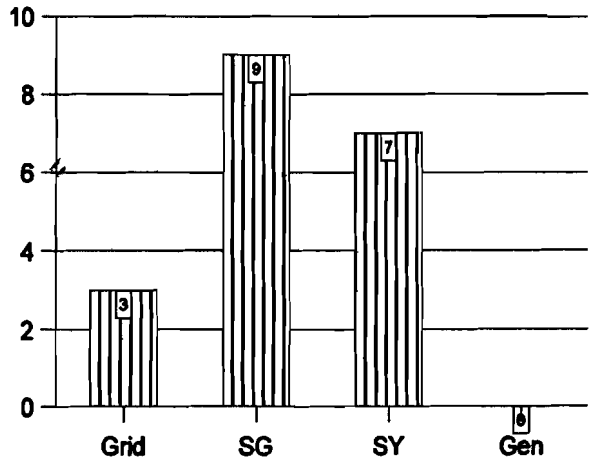


SCRAM AND ESF DATA: 1995 THROUGH FIRST QUARTER 2001

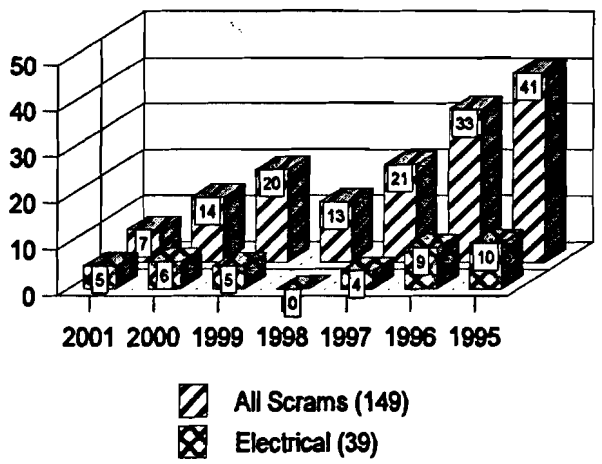
Electrical Scrams



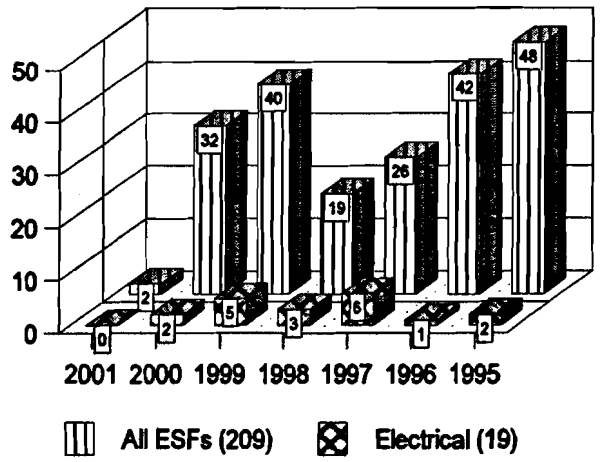
Electrical ESFs



Scrams

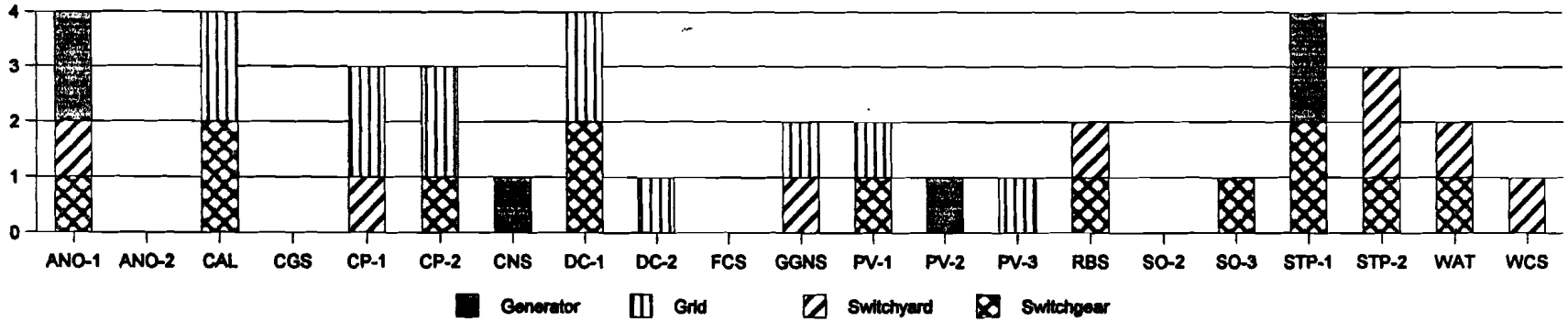


ESFs

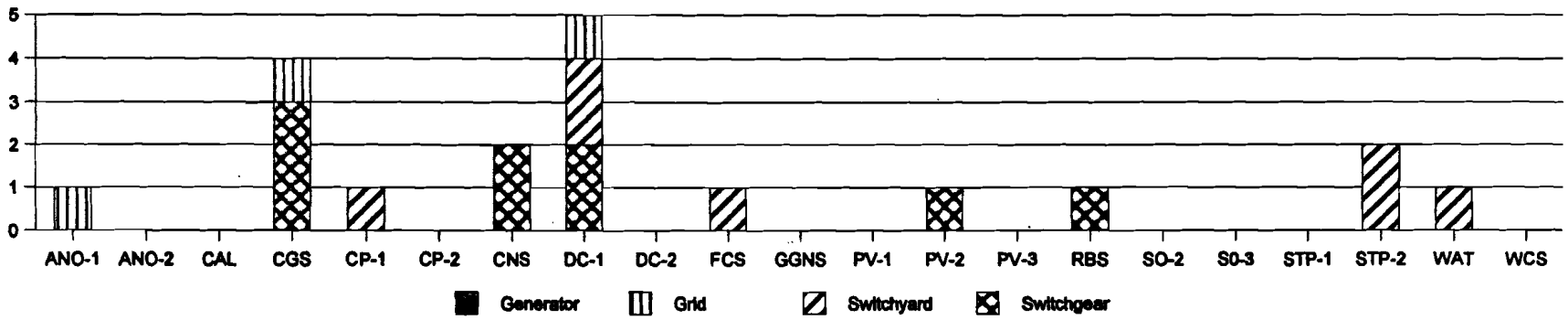


PLANT SCRAM AND ESF DATA

SCRAM DATA



ESF DATA



SONGS

UNIT 3

FIRE

Osterholtz

D

Overview

Electrical Distribution

Sequence of Events

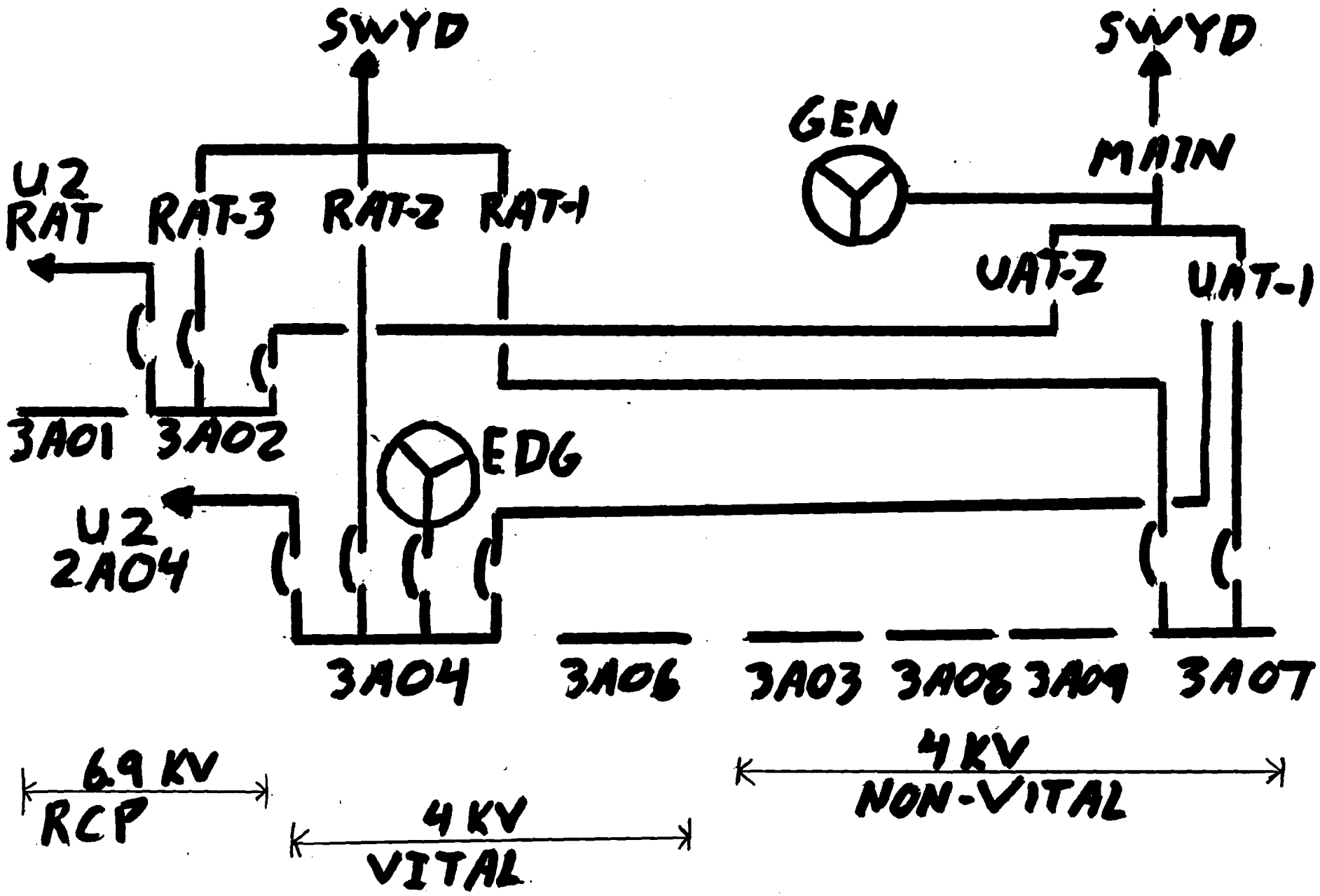
NRC Response
- Initial
- Followup

Component Failures
- UAT Breaker 3A0712
- Turbine DC lube oil pump
- Annunciators

Fire Fighting

CST overfill

Recovery


SWYD
SWYD
GEN
MAIN
U2
RAT
RAT-3
RAT-2
RAT-1
UAT-2
UAT-1
3A01
3A02
EDG
U2
2A04
3A04
3A06
3A03
3A08
3A09
3A07
69 KV
RCP
4 KV
VITAL
4 KV
NON-VITAL

REGION IV
Fire Protection Issues

Art Howell

Director, Division of Reactor Safety

Fire Protection Inspections

IP 71111.05 "FIRE PROTECTION"

- Resident Inspections
  - ➢ Quarterly Inspections
  - ➢ Annual Inspection (Fire Drill Observation)
  - ➢ 33 hours/year
- Region-based Inspections
  - ➢ Triennial Team Inspection
  - ➢ 200 hours/3 years

Fire Protection Inspections

RESIDENT INSPECTIONS

- Quarterly inspections - Residents tour 6-12 plant fire areas to observe:
  - ➢Control of transient combustible and ignition sources
  - ➢Material condition, operational lineup, operational effectiveness of fire protection (FP) systems
  - ➢Material condition and status of fire barriers
- Annually, observe a fire brigade drill

Fire Protection Inspections

REGION BASED INSPECTION

Inspection Focus:

- Verifying Licensee can Achieve and Maintain a Safe Shutdown (SSD) Condition for a Fire
  - ➢ Credited SSCs for SSD
  - ➢ Credited operator actions
  - ➢ Achievement of SSD for control room evacuation
  - ➢ Fire detection and suppression
  - ➢ Fire barriers
  - ➢ Emergency lighting and communications
  - ➢ Compensatory measures

Fire Protection Findings

OLD INSPECTION PROCEDURE 64704

- Region-based
  - ➤ 25 hours/3 years
  - ➤ Focused on:
    - Fire brigade/watch training and qualification
    - Fire protection (FP) systems
    - Control of transient combustibles
  - ➤ Typical findings
    - Fire brigade or fire watch members not qualified
    - Inadequate fire brigade drill or critique
    - Transient combustibles not on permit

Fire Protection Findings

NEW INSPECTION PROCEDURE 71111.05

- Resident Inspection
  - ➤ Typical Findings:
    - Fire doors inoperable or not latched
    - Inadequate fire brigade drill or fire watch
    - Transient combustibles not on permit

Fire Protection Findings

NEW INSPECTION PROCEDURE

- Region-based Inspections
  - ➤ Typical Findings:
    - Failure to meet separation requirements
    - Inadequate barriers
    - Inadequate emergency lighting
    - Inadequate suppression and detection

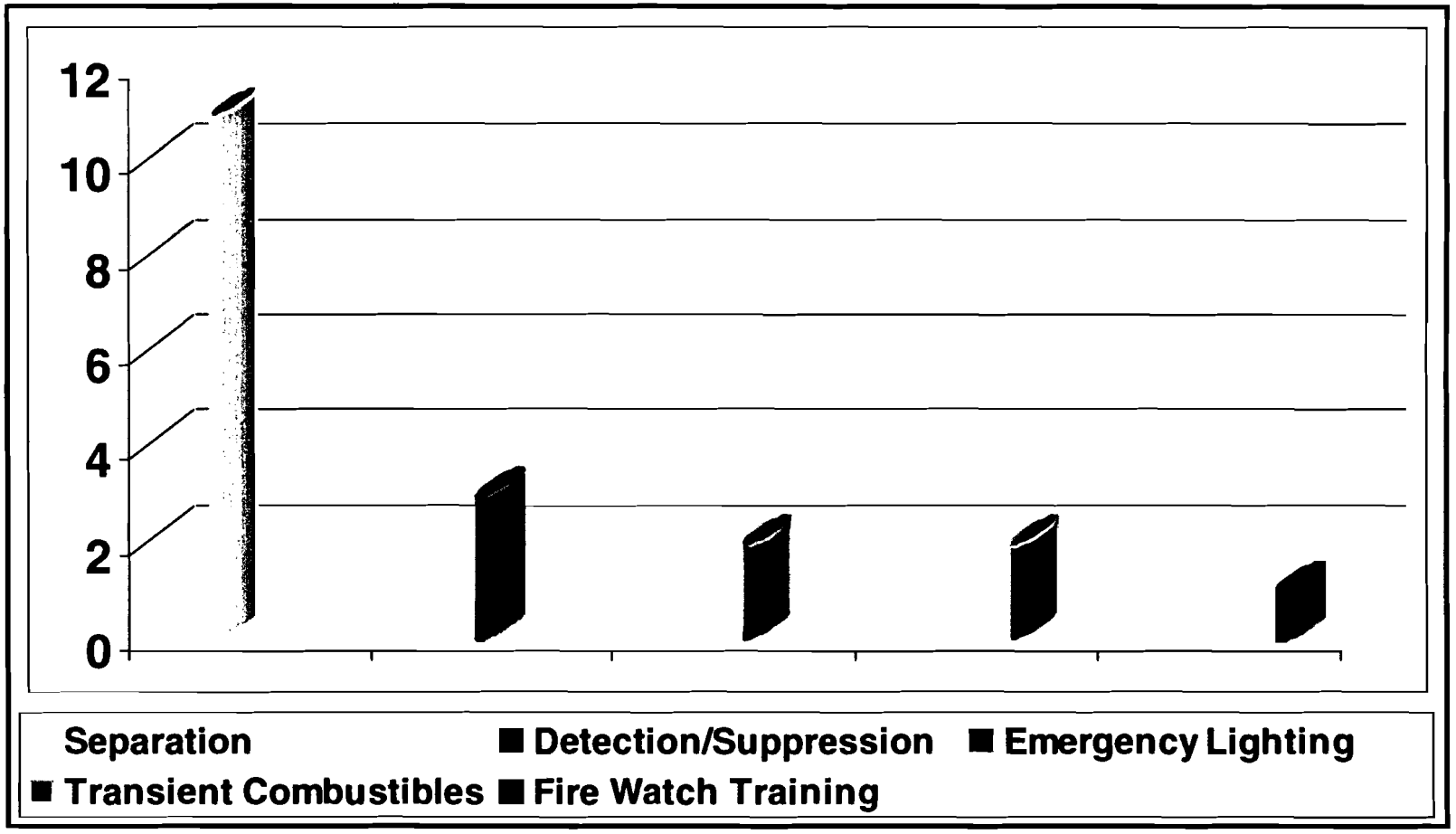
Fire Protection Inspections

- Triennial Inspections Performed to Date

➤ Fort Calhoun (Pilot)	January 2000
➤ Diablo Canyon	April 2000
➤ Columbia	May 2000
➤ Palo Verde	June 2000
➤ Callaway	August 2000
➤ Waterford	Sept. 2000
➤ Cooper	April 2001
➤ Riverbend	May 2001

Fire Protection Findings

FINDINGS GROUPED BY TYPE



Region IV's Role Under The NRC Continuity of Operations Plan

By Tom Andrews and John Pellet

The Critical Function

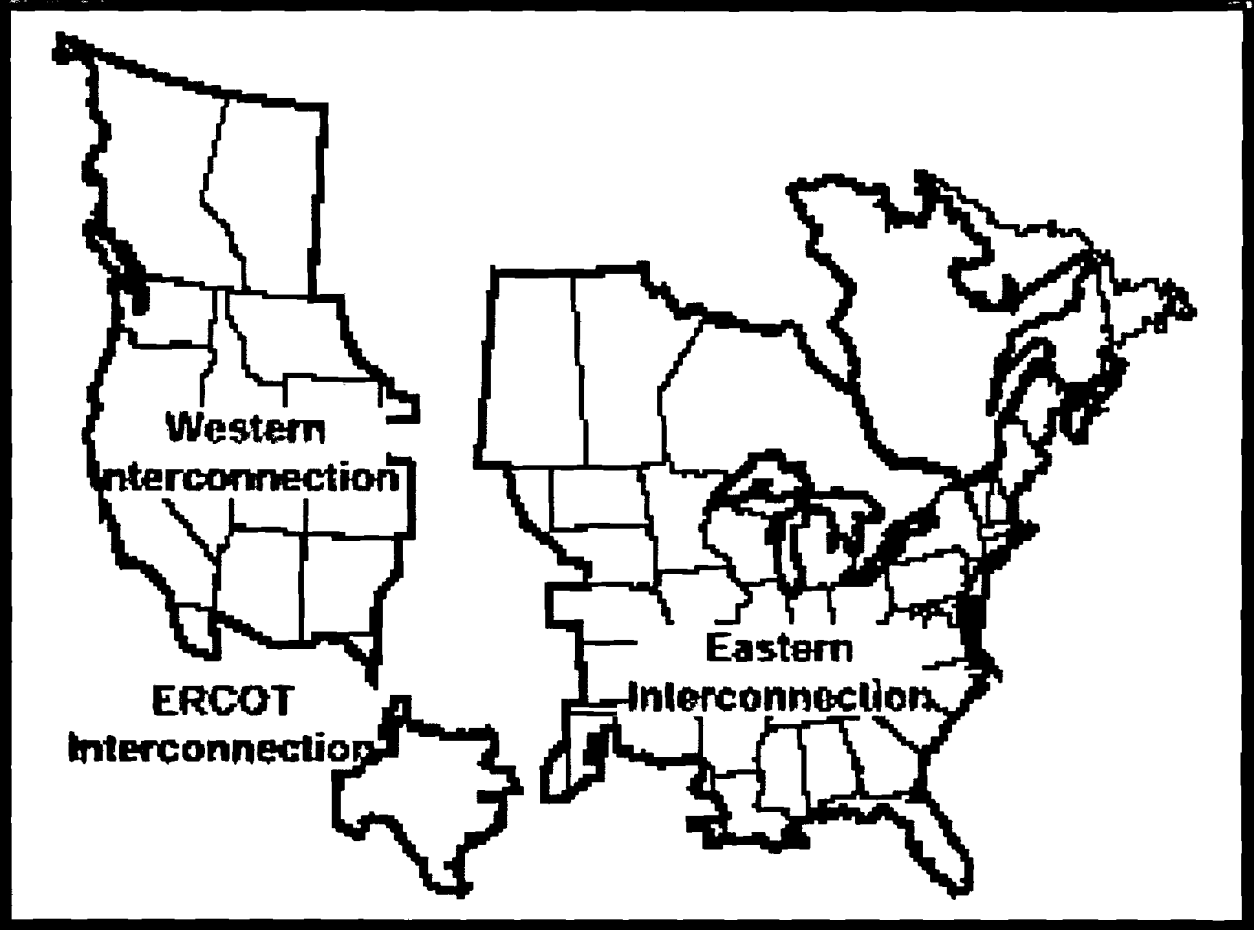
The NRC Continuity of Operations Plan identifies the emergency response function as the primary, mission-critical function of the agency

Why Region IV?

Region IV was selected as the backup for Headquarters for the following reasons:

1. Physical distance from Headquarters
2. Different weather (typically)
3. Different electrical grid

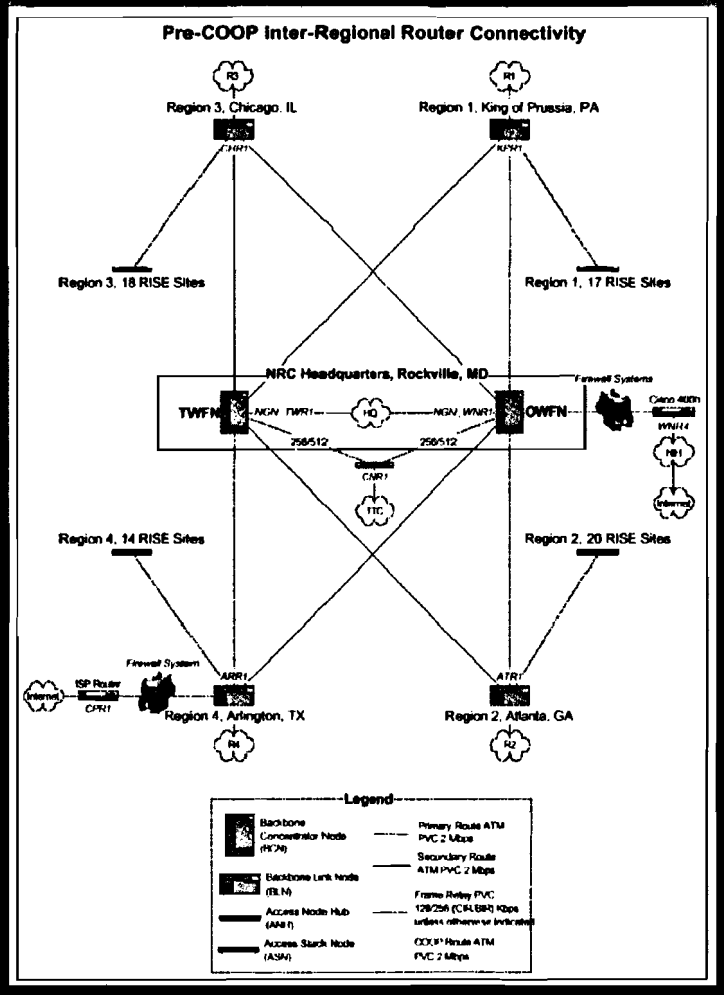
North American Electrical Grid



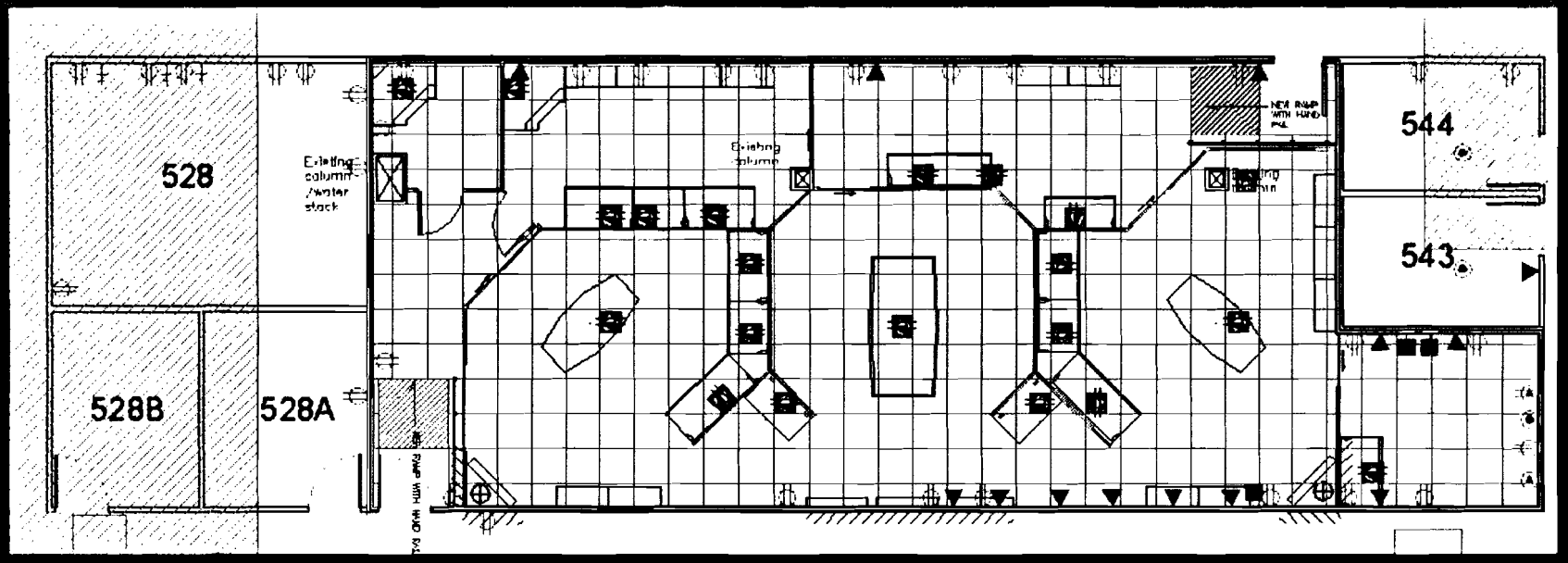
Support Functions Required for Continuity of Operations

- NRC local / wide area network
- Email (internal and external)
- External internet access

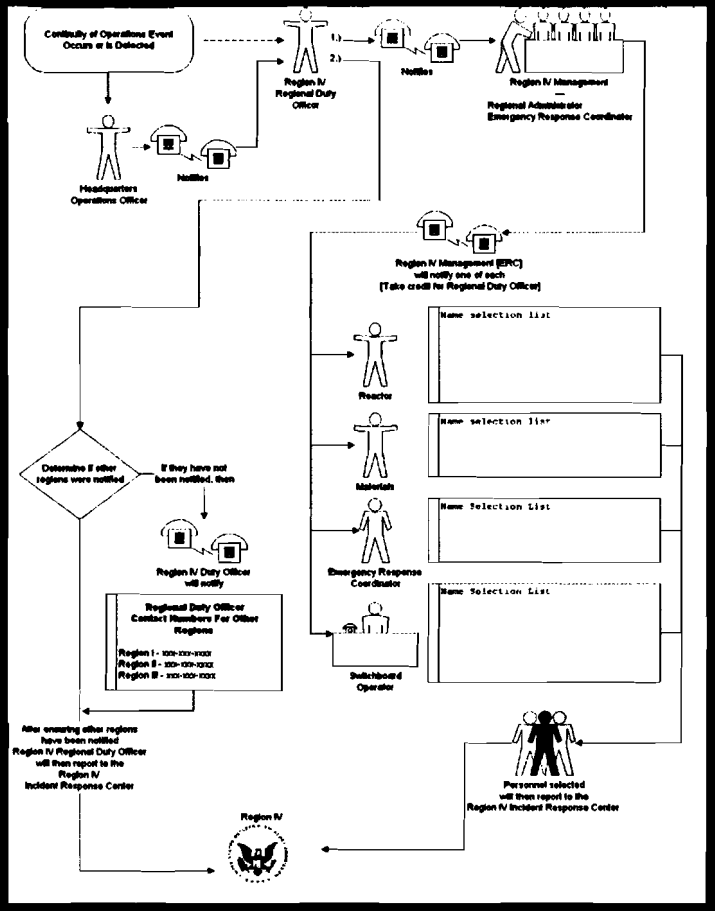
Continuity of Operations Network Connectivity



Region IV Incident Response Center Upgrade



Region IV Continuity of Operations Activation Process



Questions?

CALLAWAY ALARA

Gail M. Good, Chief
Plant Support Branch

Callaway ALARA

- Inspection Results
- Performance Problems
- Final Significance Determination & Violation
- Licensee Appeals
- NRC Response to Appeals
- Supplemental Inspection

Inspection Results

- Six jobs in Refueling Outage 10 exceeded dose projections by more than 50% and exceeded 5 person-rem per job

Six Jobs

- Scaffolding erection in reactor building
- Removal and installation of steam generator manway covers and inserts
- Eddy current testing/robotic plugging/stabilizing/electrosleeving

Six Jobs (Cont.)

- Health physics support for primary and secondary steam generator activities
- Foreign object search and retrieval
- Reactor coolant pump seal removal

Performance Problems

- Maintenance activities conducted soon after shutdown
- Maintenance activities conducted before steam generator bowl drains flushed
- Maintenance activities conducted on reactor coolant pumps and steam generators without water in secondary sides

Performance Problems
(Cont.)

- Maintenance activities conducted without sufficient practice training
- Maintenance activities performed with ineffective communications
- Collective doses increased from 1997 through 1999 - exceeded 135 person-rem industry median for pressurized water reactors

Final Significance Determination and Violation

- Three White Findings (low to moderate safety significance)
  - Two jobs accrued > 25 person-rem - 1st & 2nd White findings (scaffolding & eddy current/electrosleeving)
  - Other jobs accrued > 5 person-rem - 3rd White finding (manway covers, health physics support, seal work, foreign object)
  - Violation of 10 CFR Part 20.1101(b)

Licensee Appeals

- Imposed a staff position that was new or different (backfit)
- Denied the violation
- Significance determination process fatally flawed
- Significance determination of three White findings

NRC Response to Appeals

- Not a backfit (significance determination and violation)
- Violation occurred as described
- Occupational Radiation Safety Significance Determination Process not fatally flawed
- Region IV properly applied the significance determination process

Supplemental Inspection

- Provide assurance that root and contributing causes understood
- Independently assess extent of condition
- Provide assurance that corrective actions sufficient to address root causes and prevent recurrence

Supplemental Inspection Results

- Root Causes
  - Management's failure to establish expectations for ALARA
  - Management's failure to communicate ALARA priority
  - Culture that did not support ALARA concept
  - Administrative controls did not assure ALARA concerns would receive appropriate priority, consideration, and resolution

Supplemental Inspection Results (Cont.)

- Thorough evaluation of causes, correctly identified extent of condition, and corrective actions appropriate
- Some corrective actions not completed prior to the next outage
- Some corrective actions not incorporated into procedures to prevent recurrence

CALLAWAY GRID ISSUE

OVERVIEW

- Event Description
- Generic Concerns
- Callaway Specific Concerns

INSPECTION ISSUES

1. Callaway was supplying excess voltage to support the grid.

   - Weather Conditions
   - Power Outages
   - VAR Loading
   - Corrective Actions

2. Licensee procedures did not account for post trip voltages or instrument uncertainties.

- Post Contingency Model
- Procedure Requirements
- Corrective Actions

3. Computer alarm setpoints were nonconservative.

- Incorrect Setpoint Entry
- Equipment Deficiency
- Dispatch Alarm Setpoint
- Corrective Actions

4. Plant operators were not aware of the low voltage condition or operability requirements.

- Alarm recognition
- Contingency Model Output
- Corrective Actions

5. No agreement existed between the facility and energy supply operations personnel.

- Provider Agreement

6. Load flow analyses underestimated system loading conditions.

- Loading Assumptions
- Ongoing Loading Analyses
- Corrective Action

7. Information Notice was not dispositioned in accordance with licensee procedures.

- Corrective Actions

GENERIC COMMUNICATIONS

- Information Notice 2000-06
- Regulatory Information Summary 2000-24
- NEI Involvement